NUREG-0645
Vol. II

Report of the Bulletins and Orders Task Force

Appendices

Office of
Nuclear Reactor Regulation

U.S. Nuclear Regulatory
Commission

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NUREG-0645

Volume II

REPORT

OF THE

BULLETINS & ORDERS TASK FORCE

OF THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

NUREG-0645
Vol. II

Report of the Bulletins and Orders Task Force

Appendices

Manuscript Completed: January 1980
Date Published: January 1980

Division of Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

NUREG-0645

Volume II

APPENDICES

A

APPENDIX A

OFFICE OF INSPECTION AND ENFORCEMENT BULLETINS

Following the accident at Three Mile Island Power Plant, Unit 2, on March 28, 1979, the NRC Office of Inspection & Enforcement (I&E) issued three bulletins to licensees of operating power plants which required certain actions to be taken, based on reactor type:

IE Bulletin 79-05 (4/01/79) - Babcock & Wilcox reactors

IE Bulletin 79-06 (4/11/79) - All licensees of pressurized water reactors

IE Bulletin 79-08 (4/14/79) - All licensees of boiling water reactors

These bulletins were subsequently supplemented to provide new information, to clarify the bulletins, and/or to request other information or actions. These supplemental bulletins were:

IE Bulletin 79-05A - 4/05/79

IE Bulletin 79-05B - 4/21/79

IE Bulletin 79-05C & 79-06C - 7/26/79

IE Bulletin 79-06A - 4/14/79

IE Bulletin 79-06A, Revision No. 1 - 4/18/79

IE Bulletin 79-06B - 4/14/79

Copies of these bulletins follow.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

NUCLEAR INCIDENT AT THREE MILE ISLAND

Description of Circumstances:

On March 28, 1979 the Three Mile Island Nuclear Power Plant, Unit 2 experienced core damage which resulted from a series of events which were initiated by a loss of feedwater transient. Several aspects of the incident may have general applicability in addition to apparent generic applicability at operating Babcock and Wilcox reactors. This bulletin is provided to inform you of the nuclear incident and to request certain actions.

Actions To Be Taken By Licensees:

(Although the specific causes have not been determined for individual sequences in the Three Mile Island event, some of the following may have contributed).

For Babcock and Wilcox pressurized water reactor facilities with an operating license:

1. Review the description (Enclosure 1) of the initiating events and subsequent course of the incident. Also review the evaluation by the NRC staff of a postulated severe feedwater transient related to Babcock and Wilcox PWRs as described in Enclosure 2.

   These reviews should be directed at assessing the adequacy of your reactor systems to safely sustain cooldown transients such as these.

2. Review any transients of a similar nature which have occurred at your facility and determine whether any significant deviations from expected performance occurred. If any significant deviations are found, provide the details and an analysis of the significance and any corrective actions taken. This material may be identified by reference if previously submitted to the NRC.

3. Review the actions required by your operating procedures for coping with transients. The items that should be addressed include:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to ensure continued core cooling in the event that such voids are formed.

4. Review the actions requested by the operating procedures and the training instructions to assure that operators do not override automatic actions of engineered safety features without sufficient cause for doing so.

5. Review all safety related valve positions and positioning requirements to assure that engineered safety features and related equipment such as the auxiliary feedwater system, can perform their intended functions. Also review related procedures, such as those for maintenance and testing, to assure that such valves are returned to their correct positions following necessary manipulations.

6. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the containment to assure that undesired pumping of radioactive liquids and gases will not occur inadvertently.

   In particular assure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists and,

   b. Whether such systems are isolated by the containment isolation signal.

7. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events.

The detailed results of these reviews shall be submitted within ten (10) days of the receipt of this Bulletin.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Construction Inspection, Washington, D.C. 20555.

For all other operating reactors or reactors under construction, this Bulletin is for information purposes and no report is requested.

Approved by GAO, B180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosures:
1. Preliminary Notifications Three Mile Island - PNO-67 and 67A, B, C, D, E, F, G
2. Evaluation of Feedwater Transients w/attachment
3. List of IE Bulletins issued in last 12 months

PRELIMINARY NOTIFICATION

March 28, 1979

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-79-67

This preliminary notification constitutes EARLY notice of event of POSSIBLE safety or public interest significance. The information presented is as initially received without verification or evaluation and is basically all that is known by IE staff on this date.

Facility: Three Mile Island Unit 2
Middletown, Pennsylvania
(Docket No. 50-320)

Subject: REACTOR SCRAM FOLLOWED BY A SAFETY INJECTION AT THREE MILE ISLAND - UNIT 2

The licensee notified Region I at approximately 7:45 AM of an incident at Three Mile Island Unit 2 (TMI-2) which occurred at approximately 4:00 AM at 98% power when the secondary feed pumps tripped due to a feedwater polishing system problem. This resulted in a turbine trip and subsequent reactor trip on High Reactor Coolant Pressure. A combination of Feed Pump Operation and Pressurizer Relief - Steam Generator relief valve operation caused a Reactor Coolant System (RCS) cooldown. At 1600 psig, Emergency Safeguards Actuation occurred. All ECCS components started and operated properly. Water level increased in the Pressurizer and Safety Injection was secured manually approximately 5 minutes after actuation. It was subsequently resumed. The Reactor Coolant Pumps were secured when low net positive suction head limits were approached.

About 7:00 AM, high activity was noted in the RCS Coolant Sample Lines (approximately 600 mr/hr contact readings). A Site Emergency was then declared. At approximately 7:30 AM, a General Emergency was declared based on High Radiation levels in the Reactor Building. At 8:30 AM site boundary radiation levels were reported to not be significant (less than 1 mr/hr). The source of activity was stated to be failed fuel as a result of the transient, and due to a known previous primary to secondary leak in Steam Generator B.

The Region I Incident Response Center was activated at 8:10 AM and direct communications with the licensee and IE:Headquarters was established. The Response Team was dispatched at 8:45 AM and arrived at the site at 10:05 AM.

At 10:45 AM the Reactor Coolant System Pressure was being held at 1950 psig with temperature at 220°F in the cold leg. By 10:45 AM, radiation levels of 3 mr/hr had been detected 500 yards offsite.

CONTINUED

There is significant media interest at the present time because of concern about potential offsite radiation/contamination. The Commonwealth of Pennsylvania and EPA have been informed. Press contacts are being made by the licensee and NRC.

Contact: GKlingler, IE x28019 FNolan, IE x28019 SEBryan, IE x28019

Distribution: Transmitted H St ~~3:35~~ 3:45

Chairman Hendrie	Commissioner Bradford	S. J. Chilk, SECY
Commissioner Kennedy	Commissioner Ahearne	C. C. Kammerer, CA
Commissioner Gilinsky		(For Distribution)
Transmitted: MNBB 3:50	P. Bldg 3:40	J. G. Davis, IE
L. V. Gossick, EDO	H. R. Denton, NRR	Region I 3:55
H. L. Ornstein, EDO	R. C. DeYoung, NRR	
J. J. Fouchard, PA	R. J. Mattson, NRR	
N. M. Haller, MPA	V. Stello, NRR	(MAIL)
R. G. Ryan, OSP	R. S. Boyd, NRR	J. J. Cummings, OIA
H. K. Shapar, ELD	SS Bldg 3:52	R. Minogue, SD
	W. J. Dircks, NMSS	

PRELIMINARY NOTIFICATION

March 29, 1979

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-79-67A

This preliminary notification constitutes EARLY notice of event of POSSIBLE safety or public interest significance. The information presented is as initially received without verification or evaluation and is basically all that is known by IE staff on this date.

Facility: Three Mile Island Unit 2
Middletown, Pennsylvania (DN 50-320)

Subject: NUCLEAR INCIDENT AT THREE MILE ISLAND - UNIT 2

This supplements PNO-79-67 dated March 28, 1979.

As of 3:30 p.m., on March 28, 1979, the plant was being slowly cooled down with Reactor Coolant System (RCS) pressure at 450 psi, using normal letdown and makeup flow paths. The bubble has been collapsed in the A Reactor Coolant Loop hot leg, and some natural circulation cooling has been established. Pressurizer level has been decreased to the high range of visible indication, and some heaters are in operation. The secondary plant was being aligned to draw a vacuum in the main condenser and use the A Steam Generator for heat removal. The facility plans to continue a slow (3°F/hr) cooldown, until the Decay Heat Removal System can be placed in operation at 350 psi RCS pressure, 350°F RCS temperature in 15-18 hours.

As of 3:30 p.m., a plume approximately ½ mile wide and reading generally 1 mr/hr was moving to the north of the plant. The ARM's helicopter is being used to define the length of the plume. Airborne iodine levels of up to 1×10^{-8} uCi/ml have been detected in Middletown, Pennsylvania, which is located north of the site.

Media interest is continuing. The Commonwealth of Pennsylvania is being kept informed by plant personnel.

Contact: GKlingler, IE x28019 FNolan, IE x28019 SEBryan, IE x28019

Distribution: Transmitted H St [illegible] 10:30
Chairman Hendrie
Commissioner Kennedy
Commissioner Gilinsky

Transmitted: MNBB 10:25
L. V. Gossick, EDO
H. L. Ornstein, EDO
J. J. Fouchard, PA
N. M. Haller, MPA
R. G. Ryan, OSP
H. K. Shapar, ELD

Commissioner Bradford
Commissioner Ahearne
10:32
P. Bldg [illegible]
H. R. Denton, NRR
R. C. DeYoung, NRR
R. J. Mattson, NRR
V. Stello, NRR
R. S. Boyd, NRR
SS Bldg 10:28
W. J. Dircks, NMSS

S. J. Chilk, SECY
C. C. Kammerer, CA
(For Distribution)

J. G. Davis, IE
Region I 10:33

(MAIL)
J. J. Cummings, OIA
R. Minogue, SD

29 MAR 1979

PRELIMINARY NOTIFICATION

March 30, 1979

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-79-67B

This preliminary notification constitutes EARLY notice of event of POSSIBLE safety or public interest significance. The information presented is as initially received without verification or evaluation and is basically all that is known by IE staff on this date.

Facility: Three Mile Island Unit 2
Middletown, Pennsylvania (DN 50-320)

Subject: Nuclear Incident at Three Mile Island

Plant Status

Three Mile Island Unit 2 is continuing to remove decay heat through A-loop steam generator using one reactor coolant pump in that loop for coolant circulation. The reactor coolant pressure and temperature were stable and under control throughout the night of March 29. There has been some difficulty in maintaining coolant letdown flow due to resistance in the purification filters. The licensee notified IE at about 11:00 p.m. on March 29 that they expected to remain in this cooling mode for at least 24 hours.

The licensee's engineering staff was requested by NRR to obtain a better estimate of the volume of the noncondensible "bubbles" in the reactor coolant system. There are apparently two such bubbles one in the pressurizer that has been intentionally established for control of pressure and level, and one in the reactor vessel head caused by the accumulation of noncondensible gases from failed fuel and radiolytic decomposition of water. The estimate is to be obtained by correlating pressurizer pressure and level indications over the past hours of stable operation. The volume of the bubble in the reactor vessel is of interest in assuring that sufficient volume remains in the upper head for collection of more noncondensible gases arising from continued operation in the present cooling mode as well as to assess the potential for movement of the bubble during a switchover to decay heat removal operation.

The licensee believes it is prudent to remain in the present cooling mode due to the potential for leakage of highly radioactive coolant from the decay heat removal system into the auxiliary building, movement of noncondensible gases into the reactor coolant loop, and boiling in the core when the reactor coolant pump is shut down.

CONTINUED

Fuel Damage

Preliminary assessment of the extent of fuel damage from a reactor coolant sample taken at approximately 5:00 p.m. on March 29 indicates significant releases of iodine and noble gases from the fuel. A 100 milliliter sample taken from the primary coolant system via a letdown line was measured at about 1,000 R/hr on contact (70-80 R/hr at one foot and 10-30 R/hr at three feet). Preliminary analysis of a diluted sample in the IE mobile laboratory indicated fission product concentrations of about 8 x 10^5 microcuries per milliliter. The sample will be flown to Bettis Laboratory for further analysis.

Thermocouple readings of coolant temperature at the outlet of the instrumented fuel assemblies indicate potential local core damage, possibly in one quarter of the total of 177 fuel assemblies and generally in the center of the core. Of the 52 readings at 5:00 a.m. on March 30, one was above the coolant saturation temperature of about 550°F, 7 were above 350°F, and 2 were off-scale, indicating temperatures higher than 700°F. Upon request of NRR, Babcock and Wilcox is developing a procedure for use by the licensee in taking direct potentiometer readings from the off-scale thermocouples since the temperature scale limitation of 700°F is controlled by the process computer, not the thermocouple itself.

Reactor Coolant System (RCS) Parameters

The RCS parameters have remained relatively stable during the period. Gradual RCS cooldown continued to about 1:30 a.m., March 30, when temperature was slightly increased to allow additional margin between RCS operating parameters and Technical Specification minimum pressurization limits. Following are the primary system parameters over this period:

	10:00 a.m. 3/29/79	7:00 p.m. 3/29/79	12:01 a.m. 3/30/79	3:00 a.m. 3/30/79	5:00a.m. 3/30/79
Pressurizer Level (inches)	348	321	326	342	354
Pressurizer Pressure (psi)	863	945	1023	1055	1053
Pressurizer Temperature (°F)	529	542	551	556	557
Loop A Core Inlet Temperature (°F)	281	277	275	278	274
Loop B Core Inlet Temperature (°F)	281	277	275	278	274

CONTINUED

Environmental Status

Two aerial surveys were conducted during the evening of March 29. The first flight was made about 8:15 p.m. during which measurements were taken in a circle around the site with a radius of about eight miles. No defined plume of radioactivity was detected, but residual pockets of radioactivity were identified at various points where the measured levels ranged from .025 to .050 millircentgens per hours. (Natural background levels are about .005 to .015 millircentgens per hour.) During the second flight, at about 10:30 p.m., a plume was detected northwest of the plant with a width equal to and confined within the boundaries of the river. The plume was touching down about one mile from the plant at Hill Island and then splitting into two parts - one on each side of Hill Island. Measurements at the east shoreline of the river, opposite Hill Isalnd indicated about four milliroentgens per hour and at the shoreline on mile north of Hill Island near Olmstead Air Force Base about one milliroentgen per hour. Additional measurements at five miles from the plant were on the order of .010 milliroentgens per hour and are in agreement with the earlier flight.

During the early morning hours of March 30, an NRC monitoring team took radiation measurements from a vehicle traveling both sides of the Susquehanna River from 10 miles south of Three Mile Island to 4 miles north. Radiation levels were highest near Cly, a community just south of the facility on the west side of the river. The level at Cly was 0.15 milliroentgen per hour. All other locations had levels less than 0.05 milliroentgens per hour.

Other Information

At approximately 4:00 p.m. on March 29, two employees of Metropolitan Edison Co. received radiation exposures in excess of the quarterly limit of 3 rems. The employees, an operator and a chemist, entered the auxiliary building to collect a sample of primary coolant. Present estimates are that the operator received 3.1 rems and the chemist 3.4 rems.

The licensee released less than 50,000 gallons of slightly contaminated industrial wastes on March 29, 1979. This release was terminated at NRC request at approximately 6:00 p.m., March 29, 1979, because of concerns expressed by state representatives. At about 12:15 a.m. on March 30, NRC gave the licensee permission to resume releases of the slightly contaminated industrial wastes to the Susquehanna River. This action was coordinated with the office of the Governor of Pennsylvania and a press rlease was issued by the State. Representatives of the news media expressed concern that they were not informed of the planned resumption of the release prior to permission having been granted.

CONTINUED

At 8:40 a.m., on March 30 the licensee began venting from the gaseous waste tanks. The impact of this operation is not yet known.

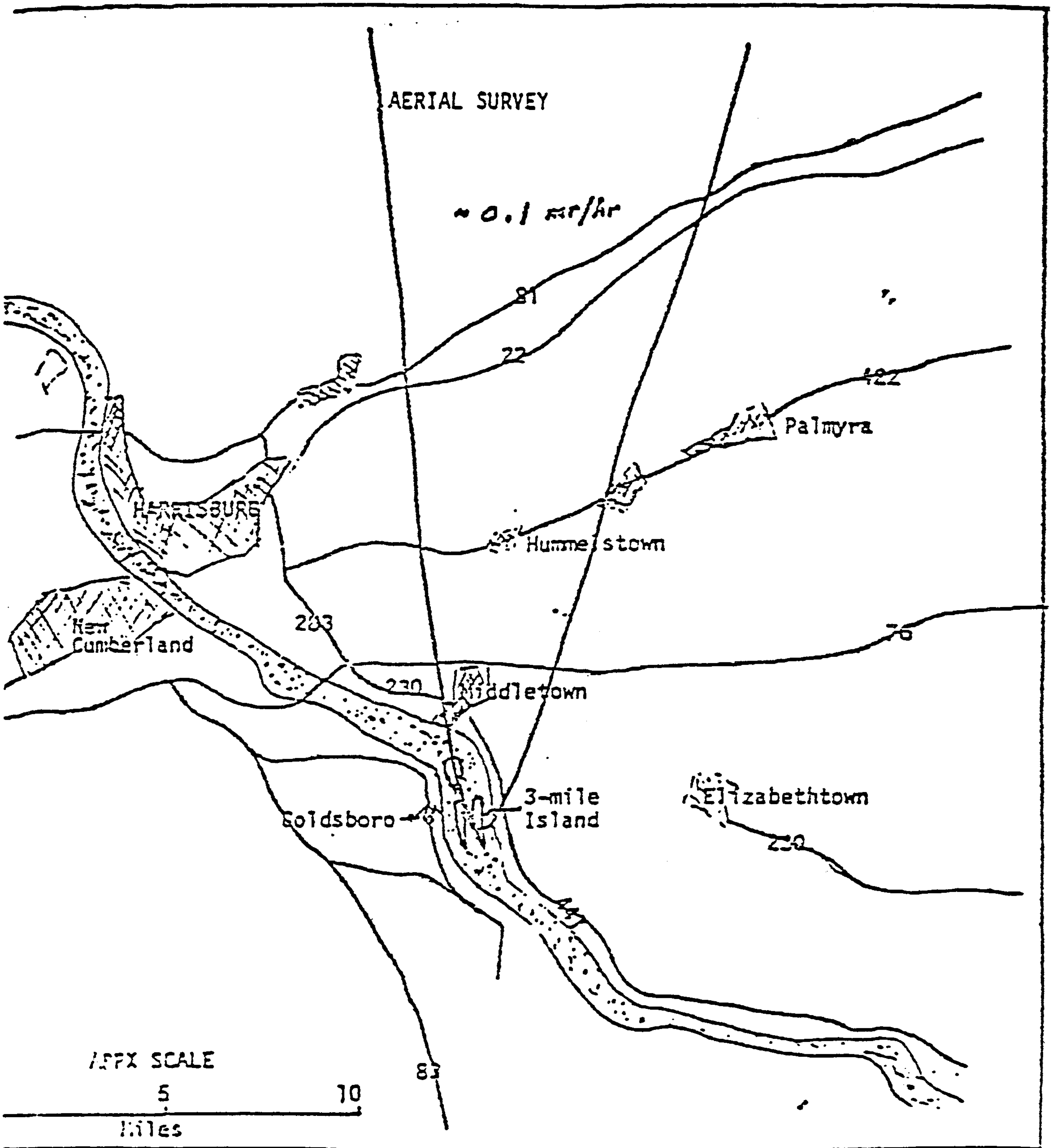
Contact: DThompson, IE x28111; EJordan, IE x 28111

Distribution: Transmitted H St 9:50

Chairman Hendrie	Commissioner Bradford	S. J. Chilk, SECY
Commissioner Kennedy	Commissioner Ahearne	C. C. Kammerer, CA
Commissioner Gilinsky		(For Distribution)
Transmitted: MNBB 10:02	P Bldg 10:15	J. G. Davis, IE
L. V. Gossick, EDO	H. R. Denton, NRR	Region ____ ____
H. L. Ornstein, EDO	R. C. DeYoung, NRR	
J. J. Fouchard, PA	R. J. Mattson, NRR	
N. M. Haller, MPA	V. Stello, NRR	(MAIL)
R. G. Ryan, OSP	R. S. Boyd, NRR	J. J. Cummings, OIA
H. K. Shapar, ELD	(SS Bldg ______	R. Minogue, SD
	W. J. Dircks, NMSS	

Attachments (7):
Aerial Survey (6)
Ground-Level Survey (1)

PRELIMINARY NOTIFICATION



28, 1979 4:30 p.m.

Plume in a N to NE direction, about 30° sector.
Primarily Xe-133. At distance of about 16 miles,
radiation measurements in the plume were about 0.1 mr/hr.

March 30, 1979

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-79-67C

This preliminary notification constitutes EARLY notice of event of POSSIBLE safety or public interest significance. The information presented is as initially received without verification or evaluation and is basically all that is known by IE staff on this date.

Facility: Three Mile Island Unit 2
Middletown, Pennsylvania (DN 50-520)

Subject: NUCLEAR INCIDENT AT THREE MILE ISLAND

Plant Status

There have been intermittent uncontrolled releases of radioactivity into the atmosphere from the primary coolant system of Unit 2 of the Three Mile Island Nuclear Power Plant near Harrisburg, Pennsylvania. The licensee is attempting to stop the intermittent gaseous releases by transferring the radioactive coolant water into the primary containment building. The levels of radioactivity being measured have been as high as 20 to 25 millirem per hour in the immediate vicinity of the site at ground level. Off-site levels were a few milliroentgen.

At about 11:30 a.m. EST, the Chairman of the NRC has suggested to Governor Thornburg of the Commonwealth of Pennsylvania that pregnant women and pre-school children in an area within five miles of the plant site be evacuated. Members of the NRC technical staff are at the site and efforts to reduce the temperatures of the reactor fuel are continuing. These temperatures have been coming down slowly and the final depressurization of the reactor vessel has been delayed. There is evidence of severe damage to the nuclear fuel. Samples of primary coolant containing high-levels of radioiodine and instruments in the core indicate high fuel temperatures in some of the fuel bundles, and the presence of a large bubble of non-condensible gases in the top of the reactor vessel.

Because of these non-condensible gases, the possiblity exists of interrupting coolant flow within the reactor when its pressure is further decreased and the contained gases expand. Several options to reach a final safe state for the fuel are under consideration. In the meantime, the reactor is being maintained in a stable condition.

Contact: SEBryan, IE x28188 ELJordan, IE x28188

Distribution: Transmitted H St 4.15

Chairman Hendrie
Commissioner Kennedy
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
S. J. Chilk, SECY
C. C. Kammerer, CA
(For Distribution)

Transmitted: MNBB ____
L. V. Gossick, EDO
H. L. Ornstein, EDO
J. J. Fouchard, PA
N. M. Haller, MPA
R. G. Ryan, OSP
H. K. Shapar, ELD

P. Bldg 4.17
H. R. Denton, NRR
R. C. DeYoung, NRR
R. J. Mattson, NRR
V. Stello, NRR
R. S. Boyd, NRR
SS Bldg ____
W. J. Dircks, NMSS

J. G. Davis, IE
Region I 4.30

(MAIL)
J. J. Cummings, OIA
R. Minogue, SD

IMMEDIATE

PRELIMINARY NOTIFICATION

March 30, 1979

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-79-67D

This preliminary notification constitutes EARLY notice of an event of POSSIBLE safety or public interest significance. The information presented is as initially received without verification or evaluation and is basically all that is known by IE staff on this date.

Facility: Three Mile Island Unit 2
Middletown, Pennsylvania (DN 50-320)

Subject: NUCLEAR INCIDENT AT THREE MILE ISLAND

Plant Status

Gaseous radioactivity from the primary coolant system letdown has been contained in waste gas decay tanks since the last gaseous release at approximately 2:50 p.m. March 30, 1979. At the present reactor coolant letdown rate of approximately 20 gpm it may be necessary to make a planned release of radioactive gas tomorrow to prevent gas decay tank relief valve operation at its setpoint of 100 psi. The licensee has installed a temporary line from the gas decay system back to reactor containment which is under evaluation before being placed in operation. Containment pressure is being maintained slightly negative (-1 psi) as a result of fan cooler operation.

Reactor coolant temperature measured at fifty-two locations at the outlet of the core have continued to come down slowly. Three Outlet temperature instruments continue to indicate above saturation temperature.

The NRC staff was informed by the licensee on Friday morning that examination of containment pressure data for March 28 indicates a pressure spike up to approximately 30 psi occurred at approximately 1:50 p.m. NRC personnel are evaluating the possibility that a hydrogen explosion was the cause of the containment internal pressure spike.

The reactor coolant path is through one reactor coolant pump and one steam generator. The steam generator is being fed by an auxiliary feed-pump. Several options for depressurizing the reactor and continuing cooldown via the residual heat removal system are under consideration.

CONTINUED

The volume of non-condensible gases in the reactor vessel has been estimated to be approximately 1000 to 1500 cubic feet at 1000 psi. This volume is estimated to result in a water level of several feet over the top of the fuel. The rate of growth of the bubble in the reactor vessel is estimated to be less than 50 cubic feet per day at 1000 psi.

The Director of the Office of Nuclear Reactor Regulation, the Director of the Region I Office of Inspection and Enforcement and the Director of the Division of Operating Reactors arrived at the site at approximately 2 p.m. today to direct NRC activities at the site and site vicinity. Representatives of HEW and EPA are providing coordination and assistance to the NRC at the Incident Response Center.

NRC personnel assembled at the TMI site and vicinity in addition to the upper management personnel consist of the following:

	RI	RII	RIII	Hq
Reactor Inspectors (IE)	8	5	4	
Health Physicists (IE)	12	12	10	
Health Physicists (SP)				4
Public Affairs	1	1		1
Reactor System Analysts (NRR)				13
Radition Waste Specialists (NRR)				4
Health Physicists (NRR)				6
Operating Licensing (NRR)				2
Total Staff			83	

CONTINUED

The following equipment has been assembled at or near the site for support of NRC operations:

Equipment	Location
1 NRC Instrument Van with 2 telephone lines	Observation Center
1 NRC Office Van	"
1 Office Trailer (Supplied by Licensee)	"
200 Hand-Held Portable Radios from US Forest Service	
Portable Health Physics Instrumentation	
3 Helicopters from DOE for survey and support	
2 Laboratory Vans DOE/Bettis	

A sophisticated communications pod from DOE/NEST will arrive tomorrow.

ENVIRONMENTAL STATUS:
At approximately 3 P.M. on March 30, 1979, NRC analysis of eight vegetation samples from the offsite areas showed no detectable activity. At 5.30 P.M. the Pennsylvania State Radiation Health Department reported that environmental water and air samples collected in the vicinity of the Three Mile Island Plant showed no detectable activity except for some Xenon-133 and Xenon-135. Milk sample analysis showed no activity levels above background.

Offsite ground level gamma surveys in the Middletown and Goldsboro areas between 3:00 and 6:00 P.M. on March 30, ranged from .01 to 1 milliroentgens per hour. An aerial survey was made by helicopter from 4:00 - 6:00 P.M. on March 30, the site was surveyed in concentric circles at approximately one mile intervals and at a height of 300 to 1,000 feet. The highest radiation readings were over the site and measured 8 to 10 milliroentgens per hour. In the plume the highest radiation readings were 6 to 8 milliroentgens per hour. The plume followed the river in a northwesterly direction and was not detectable beyond five to six miles from the site. Site ground level surveys conducted between 7:30 - 8:00 P.M. ranged from .01 to 1.8 milliroentgens per hour.

CONTINUED

At 4 P.M. March 30, upper level winds were from the southeast. Forecast indicates precipitation in the form of thunderstorms moving in after 12 midnight, March 30. At 5:00 P.M. winds onsite at Three Mile Island were reported at 2 to 3 miles per hour generally from east to west.

Contact: EMHoward, IE x28111; EJordan, IE x28111

Distribution: Transmitted H St 1:10 a 3/31

Chairman Hendrie
Commissioner Kennedy
Commissioner Gilinsky
Commissioner Bradford
Commissioner Ahearne
S. J. Chilk, SECY
C. C. Kammerer, CA
(For Distribution)

Transmitted: MNBS 1:17
L. V. Gossick, EDO
H. L. Ornstein, EDO
J. J. Fouchard, PA
N. M. Haller, MPA
R. G. Ryan, OSP
H. K. Shapar, ELD

P Bldg 1:25
H. R. Denton, NRR
R. C. DeYoung, NRR
R. J. Mattson, NRR
V. Stello, NRR
R. S. Boyd, NRR
(SS Bldg 1:33
W. J. Dircks, NMSS

J. G. Davis, IE
Region ____ ____

(MAIL)
J. J. Cummings, OIA
R. Minogue, SD

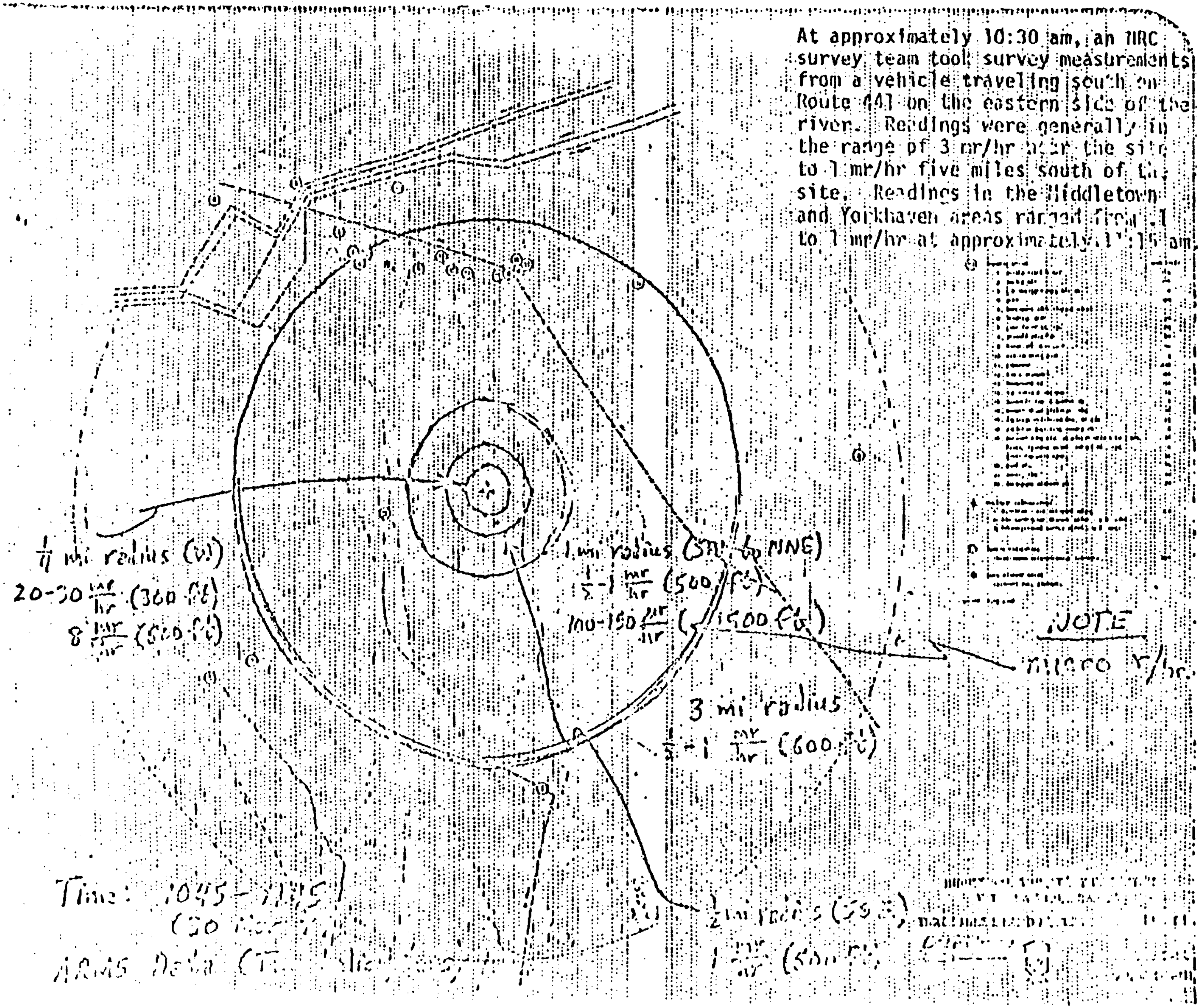
White House Situation Room 12:55 a.m. 3/31/79
EPA ________
FDA/BRH ________
DOE/EOC 2:05 am 3/31

Attachment (1)
Radiation Survey Map

IMMEDIATE

PRELIMINARY NOTIFICATION

At approximately 10:30 am, an NRC survey team took survey measurements from a vehicle traveling south on Route 441 on the eastern side of the river. Readings were generally in the range of 3 mr/hr near the site to 1 mr/hr five miles south of the site. Readings in the Middletown and Yorkhaven areas ranged from .1 to 1 mr/hr at approximately 11:15 am.



IMMEDIATE

PRELIMINARY NOTIFICATION

March 31, 1979

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-79-67E

This immediate preliminary notification constitutes an update of event of safety and public interest significance. The information presented is as initially received without verification or evaluation and is basically all that is known by NRC staff at this time.

Facility: Three Mile Island Unit 2
Middletown, Pennsylvania (DN 50-320)

Subject: NUCLEAR INCIDENT AT THREE MILE ISLAND

Plant Status

Reactor cooling continues using the 1A main reactor coolant pump with steam generator A steaming to the main condenser. Changes to this cooling method are not planned for the near term. An operability status of equipment is being compiled for use as backup in the event of failure of existing operating equipment.

The hydrogen recombiner is in an operable status; however, shielding of its piping and components is not fully installed and is presently considered inadequate. Lead for shielding has been located and will be moved to the site on an expedited basis. Calculations of hydrogen in containment show that the present concentration is less than 4%, the staff's limit on allowed concentration to ensure an explosive mixture is not obtained. Attempts are being made to obtain a containment atmosphere sample.

The waste gas decay tank pressures were 80 psi at 10:15 p.m. on March 30 and had been relatively constant for about five hours. The tank is set to relieve pressure at 100 - 110 psi. The radiation field (60 R/hr at contact) prevents resetting relief points.

Reactor coolant temperatures measured by incore thermocouples at 52 locations presently show only one location above saturation temperature. Temperatures in the core as measured from outlet thermocouples are gradually decreasing. Other system parameters are remaining stable.

Environmental Status

Three ARMS flights of one-hour length were conducted beginning at 9:30 p.m. on March 30, and at midnight and 3:00 a.m. on March 31. At a

CONTINUED

distance of one mile from the plant, maximum readings ranged from 0.5 milliroentgens per hour (mr/hr) to 1.5 mr/hr. At the 18 mile point, readings of 0.1 to 0.2 mr/hr were obtained during the two earlier surveys and 0.5 mr/hr during the latest. Flights are being made at approximately three hour intervals.

Offsite ground level gamma surveys in the Middletown area and north. between 9:30 p.m. on March 30 and 1:00 a.m. on March 31, indicated levels from 0.2 to 0.5 mr/hr. These measurements were taken in the general direction of the plume measured in aerial surveys.

At 3:00 p.m. on March 29, (prior to the releases of March 30) the licensee pulled thermoluminescent dosimeters from 17 fixed positions located within a 15 mile radius of the site. The dosimeters had been in place for three months and had been exposed for about 32 hours after the incident. Only two dosimeters showed elevated exposures above normal levels. The highest reading observed was on Three Mile Island, 0.4 miles north of the reactor at the North Weather Station. At this location, the quarterly accumulated exposure was 81 mr, approximately 65 mr above the normal quarterly exposure rate. The other high exposure was observed at North Bridge, 0.7 miles NNE of the reactor at the entrance to the site. At this location, the total quarterly accumulated exposure was 37 mr or approximately 22 mr above the normal quarterly exposure rate.

During the evening milking hours on March 30, milk samples were collected by the Pennsylvania Department of Environmental Resources at the following locations:

Harrisburg (2 sites)
York
Middletown
Bainbridge
Etters

Analyses showed no detectable radioiodine. The cows had been fed on stored feed but had been outside for exericse.

The Pennsylvania Department of Environmental Resources also collected water samples at filtration plants at Columbia, PA (for the City of Lancaster) and Wrightsville on March 30 in the morning and early afternoon. Both sample points are downstream of Three Mile Island. No detectable activity was found.

CONTINUED

Contact: DThompson, IE x28111 NCMoseley, IE x28111

Distribution: Transmitted H St 9:04

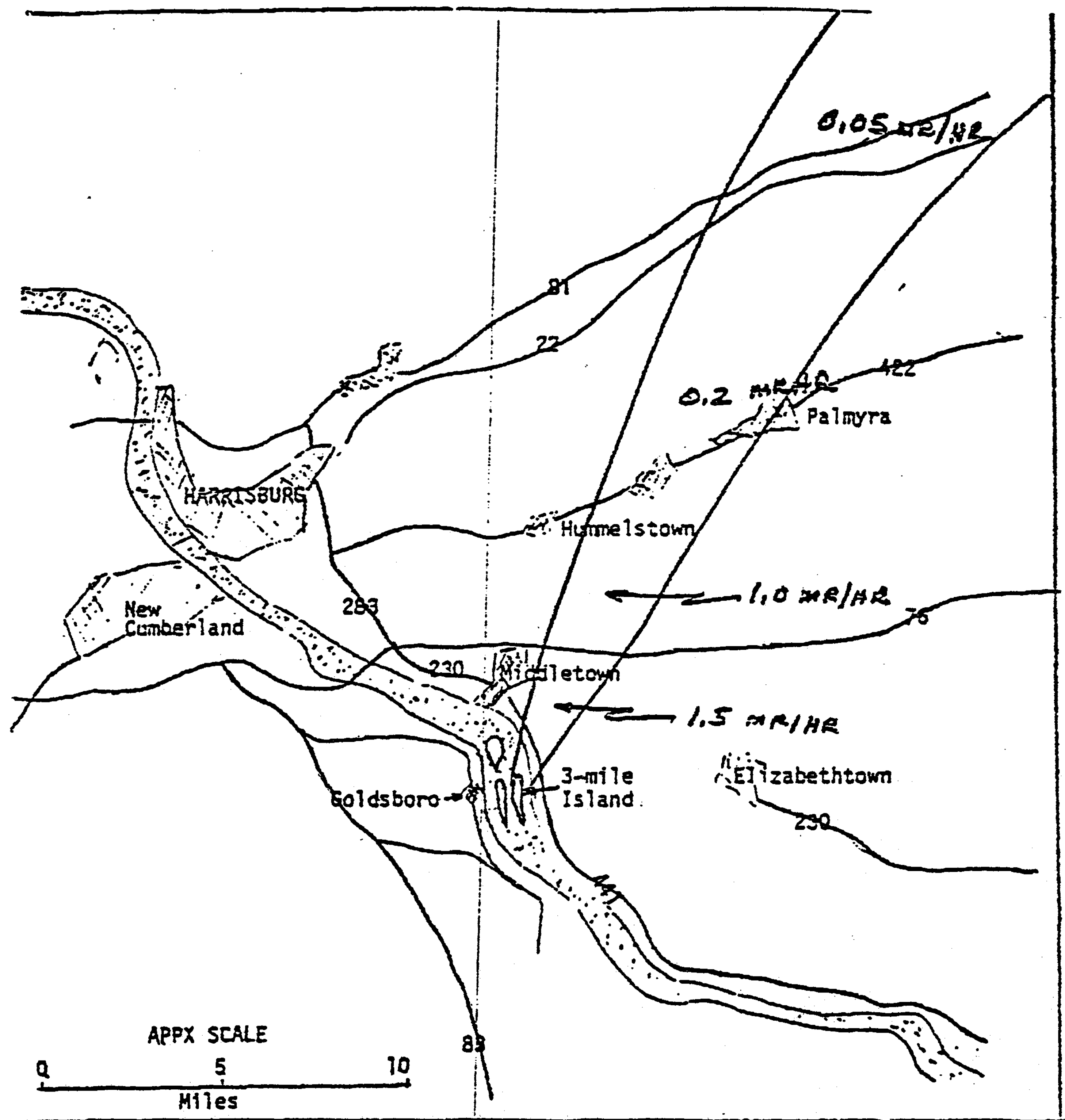
Chairman Hendrie	Commissioner Bradford	S. J. Chilk, SECY
Commissioner Kennedy	Commissioner Ahearne	C. C. Kammerer, CA
Commissioner Gilinsky		(For Distribution)
Transmitted: MNBB 9:08	P. Bldg 9:15	J. G. Davis, IE
L. V. Gossick, EDO	H. R. Denton, NRR	Region I 9:24
H. L. Ornstein, EDO	R. C. DeYoung, NRR	
J. J. Fouchard, PA	R. J. Mattson, NRR	
N. M. Haller, MPA	V. Stello, NRR	(MAIL)
R. G. Ryan, OSP	R. S. Boyd, NRR	J. J. Cummings, OIA
H. K. Shapar, ELD	SS Bldg 9:20	R. Minogue, SD
	W. J. Dircks, NMSS	

White House Situation Room ________
EPA ______
FDA/BRH ______
DOE/EOC ______

Attachment (1)
Radiation Survey Map

IMMEDIATE

PRELIMINARY NOTIFICATION



March 31, 1979 4:00 a.m. AERIAL SURVEY plume direction and radiation readings shown above.

March 31, 1979 1:00 a.m. All ground level readings were less than 0.1 mr/hr. measurements made in vehicle travelling route 441 from about ten miles south of plant to route 76 and south along roads on the west side of the river.

IMMEDIATE

PRELIMINARY NOTIFICATION

March 31, 1979

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-79-67F

This preliminary notification constitutes summary information of an event of safety or public interest significance. The information presented is a summary of information as of 5:30 pm date 3/31/79.

Facility: Three Mile Island Unit 2
Middletown, Pennsylvania (DN 50-320)

Subject: NUCLEAR INCIDENT AT THREE MILE ISLAND

Plant Status

There has been no change in the method of cooling the reactor since the previous report (PNO-79-67E). Reactor coolant temperatures measured by incore thermocouples at 52 locations have continued to decrease. At present none of the temperature readings is above saturation temperature for this pressure (554°F). System parameters remain stable. There has been a slight drop in pressurizer level from 215 to 191 inches.

Efforts continue to complete installation of components and piping on the hydrogen recombiner. Approximately 220 tons of lead shielding in various shapes and forms has arrived, or is on the way, to the site. Lead shielding is being installed around the recombiner. A decision to use the recombiner has not yet been made. Two samples of containment atmosphere have been analyzed which show hydrogen concentrations of 1.7 and 1.0%.

Efforts continue to estimate the volume of the noncondensible gas bubble above the core. Licensee calculations of the size of the bubble at 2:40 pm was 800 cubic feet at 875 psig. At about 4:20 pm this was recalculated by the licensee to be 621 cubic feet at 875 psig. This is being further evaluated.

Environmental Status

Three ARMS flights were conducted at about 6:00 a.m., 9:00 a.m., and 12:00 noon on March 31. All flights reflected a rather stable situation. Maximum readings in the plume were from 1.5 to 2.5 milliroentgens per hour (mr/hr) at a distance of one mile from the plant, from 0.5 to 1.0 mr/hr out to 7 miles, and 0.1 to 0.2 mr/hr beyond 10 miles. The plume width is about 1-1/2 to 2 miles. No radioiodines have been detected in the plume. Offsite ground level gamma surveys performed in the predominant wind direction indicated maximum levels of about 2 mr/hr at about 1/2 mile from the site in the direction of the plume. The wind was from the SSW at the time of the

CONTINUED

ARMS flights. At about 1 PM the winds shifted and are now blowing in a south easterly direction.

International Contacts

NRC's Office of International Programs (OIP) has prepared daily status reports, transmitted by Immediate Department of State telegrams to official NRC contacts in the 25 foreign countries with which NRC has regular official relations. OIP is also receiving many foreign telephone calls.

Two senior safety experts from the Federal Republic of Germany (FRG) arrived late March 30 and were briefed by NRC experts at the Operations Center, late March 30 and during March 31. Two French experts will arrive April 1. Washington Representatives or senior visitors of Japan, FRG, and Sweden also have been briefed in the Operations Center. OIP also has been briefing the President of the AECB of Canada, who offered to send any AECL or AECB experts who could be of assistance.

Contact with Licensee

NRC Regional Offices are transmitting to the utilities with operating licenses summary information (in the form of Preliminary Notifications) as they are prepared.

Contact: DThompson, IE x28111 EMHoward, IE x28111

Distribution: Transmitted H St 7:06p.

Chairman Hendrie
Commissioner Kennedy
Commissioner Gilinsky

Commissioner Bradford
Commissioner Ahearne

S. J. Chilk, SECY
C. C. Kammerer, CA
(For Distribution)

Transmitted: MNBB 7:10p
L. V. Gossick, EDO
H. L. Ornstein, EDO
J. J. Fouchard, PA
N. M. Haller, MPA
R. G. Ryan, OSP
H. K. Shapar, ELD

P. Bldg 7:15p
H. R. Denton, NRR
R. C. DeYoung, NRR
R. J. Mattson, NRR
V. Stello, NRR
R. S. Boyd, NRR
SS Bldg 7:20p
W. J. Dircks, NMSS

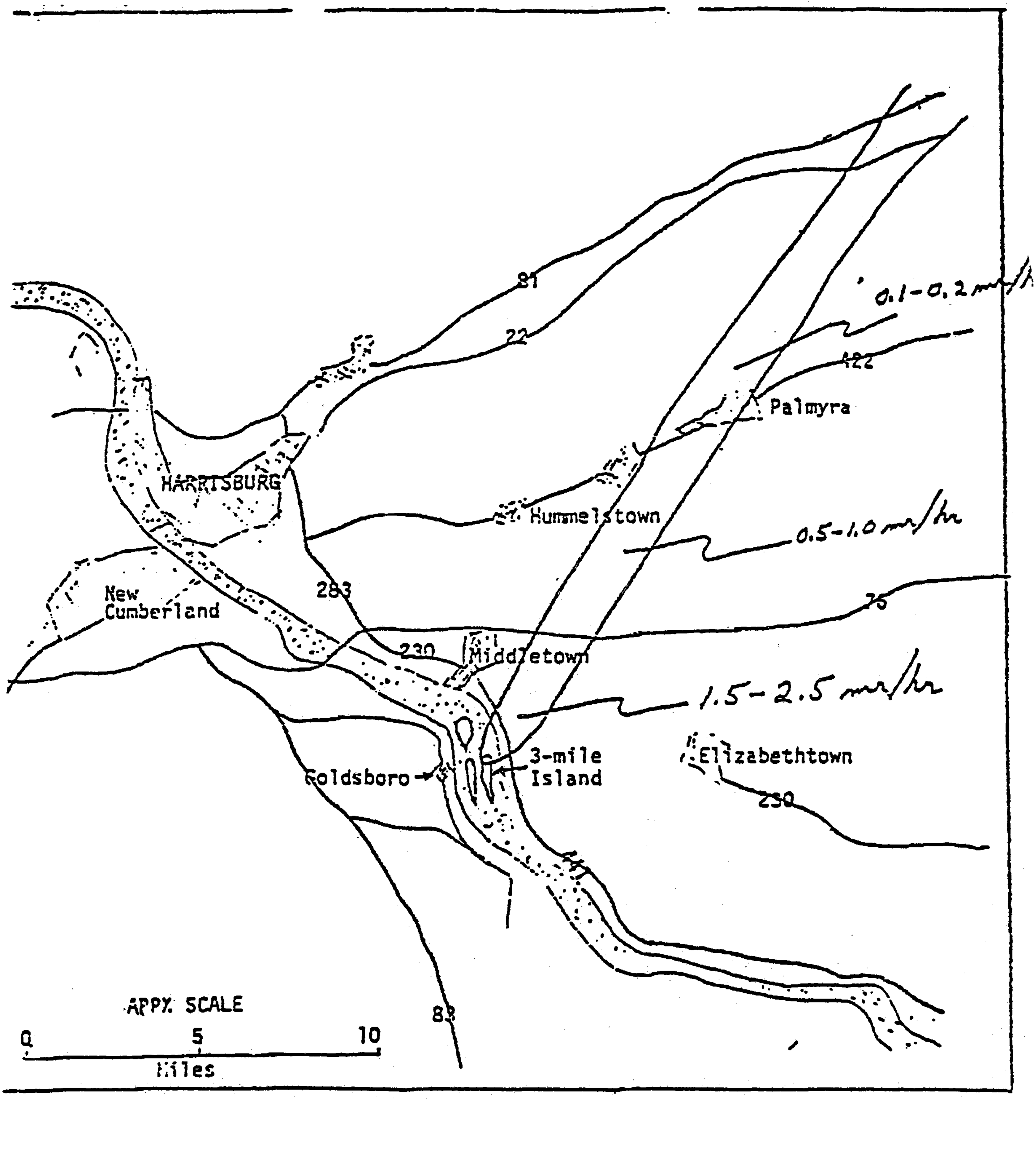
J. G. Davis, IE
Region I - 9:00
Region II
Region III
Region IV
Region V - 7:00
(MAIL)
J. J. Cummings, OIA
R. Minogue, SD

White House Situation Room 7:25p
EPA
FDA/
DOE/EOC

Attachment (1)
Radiation Survey Map

IMMEDIATE

PRELIMINARY NOTIFICATION



March 31, 1979

AERIAL SURVEY plume direction and radiation readings shown above conducted at 6:00 & 9:00 AM and 12:00 noon.

EVALUATION OF FEEDWATER TRANSIENT

A loss of offsite power occurred at Davis-Besse on November 29, 1977, which resulted in shrinkage of the primary coolant volume to the degree that pressurizer level indication was lost. A recommendation to convey this information to certain hearing boards resulted in the attached discussion and evaluation of the event. This discussion includes a review of a loss of feedwater safety analysis assuming forced flow, which predicts dispersed primary system voiding, but no loss of core cooling. During the Three Mile Island event, however, the forced flow appears to have been terminated during the transient.

Attachment:
Discussion and Evaluation of
Davis-Besse Transients

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER.

3. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% power with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Enclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

DISCUSSION AND EVALUATION

The event at Davis Besse which resulted in loss of pressurizer level indication has been reviewed by NRR and the conclusion was reached that no unreviewed safety question existed.

The pressurizer, together with the reactor coolant makeup system, is designed to maintain the primary system pressure and water level within their operational limits only during normal operating conditions. Cooldown transients, such as loss of offsite power and loss of feedwater, sometimes result in primary pressure and volume changes that are beyond the ability of this system to control. The analyses of and experience with such transients show, however, that they can be sustained without compromising the safety of the reactor. The principal concern caused by such transients is that they might cause voiding in the primary coolant system that would lead to loss of ability to adequately cool the reactor core. The safety evaluation of the loss of offsite power transient shows that, though level indication is lost, some water remains in the pressurizer and the pressure does not decrease below about 1600 psi. In order for voiding to occur, the pressure must decrease below the saturation pressure corresponding to the system temperature. 1600 psi is the saturation pressure corresponding to 605°F, which is also the maximum allowable core outlet temperature. Voiding in the primary system (excepting the pressurizer) is precluded in this case, since pressure does not decrease to saturation.

The safety analysis for more severe cooldown transients, such as the loss of feedwater event, indicates that the water volume could decrease to less than the system volume exclusive of the pressurizer. During such an event, the emptying of the pressurizer would be followed by a pressure reduction below the saturation point and the formation of small voids throughout much of the primary system. This would not result in the loss of core cooling because the voids would be dispersed over a large volume and forced flow would prevent them from coalescing sufficiently to prevent core cooling. The high pressure coolant injection pumps are started automatically when the primary pressure decreases below 1600 psi. Therefore, any pressure reduction which is sufficient to allow voiding will also result in water injection which will rapidly restore the primary water to normal levels.

For these reasons, we believe that the inability of the pressurizer and normal coolant makeup system to control some transients does not provide a basis for requiring more capacity in these systems.

General Design Criterion 13 of Appendix A to 10 CFR 50 requires instrumentation to monitor variables over their anticipated ranges for "anticipated operational occurrences". Such occurrences are specifically defined to include loss of all offsite power. The fact that T cold goes off scale at 520°F is not considered to be a deviation from this requirement because this indicator is backed up by wide range temperature indication that extends to a low limit of 50°F. Neither do we consider the makeup flow monitoring to deviate since the amount of makeup flow in excess of 160 gpm does not appear to be a significant factor in the course of these occurrences.

The loss of pressurizer water level indication could be considered to deviate from GDC 13, because this level indication provides the principal means of determining the primary coolant inventory. However, provision of a level indication that would cover all anticipated occurrences may not be practical. As discussed above, the loss of feedwater event can lead to a momentary condition wherein no meaningful level exists, because the entire primary system contains a steam water mixture.

It should be noted that the introduction to Appendix A (last paragraph) recognizes that fulfillment of some of the criteria may not always be appropriate. This introduction also states that departures from the Criteria must be identified and justified. The discussion of GDC 13 in the Davis Besse FSAR lists the water level instrumentation, but does not mention the possibility of loss of water level indication during transients. This apparent omission in the safety analysis will be subjected to further review.

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-05	Malfunctioning of Circuit Breaker Auxiliary Contact Mechanism - General Electric Model CR105X	4/14/78	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
78-06	Defective Cutler-Hammer, Type M Relays With DC Coils	5/31/78	All Power Reactor Facilities with an OL or CP
78-07	Protection afforded by Air-Line Respirators and Supplied-Air Hoods	6/12/78	All Power Reactor Facilities with an OL, all class E and F Research Reactors with an OL, all Fuel Cycle Facilities with an OL, and all Priority I Material Licensees
78-08	Radiation Levels from Fuel Element Transfer Tubes	6/12/78	All Power, Test and Research Reactor Facilities with an OL having Fuel Element Transfer Tubes
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/78	All BWR Power Reactor Facilities with an OL (for action) or CP (for information)
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All BWR Power Reactor Facilities with an OL or CP

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS (CONTINUED)

Bulletin No.	Subject	Date Issued	Issued To
78-11	Examination of Mark I Containment Torus Welds	7/24/78	BWR Power Reactor Facilities with an OL for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee. All other BWR Power Reactor Facilities with an OL for information
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	All Power Reactor Facilities with an OL or CP
78-12A	Atypical Weld Material in Reactor Pressure Vessel Welds	11/24/78	All Power Reactor Facilities with an OL or CP
78-12B	Atypical Weld Material in Reactor Pressure Vessel Welds	3/19/79	All Power Reactor Facilities with an OL or CP
78-13	Failures In Source Heads of Kay-Ray, Inc., Gauges Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B	10/27/78	All General and Specific Licensees with the subject Kay-Ray, Inc. Gauges
78-14	Deterioration of Buna-N Components In ASCO Solenoids	12/19/78	All GE BWR Facilities with an OL (for action), and all other Power Reactor Facilities with an OL or CP (for information)

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS (CONTINUED)

Bulletin No.	Subject	Date Issued	Issued to
79-01	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an OL, except the 11 Systematic Evaluation Program Plants (for action), and all other Power Reactor Facilities with an OL or CP (for information)
79-02	Pipe Support Base Plate Design Using Concrete Expansion Anchor Bolts	3/8/79	All Power Reactor Facilities with an OL or CP
79-03	Longitudinal Weld Defects in ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Company	3/12/79	All Power Reactor Facilities with an OL or CP
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an OL or CP

ENCLOSURE 2

LIST OF LICENSEES AND CONSTRUCTION PERMIT HOLDERS
RECEIVING IE BULLETIN 79-05 FOR INFORMATION

Baltimore Gas and Electric Company ATTN: Mr. A. E. Lundvall, Jr. Vice President - Supply P. O. Box 1475 Baltimore, Maryland 21203	Docket Nos. 50-317 50-318
Boston Edison Company M/C Nuclear ATTN: Mr. G. Carl Andognini, Manager Nuclear Operations Department 800 Boylston Street Boston, Massachusetts 02199	Docket No. 50-293
Connecticut Yankee Atomic Power Company ATTN: Mr. W. G. Counsil Vice President - Nuclear Engineering and Operations P. O. Box 270 Hartford, Connecticut 06101	Docket No. 50-213
Consolidated Edison Company of New York, Inc. ATTN: Mr. W. J. Cahill, Jr. Vice President 4 Irving Place New York, New York 10003	Docket Nos. 50-03 50-247
Duquesne Light Company ATTN: Mr. C. N. Dunn Vice President Operations Division 435 Sixth Avenue Pittsburgh, Pennsylvania 15219	Docket No. 50-334
Jersey Central Power and Light Company ATTN: Mr. Ivan R. Finfrock, Jr. Vice President Madison Avenue at Punch Bowl Road Morristown, New Jersey 07960	Docket No. 50-219

Maine Yankee Atomic Power Company — Docket No. 50-309
ATTN: Mr. Robert H. Groce
Licensing Engineer
20 Turnpike Road
Westborough, Massachusetts 01581

Niagara Mohawk Power Corporation — Docket No. 50-220
ATTN: Mr. R. R. Schneider
Vice President
Electric Operations
300 Erie Boulevard West
Syracuse, New York 13202

Northeast Nuclear Energy Company — Docket Nos. 50-336, 50-245, 50-423
ATTN: Mr. W. G. Counsil
Vice President - Nuclear
Engineering and Operations
P. O. Box 270
Hartford, Connecticut 06101

Philadelphia Electric Company — Docket Nos. 50-277, 50-278
ATTN: Mr. S. L. Daltroff
Vice President
Electric Production
2301 Market Street
Philadelphia, Pennsylvania 19101

Power Authority of the State of New York — Docket No. 50-286
Indian Point 3 Nuclear Power Plant
ATTN: Mr. J. P. Bayne
Resident Manager
P. O. Box 215
Buchanan, New York 10511

Power Authority of the State of New York — Docket No. 50-333
James A. FitzPatrick Nuclear Power Plant
ATTN: Mr. J. D. Leonard, Jr.
Resident Manager
P. O. Box 41
Lycoming, New York 13093

Public Service Electric and Gas Company
ATTN: Mr. F. W. Schneider
Vice President - Production
80 Park Place
Newark, New Jersey 07101

Docket No. 50-272

Rochester Gas and Electric Company
ATTN: Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
89 East Avenue
Rochester, New York 14649

Docket No. 50-244

Vermont Yankee Nuclear Power Corporation
ATTN: Mr. Robert H. Groce
Licensing Engineer
20 Turnpike Road
Westborough, Massachusetts 01581

Docket No. 50-271

Yankee Atomic Electric Company
ATTN: Mr. Robert H. Groce
Licensing Engineer
20 Turnpike Road
Westborough, Massachusetts 01581

Docket No. 50-29

Duquesne Light Company
ATTN: Mr. E. J. Woolever
Vice President
435 Sixth Avenue
Pittsburgh, Pennsylvania 15219

Docket No. 50-412

Jersey Central Power & Light Company
ATTN: Mr. I. R. Finfrock, Jr.
Vice President
260 Cherry Hill Road
Parsippany, New Jersey 07054

Docket No. 50-363

Long Island Lighting Company
ATTN: Mr. Andrew W. Wofford
Vice President
175 East Old Country Road
Hicksville, New York 11801

Docket Nos. 50-322
50-516
50-517

Niagara Mohawk Power Corporation ATTN: Mr. G. K. Rhode Vice President System Project Management 300 Erie Boulevard, West Syracuse, New York 13202	Docket No. 50-410
Pennsylvania Power & Light Company ATTN: Mr. Norman W. Curtis Vice President Engineering and Construction (N-4) 2 North Ninth Street Allentown, Pennsylvania 18101	Docket Nos. 50-387 50-388
Philadelphia Electric Company ATTN: Mr. V. S. Boyer Vice President Engineering and Research 2301 Market Street Philadelphia, Pennsylvania 19101	Docket Nos. 50-352 50-353
Public Service Electric & Gas Company ATTN: Mr. T. J. Martin Vice President Engineering and Construction 80 Park Place Newark, New Jersey 07101	Docket Nos. 50-354 50-355 50-311
Public Service Company of New Hampshire ATTN: Mr. W. C. Tallman President 1000 Elm Street Manchester, New Hampshire 03105	Docket Nos. 50-443 50-444
Rochester Gas & Electric Corporation ATTN: Mr. J. E. Arthur Chief Engineer 89 East Avenue Rochester, New York 14649	Docket No. 50-485
Metropolitan Edison Company ATTN: Mr. J. G. Herbein Vice President - Generation P. O. Box 542 Reading, Pennsylvania 19640	Docket Nos. 50-289 50-320

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

April 11, 1979

IE Bulletin No. 79-06

REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT

As previously discussed in IE Bulletin 79-05 and 79-05A, the Three Mile Island Nuclear Power Plant, Unit 2 experienced significant core damage which resulted from a series of events initiated by a loss of feedwater transient and apparently compounded by operational errors. Several aspects of the incident have generic applicability to all light water power reactor facilities, in addition to those previously identified as applicable to Babcock and Wilcox reactors. This bulletin is to identify certain actions to be taken by all other light water power reactor facilities with an operating license. Actions previously have been required of licensees with B&W reactors.

Action to be taken by licensees:

For all pressurized water power reactor facilities with an operating license except Babcock and Wilcox reactors:

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

    a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; and (3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

    b. Operations personnel should be instructed to: (1) not override automatic action of engineered safety features without careful review of plant conditions; and (2) not make operational decisions based on a single plant parameter indication when a confirmatory indication is available.

    c. All licensed operators and plant management and supervision with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

2. For pressurized water reactor facilities review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed.

3. For pressurized water reactor facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation set point whether or not the level indication has dropped to the actuation set point.

4. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to cause containment isolation of all lines whose isolation does not degrade core cooling capability upon automatic initiation of safety injection.

5. For pressurized water reactor facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

6. For all pressurized water reactors, prepare and implement immediately procedures which:

   a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open, and

b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) fail to close.

7. Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features without careful review of plant conditions.

   b. Operators are provided additional information and instructions to not rely upon any one plant parameter but to also examine other related indications in evaluating plant conditions.

8. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (daily/shift checks, etc.) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

9. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

   c. The basis on which continued operability of the above features is assured.

10. Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection per technical specifications, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operating personnel whenever a safety-related system is removed from and returned to service.

11. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events.

For all pressurized water power reactor facilities with an operating license except Babcock and Wilcox reactors, respond to Items 1-11 within 14 days of the receipt of this Bulletin.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

For all other power reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

April 14, 1979

IE Bulletin No. 79-08

EVENTS RELEVANT TO BOILING WATER POWER REACTORS IDENTIFIED DURING THREE MILE ISLAND INCIDENT

Description of Circumstances:

On march 28, 1979 the Three Mile Island Nuclear Power Plant, Unit 2 experienced core damage which resulted from a series of events which were initiated by a loss of feedwater transient. Several aspects of the incident may have general applicability to operating boiling water reactors. This bulletin requests certain actions of licensees of operating boiling water reactors.

Actions to be taken by Licensees:

For all Boiling water reactor facilities with an operating license complete the actions specified below:

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronolo'y of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

    a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both trains of a safety system at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; and (3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

    b. Operational personnel should be instructed to (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 5a of this bulletin); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

2. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to initiate containment isolation, whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

3. Describe the actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems (e.g., RCIC) that are used when the main feedwater system is not operable. For any manual action necessary, describe in summary form the procedure, by which this action is taken in a timely sense.

4. Describe all uses and types of vessel level indication for both automatic and manual initiation of safety systems. Describe other redundant instrumentation which the operator might have to give the same information regarding plant status. Instruct operators to utilize other available information to initiate safety systems.

5. Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions (e.g. vessel integrity).

   b. Operators are provided additional information and instructions to not rely upon vessel level indication alone for manual actions, but to also examine other plant parameter indications in evaluating plant conditions.

6. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks,) surveillance to to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

7. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

   c. The basis on which continued operability of the above features is assured.

8. Review and modify as necessary your maintenance and test procedures to ensure that they require:

   a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

   b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

   c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

9. Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

10. Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

11. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the items above.

For all boiling water reactor facilities with an operating license, respond to Items 1-10 within 10 days of the receipt of this Bulletin. Respond to item 11 (Technical Specification Change proposals) in 30 days.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

For all other power reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

APRIL 5, 1979

IE Bulletin 79-05A

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Preliminary information received by the NRC since issuance of IE Bulletin 79-05 on April 1, 1979 has identified six potential human, design and mechanical failures which resulted in the core damage and radiation releases at the Three Mile Island Unit 2 nuclear plant. The information and actions in this supplement clarify and extend the original Bulletin and transmit a preliminary chronology of the TMI accident through the first 16 hours (Enclosure 1).

1. At the time of the initiating event, loss of feedwater, both of the auxiliary feedwater trains were valved out of service.

2. The pressurizer electromatic relief valve, which opened during the initial pressure surge, failed to close when the pressure decreased below the actuation level.

3. Following rapid depressurization of the pressurizer, the pressurizer level indication may have led to erroneous inferences of high level in the reactor coolant system. The pressurizer level indication apparently led the operators to prematurely terminate high pressure injection flow, even though substantial voids existed in the reactor coolant system.

4. Because the containment does not isolate on high pressure injection (HPI) initiation, the highly radioactive water from the relief valve discharge was pumped out of the containment by the automatic initiation of a transfer pump. This water entered the radioactive waste treatment system in the auxiliary building where some of it overflowed to the floor. Outgassing from this water and discharge through the auxiliary building ventilation system and filters was the principal source of the offsite release of radioactive noble gases.

5. Subsequently, the high pressure injection system was intermittently operated attempting to control primary coolant inventory losses through the electromatic relief valve, apparently based on pressurizer level indication. Due to the presence of steam and/or noncondensible voids elsewhere in the reactor coolant system, this led to a further reduction in primary coolant inventory.

6. Tripping of reactor coolant pumps during the course of the transient, to protect against pump damage due to pump vibration, led to fuel damage since voids in the reactor coolant system prevented natural circulation.

Actions To Be Taken by Licensees:

For all Babcock and Wilcox pressurized water reactor facilities with an operating license (the actions specified below replace those specified in IE Bulletin 79-05):

1. (This item clarifies and expands upon item 1. of IE Bulletin 79-05.)

   In addition to the review of circumstances described in Enclosure 1 of IE Bulletin 79-05, review the enclosed preliminary chronology of the TMI-2 3/28/79 accident. This review should be directed toward understanding the sequence of events to ensure against such an accident at your facility(ies).

2. (This item clarifies and expands upon item 2. of IE Bulletin 79-05.)

   Review any transients similar to the Davis Besse event (Enclosure 2 of IE Bulletin 79-05) and any others which contain similar elements from the enclosed chronology (Enclosure 1) which have occurred at your facility(ies). If any significant deviations from expected performance are identified in your review, provide details and an analysis of the safety significance together with a description of any corrective actions taken. Reference may be made to previous information provided to the NRC, if appropriate, in responding to this item.

3. (This item clarifies item 3. of IE Bulletin 79-05.)

   Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed.

4. (This item clarifies and expands upon item 4. of IE Bulletin 79-05.)

   Review the actions directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features.

   b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

      (1) Both low pressure injection (LPI) pumps are in operation and flowing at a rate in excess of 1000 gpm each and the situation has been stable for 20 minutes, or

      (2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degree subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated.

   c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation, with reactor coolant pumps (RCP) operating, at least one RCP per loop shall remain operating.

   d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water inventory in the reactor primary system.

5. (This item revises item 5. of IE Bulletin 79-05.)

   Verify that emergency feedwater valves are in the open position in accordance with item 8 below. Also, review all safety-related valve positions and positioning requirements to assure that valves are positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance and testing, to ensure that such valves are returned to their correct positions following necessary manipulations.

6. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to cause containment isolation of all lines whose isolation does not degrade core cooling capability upon automatic initiation of safety injection.

7. For manual valves or manually-operated motor-driven valves which could defeat or compromise the flow of auxiliary feedwater to the steam generators, prepare and implement procedures which:

   a. require that such valves be locked in their correct position; or

   b. require other similar positive position controls.

8. Prepare and implement immediately procedures which assure that two independent steam generator auxiliary feedwater flow paths, each with 100% flow capacity, are operable at any time when heat removal from the primary system is through the steam generators. When two independent 100% capacity flow paths are not available, the capacity shall be restored within 72 hours or the plant shall be placed in a cooling mode which does not rely on steam generators for cooling within the next 12 hours.

   When at least one 100% capacity flow path is not available, the reactor shall be made subcritical within one hour and the facility placed in a shutdown cooling mode which does not rely on steam generators for cooling within 12 hours or at the maximum safe shutdown rate.

9. (This item revises item 6 of IE Bulletin 79-05.)

   Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

10. Review and modify as necessary your maintenance and test procedures to ensure that they require:

    a. Verification, by inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

    b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

    c. A means of notifying involved reactor operating personnel whenever a safety-related system is removed from and returned to service.

11. All operating and maintenance personnel should be made aware of the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident.

12. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events.

For Babcock and Wilcox pressurized water reactor facilities with an operating license, respond to Items 1, 2, 3, 4.a and 5 by April 11, 1979. Since these items are substantially the same as those specified in IE Bulletin 79-05, the required date for response has not been changed. Respond to Items 4.b through 4.d, and 6 through 12 by April 16, 1979.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, DC 20555.

For all other reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B 180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosures:

1. Preliminary Chronology of TMI-2 3/38/79
   Accident Until Core Cooling Restored.

Enclosure 1 to
IE Bulletin 79-05A
April 5, 1979

PRELIMINARY

CHRONOLOGY OF TMI-2 3/28/79 ACCIDENT UNTIL CORE COOLING RESTORED

TIME (Approximate)	EVENT
about 4 AM (t = 0)	Loss of Condensate Pump Loss of Feedwater Turbine Trip
t = 3-6 sec.	Electromatic relief valve opens (2255 psi) to relieve pressure in RCS
t = 9-12 sec.	Reactor trip on high RCS pressure (2355 psi)
t = 12-15 sec.	RCS pressure decays to 2205 psi (relief valve should have closed)
t = 15 sec.	RCS hot leg temperature peaks at 611 degrees F, 2147 psi (450 psi over saturation)
t = 30 sec.	All three auxiliary feedwater pumps running at pressure (Pumps 2A and 2B started at turbine trip). No flow was injected since discharge valves were closed.
t = 1 min.	Pressurizer level indication begins to rise rapidly
t = 1 min.	Steam Generators A and B secondary level very low - drying out over next couple of minutes.
t = 2 min.	ECCS initiation (HPI) at 1600 psi
t = 4 - 11 min.	Pressurizer level off scale - high - one HPI pump manually tripped at about 4 min. 30 sec. Second pump tripped at about 10 min. 30 sec.
t = 6 min.	RCS flashes as pressure bottoms out at 1350 psig (Hot leg temperature of 584 degrees F)
t = 7 min., 30 sec.	Reactor building sump pump came on.

TIME	EVENT
t = 8 min.	Auxiliary feedwater flow is initiated by opening closed valves
t = 8 min. 18 sec.	Steam Generator B pressure reached minimum
t = 8 min. 21 sec.	Steam Generator A pressure starts to recover
t = 11 min.	Pressurizer level indication comes back on scale and decreases
t = 11-12 min.	Makeup Pump (ECCS HPI flow) restarted by operators
t = 15 min.	RC Drain/Quench Tank rupture disk blows at 190 psig (setpoint 200 psig) due to continued discharge of electromatic relief valve
t = 20 - 60 min.	System parameters stabilized in saturated condition at about 1015 psig and about 550 degrees F.
t = 1 hour, 15 min.	Operator trips RC pumps in Loop B
t = 1 hour, 40 min.	Operator trips RC pumps in Loop A
t = 1-3/4 - 2 hours	CORE BEGINS HEAT UP TRANSIENT - Hot leg temperature begins to rise to 620 degrees F (off scale within 14 minutes) and cold leg temperature drops to 150 degrees F. (HPI water)
t = 2.3 hour	Electromatic relief valve isolated by operator after S.G.-B isolated to prevent leakage
t = 3 hours	RCS pressure increases to 2150 psi and electromatic relief valve opened
t = 3.25 hours	RC drain tank pressure spike of 5 psig
t = 3.8 hours	RC drain tank pressure spike of 11 psi - RCS pressure 1750; containment pressure increases from 1 to 3 psig
t = 5 hours	Peak containment pressure of 4.5 psig
t = 5 - 6 hours	RCS pressure increased from 1250 psi to to 2100 psi

TIME	EVENT
t = 7.5 hours	Operator opens electromatic relief valve to depressurize RCS to attempt initiation of RHR at 400 psi
t = 8 - 9 hours	RCS pressure decreases to about 500 psi Core Flood Tanks partially discharge
t = 10 hour	28 psig containment pressure spike, containment sprays initiated and stopped after 500 gal. of NaOH injected (about 2 minutes of operation)
t = 13.5 hours	Electromatic relief valve closed to repressurize RCS, collapse voids, and start RC pump
t = 13.5 - 16 hours	RCS pressure increased from 650 psi to 2300 psi
t = 16 hours	RC pump in Loop A started, hot leg temperature decreases to 560 degrees F, and cold leg temperature increases to 400 degrees F. indicating flow through steam generator
Thereafter	S/G "A" steaming to condenser Condenser vacuum re-established RCS cooled to about 280 degrees F., 1000 psi
Now (4/4)	High radiation in containment All core thermocouples less than 460 degrees F. Using pressurizer vent valve with small makeup flow Slow cooldown RB pressure negative

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

APRIL 21, 1979

IE Bulletin 79-05B

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Continued NRC evaluation of the nuclear incident at Three Mile Island Unit 2 has identified measures in addition to those discussed in IE Bulletin 79-05 and 79-05A which should be acted upon by licensees with reactors designed by B&W. As discussed in Item 4.c. of Actions to be taken by Licensees in IEB 79-05A, the preferred mode of core cooling following a transient or accident is to provide forced flow using reactor coolant pumps.

It appears that natural circulation was not successfully achieved upon securing the reactor coolant pumps during the first two hours of the Three Mile Island (TMI) No. 2 incident of March 28, 1979. Initiation of natural circulation was inhibited by significant coolant voids, possibly aggravated by release of noncondensible gases, in the primary coolant system. To avoid this potential for interference with natural circulation, the operator should ensure that the primary system is subcooled, and remains subcooled, before any attempt is made to establish natural circulation.

Natural circulation in Babcock and Wilcox reactor systems is enhanced by maintaining a relatively high water level on the secondary side of the once through steam generators (OTSG). It is also promoted by injection of auxiliary feedwater at the upper nozzles in the OTSGs. The integrated Control System automatically sets the OTSG level setpoint to 50% on the operating range when all reactor coolant pumps (RCP) are secured. However, in unusual or abnormal situations, manual actions by the operator to increase steam generator level will enhance natural circulation capability in anticipation of a possible loss of operation of the reactor coolant pumps. As stated previously, forced flow of primary coolant through the core is preferred to natural circulation.

Other means of reducing the possibility of void formation in the reactor coolant system are:

A. Minimize the operation of the Power Operated Relief Valve (PORV) on the pressurizer and thereby reduce the possibility of pressure reduction by a blowdown through a PORV that was stuck open.

B. Reduce the energy input to the reactor coolant system by a prompt reactor trip during transients that result in primary system pressure increases.

This bulletin addresses, among other things, the means to achieve these objectives.

Actions To Be Taken by Licensees:

For all Babcock and Wilcox pressurized water reactor facilities with an operating license: (Underlined sentences are modifications to, and supersede, IEB-79-05A).

1. Develop procedures and train operation personnel on methods of establishing and maintaining natural circulation. The procedures and training must include means of monitoring heat removal efficiency by available plant instrumentation. The procedures must also contain a method of assuring that the primary coolant system is subcooled by at least 50°F before natural circulation is initiated.

   In the event that these instructions incorporate anticipatory filling of the OTSG prior to securing the reactor coolant pumps, a detailed analysis should be done to provide guidance as to the expected system response. The instructions should include the following precautions:

   a. maintain pressurizer level sufficient to prevent loss of level indication in the pressurizer;

   b. assure availability of adequate capacity of pressurizer heaters, for pressure control and maintain primary system pressure to satisfy the subcooling criterion for natural circulation;

   c. maintain pressure - temperature envelope within Appendix G limits for vessel integrity.

   Procedures and training shall also be provided to maintain core cooling in the event both main feedwater and auxiliary feedwater are lost while in the natural circulation core cooling mode.

2. Modify the actions required in Item 4a and 4b of IE Bulletin 79-05A to take into account vessel integrity considerations.

   "4. Review the action directed by the operating procedures and training instructions to ensure that:

      a. Operators do not override automatic actions of engineered safety features, <u>unless continued operation of engineered</u>

safety features will result in unsafe plant conditions. For example, if continued operation of engineered safety features would threaten reactor vessel integrity then the HPI should be secured (as noted in b(2) below).

b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

   (1) Both low pressure injection (LPI) pumps are in operation and flowing at a rate in excess of 1000 gpm each and the situation has been stable for 20 minutes, or

   (2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degrees subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees F and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity."

3. Following detailed analysis, describe the modifications to design and procedures which you have implemented to assure the reduction of the likelihood of automatic actuation of the pressurizer PORV during anticipated transients. This analysis shall include consideration of a modification of the high pressure scram setpoint and the PORV opening setpoint such that reactor scram will preclude opening of the PORV for the spectrum of anticipated transients discussed by B&W in Enclosure 1. Changes developed by this analysis shall not result in increased frequency of pressurizer safety valve operation for these anticipated transients.

4. Provide procedures and training to operating personnel for a prompt manual trip of the reactor for transients that result in a pressure increase in the reactor coolant system. These transients include:

   a. loss of main feedwater

   b. turbine trip

   c. Main Steam Isolation Valve closure

   d. Loss of offsite power

   e. Low OTSG level

   f. low pressurizer level.

5. Provide for NRC approval a design review and schedule for implementation of a safety grade automatic anticipatory reactor scram for loss of feedwater, turbine trip, or significant reduction in steam generator level.

6. The actions required in item 12 of IE Bulletin 79-05A are modified as follows:

   Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

7. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items.

Response schedule for B&W designed facilities:

a. For Items 1, 2, 4 and 6, all facilities with an operating license respond within 14 days of receipt of this Bulletin.

b. For Item 3, all facilities currently operating, respond within 24 hours. All facilities with an operating license, not currently operating, respond before resuming operation.

c. For Items 5 and 7, all facilities with an operating license respond in 30 days.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D. C. 20555.

For all other power reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

INTRODUCTION

THE CONTINUING REVIEW OF THE SEQUENCE OF EVENTS LEADING TO THE INCIDENT AT TMI-2 ON MARCH 28, 1979 SHOWS THAT ACTION CAN BE TAKEN TO PROVIDE ASSURANCE THAT THE PILOT-OPERATED RELIEF VALVE (PORV) MOUNTED ON THE PRESSURIZER OF B&W PLANTS WILL NOT BE ACTUATED BY ANTICIPATED TRANSIENTS WHICH HAVE OCCURRED OR HAVE A SIGNIFICANT PROBABILITY OF OCCURRING IN THESE PLANTS. THIS ACTION MUST NOT DEGRADE THE SAFETY OF THE AFFECTED PLANTS WITH RESPECT TO THEIR RESPONSE TO NORMAL, UPSET OR ACCIDENT CONDITIONS NOR LEAD TO UNREVIEWED SAFETY CONCERNS. THE ANTICIPATED TRANSIENTS OF CONCERN ARE:

1. LOSS OF EXTERNAL ELECTRICAL LOAD
2. TURBINE TRIP
3. LOSS OF MAIN FEEDWATER
4. LOSS OF CONDENSER VACUUM
5. INADVERTENT CLOSURE OF MAIN STEAM ISOLATION VALVES (MSIV).

A NUMBER OF ALTERNATIVES WERE CONSIDERED IN DEVELOPING THE ACTIONS PROPOSED BELOW INCLUDING:

1. RESTRICTING REACTOR POWER TO A VALUE WHICH WOULD ASSURE NO ACTUATION OF THE PORV. THE REACTOR PROTECTION SYSTEM, DESIGN PRESSURE AND PORV SETPOINTS REMAINED AT THEIR CURRENT VALUES.

2. LOWERING THE HIGH PRESSURE REACTOR TRIP SETPOINT TO A VALUE WHICH WOULD ASSURE NO ACTUATION OF THE PORV. THE DESIGN PRESSURE OF THE REACTOR AND THE SETPOINT FOR PORV ACTUATION REMAINED AT THEIR CURRENT VALUES.

   LOWERING THE HIGH PRESSURE REACTOR TRIP SETPOINT AND ADJUSTING THE OPERATING PRESSURE (AND TEMPERATURE) OF THE REACTOR TO ASSURE NO PORV ACTUATION AND TO PROVIDE ADEQUATE MARGIN TO ACCOMMODATE VARIATIONS IN OPERATING PRESSURE. THE SETPOINT FOR PORV ACTUATION REMAINED AT ITS CURRENT VALUE. THIS ALTERNATIVE WOULD REDUCE NET ELECTRICAL OUTPUT.

4. ADJUSTING THE HIGH PRESSURE TRIP AND THE PORV SETPOINTS TO ASSURE NO PORV ACTUATION FOR THE CLASS OF ANTICIPATED EVENTS OF CONCERN. THE DESIGN PRESSURE OF THE REACTOR REMAINED AT ITS CURRENT VALUE.

AN ANALYSIS OF THE IMPACT OF THESE VARIOUS ALTERNATIVES AND THEIR CONTRIBUTION TO ASSURING THAT THE PORV WILL NOT ACTUATE FOR THE CLASS OF ANTICIPATED TRANSIENT OF CONCERN HAS BEEN COMPLETED. THE RESULTS SHOW THAT:

LOWERING THE HIGH PRESSURE REACTOR TRIP SETPOINT FROM <u>2355 PSIG TO 2300 PSIG</u>

AND

RAISING THE SETPOINT FOR THE PILOT OPERATED RELIEF VALVE FROM <u>2255 PSIG TO 2450 PSIG</u>

PROVIDES THE REQUIRED ASSURANCE. THIS ACTION HAS THE FURTHER ADVANTAGES OF:

EXTRACT OF B&W COMMUNICATION - RECEIVED BY NRC
4/20/79

1. REDUCING THE PROBABILITY OF PORV AND ASME CODE PRESSURIZER SAFETY VALVE ACTUATION FOR OTHER INCREASING PRESSURE TRANSIENTS.

2. PRESERVING PRESSURE RELIEF CAPACITY FOR ALL HIGH PRESSURE TRANSIENTS.

3. ELIMINATING THE POSSIBILITY OF INTRODUCING UNREVIEWED SAFETY CONCERNS.

4. REDUCING THE TIME AT WHICH THE STEAM SYSTEM HEAT SINK WOULD BE LOST IN THE EVENT EMERGENCY FEEDWATER FLOW WERE DELAYED.

A SUMMARY OF THE IMPACT OF THE PROPOSED SETPOINT CHANGES ON ALL ANTICIPATED TRANSIENTS IS GIVEN IN TABLE 1.

B&W PLANTS ARE CURRENTLY CAPABLE OF RUNBACK TO 15% OF FULL POWER UPON LOSS OF LOAD OR TRIP OF THE TURBINE. THIS CAPABILITY REQUIRES ACTUATION OF THE PILOT-OPERATED RELIEF VALVES. THE CAPABILITY INCREASES THE RELIABILITY OF POWER SUPPLY TO THE SYSTEM BY RETURNING THE UNITS TO POWER GENERATION MORE QUICKLY AFTER THESE TRANSIENTS. THE ACTION PROPOSED ABOVE WILL REQUIRE THAT THE REACTOR BE TRIPPED FOR THESE EVENTS.

NRC NOTE:

The effect of changing the reactor coolant system pressure trip setpoint upon peak pressurizer pressure is typified by the attached figure 1. which was developed by B&W for a loss of feedwater transient.

TABLE 1

SUMMARY OF PROTECTION AGAINST PORV ACTUATION PROVIDED BY PROPOSED SETPOINT CHANGES FOR ALL ANTICIPATED TRANSIENTS

EXTRACT OF B&W COMMUNICATION - RECEIVED BY NRC 4/20/79

1. ANTICIPATED TRANSIENTS WHICH HAVE OCCURRED AT B&W PLANTS AND WHICH WOULD NORMALLY ACTIVATE PORV AT THE <u>CURRENT</u> SETPOINT (2255 PSIG):

   A. TURBINE TRIP

   B. LOSS OF EXTERNAL ELECTRICAL LOAD

   C. LOSS OF MAIN FEEDWATER

   D. LOSS OF CONDENSER VACUUM

   E. INADVERTENT CLOSURE OF MSIV

2. ANTICIPATED TRANSIENTS WHICH HAVE OCCURRED AT B&W PLANTS AND WHICH WOULD NORMALLY ACTUATE PORV AT THE <u>PROPOSED</u> SETPOINT (2450 PSIG):

   NONE

3. ANTICIPATED TRANSIENTS WHICH HAVE NOT OCCURRED AT B&W PLANTS (LOW PROBABILITY EVENTS) AND WHICH WOULD NORMALLY ACTUATE PORV AT THE <u>CURRENT</u> SETPOINT (2255 PSIG):

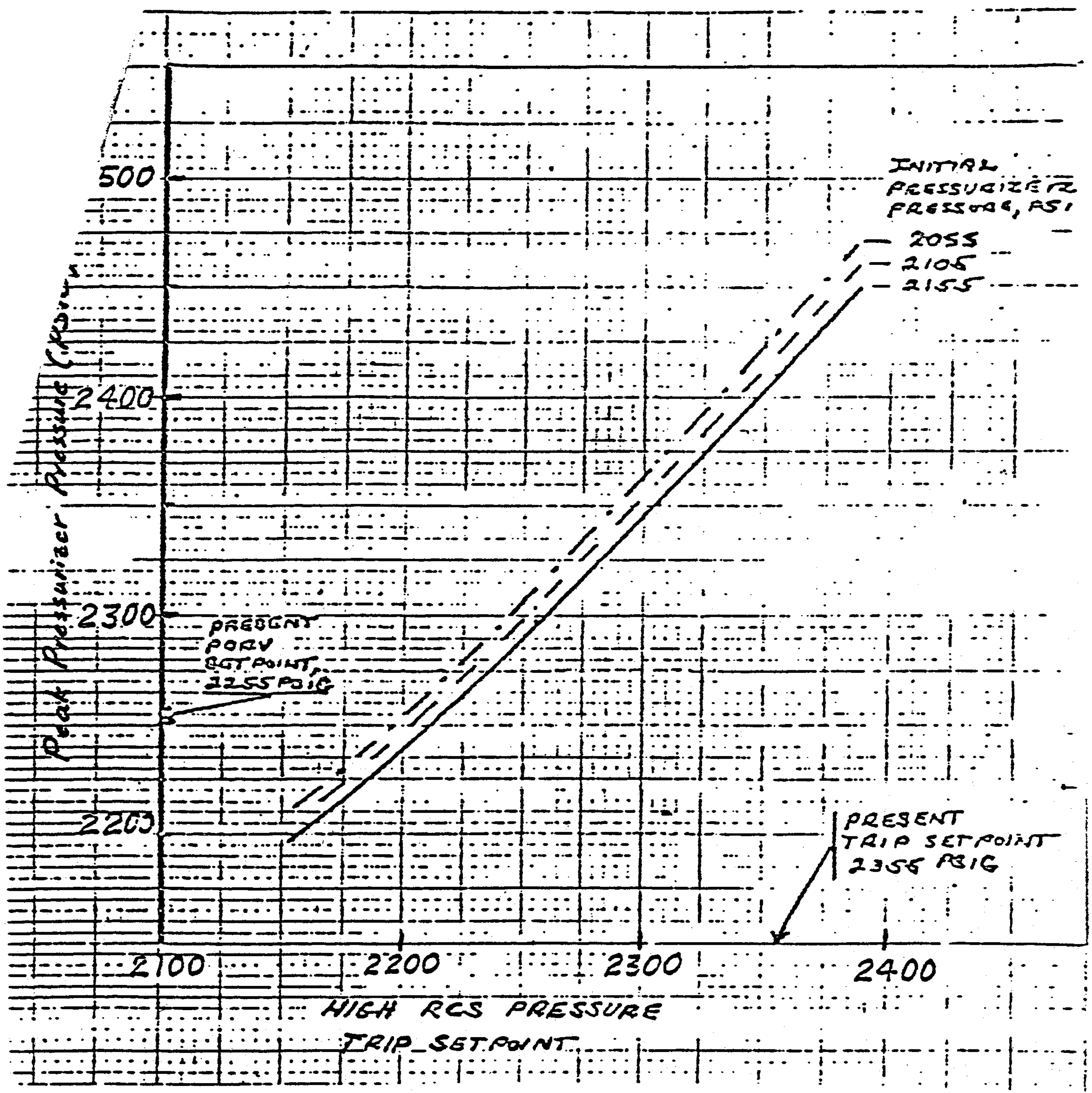
   A. SOME CONTROL ROD GROUP WITHDRAWALS (MODERATE TO HIGH REACTIVITY WORTH GROUPS NOT OTHERWISE PROTECTED BY HIGH FLUX TRIP).

   B. MODERATOR DILUTION.

4. ANTICIPATED TRANSIENTS WHICH HAVE NOT OCCURRED AT B&W PLANTS (LOW PROBABILIT EVENTS) AND WHICH WOULD ACTUATE THE PORV AT THE <u>PROPOSED</u> SETPOINT (2450 PSIG):

   A. SOME CONTROL ROD GROUP WITHDRAWALS (HIGH REACTIVITY WORTH NOT OTHERWISE PROTECTED BY HIGH FLUX TRIP).

EXTRACT OF B&W COMMUNICATION - RECEIVED BY NRC
4/20/79



Peak pressurizer pressure as a function of RCS pressure trip setpoint for a loss of feedwater transient for expected conditions and various initial pressures.

Figure 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

July 26, 1979

IE Bulletin Nos. 79-05C & 79-06C

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Information has become available to the NRC, subsequent to the issuance of IE Bulletins 79-05, 79-05A, 79-05B, 79-06, 79-06A, 79-06A (Revison 1) and 79-06B, which requires modification to the "Action To Be Taken By Licensees" portion of IE Bulletins 79-05A, 79-06A and 79-06B, for all pressurized water reactors (PWRs).

Item 4.c of Bulletin 79-05A required all holders of operating licenses for Babcock & Wilcox designed PWRs to revise their operating procedures to specify that, in the event of high pressure injection (HPI) initiation with reactor coolant pumps (RCPs) operating, at least one RCP per loop would remain operating. Similar requirements, applicable to reactors designed by other PWR vendors, were contained in Item 7.c of Bulletin 79-06A (for Westinghouse designed plants) and in Item 6.c of Bulletin 79-06B (for Combustion Engineering designed plants).

Prior to the incident at Three Mile Island Unit 2 (TMI 2), Westinghouse and its licensees generally adopted the position that the operator should promptly trip all operating RCPs in the loss of coolant accident (LOCA) situation. This Westinghouse position, has led to a series of meetings between the NRC staff and Westinghouse, as well as with other PWR vendors, to discuss this issue. In addition, more detailed analyses concerning this matter were requested by the NRC. Recent preliminary calculations performed by Babcock & Wilcox, Westinghouse and Combustion Engineering indicate that, for a certain spectrum of small breaks in the reactor coolant system, continued operation of the RCPs can increase the mass lost through the break and prolong or aggravate the uncovering of the reactor core.

The damage to the reactor core at TMI 2 followed tripping of the last operating RCP, when two phase fluid was being pumped through the reactor coolant system. It is our current understanding that all three of the nuclear steam system suppliers for PWRs now agree that an acceptable action under LOCA symptoms is to trip all operating RCPs immediately, before significant voiding in the reactor coolant system occurs.

Action To Be Taken By Licensees:

In order to alleviate the concern over delayed tripping of the RCPs after a LOCA, all holders of operating licenses for PWR facilities shall take the following actions:

Short-Term Actions

1. In the interim, until the design change required by the long-term action of this Bulletin has been incorporated, institute the following actions at your facilities:

   A. Upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, immediately trip all operating RCPs.

   B. Provide two licensed operators in the control room at all times during operation to accomplish this action and other immediate and followup actions required during such an occurence. For facilities with dual control rooms, a total of three licensed operators in the dual control room at all times meets the requirements of this Bulletin.

2. Perform and submit a report of LOCA analyses for your plants for a range of small break sizes and a range of time lapses between reactor trip and pump trip. For each pair of values of the parameters, determine the peak cladding temperature (PCT) which results. The range of values for each parameter must be wide enough to assure that the maximum PCT or, if appropriate, the region containing PCTs greater than 2200 degrees F is identified.

3. Based on the analyses done under Item 2 above, develop new guidelines for operator action, for both LOCA and non-LOCA transients, that take into account the impact of RCP trip requirements. For Babcock & Wilcox designed reactors, such guidelines should include appropriate requirements to fill the steam generators to a higher level, following RCP trip, to promote natural circulation flow.

4. Revise emergency procedures and train all licensed reactor operators and senior reactor operators based on the guidlines developed under Item 3 above.

5. Provide analyses and develop guidelines and procedures related to inadequate core cooling (as discussed in Section 2.1.9 of NUREG-0578, "TMI 2 Lessons Learned Task Force Status Report and Short-Term Recommendations") and define the conditions under which a restart of the RCPs should be attempted.

Long-Term Action

1. Propose and submit a design which will assure automatic tripping of the operating RCPs under all circumstances in which this action may be needed.

Schedule

The schedule for the short-term actions of this Bulletin is:

Item 1: Effective upon receipt of this Bulletin,

Item 2: Within 30 days of receipt of this Bulletin,

Item 3: Within 30 days of receipt of this Bulletin,

Item 4: Within 45 days of receipt of this Bulletin,

Item 5: October 31, 1979 (as noted in Table B-2 of NUREG-0578, under Item 3).

A schedule for the long-term action required by this Bulletin should be developed and submitted within 30 days of receipt of this Bulletin.

Reports should be submitted to the Director of the appropriate NRC Regional Office with copies forwarded to the Director, Office of Inspection and Enforcement and the Director, Office of Nuclear Reactor Regulation, Washington, D. C. 20555.

Approved by GAO (R0072): clearance expires 7/31/80. Approval was given under a blanket clearance specifically for generic problems.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT

Description of Circumstances:

IE Bulletin 79-06 identified actions to be taken by the licensees of all pressurized water power reactors (except Babcock & Wilcox reactors) as a result of the Three Mile Island Unit 2 incident. This Bulletin clarifies the actions of Bulletin 79-06 for reactors designed by Westinghouse, and the response to this bulletin will eliminate the need to respond to Bulletin 79-06.

Actions to be taken by Licensees:

For all Westinghouse pressurized water reactor facilities with an operating license (the actions specified below replace those identified in IE Bulletin 79-06 on an item by item basis):

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

   a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; (3) that the potential exists, under certain accident or transient conditions, to have a water level in the pressurizer simultaneously with the reactor vessel not full of water; and (4) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

   b. Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 7a.); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

2. Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed. (e.g., remote venting)

3. For your facilities that use pressurizer water level coincident pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, trip the low pressurizer level setpoint bistables such that, when the pressurizer pressure reaches the low setpoint, safety injection would be initiated regardless of the pressurizer level. In addition, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation setpoint whether or not the level indication has dropped to the actuation setpoint.

4. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to permit containment isolation whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

5. For facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

6. For your facilities, prepare and implement immediately procedures which:

   a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open, and

   b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to below the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) remain stuck open.

7. Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions. For example, if continued operation of engineered safety features would threaten reactor vessel integrity then the HPI should be secured (as noted in b(2) below).

   b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

      (1) Both low pressure injection (LPI) pumps are in operation and flowing for 20 minutes or longer; at a rate which would assure stable plant behavior; or

      (2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degress subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees F and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation with reactor coolant pumps (RCP) operating, at least one RCP shall remain operating for two loop plants and at least two RCPs shall remain operating for 3 or 4 loop plants as long as the pump(s) is providing forced flow.

d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water, inventory in the reactor primary system.

8. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks,) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

9. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

   c. The basis on which continued operability of the above features is assured.

10. Review and modify as necessary your maintenance and test procedures to ensure that they require:

    a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

11 Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

12. Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

13. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items.

For all light water reactor facilities designed by Westinghouse with an operating license, respond to Items 1-12 within 10 days of the receipt of this Bulletin. Respond to item 13 (Technical Specification Change proposals) in 30 days.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

For all other power reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT

IE Bulletin 79-06A identified actions to be taken by the licensees of all pressurized water reactors designed by Westinghouse.

Item No. 3 of the actions to be taken, as stated in the original bulletin, was:

> "3. For your facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, trip the low pressurizer level setpoint bistables such that, when the pressurizer pressure reaches the low setpoint, safety injection would be initiated regardless of the pressurizer level. In addition, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation setpoint whether or not the level indication has dropped to the actuation setpoint."

Information from licensees and Westinghouse has identified that implementation of this action would preclude the performance of surveillance testing of the pressurizer pressure bistables without initiating a safety injection.

In order to permit surveillance testing of the pressurizer pressure bistables, the low pressurizer level bistables that must operate in coincidence with the low pressurizer pressure bistables may be restored to normal operation for the duration of the surveillance test of that coincident pressurizer pressure channel. At the conclusion of the surveillance test of each pressurizer pressure channel, the coincident pressurizer level channel must be returned to the tripped mode defined in Action Item 3 of IE Bulletin 79-06A.

As a result, Item 3 should be revised as follows:

"3. For your facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, trip the low pressurizer level setpoint bistables such that, when the pressurizer pressure reaches the low setpoint, safety injection would be initiated regardless of the pressurizer level. The pressurizer level bistables may be returned to their normal operating positions during the pressurizer pressure channel functional surveillance tests. In addition, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation setpoint whether or not the level indication has dropped to the actuation setpoint."

Item 13 of the actions to be taken, as stated in the original bulletin, was:

"13. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items."

Long term resolutions of some of these required actions may require design changes. Therefore, Item 13 of actions to be taken should be revised as follows:

"13. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items and identify design changes necessary in order to effect long term resolutions of these items."

For all light water reactor facilities designed by Westinghouse with an operating license, respond to Items 1-12 within 10 days of the receipt of this Bulletin. Respond to Item 13 (Technical Specification Change proposals and identification of design changes in 30 days.)

The other requirements of IE Bulletin 79-06A remain in effect.

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IE Bulletin No. 79-06B
Date: April 14, 1979

REVIEW OF OPERATIONAL ERRORS AND SYSTEM MISALIGNMENTS IDENTIFIED DURING THE THREE MILE ISLAND INCIDENT

Description of Circumstances:

IE Bulletin 79-06 identified actions to be taken by the licensees of all pressurized water power reactors (except Babcock & Wilcox reactors) as a result of the Three Mile Island Unit 2 incident. This Bulletin clarifies the actions of Bulletin 79-06 for reactors designed by Combustion Engineering, and the response to this bulletin will eliminate the need to respond to Bulletin 79-06.

Actions to be taken by Licensees:

For all Combustion Engineering pressurized water reactor facilities with an operating license (the actions specified below replace those identified in IE Bulletin 79-06 on an item by item basis):

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.

   a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; (3) that the potential exists, under certain accident or transient conditions, to have a water level in the pressurizer simultaneously with the reactor vessel not full of water; and (4) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.

   b. Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 6a.); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.

c. All licensed operators and plant management and supervisors with operational responsibilities shall participate in this review and such participation shall be documented in plant records.

2. Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed. (e.g., remote venting)

3. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to permit containment isolation whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

4. For facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.

5. For your facilities, prepare and implement immediately procedures which:

   a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open, and

   b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to below the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) remain stuck open.

6. Review the action directed by the operating procedures and training instructions to ensure that:

   a. Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features will result in unsafe plant conditions. For example, if continued operation of engineered safety features would threaten reactor vessel integrity then the HPI should be secured (as noted in b(2) below).

   b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

      (1) Both low pressure injection (LPI) pumps are in operation and flowing for 20 minutes or longer; at a rate which would assure stable plant behavior; or

      (2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degress subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees F and the length of time HPI is in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

   c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation with reactor coolant pumps (RCP) operating, at least one RCP shall remain operating in each loop as long as the pump(s) is providing forced flow.

   d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water, inventory in the reactor primary system.

7. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks,) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

8. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

   c. The basis on which continued operability of the above features is assured.

9. Review and modify as necessary your maintenance and test procedures to ensure that they require:

   a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

   b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

   c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

10. Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

11. Review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system or be released to the containment.

12. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items.

For all light water reactor facilities designed by Combustion with an operating license, respond to Items 1-11 within 10 days of the receipt of this Bulletin. Respond to item 12 (Technical Specification Change proposals) in 30 days.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

For all other power reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

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B

APPENDIX B

NRR STATUS REPORT

ON

FEEDWATER TRANSIENTS IN BWR PLANTS

NRR STATUS REPORT
ON FEEDWATER TRANSIENTS
IN B&W PLANTS

APRIL 25, 1979

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1.0 INTRODUCTION

1.1 Statement of Problem

This paper considers the sensitivity of B&W plants to feedwater transients, and the role that this sensitivity might play as a precursor or contributor to TMI-2 type of accident. We examine the sequence of events that accompanies typical B&W feedwater transients and the role that control and safety equipment plays. We identify some design and analysis deficiencies of this class of plant and note some possible remedial measures.

There are several design differences that distinguish a B&W plant in its response to feedwater transients:

a. The mass of liquid in the secondary side of the steam generator is less than that for other PWRs. More importantly, the B&W design operates as a superheat boiler. Thus, the steam generator tubes are uncovered for a major portion of their length in steady operation. In this mode, changes in feed flow are quickly manifested as changes in heat transfer from the primary system. In this manner, absent prompt and remedial action by the control system (and in some cases a safety system), the steam generator will dry out.

b. The integrated control system is more complex than other designs and has a greater burden placed on it in terms of fast response.

c. The B&W design does not have reactor trips generated from the secondary side of the plant (for example, low steam generator level). Thus the steam generator level may drop somewhat on afeedwater transient before the reactor trips, on high pressure. (At this point, following reactor trip, the control system may overcompensate and cool to an excessive degree, with wide swings in pressure, pressurizer level, and temperature.)

In consideration of these design differences, we are concerned that a transient with a delayed or total failure of auxiliary feedwater may progress into a steam generator dryout condition. Once the steam generator substantially dries out, the reactor system will heat up. The potential for voids in the primary system increases. The reactor pressure may go up to the point where the PORV lifts. Eventually, if natural circulation is not restored or if auxiliary feedwater is not made effective, then core cooling will be dependent on initiation (manually) of the high pressure injection (HPI) system of ECCS. It is this degraded sequence which is the subject of this paper.

1.2 Meeting on April 24, 1979

We met with B&W and four utilities (Duke Power, SMUD, Toledo Edison, and AP and L) on April 24, 1979 to discuss several matters related to core coolability. We discussed the arrival rate of challenging transients, the role of the control system in responding to these transients, the analyses that exist on these transients, the mitigating equipment for plant transients, and finally we asked the utilities to propose remedial measures that might tend to make AFW more reliable such that core coolability is not so dependent on ECCS for anticipated transients.

1.3 Defense in Depth

During normal operation the reactor is cooled by the main feedwater system. This system is fairly reliable; if this were not so, the plants would not be able to produce reliable electric power. In the event of disruption of this normal cooling source, each PWR is provided with an auxiliary feedwater system. These systems differ in redundancy (some are redundant, and some are not), actuation (some are manual, and some are automatic), and in coupling with control systems (some failure modes of the B&W control function may inhibit AFW). Provided that AFW does come on, the reactor is expected to be cooled, by natural circulation if necessary. Representative tests in the natural circulation mode have been run on PWRs in the past. If AFW is not supplied, or if it is supplied too late and the natural circulation path is inhibited by voids and gases, then the system will boil off intermittently until either the HPI is initiated manually or later automatically (perhaps). If HPI is initiated, this system could operate in the inventory mode (since there is no LOCA) and balance losses through relief and safety valves. This mode of core cooling needs to be confirmed by further analyses (Section 3).

On the face of it there are thus three main systems that could remove heat from the core: main feedwater system; auxiliary feedwater, and HPI. The AFW and HPI are discussed further in Chapters 2 and 3

1.4 Conclusions

The question we address in this paper is whether there is reasonable assurance of protection of the public health and safety in continued operation of B&W plants pending improvements related to feedwater transients such as: (1) further analyses and tests on transient performance; (2) a failure modes and effects analysis on the Integrated Control System; (3) system design changes based on the results of these first items; (4) design and installation of additional reactor trip circuits for faults originating in the secondary side of the system; and (5) operator training, including stationing of a full-time dedicated operator assigned to take any needed prompt manual actions. We have considered three alternatives (and they are documented in further detail in Chapter 4):

1. Issue further bulletins to obtain more knowledge about the four items listed above, and implement design and procedural changes on a schedule consistent with the arrival of and evaluation of information.
2. Specify needed design and procedural changes now, and place continued operation as being contingent on implementation within a specified period of time.
3. Require plant shutdown until satisfactory answers to the items 1-4 are provided and evaluated.

These alternatives have been evaluated solely on the basis of safety considerations; i.e., whether there is adequate assurance that the facilities can be operated without endangering the health and safety of the public. We considered the following questions:

1. Do challenging transients arrive at a frequency high enough to be of concern?

   Our answer is yes (Section 2.3.1)

2. Does the ICS perform satisfactorily?

   a. B&W has stated and we agree, that "we are not satisfied with the reliability of the integrated control system".

   b. The failure modes and effects have not been systematically analyzed (Section 2.3.5).

   c. The ICS may initiate a feedwater transient (on the order of 10-15% of all events in the past).

   d. The ICS controls AFW in some plants (Section 2.2.5) and could contribute to loss of AFW.

   e. Even when the ICS works well there may be, in response to a feedwater transient, wide swings in reactor pressure, pressurizer level, and average reactor coolant temperature.

3. Is the system response to loss-of-feedwater transient well known? Again, we split our answer in several parts:

   a. Detailed analyses on loss or delay of AFW, with or without PORV operation, of the system response have not yet been made available to us (Section 3.1).

   b. For very small breaks (e.g., stuck-open PORV) the role of HPI in maintaining core cooling is not well analyzed (Section 3.2)

   c. The heat removal path by natural circulation is not well understood, especially when it is aggravated by void formation (Section 3.3).

4. Are the plant mitigating systems (AFW, ECCS) generally reliable? Our answer is that in most plants these systems are reliable; i.e., state of the art (Section 2.4.2). An exception is the AFW systems which are active at Oconee, which have only one pump per unit. Some other old B&W plants have lesser single failure vulnerabilities.

On the basis of the foregoing it appears that Alternative 1 should not be selected. There is too much unknown about the two items (ICS, plant transient response) to await the several months necessary to generate and evaluate the information.

Thus the choices are whether to shut down the plants now (for one or more months) or whether remedial measures exist or can be generated shortly so that interim operation poses no undue risk.

We asked the industry to propose remedial measures, and have received little to date. We note that Duke Power is considering some AFW redundancy measures (Section 2.3.3). Remedial measures could include improved operator manning, partial power or other changes to increase the thermal margin of reactor operations to reduce the boil-off rate of the steam generator and subsequent core heatup rate); increased testing of AFW; or, in the case of Oconee, perhaps full-time ~~operation of one AFW;~~ removal of AFW from ICS control, if possible, and placement on a separate and independent control system of high reliability; escalated delivery of analyses. ~~However,~~ we believe that our role is to diagnose the ailment (this we have done); it is up to the utilities to propose the treatment.

We conclude that we do not now have reasonable assurance that these B&W plants can continue to operate without undue risk. We believe that these plants should be shutdown now, and that the following information is necessary before restart can be permitted.

In the short-term, we must take all reasonable steps to reduce the likelihood of occurrence of transients at B&W plants and to improve standing instructions, training and emergency procedures available to plant operators. This can be accomplished by:

a. Reviewing and upgrading, as appropriate, auxiliary feed reliability and performance (timeliness);
b. Reviewing results of FMEA analysis of ICS and taking actions, as to reduce its likelihood of initiating or exacerbating transients;
c. Hard wiring anticipatory scram based on FW transients;
d. Reviewing detailed analyses of plant response to transients to effects of HPI injection, and return to natural circulation cooling and
e. Reviewing new and augmented standing instructions and emergency procedures for plant operators developed as a result of a-d above, and training plant operators and the new and augmented instructions and procedures including the stationing of a full-time dedicated operator to take appropriate prompt manual actions.

In the long-term, we must either reduce the sensitivity of the response of B&W plants to transients by design changes, or substantially upgrade the instrumentation and controls available to the plant operator and substantially upgrade plant operator education training and experience.

2. AUXILIARY FEEDWATER REQUIREMENTS

2.1 Overview

The auxiliary feedwater system (AFW) requirements are related to its performance and reliability. In this context, reliability measures the probability that the system will function when called upon, whereas performance measures the adequacy of the amount, rate, and timeliness of the water actually supplied to the steam generators.

Both the performance and the reliability of installed AFW systems vary from plant to plant. The principal differences are related to (1) differences in plant parameters, (2) differences in system configurations, and (3) differences in regulatory requirements over the years. The characteristics of AFW in the operating B&W plants are given in Table 2.1. The AFW is not in the B&W scope of supply, so the different plants have quite different AFW configurations, as is evident from the table.

2.2 Performance

The performance requirements of an AFW are derived from its design basis and the assumptions made. Loss of main feedwater (LOFW) is the initiating event. The steam generator inventory decreases at a rate determined by the heat input rate, the heat removal rate

April 26, 1979

TABLE 2.1 AFW SYSTEMS

	OCONEE	CRYSTAL RIVER	RANCHO SECO	DAVIS BESSE	ARKANSAS
Auto FW Isolation Signal	None	Steam Line Failure Matrix. Closes FW block valve at P<600 psig. (Includes AFW valves.) Also MSIV's. (Isolates faulted steam generator only.)	MSL Failure-Logic: Isolates main FW from faulted SG at P<435 Psig	STM & FW Rupture Control System (IE) (1) Stm P-FW P≥170 psi (2) Steam Generator Low Level (3) Loss of all RCP's (Power Monitor) (4) Low Steam Generator Pressure in either SG. (600 psig) (1) or (2) or (4) isolates main FW to both SG's, closes MSIV's (4) Also aligns both AFW PPs to the good SG. (1) or (2) or (3) or (4) starts both AFW PPs	Steam line break inst. & control (SLBIC) isolates both steam generators' main FW & MSIV's at ≤600 psig Does not isolate emergency FW. (SLBIC is IE)
Auxiliary Feedwater					System is seismically designed. Valves are Class IE; most instr. is not IE.
Pumps: Type/No./Strainers	(Emergency FW pumps) Located Turbine Bldg. 2 floors under grade. centrifugal/ 1 per unit/No	Located near grade level in intermediate bldg. (Seismic category I). Centrifugal/2/No	Located at CST in Missile Enclosure centrifugal/2/No	Does not start on SFAS. centrifugal/2/suction strainers	"EMERGENCY" FW on this plant centrifugal/2/none
Drive: Type	steam driven	1-motor driven 1-steam driven	1-motor driven 1-motor & turbine tanden	(800 HP) turbines (Terry/ Woodward)	1-Turbine (Terry) 1-Motor (Normal supply not Class IE. Can be put on Class IE-15 min.)
Supply/Exhaust	Main Steam/ Atmosphere (≥10 min.)	Motor:Class IE Main steam either SG upstream MSIV/Atmosphere	motors-Class IE steam from MSL/ Atmosphere	main steam/ atmosphere	main steam/atmosphere

Orientation of Pumps (Self Venting?)	Horizontal Yes (low point in system)	Horizontal Possibly not self-venting. Elevation same as bottom of condenser	Yes-thru mini-flow recirc line	horizontal (yes)	horizontal/yes low point in system)
Capacity	1080 gpm at 1 1065 psia	740 gpm each @ 3000 ft.	motor 840 GPM @2700ft. turbine 840 GPM @2650 ft.	1050 GPM @2500 ft. (260 GPM of this is recirc)	780 GPM @2600 ft.
Shutoff Head	1465 psia	motor: 3400 ft. steam: 3500 ft.	steam: 3050' @ 3560 rpm motor: 3100' @ 3560 rpm	≈3150 ft. @ 3600 rpm	
Suction Sources/Seismic Category	(1) Upper Surge Tank/ ASME VIII (3) Other Units' Surge Tanks/ ASME VIII (2) hotwell/no, Aux. SW pumps (3000 gpm @ 75 psig) (from emergency power-1 per site) Suction from CW intake located in Aux. Bldg. 1 floor below grade.	(1) CST/No. ASME Class 3, B31.1 (2) hotwell/No-These suction valves int'lked with vac. brkr. valve position (3) makeup from fossil units' demin./No (1 min)	Condensate storage tank - Seismic Cat. 1 Canal-Non-seismic (5 min.) Reservoir-Non-seismic	(1) CST/No (auto XFER to SW on low suction P-Class 1E, redundant instr.) (2) Deaerator/No (3) Fire Water System/No Last: SW pump discharges/Yes	(1) CST/No (2) SW pp disch./ Yes Suction pressure switch (common-Non 1E) Remote Manual MOV's (requires only seconds to switch-Class 1E valves)
Turbine Driven Pumps Operable at What Range Of Steam Pressures	$\geq$300 psig	$\geq$200 psig	$\geq$213 psig (tested 1124 gpm at 213 psig)	$\geq$50 psia (Psat for 280°F)	$\geq$270 psig
Trips	(1) Overspeed (2) Low Hydraulic Pressure (shaft driven pump)	Overspeed/Motor trips on closed suction valve. Overcurrent	Manual (local or remote) Bus unloading Overcurrent: Inst 2000A, OST 4450 RPM, 960 for 5.15 sec, 640 for 6.43, 320 for 11.39	OST, Low Suction P, Low Steam Inlet P at >25 sec., Manual	Turbine-OST Motor-None
Instrumentation	EFW pp disch. P & Flow, SG level, SG Pressure	Drive turbine SV position Motor on-off lights Flow in SU FW line Ammeter	On-off lights for motor drive Ammeters Steam Supply valve position	Each pump: Discharge P Speed indication	Discharge P each pump

Normal Lineup	Suction valves from surge TK open Disch. valves N.O. (check valves prevent back-flow)	All injection valves N.O. (check valve prevent back-flow)	FCV's & Bypasses N.C./ Cross-tie N.O. Suction from CST:N.O.	Suction Valves N.O. from CST Two series MOV's closed in each pump's discharge. One pump feeds one SG.	Discharge valves (MOV's-Class 1E) Closed. Cross-tie valves open.
Auto Initiation	Loss of both main FW pumps (detected by discharge header pressure <760 psig or FW pump turbine stop valve position) EFW does not start on ECCS.	Loss of both main FW pumps (as indicated by low control oil pressure) AFW does not start from ECCS initiation. Motor Driven pump-no auto start	Loss of both FW pumps (<850 psig on each pump disch.) These switches reset but pumps continue to run. (Single fail. proof) All RCP's off (Power monitor-current, volts, phase-same as RPS) Turbine only starts on ECCS initiation	Stm & FW Rupture Control System (see description under Auto FW Isolation) Does not start on SFAS.	Turbine: (1) SLBIC (see Auto FW Isol.) (2) Loss of FW sensed by governor latch on main FW pumps and "auxiliary" FW pump low disch.P (thru ICS) (3) loss of all RCP's (breaker position) Motor: No auto start. (No starts on ECCS)
Failure Mode on Loss of Air/Power	Loss of air switches 14" main header/ Valves & solenoids powered by batteries (non 1E)	FCV's lock in position Reservoir for 3 cycles/Emergency buses	Class 1E MOV Bypasses FCV on SFAS. FCV Fails Open/FCV Fails to 50%	No air-OP. Valves/MOV's fail as is - but all are powered by 1E	No air op. valves/ As-is (all valves are MOV's)
ICS Control Level: RCP/No RCP	25"/260" Sensed from breaker positions	30"/250"	30"/~318"	Not ICS. Auto essential level control system 120" from redundant, Class 1E instrumentation (Pump speed)	20" & 24"/~300" (50% OP. range)
Procedure/Practice	Same/Same	Same/Same	Same/Same	Licensee is installing dual level setpoint controller to relieve operator of this duty. 35" unless there is a loca/Same	Control RCS Temperature/Same

Surveillance Test Method	Close manual AFW supply block valve. Recirc from/to upper surge tank. Valves do not realign automatically on SFAS.	Close discharge MOV's and recirc from/to CST thru, mini-flow lne. Valves do not realign automatically on SFAS.	Close FCV & x-tie from C.R. Pump from CST to cond. through test line. Valves do not realign automatically on SFAS.	From CST to drain thru normal recirc line (250 gpm). No valve realignment necessary.	Recirc to condenser or CST. Injection valves already closed. Operator opens the manual valve.
Steam Generator: Distance between tube sheets/AFW inlet/Main Feedring	625"/590"/362"	625"/590"/382"	625"/603"/338"	625"/608"/388"	625"/590"/382"
Method to Protect Good S.G.	Operator action from control room.	Steam line failure matrix isolates all FW from SG if P<600 psig	Main steam line failure logic. (<435 psig isolates SG) Does not isolate AFW.	Stm & FW Rupture Control System (see description under Auto FW Isol.)	SLBIC (See Auto FW Isolation) Does not isolate EFW.

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Art Oxfurth x27741

and the primary-to-secondary heat transfer. The steam generator inventory as a function of time, and the time to steam generator dryout, depend on these rates and on the initial inventory.

2.2.1 Initial Inventory

We have had little discussion on whether it is practical to increase the time to steam generator dryout by increasing during normal operation the amount of fluid in the secondary side of the steam generator. As presently operated the collapsed water level at full power is quite low. The potential problems of increasing this inventory have not been discussed with NRC.

2.2.2 Scram

The scram decreases the heat input. Present B&W designs scram on primary system high pressure for LOFW transients. This typically occurs 8-10 seconds after LOFW. Alternatively, an anticipatory scram signal could be derived from one or more secondary system parameters (e.g., steam generator water level, turbine stop valve closure). This would initiate a scram ~6 seconds sooner than the present design, increasing the time to reach steam generator dryout by 1 minute or more. NRC Bulletin No. 79-05B requires B&W plants to provide for NRC approval a design review and schedule for implementation of a safety grade automatic anticipatory reactor scram for loss of feedwater, turbine trip, or significant reduction in steam generator level.

The realignment of primary pressure scram and relief-valve setpoints mandated in Bulletin 79-05B also have the effect of decreasing the scram delay and delaying dryout. The increment, whose value has not been calculated, is smaller than would be provided by the anticipatory scram. However, the setpoint changes have already been implemented on the plants whereas the anticipatory scram will be added in the future.

2.2.3 Time to Steam Generator Dryout

Table 2.1 gives the time to dryout of the steam generators of the operating B&W plants - about one-half minute at full power. Westinghouse steam generators have 2-3 times as much water in the secondary side of the steam generators, proportionately, as B&W plants; CE plants have 3-4 times as much as B&W plants.

However, these plants (W & CE) have anticipatory scram which extend the dryout times to many minutes.

After the scram, the heat input decreases rapidly and the water in the steam generator secondary boils off more slowly. Calculations for LOFW give B&W dryout times of 1-2 minutes for present B&W designs, depending on the course of the event. It is this fast dryout compared to other PWRs that makes B&W plants unique. The factor of 2-4 larger inventory and the anticipatory scram in non-B&W plants give calculated dryout times of many minutes. Thus the timing requirements for AFW delivery are substantially more stringent for B&W plants than for others. This increases the importance of timely manual initiation of AFW in B&W plants compared to the others. Moreover, there is less time to rectify operating or maintenance errors and get the AFW operational if it doesn't start initially.

2.2.4 AFW Delivery Rate

Table 2.1 shows the differences in AFW flow rate for the different plants. The actual flow will depend on the number of pumps running, the pressure in the steam generator against which they have to pump, and the action of control devices. These last are flow control valves in the AFW lines or throttle valves in the steam lines to the turbines on steam-driven pumps.

On all B&W operating plants but Davis-Besse, AFW flow is controlled automatically by values receiving a signal from the integrated control system. The controlled variable is water level, as shown in Table 2.1. A low level setpoint (2-3 feet above the tubesheet) is used when the reactor coolant recirculation pumps (RCP) are operating. This is switched automatically to a high level setpoint (21-26 feet) to enhance natural circulation when the RCP are not operating.

On Davis-Besse, a separate, safety grade, control system controls pump speed (via steam throttle valves) to maintain a level 10 feet above the tubesheet. For the "raised steam generator" configuration in this plant, the 10-feet level is sufficient to maintain natural circulation.

After a LOFW and scram, the steam-water mixture normally present in these once-through steam generators collapses to a liquid level typically 3 ft or lower. The level then decreases, and later increases as AFW comes on.

2.2.5 Long-Term Considerations - HPI

Recent operating data obtained informally from Oconee show the following:

	Unit 1	Unit 2	Unit 3
Automatic Initiation of HPI	1	1	2
Manual initiation of HPI	16	9	7

Thus HPI was initiated at a frequency of about two times per reactor-year. Not all of these initiations were for LOFW events, but some were. Manual initiations were said to have been accomplished in order to maintain pressurizer level. Evidently the primary system shrinkage after a successfully controlled transient involves HPI action.

This raises questions about the role and requirements for HPI. Rather than just being part of the ECCS, which was put in to control small breaks. it is used routinely for frequent anticipated transients. Its failure modes and the consequences of its failure should therefore be analyzed in that context in addition to reviews conducted in the LOCA context.

2.3 Reliability

Numerical criteria for AFW reliability do not exist, and estimates of the reliability actually achieved are also not available. The following discussions are therefore qualitative only.

2.3.1 Challenge Rate

Estimates by B&W and others give about two per reactor-year as the rate of LOFW events. B&W states that the rate, for all PWRs and for B&W plants, decreases to ~1.5 per reactor-year after an initial period of operation. We have no reason to doubt these values.

The HPI initiation rate reported in Section 2.2.5 above is also about 2 per reactor-year.

For a LOFW event, either AFW or HPI must function to protect the core. (There are some other alternatives, such as restoring main feedwater flow, but they do not significantly change the picture). The rate of accidents (full damage) would therefore be:

A(BC)

where A = challenge rate

B = failure probability of AFW

C = failure probability of HPI

Hence, "failure" means insufficient functioning to cool the core, and involves consideration of performance, timing, and reliability. Given A=2 per reactor-year, the product BC must be adequately low; numerical guidance is not currently available.

2.3.2 Source of Water

Table 2.1 shows the sources of water available to the AFW. Each plant has multiple sources, but in some older plants they are not seismic Category 1. Abundant quantities of water are available from these sources.

2.3.3 Pump redundancy

All plants except Oconee have redundant AFW pumps. All plants except Oconee and Davis-Besse have diverse prime movers - steam and electric.

Oconee has one steam-driven pump per unit. The three pumps for the three units can be interconnected through normally closed valves (remote manual control); two pumps are stated to be sufficient in capacity for all these units. The potential redundancy in this arrangement has not so far been exploited. Davis-Besse has two identical steam-driven AFW pumps.

2.3.4 Valves and Piping

Table 2.1 does not list the valve arrangement. In general, separate valves are provided to control AFW to the two steam generators. We have not yet evaluated whether a single failure - control, valve or pipe break - could inhibit all AFW; this was not a requirement when these old plants were licensed. In some plants, common pipes and relief valves exist whose failure could inhibit all AFW.

2.3.5 Controls

In all plants except Davis-Besse, the Integrated Control System actuates the AFW flow control valves. On some plants, these control valves can be bypassed (remote manual control) to allow AFW flow in the event of control system failure.

B&W was unable to state whether failures in the Integrated Control System could initiate a LOFW event and also inhibit AFW via the flow control values. We have asked B&W to analyze this question promptly. If this common-mode failure can occur, and we see no reason why it is impossible, then the combined frequency AB (see Section 2.3.1) could be high because, for these events, B = 1.

2.4 Conclusions regarding AFW

2.4.1 Performance

AFW performance in operating B&W plants appears marginal, in that dryout would occur rapidly (1-2 min) unless AFW is initiated at its design time of 40 seconds after a LOFW.

2.4.2 Reliability

AFW reliability in operating B&W plants varies widely among different designs. The older plants are not in conformance with SRP 10.4.9, for example, by requiring redundancy, diversity, and single failure criterion, etc. Improvements are needed in some plants.

2.4.3 Dependence on HPI

Successful recovery for most LOFW events appears to require HPI even if AFW functions as desired. This requirement to use HPI for an anticipated transient, and its failure modes and consequences of failure, should be analyzed in this context of use as inventory control.

3.0 TRANSIENT ANALYSIS

3.1 General

In general, the loss of feedwater transient analyses performed and reported in the Final Safety Analysis Reports for B&W reactors considered the event to be a loss of main feedwater only. A loss of all (i.e., main and auxiliary) feedwater has not been considered in the course of a usual case review. This is consistent with current and past practices because it was believed that a total loss of all feedwater could only occur after multiple and unlikely equipment failures. Operator error to lock-out a system had not been considered. Single failures were generally considered to be a loss of a redundant component to establish minimum system performance requirements.

An evaluation of a feedwater transient was performed for Three Mile Island Unit 2 as reported in the SAP and the results are typical for all B&W plants. However, feedwater transient analyses that take the lessons learned from TMI-2 have not yet been provided.

During a LOFW transient, the loss of main feedwater reduces the capability to dissipate heat-flow from the primary to secondary system. The primary system heats up, the power operated relief valve is actuated, and the reactor trips on overpressure in the primary system. [There are safety valves installed on the pressurizer to limit the pressure excursion to code design limits.] The emergency feedwater system refills the steam generator and dissipates the decay heat. The reactor core remains covered, no fuel damage occurs and calculated offsite radiological doses are well within the guidelines of 10 CFR 100. The actual analysis presented in the SAR spans a time period of about one minute. In this time, it indicates that core power and primary sytem pressure are moving in a safe direction relative to fuel damage and system overpressure.

The SAR analysis that was performed did not include delay of AFW or failure of the power operated relief valve to reclose when the pressure decreased further. Further long term cooling aspects were not addressed. However, the Standard Review Plan (SRP 15.2.7) indicates that there should be no loss of function for any barrier other than the fuel cladding for such a feedwater transient, even when accompanied by a single failure.

The analyses of situations involving a release of reactor coolant from the system through a failure of a relief valve were based on small break ECCS studies and not as a consequence of an operational transient.

3.2 Small Break Analysis

The models that are used for small breaks analysis are usually Appendix K type with the emphasis on conservatism; e.g., loss-of-offsite power, minimum core cooling and no short term operator actions. More realistic studies of the reactor plant dynamic response are needed to ensure proper tracking and understanding of the event being analyzed.

The blowdown codes used by B&W are CRAFT and TRAP. CRAFT has been approved by the NRC for ECCS analysis of large and small breaks in the primary system. TRAP is a modified version of CRAFT with a detailed secondary model and a simplified primary model and is used for steam and feedwater line break analysis. TRAP is currently under review by the NRC.

The transient codes used by B&W are NATURAL, CADD and POWER TRAIN. CADD has been approved by the NRC for ATWS analysis. NATURAL, which would be used for natural circulation calculations, has not been submitted and POWER TRAIN is under review.

In response to staff requests, the Duke Power Company (Oconee Nuclear Station, Units 1, 2 and 3) provided (April 21, 1979) the results of an evaluation of small break events in conjunction with the loss of emergency feedwater flow for 20 minutes.

Operator actions are assumed to initiate HPl and restore emergency feedwater flow to the steam generators. The analyses indicate, in the licensee's opinion, acceptable results. The core uncovery is not predicted to occur and therefore adequate core cooling was available. The analyses covered various small break sizes of 0.07 ft^2; 0.02 ft^2 and 0.01 ft^2.

At a meeting held on April 24, 1979 the staff indicated its need for additional information for its review concerning the analyses; e.g., the ability of a HPI to provide adequate core cooling without short term operation of the AFW, break locations such as in the pressurizer should be considered; the analyses should extend into the long term cooling mode, and the systems effects of a stuck-open relief valve need to be discussed.

At this meeting the B&W representatives stated that further small break analyses had been performed that covered some of the staff's concerns. B&W agreed to provide the results of such analyses to the staff in two weeks. The analyses would include sensitivity studies on the delay of AFW, one and two HPI pumps in operation, and long term cooling capability.

Table 3.1, obtained from B&W, states those analyses done or a process that is relevant to transient analyses.

TABLE 3.1

CRAFT-II ANALYSES

		STATUS	RESULTS
1.	PORV stuck open; 2 HPI; RC pumps on + autofeed	Done	OK
2.	PORV stuck open; 1 HPI; RC pumps on + auxiliary feed	Done	OK
3.	PORV stuck open; 200 gpm; RC pumps on + auxiliary feed	Done	Melt
4.	TMI-II actual transient best estimate prediction	1/2 done we have it to one hour we will finish it to core uncovery	Melt
5.	.07, .02, + .01 Small breaks; no RC pumps, no auxiliary feedwater no 20 min.; 2 HPI	Done	OK
6.	Zero break with manual actuation of 2 HPI @ 20 min.; no RC pumps	Reconfirm old analysis	OK
7.	Small break in steam space of pressurizer 1.05 in^2. PORV break treated as normal small break; no RC pumps; auxiliary feedwater, 1 HPI	Done	OK
	Note: Additionally all analyses previously submitted in support of our FAC evaluation model. These make use of the three forms of natural circulation described.	Done	

CADDS SENSITIVITY STUDIES

		STATUS	RESULTS
1.	TMI-2 incident benchmark (~6 min.)	Done	
2.	Best Estimate Model Studies	Done	
	• AFW Actuation delay (40 sec.; 120 sec + delay)	Done	
	• Reactor trip coincident with LDFW/turbine trip	To do	
	• Studies supporting changes recommended in high RC pressure trip setpoint and PORV setpoint.	Done	

3.3 Natural Circulation Cooling in a B&W Plant

For most B&W plants, the safety analyses are carried out in time only long enough to indicate that pertinent parameters relative to core damage or overpressurization are proceeding in a safe direction. Analyses are seldom pursued out in time to evaluate operator actions, inactions, or error in judgment, or the course of natural circulation cooling in the event of a loss-of-offsite power. The concerns on natural circulation cooling have been raised by the ACRS and C. Michelson, a consultant to the ACRS.

A report entitled, "DECAY HEAT REMOVAL DURING A VERY SMALL BREAK LOCA FOR A B&W 205-FUEL-ASSEMBLY PWR," by C. Michelson (January 1978) has recently been provided to the staff. In this report Mr. Michelson described concerns regarding small breaks (~ .5 ft^2 range) and the ability of the plant's heat removal systems to remove adequate decay heat to prevent system repressurization in the event of a loss-of-natural circulation or break isolation by operator action. He has also discussed concerns on slug or two-phase flow through a PORV. This report is presently being reviewed by the staff and B&W. The staff is pursuing with B&W and the owners of B&W plants those aspects of concern raised in this report.

Studies by B&W indicate that natural circulation should not be significantly affected due to the formation of steam spaces in the upper portions of the hot leg piping and upper plenum of the reactor vessel.

B&W has conducted tests to determine the amount of natural circulation. The tests are normally done during startup testing from an initial power level of about 20-25%. The reactor is scrammed, the RCPs are tripped, the emergncy diesel generator comes on, the steam and motor driven AFW pumps start, the ICS raises OTSG level to the 50% value, and the plant is verified to be operating on natural circulation, without any operator action.

These tests have been conducted at Davis-Besse and Oconee. Also, Arkansas-1 suffered a loss of offsite power from 100% on 7/25/75 and natural circulation was established, without any operator action. We were not provided with these data. TMI-2 also had two (2) unscheduled events in their startup testing program which resulted in natural circulation.

The staff requested as much detail and description as possible on all the natural circulation tests and events. B&W has agreed to provide the requested information to the staff including verification of its computer code to calculate natural circulation cooling. Such studies will include recent TMI-2 results.

While the staff believes that natural circulation cooling is effective, further evaluation of the B&W analyses and test information will be necessary to confirm the adequacy of this cooling mode.

4.0 ALTERNATIVES

We have briefly considered the pro-and-con of three alternatives related to the safety of continued operation of the B&W plants. They are listed below.

4.1 Further Bulletins

Pro

1. Bulletin process is simple for NRC, and has not proved a burden to industry (according to industry)
2. Temporary improvements can be implemented quickly.
3. We need more information of FMEA of ICS and plant transient behavior in order to make an informed decision; the bulletin is a fast and effective way to obtain information.

Con

1. Multiple bulletins on some subject poses potential for overloading operator.
2. Technical merits of revised designs not subject to usual thorough scrutiny of staff and applicant.
3. Needed information may take 1-2 months; delay in decision-making is not the most cautious thing to do.
4. Plant responses to bulletins are varied in substance.

4.2 Immediate Remedial Measures

Pro

1. Faster implementation of needed safety measures reduces the likelihood of another TMI in the interim.

Con

1. May not be enough time or adequate information for careful staff consideration.

4.3 Plant Shutdown

Pro

1. Conservative course of action.
2. Gives time for staff and industry to work in more orderly fashion.

Con

1. Difficult to enumerate the restart criteria.

C

APPENDIX C

ORDERS ON BABCOCK & WILCOX COMPANY PLANTS

After a series of discussions between the NRC staff and licensees of operating Babcock & Wilcox-designed plants, the licensees agreed to shut down these plants and keep them shut down until the actions identified in an April 25, 1979 status report to the Commission could be completed. This agreement was confirmed by a Commission Order to each licensee. The Orders contained both short-term and long-term modifications to be made by the licensees. Copies of the Orders are contained in this appendix. They are as follows:

Arkansas 1	-	5/17/79
Crystal River 3	-	5/16/79
Davis-Besse 1	-	5/16/79
Oconee	-	5/07/79
Rancho Seco	-	5/07/79

7590-01

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of ARKANSAS POWER & LIGHT COMPANY ARKANSAS NUCLEAR ONE, UNIT 1	)))))	Docket No. 50-313

<u>ORDER</u>

I.

The Arkansas Power & Light Company (the licensee or AP&L) is the holder of Facility Operating License No. DPR-51 which authorizes the operation of the nuclear power reactor known as the Arkansas Nuclear One, Unit 1 (the facility or ANO-1), at steady state power levels not in excess of 2568 megawatts thermal (rated power). The facility is a Babcock & Wilcox (B&W) designed pressurized water reactor (PWR) located at the licensee's site in Pope County, Arkansas.

II.

In the course of its evaluation to date of the accident at the Three Mile Island Unit No. 2 facility, which utilizes a B&W designed PWR, the Nuclear Regulatory Commission staff has ascertained that B&W designed reactors appear to be unusually sensitive to certain off-normal transient conditions originating in the secondary system. The features of the B&W design that contribute to this sensitivity are: (1) design of the steam generators to operate with relatively small liquid volumes in the secondary side; (2) the lack of direct initiation of reactor trip upon the occurrence of off-normal conditions in the feedwater system; (3) reliance on an integrated control system (ICS) to automatically regulate feedwater flow; (4) actuation before reactor trip of a pilot-operated relief valve on the primary system pressurizer (which, if the valve sticks open, can aggravate the event); and (5) a low steam generator elevation (relative to the reactor vessel) which provides a smaller driving head for natural circulation.

Because of these features, B&W designed reactors place more reliance on the reliability and performance characteristics of the auxiliary feedwater system, the integrated control system, and the emergency core cooling system (ECCS) performance to recover from frequent anticipated transients, such as loss of offsite power and loss of normal feedwater, than do other PWR designs. This, in turn, places a large burden on the plant operators in the event of off-normal system behavior during such anticipated transients.

As a result of a preliminary review of the Three Mile Island Unit No. 2 accident chronology, the NRC staff initially identified several human errors that occurred during the accident and contributed significantly to its severity. All holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's

Office of Inspection and Enforcement (IE). In addition, the NRC staff began an immediate reevaluation of the design features of B&W reactors to determine whether additional safety corrections or improvements were necessary with respect to these reactors. This evaluation involved numerous meetings with B&W and certain of the affected licensees.

The evaluation identified design features as discussed above which indicated that B&W designed reactors are unusually sensitive to certain off-normal transient conditions originating in the secondary system. As a result, an additional bulletin was issued by IE which instructed holders of operating licenses for B&W designed reactors to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer pilot-operated relief valve setting. Also, as a result of this evaluation, the NRC staff identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors. These were identified as items (a) through (e) on page 1-7 of the Office of Nuclear Reactor Regulation Status Report to the Commission of April 25, 1979.

After a series of discussions between the NRC staff and the licensee concerning possible design modifications and changes in operating procedures, the licensee agreed in a letter dated May 11, 1979, to perform promptly the following actions:

(a) Upgrade of the timeliness and reliability of the Emergency Feedwater (EFW) system by performing the items specified in Enclosure 1 of the licensee's May 11, 1979, letter. Changes in design will be submitted to the NRC staff for review.

(b) Develop and implement operating procedures for initiating and controlling EFW independent of Integrated Control System (ICS) control.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or on turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) At least one Licensed Operator who has had Three Mile Island Unit No. 2 (TMI-2) training on the B&W simulator will be assigned to the control room (one each shift).

In its letter the licensee also stated that ANO-1 was currently shut down and would remain shut down until (a) through (e) above are completed.

In addition to these modifications to be implemented promptly, the licensee has also proposed to carry out certain additional long-term modifications to further enhance the capability and reliability of the reactor to respond to various transient events. These are:

1) The items in Enclosure 2 of the licensee's letter of May 11, 1979, will be implemented during the next outage (following completion of the design change engineering) to cold shutdown conditions which is of sufficient length to accommodate the change, but no later than the next refueling outage. Further, the licensee will provide a schedule for implementing any other modifications identified as necessary as a result of the licensee's reviews shown on Enclosure 1 of the licensee's letter. The design changes will be submitted to the NRC staff for review.

2) The failure modes and effects analysis (FMEA) of the ICS is underway with high priority by B&W and will be submitted as soon as practicable.

3) The hard-wired trips addressed in Item (c) above will be upgraded to safety grade. This design change will be submitted to the NRC staff for review.

4) The licensee will continue operator training and drilling of response procedures as a part of an ongoing program to assure the high state of readiness and safe operation at ANO-1.

The Commission has concluded that the prompt actions set forth as (a) through (e) above are necessary to provide added reliability to the reactor system to respond safely to feedwater transients and should be confirmed by a Commission order.

The Commission finds that operation of ANO-1 should not be resumed until the actions described in paragraphs (a) through (e) above have been satisfactorily completed.

For the foregoing reasons, the Commission has found that the public health, safety and interest require that this Order be effective immediately.

III.

Copies of the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. 20555, and are being placed in the Commission's local public document room at Arkansas Polytechnic College, Russellville, Arkansas:

(1) Office of Nuclear Reactor Regulation Status Report on Feedwater Transients in B&W Plants, April 25, 1979.

(2) Letter from William Cavanaugh III (AP&L) to Harold Denton (NRR) dated May 11, 1979.

IV.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED THAT:

(1) The licensee shall take the following actions with respect to ANO-1:

(a) Upgrade of the timeliness and reliability of the EFW system by performing the items specified in Enclosure 1 of the licensee's letter of May 11, 1979. Provide changes in design for NRC review.

(b) Develop and implement operating procedures for initiating and controlling EFW independent of Integrated Control System control.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or on turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) Assign at least one Licensed Operator who has had TMI-2 training on the B&W simulator to the control room (one each shift).

(2) The licensee shall maintain ANO-1 in a shutdown condition until items (a) through (e) in paragraph (1) above are satisfactorily completed. Satisfactory completion will require confirmation by the Director, Office of Nuclear Reactor Regulation, that the actions specified have been taken, the specified analyses are acceptable, and the specified implementing procedures are appropriate.

(3) The licensee shall as promptly as practicable also accomplish the long-term modifications set forth in Section II of this Order.

V.

Within twenty (20) days of the date of this Order, the licensee or any person whose interest may be affected by this Order may request a hearing with respect to this Order. Any such request shall not stay the immediate effectiveness of this Order.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Chilk
Secretary of the Commission

Dated at Washington, D. C.
this 17th day of May 1979.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
FLORIDA POWER CORPORATION, ET AL	)	Docket No. 50-302
	)	
Crystal River Unit No. 3	)	
Nuclear Generating Plant	)	

ORDER

I.

Florida Power Corporation (FPC or the licensee) and eleven other co-owners are the holders of Facility Operating License No. DPR-72 which authorizes the operation of the nuclear power reactor known as Crystal River Unit No. 3 Nuclear Generating Plant (the facility or Crystal River Unit 3), at steady state power levels not in excess of 2452 megawatts thermal (rated power). The facility is a Babcock & Wilcox (B&W) designed pressurized water reactor (PWR) located at the licensees' site in Citrus County, Florida.

II.

In the course of its evaluation to date of the accident at the Three Mile Island Unit No. 2 facility, which utilizes a B&W designed PWR, the Nuclear Regulatory Commission staff has ascertained that B&W designed reactors appear to be unusually sensitive to certain off-normal transient conditions originating in the secondary system. The features of the B&W design that contribute to this sensitivity are: (1) design of the steam generators to operate with relatively small liquid volumes in the secondary side; (2) the lack of direct initiation of reactor trip upon the occurrence of off-normal conditions in the feedwater system; (3) reliance on an integrated control system (ICS) to automatically regulate feedwater flow; (4) actuation before reactor trip of a pilot-operated relief valve on the primary system pressurizer (which, if the valve sticks open, can aggravate the event); and (5) a low steam generator elevation (relative to the reactor vessel) which provides a smaller driving head for natural circulation.

Because of these features, B&W designed reactors place more reliance on the reliability and performance characteristics of the auxiliary feedwater system, the integrated control system, and the emergency core cooling system (ECCS) performance to recover from frequent anticipated transients, such as loss of offsite power and loss of normal feedwater, than do other PWR designs. This, in turn, places a large burden on the plant operators in the event of off-normal system behavior during such anticipated transients.

As a result of a preliminary review of the Three Mile Island Unit No. 2 accident chronology, the NRC staff initially identified several human errors that occurred during the accident and contributed significantly to its severity. All holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's Office of Inspection and Enforcement (IE). In addition, the NRC staff began an immediate reevaluation of the design features of B&W

reactors to determine whether additional safety corrections or improvements were necessary with respect to these reactors. This evaluation involved numerous meetings with B&W and certain of the affected licensees.

The evaluation identified design features as discussed above which indicated that B&W designed reactors are unusually sensitive to certain off-normal transient conditions originating in the secondary system. As a result, an additional bulletin was issued by IE which instructed holders of operating licenses for B&W designed reactors to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer pilot-operated relief valve setting. Also, as a result of this evaluation, the NRC staff identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors. These were identified as items (a) through (e) on page 1-7 of the Office of Nuclear Reactor Regulation Status Report to the Commission of April 25, 1979.

After a series of discussions between the NRC staff and the licensee concerning possible design modifications and changes in operating procedures, the licensee agreed in a letter dated May 1, 1979, to perform promptly the following actions:

(a) Upgrade the timeliness and reliability of delivery from the Emergency Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of May 1, 1979.

(b) Develop and implement operating procedures for initiating and controlling emergency feedwater independent of Integrated Control System control.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) All licensed reactor operators and senior reactor operators will have completed the Three Mile Island Unit No. 2 (TMI-2) simulator training at B&W.

In its letter the licensee also stated that the facility is shut down and would remain shut down until (a) through (e) above are completed.

In addition to these modifications to be implemented promptly, the licensee has also proposed to carry out certain additional long-term modifications to further enhance the capability and reliability of the reactor to respond to various transient events. These are:

- The licensee will make modifications to provide verification in the control room of emergency feedwater flow to each steam generator.

- The licensee will submit a failure mode and effects analysis of the Integrated Control System to the NRC staff as soon as practicable. The licensee stated that this analysis is now underway with high priority by B&W.

- The reactor trip following loss of main feedwater and/or trip of the turbine to be installed promptly pursuant to this Order will thereafter be upgraded so that the components are safety grade. The licensee will submit this design to the NRC staff for review.

- The licensee will continue reactor operator training and drilling of response procedures to assure a high state of preparedness.

The Commission has concluded that the prompt actions set forth as (a) through (e) above are necessary to provide added reliability to the reactor system to respond safely to feedwater transients and should be confirmed by a Commission order.

The Commission finds that operation of the facility should not be resumed until the actions described in paragraphs (a) though (e) above have been satisfactorily completed.

For the foregoing reasons, the Commission has found that the public health, safety and interest require that this Order be effective immediately.

III.

Copies of the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555, and are being placed in the Commission's local public document room in the Crystal River Public Library, Crystal River, Florida, 32629:

(1) Office of Nuclear Reactor Regulation Status Report on Feedwater Transients in B&W Plants, April 25, 1979.

(2) Letter from B. L. Griffin (FPC) to Harold Denton (NRR) dated May 1, 1979.

IV.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED THAT:

(1) The licensee shall take the following actions with respect to Crystal River Unit 3:

(a) Upgrade the timeliness and reliability of delivery from the Emergency Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of May 1, 1979.

(b) Develop and implement operating procedures for initiating and controlling emergency feedwater independent of Integrated Control System control.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) All licensed reactor operators and senior reactor operators will have completed the TMI-2 simulator training at B&W.

(2) The licensee shall maintain Crystal River Unit 3 in a shutdown condition (the facility was shut down on April 23, 1979) until items (a) through (e) in paragraph (1) above are satisfactorily completed. Satisfactory completion will require confirmation by the Director, Office of Nuclear Reactor Regulation, that the actions specified have been taken, the specified analyses are acceptable, and the specified implementing procedures are appropriate.

(3) The licensee shall as promptly as practicable also accomplish the long-term modifications set forth in Section II of this Order.

V.

Within twenty (20) days of the date of this Order, the licensees or any person whose interest may be affected by this Order may request a hearing with respect to this Order. Any such request shall not stay the immediate effectiveness of this Order.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Chilk
Secretary of the Commission

Dated at Washington, D.C.
this 16th day of May 1979.

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Office of the Secretary
Docketing & Service
Section

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of THE TOLEDO EDISON COMPANY AND THE CLEVELAND ELECTRIC ILLUMINATING COMPANY Davis-Besse Nuclear Power Station, Unit No. 1	))))))))	Docket No. 50-346

ORDER

I.

The Toledo Edison Company (TECO) and The Cleveland Electric Illuminating Company (the licensees), are holders of Facility Operating License No. NPF-3 which authorizes the operation of the nuclear power reactor known as Davis-Besse Nuclear Power Station, Unit No. 1 (the facility or Davis-Besse 1), at steady state power levels not in excess of 2772 megawatts thermal (rated power). The facility is a Babcock & Wilcox (B&W) designed pressurized water reactor (PWR) located at the licensees' site in Ottawa County, Ohio.

II.

In the course of its evaluation to date of the accident at the Three Mile Island Unit No. 2 facility, which utilizes a B&W designed PWR, the Nuclear Regulatory Commission staff has ascertained that B&W designed reactors appear to be unusually sensitive to certain off-normal transient conditions originating in the secondary system. The features of the B&W design that contribute to this sensitivity are: (1) design of the steam generators to operate with relatively small liquid volumes in the secondary side; (2) the lack of direct initiation of reactor trip upon the occurrence of off-normal conditions in the feedwater system; (3) reliance on an integrated control system (ICS) to automatically regulate feedwater flow; (4) actuation before reactor trip of a pilot-operated relief valve on the primary system pressurizer (which, if the valve sticks open, can aggravate the event); and (5) a low steam generator elevation (relative to the reactor vessel) which provides a smaller driving head for natural circulation.*

Because of these features, B&W designed reactors place more reliance on the reliability and performance characteristics of the auxiliary feedwater system, the ICS, and the emergency core cooling system (ECCS) performance to recover from frequent anticipated transients, such as loss of offsite power and loss of normal feedwater, than do other PWR designs. This, in turn, places a large burden on the plant operators in the event of off-normal system behavior during such anticipated transients.

*It is noted that although features numbers 3 and 5 do not apply to Davis-Besse 1 to the same extent as they apply to other currently licensed B&W designed reactors, the other features are fully applicable.

As a result of a preliminary review of the Three Mile Island Unit No. 2 accident chronology, the NRC staff initially identified several human errors that occurred during the accident and contributed significantly to its severity. All holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's Office of Inspection and Enforcement (IE). In addition, the NRC staff began an immediate reevaluation of the design features of B&W reactors to determine whether additional safety corrections or improvements were necessary with respect to these reactors. This evaluation involved numerous meetings with B&W and certain of the affected licensees.

The evaluation identified design features as discussed above which indicated that B&W designed reactors are unusually sensitive to certain off-normal transient conditions originating in the secondary system. As a result, an additional bulletin was issued by IE which instructed holders of operating licenses for B&W reactors to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer pilot-operated relief valve setting. Also, as a result of this evaluation, the NRC staff identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors.

These were identified as items (a) through (e) on page 1-7 of the Office of Nuclear Reactor Regulation Status Report to the Commission of April 25, 1979.

After a series of discussions between the NRC staff and the licensees concerning possible design modifications and changes in operating procedures, the licensees agreed in letters dated April 27 and May 4, 1979, to implement promptly the following actions:

(a) Review all aspects of the safety grade auxiliary feedwater system to further upgrade components for added reliability and performance. Present modifications will include the addition of dynamic braking on the auxiliary feedpump turbine speed changer and provision of means for control room verification of the auxiliary feedwater flow to the steam generators. This means of verification will be provided for one steam generator prior to startup from the present maintenance outage and for the other steam generator as soon as vendor-supplied equipment is available (estimated date is June 1, 1979). In addition, the licensees will review and verify the adequacy of the auxiliary feedwater system capacity.

(b) Revise operating procedures as necessary to eliminate the option of using the Integrated Control System as a backup means for controlling auxiliary feedwater flow.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) All licensed reactor operators and senior reactor operators will have completed the Three Mile Island Unit No. 2 simulator training at B&W.

(f) Submit a reevaluation of the TECO analysis of the need for automatic or administrative control of steam generator level setpoints during auxiliary feedwater system operation, previously submitted by TECO letter of December 22, 1978, in light of the Three Mile Island Unit No. 2 incident.

(g) Submit a review of the previous TECO evaluation of the September 24, 1977 event involving equipment problems and depressurization of the primary system at Davis-Besse 1 in light of the Three Mile Island Unit No. 2 incident.

In its letters the licensees also stated that the actions listed in (a) through (g) above would, except as noted in item (a), be completed prior to startup from the current maintenance outage.

In addition to these modifications to be implemented promptly, the licensees have also proposed to carry out certain additional long-term modifications to further enhance the capability and reliability of the reactor to respond to various transient events. These are:

- The licensees will continue to review performance of the auxiliary feedwater system for assurance of reliability and performance.

- The licensees will submit a failure mode and effects analysis of the ICS to the NRC staff as soon as practicable. The licensees stated that this analysis is now underway with high priority by B&W.

- The reactor trip following loss of main feedwater and/or trip of the turbine to be installed promptly pursuant to this Order will thereafter be upgraded so that the components are safety grade. The licensees will submit this design to the NRC staff for review.

- Continued attention will be given to transient analysis and procedures for management of small breaks.

- The licensees will continue reactor operator training and drilling of response procedures to assure a high state of preparedness.

The Commission has concluded that the prompt actions set forth as (a) through (g) above are necessary to provide added reliability to the reactor system to respond safely to feedwater transients and should be confirmed by a Commission order.

The Commission finds that operation of Davis-Besse 1 should not be resumed until the actions described in paragraphs (a) through (g) above, with the exception as noted in item (a), have been satisfactorily completed.

For the foregoing reasons, the Commission has found that the public health, safety and interest require that this Order be effective immediately.

III.

Copies of the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555, and are being placed in the Commission's local public document room in the Ida Rupp Public Library, 310 Madison Street, Port Clinton, Ohio 43452:

(1) Office of Nuclear Reactor Regulation Status Report on Feedwater Transients in B&W Plants, April 25, 1979.

(2) Letters from Lowell E. Roe (TECO) to Harold Denton (NRR) dated April 27 and May 4, 1979.

IV.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED THAT:

(1) The licensees shall take the following actions with respect to Davis-Besse 1:

(a) Review all aspects of the safety grade auxiliary feedwater system to further upgrade components for added reliability and performance. Present modifications will include the addition of dynamic braking on the auxiliary feedpump turbine speed changer and provision of means for control room verification of the auxiliary feedwater flow to the steam generators. This means of verification will be provided for one steam generator prior to startup from the present maintenance outage and for the other steam generator as soon as vendor-supplied equipment is available (estimated date is June 1, 1979). In addition, the licensees will review and verify the adequacy of the auxiliary feedwater system capacity.

(b) Revise operating procedures as necessary to eliminate the option of using the Integrated Control System as a backup means for controlling the auxiliary feedwater system.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) All licensed reactor operators and senior reactor operators will have completed the Three Mile Island Unit No. 2 simulator training at B&W.

(f) Submit a reevaluation of the TECO analysis of the need for automatic or administrative control of steam generator level setpoints during auxiliary feedwater system operation previously submitted by TECO letter dated December 22, 1978, in light of the Three Mile Island No. 2 incident.

(g) Submit a review of the previous TECO evaluation of the September 24, 1977 event involving equipment problems and depressurization of the primary system at Davis-Besse 1 in light of the Three Mile Island Unit No. 2 incident.

(2) The licensees shall maintain Davis-Besse 1 in a shutdown condition until items (a) through (g) in paragraph (1), except as noted in item (a), above are satisfactorily completed. Satisfactory completion will require confirmation by the Director, Office of Nuclear Reactor Regulation, that the

actions specified have been taken, the specified analyses are acceptable, and the specified implementing procedures are appropriate.

(3) The licensees shall as promptly as practicable also accomplish the long-term modifications set forth in Section II of this Order.

V.

Within twenty (20) days of the date of this Order, the licensees or any person whose interest may be affected by this Order may request a hearing with respect to this Order. Any such request shall not stay the immediate effectiveness of this Order.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Chilk
Secretary of the Commission

Dated at Washington, D.C.,
this 16th day of May 1979.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	Dockets Nos. 50-269
	)	50-270
DUKE POWER COMPANY	)	and 50-287
	)	
Oconee Nuclear Station, Units Nos. 1, 2 and 3	)	

ORDER

I.

The Duke Power Company (the licensee), is the holder of Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 which authorize the operation of the nuclear power reactors known as Oconee Nuclear Station, Units Nos. 1, 2 and 3 (the facilities, or Oconee 1, 2 and 3), at steady state power levels not in excess of 2568 megawatts thermal (rated power) for each unit. The facilities are Babcock & Wilcox (B&W) designed pressurized water reactors (PWR's) located at the licensee's site in Oconee County, South Carolina.

II.

In the course of its evaluation to date of the accident at the Three Mile Island Unit No. 2 facility, which utilizes a B&W designed PWR, the Nuclear Regulatory Commission staff has ascertained that B&W designed reactors appear to be unusually sensitive to certain off-normal transient conditions originating in the secondary system. The features of the B&W design that contribute to this sensitivity are: (1) the design of steam generators to operate with relatively small liquid volumes in the secondary side; (2) the lack of direct initiation of reactor trip upon the occurrence of off-normal conditions in the feedwater system; (3) reliance on an integrated control system (ICS) to automatically regulate feedwater flow; (4) actuation before reactor trip of a pilot-operated relief valve on the primary system pressurizer (which, if the valve sticks open, can aggravate the event); and (5) a low steam generator elevation (relative to the reactor vessel) which provides a smaller driving head for natural circulation.

Because of these features, B&W designed reactors place more reliance on the reliability and performance characteristics of the auxiliary feedwater system, the ICS, and the emergency core cooling system (ECCS) performance to recover from frequent anticipated transients, such as loss of offsite power and loss of normal feedwater, than do other PWR designs. This, in turn, places a large burden on the plant operators in the event of off-normal system behavior during such anticipated transients.

As a result of a preliminary review of the Three Mile Island Unit No. 2 accident chronology, the NRC staff initially identified several human errors that occurred during the accident and contributed significantly to its severity. All holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's Office of Inspection and Enforcement (IE). In addition, the NRC staff began an immediate reevaluation of the design features of B&W reactors to determine whether additional safety corrections or improvements were necessary with respect to these reactors. This evaluation involved numerous meetings with B&W and certain of the affected licensees.

The evaluation identified design features as discussed above which indicated that B&W designed reactors are unusually sensitive to certain off-normal transient conditions originating in the secondary system. As a result, an additional bulletin was issued by IE which instructed holders of operating licenses for B&W designed reactors to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer pilot-operated relief valve setting. Also, as a result of this evaluation, the NRC staff identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors. These were identified as items (a) through (e) on page 1-7 of the Office of Nuclear Reactor Regulation Status Report to the Commission on April 25, 1979.

After a series of discussions between the NRC staff and the licensee concerning possible design modifications and changes in operating procedures, the licensee agreed in letters dated April 25, 26, and May 4, 1979 to perform promptly the following actions:

(a) Install automatic starting of the interconnected emergency feedwater system so that all three pumps will receive a start signal from any affected unit, and test the system for stability. The emergency feedwater pump discharge flow will be connected to the interconnection headers such that each or all emergency feedwater pumps can supply water to any unit. Until these modifications and tests are completed, operating personnel have been stationed at each emergency feedwater pump with a direct communication link to that unit's control room. In addition, the following procedural changes, put into effect on April 25, 1979 to enhance the reliability of the emergency feedwater system, will remain in force:

   (1) The discharges of these pumps have been tied together by alignment of manual valves such that each and all of the pumps can supply emergency feedwater to any Oconee Unit requiring it.

(2) Administrative controls have been established so that in the event of loss of both main feedwater pumps on an affected unit, that unit's emergency feedwater pump will start automatically, backed up by remote manual start from the control room. If the pump fails to start automatically, the operator stationed at that pump will start the pump locally, and has been trained to do so. In addition, the other two available emergency feedwater pumps will be started remotely from their unit's control room or locally if required to provide two more redundant sources of feedwater to the affected unit.

(3) Emergency feedwater flow to the steam generators will be assured by the control room operator who has been trained to maintain the necessary level.

(b) Develop and implement operating procedures for initiating and controlling emergency feedwater independent of Integrated Control System control.

(c) Implement a hard-wired control-grade reactor trip on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) All licensed reactor operators and senior reactor operators will have completed the TMI-2 simulator training at B&W.

(f) Station in the control room an additional full-time Senior Reactor Operator (SRO) (or previously licensed SRO) with Three Mile Island training for each operating unit to assist with guidance and possible manual action in case of transients until items (a) through (e) are completed.

In its letters the licensee also stated that (1) Oconee 3 would be shut down on April 28, 1979, and remain shutdown until (a) through (e) above are completed (the facility was shut down on April 28, 1979 as stated); (2) a second Oconee unit would be shut down on May 12, 1979, if items (a) through (e) have not been previously accomplished and remain shut down until items (a) through (e) have been completed; and, (3) a third Oconee unit would be shut down on May 19, 1979, if items (a) through (e) have not been previously accomplished and will remain shut down until completion of items (a) through (e).

In addition to these modifications to be implemented promptly, the licensee has also proposed to carry out certain additional long-term actions to increase the capability and reliability of the reactors to respond to various transient events. These are:

- The licensee will install two motor driven pumps for each Oconee unit, as more particularly described as Part III of a letter from W.O. Parker to the NRC of April 25, 1979, to provide greater assurance of emergency feedwater supply. The licensee will submit this system concept and analysis to the NRC staff for review.

- The licensee will submit a failure mode and effects analysis of the Integrated Control System to the NRC staff as soon as practicable. The licensee states that this analysis is now underway with high priority by B&W.

- The reactor trip on loss of the main feedwater and/or trip of the turbine to be installed promptly pursuant to this Order will thereafter be upgraded so that the components are safety grade. The licensee will submit this design to the NRC staff for review.

- The licensee will continue reactor operator training and drilling of response procedures to assure a high state of preparedness.

The Commission has concluded that the prompt actions set forth as (a) through (e) above are necessary to provide added reliability to the reactor system to respond safely to feedwater transients and should be confirmed by a Commission order. The immediate procedural changes to assure redundant sources of auxiliary feedwater that were put into effect on April 25 at the two operating Oconee units, as described in paragraph (a) above, and the immediate additions to the operating staff, as described in paragraph (f) above, provide the bases for continued safe operation of those facilities during the interim period until May 12 and May 19, 1979, respectively. The Commission finds, however, that operation of all units should not be resumed or continued on an indefinite basis until actions described in paragraphs (a) through (e) above have been satisfactorily completed.

For the foregoing reasons, the Commission has found that the public health, safety and interest require that this Order be effective immediately.

III.

Copies of the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555, and are being placed in the Commission's local public document room at the Oconee County Library, 201 South Spring, Walhalla, South Carolina 29691:

(1) Office of Nuclear Reactor Regulation Status Report on Feedwater Transients in B&W Plants, April 25, 1979.

(2) Letter from W. S. Lee (Duke Power Company) to Harold Denton (NRR), dated April 25, 1979.

(3) Two letters from W. O. Parker, Jr. (Duke Power Company) to Harold Denton (NRR), dated April 25, 1979.

(4) Letter from W. H. Owens (Duke Power Company) to Roger J. Mattson (NRR), dated April 25, 1979.

(5) Letter from W. S. Lee (Duke Power Company) to Harold Denton (NRR), dated April 26, 1979.

(6) Letter from W. O. Parker, Jr. (Duke Power Company) to James P. O'Reilly (IE), dated May 4, 1979.

IV.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED THAT:

(1) The licensee shall take the following actions with respect to Oconee 1, 2 and 3:

(a) Install automatic starting of the interconnected emergency feedwater system so that all three pumps will receive a start signal from any affected unit, and test the system for stability. The emergency feedwater pump discharge flow will be connected to the interconnection headers such that each or all of the emergency feedwater pumps can supply water to any unit. Until these modifications and tests are completed, operating personnel will be stationed at each emergency feedwater pump with a direct communication link to that unit's control room. In addition, the following procedural changes, put into effect on April 25, 1979 to enhance the reliability of the emergency feedwater system, will remain in force:

(1) The discharges of these pumps have been tied together by alignment of manual valves such that each and all of the pumps can supply emergency feedwater to any Oconee Unit requiring it.

(2) Administrative controls have been established so that in the event of loss of both main feedwater pumps on an affected unit, that unit's emergency feedwater pump will start automatically, backed up by remote manual start from the control room. If the pump fails to start automatically, the operator stationed at that pump will start the pump locally, and has been trained

to do so. In addition, the other two available emergency feedwater pumps will be started remotely from their unit's control room or locally if required to provide two more sources of feedwater to the affected unit.

(3) Emergency feedwater flow to the steam generators will be assured by the control room operator who has been trained to maintain the necessary level.

(b) Develop and implement operating procedures for initiating and controlling emergency feedwater independent of Integrated Control System control.

(c) Implement a hard-wired control-grade reactor trip on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) All licensed reactor operators and senior reactor operators assigned to the Oconee control rooms will have completed the TMI-2 simulator training at B&W.

(f) Station in the control room an additional full-time Senior Reactor Operator (SRO) (or previously licensed SRO) with Three Mile Island training for each operating unit to assist with guidance and possible manual actions until items (a) through (e) are completed.

(2) The licensee shall maintain Oconee 3 in a shut down condition (the facility was shut down on April 28, 1979) until items (a) through (e) in paragraph (1) above are satisfactorily completed and such completion has been confirmed by the Director, Office of Nuclear Reactor Regulation.

(3) The licensee shall shut down a second of the three Oconee units on May 12, 1979, unless items (a) through (e) in paragraph (1) above have been satisfactorily completed and the completion has been confirmed by the Director, Office of Nuclear Reactor Regulation, before that date. In the event the second unit is shut down on May 12, 1979, it will remain shutdown until items (a) through (e) in paragraph (1) above are satisfactorily completed and such completion has been confirmed by the Director, Office of Nuclear Reactor Regulation.

(4) The licensee shall shut down the third of the three Oconee units on May 19, 1979, unless items (a) through (e) in paragraph (1) above have been satisfactorily completed and the completion has been confirmed by the Director, Office of Nuclear Reactor Regulation, before that date. In the event the third unit is shut down on May 19, 1979, it shall remain shut down until items (a) through (e) in paragraph (1) above are satisfactorily completed and such completion has been confirmed by the Director, Office of Nuclear Reactor Regulation.

(5) The licensee shall as promptly as practicable also accomplish the long-term modifications set forth in Section II of this Order.

Satisfactory completion of items (a) through (e) in paragraph (1) and in paragraphs (2) through (4) above will require confirmation by the Director, Office of Nuclear Reactor Regulation, that the actions specified have been taken, the specified analyses are acceptable, and the specified implementing procedures are appropriate.

V.

Within twenty (20) days of the date of this Order, the licensee or any person whose interest may be affected by this Order may request a hearing with respect to this Order. Any such request shall not stay the immediate effectiveness of this Order.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Chilk
Secretary of the Commission

Dated at Washington, DC
this 7th day of May 1979.

7590-01

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
SACRAMENTO MUNICIPAL UTILITY DISTRICT	)	Docket No. 50-312
	)	
Rancho Seco Nuclear Generating Station	)	

ORDER

I.

The Sacramento Municipal Utility District (the licensee or SMUD) is the holder of Facility Operating License No. DPR-54 which authorizes the operation of the nuclear power reactor known as the Rancho Seco Nuclear Generating Station (the facility or Rancho Seco), at steady state power levels not in excess of 2772 megawatts thermal (rated power). The facility is a Babcock & Wilcox (B&W) designed pressurized water reactor (PWR) located at the licensee's site in Sacramento County, California.

II.

In the course of its evaluation to date of the accident at the Three Mile Island Unit No. 2 facility, which utilizes a B&W designed PWR, the Nuclear Regulatory Commission staff has ascertained that B&W designed reactors appear to be unusually sensitive to certain off-normal transient conditions originating in the secondary system. The features of the B&W design that contribute to this sensitivity are: (1) design of the steam generators to operate with relatively small liquid volumes in the secondary side; (2) the lack of direct initiation of reactor trip upon the occurrence of off-normal conditions in the feedwater system; (3) reliance on an integrated control system (ICS) to automatically regulate feedwater flow; (4) actuation before reactor trip of a pilot-operated relief valve on the primary system pressurizer (which, if the valve sticks open, can aggravate the event); and (5) a low steam generator elevation (relative to the reactor vessel) which provides a smaller driving head for natural circulation.

Because of these features, B&W designed reactors place more reliance on the reliability and performance characteristics of the auxiliary feedwater system, the integrated control system, and the emergency core cooling system (ECCS) performance to recover from frequent anticipated transients, such as loss of offsite power and loss of normal feedwater, than do other PWR designs. This, in turn, places a large burden on the plant operators in the event of off-normal system behavior during such anticipated transients.

As a result of a preliminary review of the Three Mile Island Unit No. 2 accident chronology, the NRC staff initially identified several human errors that occurred during the accident and contributed significantly to its severity All holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's Office of Inspection and Enforcement (IE). In addition, the NRC staff began an immediate reevaluation of the design features of B&W reactors to determine whether additional safety corrections or improvements were necessary with respect to these reactors. This evaluation involved numerous meetings with B&W and certain of the affected licensees.

The evaluation identified design features as discussed above which indicated that B&W designed reactors are unusually sensitive to certain off-normal transient conditions originating in the secondary system. As a result, an additional bulletin was issued by IE which instructed holders of operating licenses for B&W designed reactors to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer pilot-operated relief valve setting. Also, as a result of this evaluation, the NRC staff identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors. These were identified as items (a) through (e) on page 1-7 of the Office of Nuclear Reactor Regulation Status Report to the Commission of April 25, 1979.

After a series of discussions between the NRC staff and the licensee concerning possible design modifications and changes in operating procedures, the licensee agreed in a letter dated April 27, 1979, to perform promptly the following actions:

(a) Upgrade the timeliness and reliability of delivery from the Auxiliary Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of April 27, 1979.

(b) Develop and implement operating procedures for initiating and controlling auxiliary feedwater independent of Integrated Control System control.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) Provide for one Senior Licensed Operator assigned to the control room who has had Three Mile Island Unit No. 2 (TMI-2) training on the B&W simulator.

In its letter the licensee also stated that Rancho Seco would be shut down on April 28, 1979 and would remain shut down until (a) through (e) above are completed (The facility was shut down on April 28, 1979 as stated).

In addition to these modifications to be implemented promptly, the licensee has also proposed to carry out certain additional long-term modifications to further enhance the capability and reliability of the reactor to respond to various transient events. These are:

- The licensee will provide to the NRC staff a proposed schedule for implementation of identified design modifications which specifically relate to items 1 through 9 of Enclosure 1 to the licensee's letter of April 27, 1979, and would significantly improve safety.

- The licensee will submit a failure mode and effects analysis of the Integrated Control System to the NRC staff as soon as practicable. The licensee stated that this analysis is now underway with high priority by B&W.

- The reactor trip following loss of main feedwater and/or trip of the turbine to be installed promptly pursuant to this Order will thereafter be upgraded so that the components are safety grade. The licensee will submit this design to the NRC staff for review.

- The licensee will continue operator training and have a minimum of two licensed operators per shift with TMI-2 simulator training at B&W by June 1, 1979. Thereafter, at least one licensed operator with TMI-2 simulator training at B&W will be assigned to the control room. All training of licensed personnel will be completed by June 28, 1979.

The Commission has concluded that the prompt actions set forth as (a) through (e) above are necessary to provide added reliability to the reactor system to respond safely to feedwater transients and should be confirmed by a Commission order.

The Commission finds that operation of Rancho Seco should not be resumed until the actions described in paragraphs (a) through (e) above have been satisfactorily completed.

For the foregoing reasons, the Commission has found that the public health, safety and interest require that this Order be effective immediately.

III.

Copies of the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555, and are being placed in the Commission's local public document room in the Business and Municipal Department, Sacramento City - County Library, 828 I Street, Sacramento, California 95814:

(1) Office of Nuclear Reactor Regulation Status Report on Feedwater Transients in B&W Plants, April 25, 1979.

(2) Letter from J. J. Mattimoe (SMUD) to Harold Denton (NRR) dated April 27, 1979.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED THAT:

(1) The licensee shall take the following actions with respect to Rancho Seco:

(a) Upgrade the timeliness and reliability of delivery from the Auxiliary Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of April 27, 1979.

(b) Develop and implement operating procedures for initiating and controlling auxiliary feedwater independent of Integrated Control System control.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) Provide for one Senior Licensed Operator assigned to the control room who has had Three Mile Island Unit No. 2 (TMI-2) training on the B&W simulator.

(2) The licensee shall maintain Rancho Seco in a shutdown condition (the facility was shut down on April 28, 1979) until items (a) through (e) in paragraph (1) above are satisfactorily completed. Satisfactory completion will require confirmation by the Director, Office of Nuclear Reactor Regulation, that the actions specified have been taken, the specified analyses are acceptable, and the specified implementing procedures are appropriate.

(3) The licensee shall as promptly as practicable also accomplish the long-term modifications set forth in Section II of this Order.

V.

Within twenty (20) days of the date of this Order, the licensee or any person whose interest may be affected by this Order may request a hearing with respect to this Order. Any such request shall not stay the immediate effectiveness of this Order.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Chilk
Secretary of the Commission

Dated at Washington, D.C.
this 7th day of May 1979.

D

APPENDIX D

LETTERS LIFTING ORDERS

The licensees of the five operating reactors submitted responses to the confirmatory orders (see Appendix C) indicating actions taken to implement short-term modifications. Following review and evaluation of the responses, it was determined that the licensees had satisfactorily completed the short-term requirements. Subsequently, the NRC issued letters lifting the Orders with respect to short-term modifications. The letters were as follows:

Arkansas 1	- 5/31/79
Davis-Besse 1	- 7/06/79
Crystal River 3	- 7/06/79
Oconee	- 5/18/79
Oconee, Supplement 1	- 10/10/79
Rancho Seco	- 6/27/79

A copy of each follows in this appendix.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 31, 1979

Docket No.: 50-313

Mr. William Cavanaugh, III
Vice President, Generation
and Construction
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Cavanaugh:

By Order of May 17, 1979, the Commission confirmed your undertaking a series of actions, both immediate and long term, to increase the capability and reliability of Arkansas Nuclear One, Unit No. 1 (ANO-1) to respond to various transient events. In addition, the Order confirmed that ANO-1 was shutdown and would not be restarted until the following actions had been accomplished:

(a) Upgrade of the timeliness and reliability of the Emergency Feedwater System (EFW) by performing the items specified in Enclosure 1 of the licensee's letter of May 11, 1979. Provide changes in design for NRC review.

(b) Develop and implement operating procedures for initiating and controlling EFW independent of Integrated Control System (ICS) control.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or on turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) Assign at least one Licensed Operator who has had Three Mile Island, Unit No. 2 training on the Babcock & Wilcox simulator to the control room (one each shift).

By submittal of May 17, 1979, as supplemented by letters dated May 21, 22, 23, 24 and 29, 1979, you have documented the actions taken in response to the May 17 Order. I have reviewed this submittal, and am satisfied that, with respect to ANO-1, you have satisfactorily completed the actions prescribed in items (a) through (e) of paragraph (1) of Section IV of the Order, the specified modifications and analyses are acceptable, and the specified implementing procedures are appropriate. The bases for these conclusions are set forth in the enclosed Safety Evaluation.

As noted on page 5 of the Safety Evaluation you will be required to conduct a test during power operation to demonstrate operator capability to assume manual control of the EFW system independent of ICS. In addition, we have discussed the need for monitoring core exit temperature with your staff and they have agreed to provide a minimum of sixteen thermocouple indications of core exit temperature in the control room prior to startup. Also, your staff has agreed to provide an additional sixteen thermocouple indications of core exit temperature in the control room by October 31, 1979.

Appropriate Technical Specifications for Limiting Conditions for Operation and for surveillance requirements should be developed as soon as practicable and provided to the staff within seven days with regard to the design and procedural changes which have been completed in compliance with the provisions of the May 17, 1979 Commission Order. The revised Technical Specifications should cover:

(1) Changes to the EFW System;

(2) Plant alignment changes made to ensure control of the EFW independnt of the ICS;

(3) Addition of the Anticipatory Reactor Trip; and

(4) EFW capacity.

We note that by letter dated April 24, 1979, you have submitted proposed Technical Specifications for changes in setpoints for high pressure reactor trip and pilot operated relief valve actuation.

Also by letter dated May 16, 1979 you have submitted proposed changes to the Technical Specifications which define limiting conditions of operation upon loss of EFW equipment.

Within 30 days of receipt of this letter, you should provide us with your schedule for completion of the long-term modifications described in Section II of the May 17 Order.

My finding of satisfactory compliance with the requirements of items (a) through (e) of paragraph (1) of Section IV of the Order will permit resumption of operation in accordance with the terms of the Commission's Order; it in no way affects your duty to continue in effect all of the above provisions of the Order pending your submission and approval by the Commission of the Technical Specification changes necessary for each of the required modifications.

Sincerely,

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Enclosure:
Notice of Authorization
to Resume Operation
Safety Evaluation
cc w/enclosure:
See next page

cc w/enclosure(s):
Phillip K. Lyon, Esq.
House, Holms & Jewell
1550 Tower Building
Little Rock, Arkansas 72201

Mr. David C. Trimble
Manager, Licensing
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Mr. James P. O'Hanlon
General Manager
Arkansas Nuclear One
P. O. Box 608
Russellville, Arkansas 72801

Mr. William Johnson
U. S. Nuclear Regulatory Commission
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1747 Pennsylvania Avenue, N.W.
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Arkansas Polytechnic College
Russellville, Arkansas 72801

Honorable Ermil Grant
Acting County Judge of Pope County
Pope County Courthouse
Russellville, Arkansas 72801

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Region VI Office
ATTN: EIS COORDINATOR
1201 Elm Street
First International Building
Dallas, Texas 75270

Director, Bureau of Environmental
Health Services
4815 West Markham Street
Little Rock, Arkansas 72201

UNITED STATES NUCLEAR REGULATORY COMMISSION

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

NOTICE OF AUTHORIZATION TO RESUME OPERATION

The United States Nuclear Regulatory Commission issued an Order on May 17, 1979 (44 FR 29997, May 23, 1979), to Arkansas Power & Light Company (the licensee), holder of Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1), confirming that the licensee accomplish a series of actions, both immediate and long term, to increase the capability and reliability of ANO-1 to respond to various transient events. In addition, the Order confirmed that the licensee would maintain ANO-1 in a shutdown condition until the following actions had been satisfactorily completed:

(a) Upgrade of the timeliness and reliability of the Emergency Feedwater (EFW) System by performing the items specified in Enclosure 1 of the licensee's letter of May 11, 1979. Provide changes in design for NRC review.

(b) Develop and implement operating procedures for initiating and controlling EFW independent of Integrated Control System control.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or on turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) Assign at least one Licensed Operator who has had Three Mile Island Unit No. 2 training on the Babcock & Wilcox simulator to the control room (one each shift).

By submittal of May 17, 1979, as supplemented by letters dated May 21 and 22, 1979, the licensee has documented the actions taken in response to the May 17, Order. Notice is hereby given that the Director of Nuclear Reactor Regulation (the Director) has reviewed this submittal and has concluded that the licensee has satisfactorily completed the actions prescribed in items (a) through (e) of paragraph (1) of Section IV of the Order, that the specified modifications and analyses are acceptable and the specified implementing procedures are appropriate. Accordingly, by letter dated May 31, 1979, the Director has authorized the licensee to resume operation of ANO-1. The bases for the Director's conclusions are more fully set forth in a Safety Evaluation dated May 31, 1979.

Copies of (1) the licensee's letters dated May 17, 21 and 22, 1979, (2) the Director's letter dated May 31, 1979, and (3) the Safety Evaluation dated May 31, 1979 are available for inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. 20555, and are being placed in the Commission's local public document room at the Arkansas Polytechnic College, Russellville, Arkansas. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Dated at Bethesda, Maryland
this 31st day of May 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

EVALUATION OF LICENSEE'S COMPLIANCE WITH THE NRC ORDER DATED MAY 17, 1979 ARKANSAS POWER & LIGHT COMPANY ARKANSAS NUCLEAR ONE, UNIT 1 DOCKET NO. 50-313

INTRODUCTION

By order dated May 17, 1979, (the order) the Arkansas Power & Light Company (AP&L or the licensee) was directed by the NRC to take certain actions with respect to Arkansas Nuclear One, Unit 1. Prior to this order and as a result of a preliminary review of the Three Mile Island Unit No. 2 accident, the NRC staff initially identified several human errors that contributed significantly to the severity of the event. All holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's Office of Inspection and Enforcement (IE). Subsequently, an additional bulletin was issued by IE which instructed holders of operating licenses for B&W designed reactors to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer power-operated relief valve (PORV) setting.

The NRC staff identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors. Those were identified as items (a) through (e) in page 1-7 of the Office of Nuclear Reactor Regulation Status Report to the Commission on April 25, 1979. After a series of discussions between the NRC staff and the licensee concerning possible design modifications and changes in operating procedures, the licensee agreed in a letter dated May 11, 1979 to perform promptly certain actions. The Commission found that operation of the plant should not be resumed or continued on an indefinite basis until actions described in paragraphs (a) through (e) of paragraph (1) of Section IV of the order were satisfactorily completed.

Our evaluation of the licensee's compliance with items (a) through (e) of paragraph (1) of Section IV of the order is given below. In performing this evaluation we have utilized additional information provided by the licensee on May 11, 16, 17, 21, 22, 23, 24, and 29, 1979 and numerous discussions with the licensee's staff. Confirmation of design and procedure changes was made by members of the NRC staff at the ANO-1 site. An audit of the ANO-1 reactor operators was also performed by the NRC staff to assure that the design and procedure changes were understood and were being correctly implemented by the operators.

EVALUATION

Item a

It was ordered that the licensee take the following action;

"Upgrade of the timeliness and reliability of the EFW system by performing the items specified in Enclosure 1 of the licensee's letter of May 11, 1979."

The ANO-1 design has one turbine-driven emergency feedwater (EFW) pump that is automatically actuated and controlled independent of offsite power, and one motor-driven EFW pump that must be manually transferred to a vital AC bus if offsite power is lost. By reference above to Enclosure (1) of the licensee's letter of May 11, 1979, it was ordered that the licensee;

"1. Review procedures, revise as necessary and conduct training to ensure timely and proper starting of motor driven emergency feedwater (EFW) pump from an engineered safeguards bus upon loss of offsite power. Conduct a test of the manual startup of the motor driven EFW pump from a vital AC power supply."

Tests were conducted by the licensee and witnessed by a member of the NRC staff. The test described in Item 1 above was conducted four times. During the conduct of the first test to transfer to a vital AC power supply, a breakdown in communication between the two operators performing the test resulted in a skipped step in the test procedure. A second test was then successfully performed in less than five minutes. However, the NRC staff subsequently required that the licensee repeat the test a third time, using the actual procedure available in the control room instead of the test procedure. This control room procedure was reviewed and modified at our request prior to the third test which was conducted subsequent to the addition of automatic start circuitry described in Part 6. The results of this third test were incomplete due to a feature built into the new automatic start design of the motor-operated EFW pump which required an additional manual switching operation not previously included in the emergency procedure. The procedure was again revised and the fourth test conducted satisfactorily within five minutes. Subsequently, the design of the automatic start circuitry was modified so as to not require this additional manual switching operation, and the procedure was changed accordingly. Members of the NRC staff on site have verified that the control room operators are properly trained to carry out this revised procedure. The licensee has also agreed to have two operators stationed in the control room at all times until the electric driven EFW pump is permanently connected to vital power. Since the time frame of five minutes is well within the allowable delay of 20 minutes indicated by the generic B&W analyses discussed in Item (d), we conclude that the licensee has complied with the requirement for demonstrating manual startup of the motor-driven EFW pump from a vital AC power supply.

It was also ordered that;

"2. To assure that EFW be aligned in a timely manner to inject on all EFW demand events when in the surveillance test mode, procedures will be implemented and training conducted to provide an operator at the necessary valves in communication with the control room during the surveillance mode to carry out the valve alignment changes upon EFW demand events."

The ANO-1 staff has revised OP 1106.06 "Emergency Feedwater Pump Operation." Supplements I and II provide procedures for conducting the Electric and Steam Driven

Emergency Feedwater Pump surveillance test, respectively. The NRC staff has reviewed these procedures which require in part; "Operator shall remain in area for duration of test in communication with the control room to align system in the event of an EFW demand." The NRC staff has also determined that training of operators in use of this procedure has been conducted and is adequate. Subject to confirmation by a member of the NRC staff that noise levels in this area during plant operation are conducive to communications with the control room, we conclude that the licensee has complied with the order.

It was also ordered that the licensee;

"3. Write and implement procedures for the manual initiation and control of the EFW System following failure of the Integrated Control System."

The licensee has revised OP 1106.06 (Emergency Feedwater Pump Operation) and this procedure has been reviewed by the NRC staff. This procedure provides operator guidance concerning manual initiation and control of the EFW System following failure of the Integrated Control System.

The procedures were reviewed by the NRC staff to assure that feedwater from both the motor-driven pump and the steam-driven pump would be available in a timely manner. The procedures provide for verification of pump start, either automatic or manual. If offsite power is not available to the motor-driven pump, EP 1202.05 (Degraded Power) provides operator guidance to provide diesel generator power for this pump. If manual intervention to control cooldown rate is required, procedures provide for initiation and control of emergency feedwater flow through the bypass valves. These procedures would be implemented by the operator in the event of failure of the Integrated Control System. Specific procedural steps provide for:

- Startup of the electric driven EFW pump (including procedures to provide power supply from the diesel generator, if normal offsite power is not available).

- Startup of the steam driven EFW pump by opening the steam supply valves.

- Closing the ICS-controlled EFW valves (using the control room handswitch).

- Opening, and modulating as necessary, the emergency feedwater bypass valves to control EFW to the steam generator (using their control room handswitches).

- Verifying system operation by observation of EFW flow, EFW pump discharge pressure, steam generator pressure, and steam generator level.

We have reviewed these revised procedures for manual initiation and control of the EFW system and conclude that there is sufficient guidance to the operator to perform these actions to control and maintain level in the steam generators to specified values.

In addition, the NRC staff required that a test be conducted to demonstrate the capability to provide and control emergency flow to the steam generators. The licensee has committed to perform a test at low power operation (10-15%) during power ascension. The primary objective of the test will be to further verify the capability to manually control steam generator level independent of ICS. A member of the NRC staff at the ANO-1 site will witness the test and will verify acceptance prior to proceding to full power operation. Subject to the successful completion of this test, we conclude that the licensee has complied with this portion of the order.

It was also ordered that;

"4. The EFW pumps will be verified operable in accordance with the ANO-1 Technical Specifications and Surveillance Procedures."

The ANO-1 Technical Specifications provide for EFW surveillance and limiting conditions of operation. Consistent with the cover letter for this evaluation, the NRC staff will receive from the licensee within seven days revised proposed Technical Specifications with regard to design and procedural changes.

It was also ordered that the licensee;

"5. Review and revise, as necessary, the procedures and conduct training for providing alternate sources of water to the suction of the EFW pumps."

The means available to alert the operator to perform the manual transfer of EFW from the condensate storage tank (CST) to the service water system consists of an alarm in the control room which annunciates on low EFW pump suction pressure. The licensee has an additional annunciation in the control room on low level in the condensate storage tank. This new feature allows direct control room annunciation that is redundant to the existing low suction pressure switch annunciation. The NRC staff reviewed procedure OP 1106.06 "Emergency FW Pump Operation" and requested revision of the guidance to the operator for providing alternate sources of water to the suction of the EFW pumps. The revision has been made to provide additional guidance to the operator for alternate means of verifying low level in the condensate storage tank. The NRC staff at the site has verified that the control room operators are properly trained to carry out these procedures. We conclude that the licensee has complied with the requirements to review and revise procedures and has conducted operations personnel training for providing alternate sources of water.

It was also ordered that;

"6. In the event emergency feedwater is necessary and offsite power is available, an auto start signal will be provided to the motor driven emergency feedwater pump."

The licensee has installed an automatic start of the motor-operated EFW pump on loss of all RC pumps or loss of both main feedwater pumps. Relay contacts associated with existing relays within the integrated control system cabinet, additional relays and contacts, and wiring are arranged in the final actuation control circuitry for the motor-driven emergency feedwater pump such that, if offsite power is available, the motor is provided a signal to start automatically. Further, manual capability to initiate and/or override this automatic circuitry is included in the design. In addition, annunciation within the control room has been provided whenever this pump is started by the automatic circuitry.

Based on our review of this aspect of the design, we conclude that it is in accordance with the order.

It was also ordered that;

"7. Procedures will be developed and implemented and training conducted to provide guidance for timely operator verification of any automatic initiation of EFW."

The licensee has revised procedure OP 1106.06 (Emergency Feedwater Pump Operation) to provide specific operator guidance as to the methods for confirming automatic initiation of EFW. This includes:

- Verification that pump discharge pressure is greater than OTSG pressure.

- Verification of feedwater flow (on the flow indicator installed pursuant to Part 9, below).

- Observation of steam generator levels.

Emergency procedures for plant transients requiring initiation of emergency feedwater (such as loss of normal feedwater or loss of reactor coolant flow) require the operator to verify the initiation of emergency feedwater. Additionally, the operator is required to observe alternate instrumentation channels to provide further assurance. The NRC staff has confirmed that control room operators are properly trained to carry out these procedures.

It was also ordered that;

"8. Verification that Technical Specification requirements for EFW capacity are in accordance with the accident analysis will be conducted."

The licensee has stated that a minimum flow of 550 gpm is required to support the accident analyses. Low power testing will substantiate the availability of at least this flow capacity by each EFW train (see Part 3). Consistent with the cover letter to this evaluation, we will require submittal of a Technical Specification change concerning EFW capacity. This change will be a limiting condition of reactor operation in the event the minimum allowable value assumed in the accident analysis is not met, and will provide for periodic surveillance.

It was also ordered that;

"9. Modifications will be made to provide verification in the control room of EFW flow to each steam generator."

To verify that emergency feedwater is being pumped to the steam generators, the licensee is providing two orifice plates and differential pressure sensing equipment. These flow devices will be installed on each of the EFW injection flow paths downstream of the crossover line, so that flow to each steam generator will be measured. The output of the differential pressure transmitter will be displayed in the control room, indicated in gallons-per-minute.

A verification test will be performed to assure performance of this design modification. This will be performed as part of the test described in Part (3) in this report. The test procedure has been reviewed by the NRC staff and verified as acceptable.

It was also ordered that the licensee;

> "10. Provide a means of notification to the control room that the EFW system has auto started. This notification can be provided from a temporary modification or a dedicated operator."

As described in Part 7, above, the control room operator can determine the initiation of emergency feed by observation of pump discharge pressure (as compared to steam generator pressure), emergency feed flow, and steam generator level. In addition, annunciation has been provided in the control room whenever either pump is automatically started. Based on our review of this design, we conclude that it is in accordance with the order.

<u>Item b</u>

It was ordered that the licensee;

> "Develop and implement operating procedures for initiating and controlling EFW independent of Integrated Control System (ICS) control."

Several components in each EFW train are provided with an automatic initiation signal. Four components in one train are one steam-driven pump controller, one motor-operated valve located at the discharge of this pump, and two motor-operated valves associated with the steam supply for this turbine-driven pump. Two components for the other EFW train are the motor-driven pump and one motor-operated valve at the pump discharge. Although the automatic actuation signal is provided by common circuitry within the integrated control system cabinet, provisions exist to manually control these components from the control room. This manual provision provides overriding control of the automatic signal (from the Integrated Control System cabinet). We conclude that manual means exist in the design whereby the operator can initiate and control emergency feedwater following failure of the Integrated Control System automatic initiation circuitry.

We have reviewed the revised procedures for the emergency feedwater system to assure that there is sufficient guidance to the operator to actuate the system if the automatic initiation failed and to control the steam generator level to specified values. The review of the procedures focused on whether the operator was directed to observe the proper instruments and whether the operator was given specific values of parameters, such as steam generator level, to maintain by operating controls. The review also determined that the operator should confirm the validity of the instrument readings of certain key parameters such as steam generator level. The necessary modifications to the procedures to satisfy these determinations were presented to the licensee, and the NRC staff has verified that the modifications have been incorporated in the procedures. (See further discussion of these procedures and test requirements in Part 3 of Item a).

The NRC staff at the ANO-1 site walked through the emergency feedwater procedures with ANO-1 operators to evaluate whether the procedures were functionally adequate. In addition, the NRC staff audited a sample of ANO-1 operators to determine if they were familiar with the revised procedures and could implement them correctly. Based on the NRC staff audit, we conclude that the revised procedures and operator training are satisfactory.

Item c

The order requires that the licensee;

> "Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or on turbine trip."

The Arkansas Nuclear One Unit 1 original design did not have a direct reactor trip from a malfunction in the secondary system (loss of main feedwater and/or turbine trip). To obtain an earlier reactor trip (rather than delaying the trip until an operator took action or until a primary system parameter exceeded its trip setpoint), the licensee committed to install a hardwired control grade reactor trip on the loss of all main feedwater and/or on turbine trip (letter from William Cavanaugh III (AP&L) to H. Denton (NRC) dated May 11, 1979).

The purpose of this anticipatory trip is to minimize the potential for opening of the power-operated relief valve (PORV) and/or the safety valves on the pressurizer. The licensee has indicated that this new circuitry meets this objective by providing a reactor trip during the incipient stage of the related transients (turbine trip and/or loss of main feedwater).

AP&L has added control grade circuitry to ANO-1 which is designed to provide an automatic reactor trip when either the main turbine trips or both of the two main feedwater pumps trip. The main turbine trip is sensed by a normally de-energized auxiliary relay associated with the main turbine Electro-Hydraulic master trip circuitry. The power for this circuitry is provided from a Class 1E 125 volt direct current bus by way of a 125 volt distribution panel. A contact from this auxiliary relay is arranged into a 118 volt alternating current circuit containing a normally de-energized relay. This alternating current relay is physically located within the Integrated Control System cabinet and is provided power from the associated Integrated Control System power supply. A contact from this alternating current relay is arranged into a normally energized 24 volt direct current circuit containing two additional relays. This 24 volt power supply is derived within the Integrated Control System cabinet. To open each of the breakers and trip the reactor, two associated direct current relays provide four contact closures to energize two direct current shunt coils (two contact closures per shunt trip coil and one shunt trip coil for each of the two reactor trip alternating current circuit breakers). Power is provided to the shunt trip coils from Class 1E 125 volt direct current buses.

The main feedwater pump trip is sensed by two normally de-energized auxiliary relays associated with the main feedwater pumps master trip circuitry (one relay associated with each of the two main feedwater pumps). The remaining circuitry associated with this trip is identical to that described above for the turbine trip including power supplies, with the exception that two corresponding relays and contacts are provided. Also, the two associated contacts (these contacts are arranged in parallel) within the 24 volt direct current circuit are in series with the associated turbine trip contact.

Provisions have been included to automatically bypass and re-instate these additional trips at low power to allow a normal startup and shutdown. Operator verification of the bypass removal is required procedurally during power escalation. The NRC staff at the ANO-1 site audited a sample of ANO-1 operators and concluded that they were familar with the functions of these trips and associated procedural requirements.

The licensee has analyzed this additional circuitry with respect to its independence from the existing reactor trip system. The licensee has stated that the shunt coil is part of the existing AC reactor trip breaker. However, it is separate and operates independently from the 120 volt alternating current undervoltage trip coil of the associated breaker. The reactor trip safety-grade signal de-energizes the 120 volt alternating current undervoltage coil to produce a trip of the associated alternating current breaker.

Based on our review of the implementation of the trip circuitry with respect to its independence from the existing trip circuitry, we conclude that this addition will not degrade the existing reactor protection system design. The licensee has installed and completed checkout of the trip circuitry.

The licensee has committed to perform a monthly periodic test on the added circuitry to demonstrate its ability to open the AC circuit breakers (tripping the AC breakers via the shunt trip circuit). Additionally, the licensee has committed to perform a more complete test of this additional circuitry whenever the reactor is brought to a hot shutdown condition as the result of a normal outage or reactor trip (but not more frequently than once per 31 days). We conclude that there is reasonable assurance that the additional circuitry will perform its function. Accordingly, on the basis of the above, we conclude that this additional circuitry is in accordance with the requirements of item (c) of the order.

Item d

This item in the order requires the licensee to:

> "Complete analyses for potential small breaks and implement operating instructions to define operator action."

By letter from William Cavanaugh III (AP&L) to H. Denton (NRC) dated May 11, 1979, the licensee committed to providing the analyses and operating procedures of this requirement.

Babcock and Wilcox, the reactor vendor for the ANO-1 plant, submitted an analysis entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" and supplements to these analyses (References 1 through 6). The major parameters used in this generic study, with the exception of emergency feedwater flow, conservatively bound the ANO-1 plant. An additional analysis assuming a bounding value for emergency feedwater flow was subsequently submitted (Reference 6). In a letter dated May 16, and 22, 1979, AP&L has referenced these analyses as appropriate for ANO-1. The staff evaluation of the B&W generic study has been completed and the results of the evaluation will be issued as a NUREG report in June 1979.

A principal finding of our generic review is a reconfirmation that Loss-of-Coolant Accident (LOCA) analyses of breaks at the lower end of the small break spectrum (smaller than 0.04 sq. ft.) demonstrate that a combination of heat removal by the steam generators, high pressure injection system and operator action ensure adequate core cooling. The emergency feedwater system used to remove heat through the steam generators has been modified to enhance its reliability as discussed in item (a). The high pressure injection system is capable of providing emergency core cooling even at the safety valve pressure setpoint. Reactor core uncovery is not predicted for these events. The calculated peak cladding temperature was less than 800°F, well below the 10 CFR 50.46 requirement of 2200°F. The ability to remove heat via the steam generators has always been recognized to be an important consideration when analyzing very small breaks. Sensitivity analyses were performed with acceptable results assuming permanent loss of all feedwater (with operator initiation of the high pressure injection system at 20 minutes) and loss of feedwater for only the first 20 minutes of the accident. These results are appropriate for ANO-1 considering the ability to manually start the EFW pumps within 20 minutes as discussed under item (a) and (b) of this evaluation, assuming failure of automatic EFW actuation.

Another aspect of the studies was the assessment of recent design changes on the lift frequency of pressurizer safety and relief valves. The design changes included change in the setpoint of the pressurizer power-operated relief valve (PORV) from 2255 psi to 2450 psi, change in the high pressure reactor trip setpoint from 2355 psi to 2300 psi and the installation of anticipatory reactor trips on turbine trip and on loss of feedwater. In the past, during turbine trip and loss of feedwater transients the PORV was lifted. With the new design these transients do not result in lifting of this valve. However, lifting of both PORV and safety valves might occur in case of rod withdrawal and inadvertant boron dilution transients, using the normally conservative assumptions found in the Chapter 15 safety analysis. The above design changes did not effect the lift frequency of the valves for these Chapter 15 safety analyses.

Based on our review of the small break analyses presented by B&W, the staff has determined that a loss of all main feedwater with (a) an isolated PORV, but safety valves opening and closing as designed, or (b) a stuck open PORV consequentially does not result in core uncovery, provided either EFW or 2 HPI pumps are initiated within 20 minutes. Based on the acceptable consequences calculated for small break LOCAs and loss of all main feedwater events and the expected reliability of the EFW and high pressure injection systems, we conclude that the licensee has complied with the ana lysis portion of paragraph (1)(d) of the Order.

To support longer term operation of the facility, requirements will be developed for additional and more detailed analyses of loss of feedwater and other anticipated transients. More detailed analysis of small break LOCA events are also needed for this purpose. Accordingly, the licensee will be required to provide the analyses discussed in Sections 8.4.1 and 8.4.2 of the recent NRC Staff Report of the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company (NUREG 0560). Further details on these analyses and their applicability to other PWRs and BWRs will be specified by the staff in the near future. In addition, to assist the staff in developing more detailed guidance on design requirements of relief and safety valve reliability during anticipated transients, as discussed in Section 8.4.6 of the NUREG report, the licensee will be required to provide analyses of the mechanical reliability of the pressurizer relief and safety valves of the ANO-1 facility.

The B&W analyses show that some operator action, both immediate and followup, is required under certain circumstances for a small break accident. Immediate operator action is defined as those actions committed to memory by the operators which are necessary to take as soon as the problem is diagnosed. To perform followup actions, operators must consult and follow instructions in written and approved procedures. These procedures must always be readily available in the control room for the operators use. Guidelines were developed by B&W to assist the operating B&W facilities to develop emergency procedures for the small break accident.

The Operating Guidelines for Small Breaks were issued by B&W on May 5, 1979 and rev iewed by the NRC staff. Revisions recommended by the staff were incorporated in the guidelines. In response to these guidelines, the licensee made substantial revisions to EP 1202.06 (Loss of Reactor Coolant/RC Pressure), EP1202.14 (Loss of Reactor Coolant Flow-RCP Trip), EP 1202.26 (Loss of Steam Generator Feed), EP 1202.23 (Steam Generator Tube Rupture), and EP 1202.05 (Degraded Power). These emergency procedures define the required operator action in response to a spectrum of break sizes for a loss-of-coolant accident in conjunction with various equipment availability and failures.

The procedure dealing with loss-of-reactor coolant (EP 1202.06) is divided into three sections. The first deals with a rupture well in excess of the capability of the high pressure injection pumps (a large break in which the system depressurizes to the point of low pressure injection). An automatic reactor trip is assumed. The second section of this procedure assumes the small break is within the capacity of the high pressure injection system and the reactor may not automatically trip. The third section assumes reactor coolant system leakage within the capacity of a single makeup pump and no automatic reactor trip. A separate procedure (EP 1202.23) provides guidance to the operator in the event of a steam generator tube rupture. In all cases dealing with a small break, the operator actions are aimed at achieving a safe cold shutdown in accordance with the normal cooldown procedure.

As indicated above, other procedures provide guidance to the operators for dealing with small breaks in the event of a degraded condition (such as a loss-of-feedwater and/or loss of reactor coolant pumps). These procedures are EP 1202.05, EP 1202.14, and EP1202.26. If all feedwater is lost, a heat removal path is established from the high pressure injection system through the break and the pressurizer power-operated relief valve or the safety valves. Once feedwater is reestablished, the steam generators can be used as a heat sink. If the reactor coolant pumps are not available, the operator is directed to establish and verify natural circulation. Additional guidance is provided if natural circulation is not immediately achieved. If normal power to the motor-driven emergency feed pump is lost, guidance is provided to the operator to power this pump from the diesel generator.

For all cases in which high pressure injection is manually or automatically initiated, the operators are specifically instructed to maintain maximum HPI flow unless two criteria are met. These criteria are:

1. LPI has been operating for greater than 20 minutes with flow rates in excess of 2650 gpm per train, or greater than 3100 gpm with one train operating.

2. All hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If the 50 degrees subcooling cannot be maintained after HPI cutoff, the HPI shall be reactuated.

The requirement to determine and maintain 50°F subcooling has been incorporated in all other procedures in which HPI has been manually or automatically initiated. These procedures include, Steam Supply System Rupture, Steam Generator Tube Rupture, Loss of Reactor Coolant Flow and Loss of Steam Generator Feedwater. Each of these procedures, in addition to the Loss of Reactor Coolant procedure, provide additional instructions to the operators in the event of faulty or misleading indications. A subsequent action statement directs the operators to check alternate instrumentation channels to confirm the key parameter readings. The ANO-1 staff have made revisions to all of these emergency procedures to include this requirement. Also, the licensee has provided for computer readout of 16 thermocouple indications of core exit temperatures available to the operator in the control room. The licensee further committed to installation of an additional 16 thermocouples to be available before October 31, 1979. The staff has reviewed the additional information to be gained with regard to providing additional verification of reactor coolant system temperature and finds the modifications acceptable.

The Loss of Reactor Coolant procedure was reviewed by the NRC staff to determine its conformance with the B&W guidelines. Comments generated as a result of this review were incorporated in a further revision to the procedure. A member of the NRC staff walked through this emergency procedure in the ANO-1 control room. The procedure was judged to provide adequate guidance to the operators to cope with a small break loss of coolant accident. The instrumentation necessary to diagnose the break, the indications and controls required by the action statements, and the administrative controls which prevent unacceptable limits from being exceeded are readily available to the operators. We conclude that the operators should be able to use this procedure to bring the plant to a safe shutdown condition in the event of a small break accident.

An audit of nine of the 27 licensed operators and senior operators was conducted by the NRC staff to determine the operators' understanding of the small break accident, including how they are required to diagnose and respond to it. The ANO-1 staff has conducted special training sessions for the operators on the concept of and use of emergency procedure 1202.06. The operators were found to have sufficient knowledge of the small break phenomenon and the general requirements of the emergency procedure. Each licensed individual will also receive additional training on the approved procedure prior to power operation.

The audit of the operators also included questioning about the TMI-2 incident and the resulting design changes made at ANO-1. The discussions covered the initiating events of the incident, the response of the plant to the simultaneous loss of feedwater and small break LOCA (PORV stuck open), and the operational actions that were taken during the course of the incident. We found their level of understanding sufficient to be able to respond to a similar situation if it happened at ANO-1. We also concluded that they have adequate knowledge of subcooling and saturated conditions and are able to recognize each condition in the primary coolant system by various methods. The emergency feedwater system was also discussed during the audit to determine the operators' ability to assure proper starting and operation of the system during normal conditions, as well as during adverse conditions such as loss of offsite power or loss of normal feedwater. The long term operation of the system was examined to evaluate the operators' ability to use available manual controls and water supplies. The level of understanding was found to be sufficient to assure proper short and long term emergency feedwater flow to the steam generators.

The licensed operators and senior operators have received training concerning the TMI-2 accident, small break LOCA recognition, design modifications, and procedure changes. To determine the effectiveness of this training program a written exam was administered to all licensed personnel by the licensee. Individuals scoring less than 90 percent on the exam will receive additional training and will not assume licensed duties until a score of at least 90 percent is attained on an equivalent, but different exam. Arkansas Power and Light also contracted with B&W and NUS Corporation to conduct audits to determine the effectiveness of the training program. The NRC staff also conducted audits which were judged satisfactory with some deficiencies noted to the ANO-1 staff. The ANO-1 staff will use the results of these audits and any generic weaknesses discovered on the written exams in their development of future training and requalification programs. The NRC staff will review all results and records as part of the normal inspection function of the ANO-1 requalification program. We conclude that there is adequate assurance that the operators at ANO-1 have and will continue to receive a sufficient level of training concerning the TMI-2 accident.

Based on the foregoing evaluation, we conclude that the licensee has complied with the requirements of item (d) of Paragraph (1) of the order.

Item e

The order required that;

> "At least one Licensed Operator who has had TMI-2 training on the B&W simulator will be assigned to the control room (one each shift)."

The licensee has confirmed that all reactor operators and senior operators have completed the TMI-2 simulator training at B&W as required by the Order. This training consisted of a class discussion of the TMI-2 event and a demonstration of the event on the simulator as it occurred and how it should have been controlled. The class discussion was about one hour long and the remainder of the four hour session was conducted on the simulator. The TMI-2 event, including operational errors, was demonstrated to each operator. The event was again initiated and the operators were given "hands-on" experience in successfully regaining control of the plant by several methods. Other transients which resulted in depressurization and saturation conditions were presented to the operators in which they maneuvered the plant to a stable, subcooled condition.

CONCLUSION

We conclude that the actions described above fulfill the requirements of our Order of May 17, 1979 in regard to Paragraph (1) of Section IV. The licensee having met the requirements of Paragraph (1) may restart ANO-1 as provided by Paragraph 2. Paragraph 3 of Section IV of the Order remains in force until the long term modifications set forth in Section II of the Order are completed and approved by the NRC.

Dated: May 31, 1979

REFERENCES

1. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting report entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," dated May 7, 1979.

2. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting revised Appendix 1, "Natural Ciculation in B&W Operating Plants (Revision 1)," dated May 8, 1979.

3. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting additional information regarding Appendix 2, "Steam Generator Tube Thermal Stress Evaluation," to report identified in Item 2 above, dated May 10, 1979.

4. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing an analysis for "Small Break in the Pressurizer (PORV) with no Auxiliary Feedwater and Single Failure of the ECCS," identified as Supplements 1 and 2 to Section 6.0 of report in Item 2, dated May 12, 1979.

5. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing an analysis for "Small Break in the Pressurizer (PROV) with no Auxiliary Feedwater and Single Failure of the ECCS" identified as Supplements 1 and 2 to Section 6.0 of report in Item 2, dated May 12, 1979.

6. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing Supplement 3 to Section 6 of report in Item 2, dated May 24, 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

July 6, 1979

Docket No.: 50-346

Mr. Lowell E. Roe
Vice President, Facilities Development
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Roe:

By Order of May 16, 1979, the Commission confirmed your undertaking a series of actions, both immediate and long-term, to increase the capability and reliability of the Davis-Besse Nuclear Power Station, Unit No. 1 to respond to various transient events. In addition, the Order confirmed that you would maintain the plant in a shutdown condition until the following actions had been satisfactorily completed:

(a) Review all aspects of the safety grade auxiliary feedwater system to further upgrade components for added reliability and performance. Present modifications will include the addition of dynamic braking on the auxiliary feedpump turbine speed changer and provision of means for control room verification of the auxiliary feedwater flow to the steam generators. This means of verification will be provided for one steam generator prior to startup from the present maintenance outage and for the other steam generator as soon as vendor-supplied equipment is available (estimated date is June 1, 1979). In addition, the licensees will review and verify the adequacy of the auxiliary feedwater system capacity.

(b) Revise operating procedures as necessary to eliminate the option of using the Integrated Control System as a backup means for controlling auxiliary feedwater flow.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) All licensed reactor operators and senior reactor operators will have completed the Three Mile Island Unit No. 2 simulator training at B&W.

(f) Submit a reevaluation of the TECO analysis of the need for automatic or administrative control of steam generator level setpoints during auxiliary feedwater system operation, previously submitted by TECO letter of December 22, 1978, in light of the Three Mile Island Unit No. 2 incident.

(g) Submit a review of the previous TECO evaluation of the September 24, 1977 event involving equipment problems and depressurization of the primary system at Davis-Besse 1 in light of the Three Mile Island Unit No. 2 incident.

By your letters dated April 27 and May 4, 1979 and supplemented by sixteen letters dated May 11, 18, 19, 22(2), 23(2), 26(2), 29 and June 15(2), 18, 21, 23 and 25, 1979, you have documented the actions taken in response to the May 16 Order. We have reviewed this submittal, and are satisfied that, with respect to Davis-Besse, Unit 1, you have satisfactorily completed the actions prescribed in items (a) through (g) of paragraph (1) of Section IV of the Order, the specified analyses are acceptable, and the specified implementing procedures are appropriate. The bases for these conclusions are set forth in the enclosed Safety Evaluation.

Appropriate Technical Specifications for Limiting Conditions for Operation and for surveillance requirements should be developed as soon as practicable and provided to the staff within seven days with regard to the design and procedural changes which have been completed in compliance with the provisions of the May 16, 1979 Commission Order. The revised Technical Specifications should cover:

(1) Addition of flow rate indication for the auxiliary feedwater system;

(2) Addition of the anticipatory reactor trips; and

(3) Changes in set points for high pressure reactor trip and PORV actuation.

Within 30 days of receipt of this letter, you should provide us with your schedule for completion of the long-term modifications described in Section II of the May 16 Order.

My finding of satisfactory compliance with the requirements of items (a) through (g) of paragraph (1) of Section IV of the Order will permit resumption of operation in accordance with the terms of the Commission's Order; it in no way affects your duty to continue in effect all of the above provisions of the Order pending your submission and approval by the Commission of the Technical Specification changes necessary for each of the required modifications.

Sincerely,

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Safety Evaluation
2. Notice

cc w/encls:
See next page

Toledo Edison Company

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July 6, 1979

EVALUATION OF LICENSEE'S COMPLIANCE
WITH THE NRC ORDER DATED MAY 16, 1979
TOLEDO EDISON COMPANY AND
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
DAVIS-BESSE NUCLEAR POWER STATION, UNIT No. 1
DOCKET NO. 50-346

INTRODUCTION

By Order dated May 16, 1979, (the Order) the Toledo Edison Company and the Cleveland Electric Illuminating Company (TECO or the licensee) were directed by the NRC to take certain actions with respect to Davis-Besse Nuclear Power Station, Unit 1 (DB-1). Prior to this Order and as a result of a preliminary review of the Three Mile Island, Unit No. 2 (TMI-2) accident, the NRC staff initially identified several human errors that contributed significantly to the severity of the event. All holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's Office of Inspection and Enforcement (IE). Subsequently, an additional bulletin was issued by IE which instructed holders of operating licenses for Babcock & Wilcox (B&W) designed reactors to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer power-operated relief valve (PORV) setting.*

*[IE Bulletins Nos. 79-05 (April 1, 1979), 79-05A (April 5, 1979), and 79-05B (April 21, 1979) apply to all B&W facilities.]

The NRC staff identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors. Those were identified as items (a) through (e) on page 1-7 of the "Office of Nuclear Reactor Regulation Status Report to the Commission" dated April 25, 1979. After a series of discussions between the NRC staff and the licensee concerning possible design modifications and changes in operating procedures, the licensee agreed, in letters dated April 27, 1979 and May 4, 1979, to perform promptly certain actions. The Commission found that operation of the plant should not be resumed until the actions described in Items (a) through (g) of paragraph (1) of Section IV of the Order are satisfactorily completed.

Our evaluation of the licensee's compliance with items (a) through (g) of paragraph (1) of Section IV of the Order is given below. In performing this evaluation we have utilized additional information provided by the licensee in letters dated May 11, 18, 19, 22 (2), 23 (2), 26 (2), 29 and June 15 (2), 18, 21, 23 and 25, 1979 and numerous discussions with the licensee's staff. Confirmation of design and procedural changes was made by members of the NRC staff at the DB-1 site. An audit of the training and performance of the DB-1 reactor operators was also performed by the NRC staff to assure that the design and procedural changes were understood and were being correctly implemented by the operators.

EVALUATION

Item (a)

It was ordered that the licensee take the following action:

> "Review all aspects of the safety grade auxiliary feedwater system to further upgrade components for added reliability and performance. Present modifications will include the addition of dynamic braking on the auxiliary feedpump turbine speed changer and provision of means for control room verification of the auxiliary feedwater flow to the steam generators. This means of verification will be provided for one steam generator prior to startup from the present maintenance outage and for the other steam generator as soon as vendor-supplied equipment is available (estimated date is June 1, 1979). In addition, the licensees will review and verify the adequacy of the auxiliary feedwater system capacity."

The auxiliary feedwater (AFW) system at DB-1 consists of two safety-grade AFW pumps capable of being actuated and controlled by safety-grade signals that ensure the availability of feedwater to at least one steam generator, under the assumed conditions of a single failure. In addition, the capability to manually actuate and control AFW is available in the control room. The sources of water include two condensate storage tanks (CST), the service water system and the fire protection system. The CSTs provide the normal supply (non-safety-grade) and the service water system is used as a backup safety-grade supply.

A low level in either CST is alarmed to the operator and a continuous level is displayed inside the control room. Low pressure switches on the AFW pump suction provide safety-grade signals to automatically shift suction for the pump from the CSTs to the backup service water supply. Additionally, the operator could also manually transfer the AFW suction to the fire water storage tank (FWST) in the fire protection system.

Both steam-driven auxiliary feedwater pump turbines at DB-1 are provided with a governor used for variable pump speed control. The governor is equipped with a small DC motor which changes the speed setpoint on the turbine control valve, thereby controlling steam flow which regulates the turbine and pump speed. This DC motor receives "raise-and-lower" pulses from the safety-grade steam generator level control system or the manual control switches (located in the control room), which change the turbine speed as required. Pulse length is automatically increased the further steam generator level deviates from its setpoint. These changes in pump speed alter the AFW flow and thus control the water level in the steam generators.

A "dynamic brake" feature has been added, which consists of a resistor and electrical contacts in parallel with the windings of the DC motor. When the control pulse is terminated, the braking resistor is placed in parallel with the motor windings, causing rapid dissipation of the energy associated with the motor momentum (thus reducing the amount of motor coast). This, in turn, reduces the amount of pump speed overshoot, thereby allowing fewer speed changes to match the AFW flow rate to the steaming rate of the steam generators.

The licensee has also added flow rate indication for both steam generator AFW inlet lines. Each inlet line has a pipe-mounted ultrasonic flow transducer and signal conditioner. These are located in the auxiliary building and are accessible during normal plant operations. The signal conditioners provide outputs both locally and in the control room on the AFW pump section of the main control console. Each device is designed to provide flow rate indication to each steam generator from 0 to 1000 gpm. The systems are powered from 120 VAC, 60 Hz buses which are fed by redundant non-Class IE station inverters. Functional testing of the installed auxiliary feedwater flow rate indication is to be conducted in conjunction with the functional testing of the dynamic braking modification of AFW pump turbine controls. The staff concludes that the dynamic brake and AFW flow rate indication modifications are acceptable contingent upon successful testing prior to restart.

We have reviewed the piping and instrumentation diagrams and have determined that no active failure of a mechanical component, such as a pump or valve, would preclude obtaining the required AFW flow rate. The licensee has previously performed tests of the manual and automatic level control system. The test results showed that the control system functioned as designed to control steam generator level. Verification of acceptable flow capacity for each of the two AFW pumps was based upon recorded steam generator level changes following a previous reactor trip. These data showed that each pump exceeded the design flow rate of 800 gpm at a steam generator pressure of 1050 psig. (The 800 gpm is the flow rate delivered to the steam generators and does not include the approximately 250 gpm recirculation flow rate.)

Additional information submitted by the licensee (letter from Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated May 23, 1979) shows that a total minimum flow, to one or both steam generators, of 550 gpm is required to support the accident analyses. Based on these data and analyses, and the agreement by the licensee to perform checkout testing of the dynamic braking and flow rate indication modifications prior to restart, we conclude that adequate assurance exists that the AFW system will deliver the required flow rate upon demand.

By letter (Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated May 23, 1979), the licensee provided results of a review of the operating history of the AFW system at DB-1. The largest number of failures* occurred during the initial operating and debugging phase of the facility. Fourteen (14) of the seventeen (17) reported failures occurred prior to January, 1978. Subsequent to implementing system design changes as a result of several of these failures, the systems failure rate has been reduced and its reliability enhanced. There were 3 failures of AFW system components from January 1978 to May 1979. (There were a total of 65 actuations of the AFW system in this time period.) Three different components in the AFW system were involved in these three failures: (1) the speed control circuit for #1 AFW pump turbine, (2) a faulty limit switch on an AFW discharge valve, and (3) two sticky AFW pump turbine steam supply valves. In each case, the licensee performed corrective actions.

*[For the purpose of demonstrating improvement in the performance of the AFW system, the licensee has defined a failure of the AFW system to be any event for which at least one train of the AFW system is not delivering design flow to a steam generator.]

A later letter (Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated June 29, 1979) addressed a series of pressure switch failures which were discovered on May 21, 1979, and which affected both AFW trains. An evaluation of these failures by the licensee concluded that both trains would have automatically actuated if required, but that one train would not have shifted automatically to the service water supply. The NRC staff has discussed these failures with TECO and has requested that an improved surveillance program for these pressure switches be initiated to determine the cause of the failures and the optimum calibration interval. The licensee has agreed to an increased frequency of switch calibration. In addition, the licensee has made procedural changes, requested by the staff, to instruct the operator to manually shift to the alternate supply of water for the AFW pumps, when the CST level drops to three feet (if automatic switchover has not occurred). This procedure provides greater assurance that, even with failures of this nature, the AFW system is available during the longer term. More recently (July 5, 1979), the NRC staff was verbally informed by TECO (Mr. G. Novak) of a valve malfunction which took place in an AFW system pump discharge line on July 4, 1979. The cause of the valve failure (failed closed) was apparently due to an electrical malfunction. TECO stated that they would request the motor vendor to examine the failed motor to determine the cause of the malfunction. The IE site inspector has been requested to follow this evaluation and to determine the need for further study and corrective action if necessary. The licensee has noted that manual capability (local handwheel) to open the valve existed at the time of the failure and that the redundant AFW train was available.

With regard to the operating history of the AFW system, the staff concludes that the licensee has increased the reliability of the AFW system by implementing appropriate corrective actions and design modifications. With regard to the more recent pressure switch and valve failures, the staff concludes that adequate assurance exists that the causes of the failures are being pursued by the licensee in a timely manner, and that the IE site inspector will follow the need for further corrective action.

In addition, the licensee has revised the administrative procedure pertaining to valve alignment and control. These revisions to AD 1839.02 ("Operation and Control of Locked Valves") provide further assurance that mispositioning of AFW system valves would be detected.

Based on the above evaluation, the NRC staff concludes that the licensee has complied with the requirement of Item (a) of the Order.

Item (b)

It was also ordered that the licensee:

> "Revise operating procedures as necessary to eliminate the option of using the Integrated Control System as a backup means for controlling auxiliary feedwater flow."

As indicated in Item (a), the DB-1 AFW system has been designed as a safety grade system and, as such, is separate from the integrated control system (ICS); however, the licensee has indicated that the AFW system is capable of being switched to the ICS mode for a backup means of control. As currently designed, the AFW system has three operational modes of controlling flow: "ICS control", "auto-essential" and "manual." We requested that the licensee consider a more positive means to assure the continued separability of the ICS control position of the mode selector switches. The licensee agreed (letter from Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated June 15, 1979) to install a mechanical stop on these switches to further deter use of the ICS control position. The IE site inspector has verified the installation of this mechanical stop.

The licensee has revised SP 1106.06 ("Auxiliary Feedwater System"), which describes procedures for AFW system operation. This procedure specifically prohibits the use of the ICS control position on the mode selector switches. Procedural steps for placing the AFW system in service for plant startup require the operator to place the AFW mode selector switches in the auto-essential position. We have reviewed the revised procedure for AFW switch operation and conclude there is sufficient guidance to prevent use of the AFW system in the ICS mode of control.

Other plant procedures that made reference to the ICS control mode of AFW have been revised by the licensee to no longer authorize that mode of control. The staff has reviewed those procedures and concludes that those revisions are adequate. In addition, the NRC staff audit confirmed that the control room operators are aware that ICS control of AFW is prohibited.

Based on the above evaluation, we conclude that the licensee has complied with the requirements of Item (b) of the Order.

Item(c)

The Order requires that the licensee:

> "Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip."

The DB-1 original design did not have a direct reactor trip from a malfunction in the secondary system (loss of main feedwater and/or turbine trip). To obtain an earlier reactor trip (rather than delaying the trip until an operator took action or until a primary system parameter exceeded its trip setpoint), the licensee committed to install a hard-wired, control-grade reactor trip on the loss of all main feedwater and/or on turbine trip (letter from Lowell E. Roe (TECO) to H. Denton (NRC) dated April 27, 1979). The purpose of this anticipatory trip is to minimize the potential for opening of the power-operated relief valve (PORV) and/or the safety valves on the pressurizer. This new circuitry meets this objective by providing a reactor trip during the incipient stage of the related transients (turbine trip and/or loss of main feedwater).

TECO has added control-grade circuitry to DB-1 which is designed to provide an automatic reactor trip when either the main turbine trips or there is a reverse differential pressure of 177 psid across both of the two main feedwater check valves (one check valve is located in the main feedwater discharge piping associated with each steam generator). The main turbine trip is sensed by a normally deenergized auxiliary relay associated with the main turbine generator master trip bus. The power for this bus is provided from a 24 volt DC source, which in turn is provided power (through rectifier circuitry) from a non-Class 1E inverter supplied 120 volt AC distribution panel. A contact from the above auxiliary relay is arranged into a 120 volt AC circuit containing four normally deenergized relays. Power for this 120 volt circuit is provided from a Class 1E inverter supplied distribution panel. The design for these four relays and appropriate associated circuitry conform to Class 1E requirements, including physical independence and provisions for testing. Each of these four relays provide one contact which is arranged in series with one of the four Class 1E undervoltage coils associated with one of the four AC reactor trip circuit breakers (one undervoltage coil associated with each AC reactor trip circuit breaker). When these relays are energized, power to the associated Class 1E undervoltage coils is interrupted so as to produce the desired reactor trip.

As indicated above, differential pressure switches across check valves, located in the main feedwater pump discharge piping, actuate upon sensing a reverse differential pressure across these check valves. Two contacts from these differential pressure switches are arranged into a 125 volt DC circuit, which is provided power from a Class 1E 125 volt distribution panel. This circuit contains two associated DC relays. Two contacts (one contact per relay) associated with these relays are arranged in series. This series contact arrangement is provided in parallel with the contact associated with the main turbine generator master trip bus. The remaining circuitry associated with this trip is identical and common (shared) to that described above for the turbine trip (including power supply identification).

Provisions have been included in the design to manually bypass and to reinstate the reactor trip feature associated with the main turbine generator trip. To supplement this feature, the design includes an annunciator which actuates whenever this reactor trip is bypassed and the reactor power level is above 15 percent. Access to this bypass switch will require a key which is under suitable administrative control. Operator verification of the bypass removal is required by procedure during power escalation. The NRC staff has reviewed these procedures and concludes that sufficient administrative control exists. No bypass features are included in the design for the reactor trip feature associated with the loss of main feedwater circuitry. During normal startup or shutdown, an electric auxiliary pump is used when the steam driven main feedwater pumps are not available.

The licensee has analyzed this additional circuitry with respect to its independence from the existing reactor trip system and to assure that the design and operation of this additional circuitry will neither degrade the reliability of the existing reactor protection system nor create any new adverse safety system interactions. Based on our review of the implementation of the added trip circuitry, with respect to its independence from the existing trip circuitry, we conclude that this addition will not degrade the existing reactor protection system design. In addition, the licensee has satisfactorily completed testing of this trip circuitry.

The licensee has committed to perform a monthly periodic test of the added circuitry to demonstrate its ability to open the AC reactor trip circuit breakers (tripping of the AC reactor trip circuit breakers via the under-voltage trip circuit). We conclude that there is reasonable assurance that the additional circuitry will perform its intended function.

Based on the above evaluation, we conclude that the licensee has complied with the requirements of Item (c) of the Order.

Item(d)

This Item in the Order requires the licensee to:

"Complete analyses for potential small breaks and develop and implement operating instructions to define operator action."

By letter, (Lowell E. Roe (TECO) to H. Denton (NRC) dated April 27, 1979), the licensee agreed to provide the analyses and operating procedures of this requirement.

B&W, the reactor vendor for the DB-1 plant, submitted generic analyses for B&W plants entitled, "Evaluation of Transient Behavior and Small Reactor Coolant Systems Breaks in the 177 Fuel Assembly Plant," and supplements to these analyses (References 1 through 5). Additional information specific to DB-1 was transmitted in References 6 to 8. The transmittal under Reference 6 contains Volume III for the B&W generic study covering raised-loop plants. Reference 7 provides additional analytical results specific to DB-1 with appropriate auxiliary feedwater flow rates. Reference 8 provides additional analytical results for the loss of all main feedwater flow accident with loss of all AFW. This latter analysis demonstrates that capability exists at DB-1 which the operator could use in the unlikely event of a loss of main feedwater and a loss of both safety grade AFW trains. This capability consists of using the combined functions the makeup pumps,* the electric startup auxiliary feedwater pump and the PORV to achieve depressurization (only if necessary). We requested that the availability of this option be incorporated in procedures at DB-1. The NRC staff will review these procedural changes prior to startup.

*At DB-1, the makeup pumps are separate from the HPI pumps.

By letter, (Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated May 22, 1979), TECO referenced the analyses as appropriate for DB-1. The staff evaluation of the B&W generic study has been completed and the results of the evaluation will be issued as a NUREG report in July 1979. A principal finding of our review of the DB-1 submittals and the generic study is a reconfirmation that loss-of-coolant accident (LOCA) analyses of breaks at the lower end of the small breaks spectrum (smaller than 0.04 $ft.^2$) demonstrate that a combination of heat removal by the steam generators, high pressure injection (HPI) system and through the break ensure adequate core cooling. The AFW system used to remove heat through the steam generators has been modified to enhance its reliability as discussed in Item (a).

Uncovering of the reactor core is not predicted for breaks at this end of the small break spectrum with these features available, therefore, cladding temperatures do not rise significantly above pre-reactor trip temperatures (less than 800°F), and remain well within the 10 CFR 50.46 limit of 2200°F. The ability to remove heat via the steam generators has always been recognized to be an important consideration when analyzing very small breaks. The licensee demonstrated that permanent loss of main feedwater and loss of AFW for the first 20 minutes of a small LOCA will not result in uncovering the reactor core. However, when AFW is delayed beyond this time, a positive reliance on AFW actuation exists as a result of the relatively low (1600 psig) HPI system shutoff head for DB-1. Thus permanent loss of both main and auxiliary feedwater could result in uncovering the core and fuel damage for the facility because of the unavailability of the high pressure injection pumps. Makeup pump and startup feedwater pump actuation, as discussed in the analysis of Reference 8 for the loss of feedwater accident with permanent loss of AFW, are considered potentially capable of maintaining the vessel mixture above the core for a small break, but this scenario was not confirmed in the small break analyses. The licensee's position is that such analyses are unwarranted in light of the safety-grade design of the AFW system. Since the additional heat removal and coolant makeup capability does exist at DB-1, we requested that the procedures identify the availability of this option. Implementation of this procedural change will be verified by the staff prior to restart. While the staff recognizes that the AFW system is safety-grade, we also note that the licensee has agreed to continue to review performance of the AFW system for assurance of reliability and performance. Consistent with this long-term agreement, we will require that the licensee modify the plant to provide the greater degree of diversity offered by a 100% capacity motor-operated AFW pump, or an alternative acceptable to the staff.

Another aspect of the analytical studies conducted was an assessment of the effect of recent design changes on the lift frequency of pressurizer safety and relief valves. The design changes included: (1) a change in the setpoint of the PORV from 2255 psig to 2400 psig, (2) a change in the high pressure reactor trip setpoint from 2355 psig to 2300 psig, and (3) the installation of anticipatory reactor trips on turbine trip and/or loss of main feedwater. In the past, during turbine trip and loss of feedwater transients, the PORV was lifted. With the new design, these transients do not result in lifting of this valve. However, lifting of both PORV and safety valves might occur in the cases of rod withdrawal or inadvertant boron dilution transients, using the normally conservative assumptions presented in Chapter 15 of the Final Safety Analysis Report (FSAR). The above design changes did not affect the lift frequency of the valves for these Chapter 15 safety analyses.

Based on our review of the analyses presented by B&W, the staff has determined that a loss of all main feedwater with (1) an isolated PORV (closed block valve), but safety valves opening and closing as designed, or (2) a stuck open PORV consequentially does not result in uncovering the reactor core, provided AFW pumps are initiated within 20 minutes. It is also concluded, that in the event of a loss of all AFW for either case, covering of the core would be sustained to long-term cooling by operator actions described in the analysis of Reference 8. These actions consist of starting at least one of the two makeup pumps, starting the startup feedwater pump, and opening the PORV (only if needed).

Based on the consequences calculated for small break LOCAs and loss of all main feedwater events, and taking into account the expected reliability of the AFW and HPI systems for DB-1, we conclude that the licensee has complied with the analyses portion of Item (d) of the Order.

To support long-term operation of the facility, requirements will be developed for additional and more detailed analyses of loss of feedwater and other anticipated transients. More detailed analyses of small break LOCA events are also needed for this purpose. Accordingly, the licensee will be required to provide the analyses discussed in Sections 8.4.1 and 8.4.2 of the recent NRC "Staff Report of the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company" (NUREG 0560). Further details on these analyses and their applicability to other PWRs and BWRs will be specified by the staff in the near future. In addition, to assist the staff in developing more detailed guidance on design requirements of relief and safety valve reliability during anticipated transients, as discussed in Section 8.4.6 of NUREG 0560, the licensee will be required to provide analyses of the lift frequency and the mechanical reliability of the pressurizer relief and safety valves of the DB-1 facility.

The B&W analyses show that some operator actions, both immediate and followup, are required under certain circumstances for a small break accident. Immediate operator actions are defined as those actions, committed to memory by the operators, which must be carried out as soon as the problem is diagnosed. Followup actions require operators to consult and follow steps in written and approved procedures. These procedures must always be readily available in the control room for the operators' use. Guidelines were developed by B&W to assist the operating B&W facilities to develop emergency procedures for the small break accident.

The "Operating Guidelines for Small Breaks" were issued by B&W on May 5, 1979 and reviewed by the NRC staff. Revisions recommended by the staff were incorporated in the guidelines.* In addition, by letter, the licensee submitted supplemental guidelines (Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated May 22, 1979). In response to these guidelines, the licensee made substantial revisions to EP 1202.06 ("Loss of Reactor Coolant and Reactor Coolant Pressure"), EP 1202.14 ("Loss of Reactor Coolant Flow/RCP Trip"), and EP 1202.26 ("Loss of Steam Generator Feed"). These emergency procedures define the required operator action in response to a spectrum of accidents including a LOCA in conjunction with various equipment availability and failures.

The procedure dealing with loss of reactor coolant (EP 1202.06) is divided into three sections. The first section deals with small reactor coolant system leaks within the capacity of the makeup pumps and assumes the reactor does not automatically trip. The second section assumes a small break within the capacity of the HPI system and a situation where the SFAS** and reactor trips may or may not automatically occur. This section incorporates the B&W small break guidance and provides for operator actions in the event other

*[Letter from J. Taylor (B&W) to Z. Rosztoczy (NRC) dated May 16, 1979]

**[The safety features actuation system (SFAS) monitors variables to detect loss of reactor coolant system boundary integrity. Upon detection of "out-of-limit" conditions of these variables, the system initiates various actions, depending upon the location and severity of the "out-of-limit" conditions measured. These actions can include: initiation of emergency core cooling (ECC), which consists of high pressure injection (HPI) and low pressure injection (LPI); containment vessel cooling and isolation; containment vessel spray systems; and starting of the emergency diesel generators.]

systems (such as reactor coolant pumps) do not operate as expected. The third section of this procedure deals with a pipe rupture well in excess of the capability of the makeup and/or HPI pumps (a large break in which the system depressurizes to the point of low pressure injection). Automatic reactor trip and SFAS actuation are assumed. In all cases dealing with a small break, the operator actions are aimed at achieving a safe cold shutdown in accordance with the normal cooldown procedure.

As indicated above, procedures provide guidance to the operators for dealing with small breaks in the event of a degraded condition (such as loss of reactor coolant pumps). If the reactor coolant pumps are inoperable, the operator is directed to establish and verify natural circulation. Procedural steps to restore reactor coolant pump operation, once a pump becomes available, are provided. In the event natural circulation cannot be established and a reactor coolant pump cannot be restarted and plant pressure reaches 2300 psig, the operator is provided procedural steps to relieve the heat energy via the PORV. (Additional relief capacity is provided via the code safety valves if the PORV is inoperable).

In the event that normal feedwater is lost to the steam generators, auxiliary feedwater is automatically initiated via the safety-grade AFW system. EP 1202.26 provides operator guidance in this event. With SFAS actuation, steam generator level is automatically maintained at 96 inches on the startup range to assure adequate heat removal during the small break event.

For all cases in which HPI is manually or automatically initiated, the operators are specifically instructed to maintain maximum HPI flow unless one of the two following criteria is met:

(1) Low pressure injection has been operating for greater than 20 minutes with flow rates in excess of 1000 gallons per minute per train, or

(2) All hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing reactor coolant system pressure. If the 50 degrees subcooling cannot be maintained after high pressure injection cutoff, the high pressure injection shall be reactuated.

This requirement to determine and maintain 50°F subcooling has been incorporated into EP 1202.06 ("Loss of Reactor Coolant and Reactor Coolant Pressure") and EP 1202.24 ("Steam Supply System Rupture"). The procedures also provide instructions to the operators to check alternate instrumentation channels to confirm key parameter readings, such as the degree of subcooling. Accordingly, the use of core exit thermocouples as alternate temperature indicators is addressed in the procedures. Under degraded cooling conditions (such as a LOCA), the pressure-temperature limits considered in the Technical Specifications are not applicable to the ensuing depressurization and cooldown because these limits were developed for normal and upset operating conditions only. Density differences between the downcomer and reactor core will cause recirculation flow between the core exit and downcomer via the vent valves.

Mixing of the hot core exit water with the cold HPI water (or makeup water) will provide sufficiently warm vessel temperatures to preclude any significant thermal shock effects to the vessel. Subsequent restoration of AFW would depressurize the reactor coolant system to below 600 psig where pressure vessel integrity is assured for any reasonable thermal transients that might subsequently occur. B&W has agreed to provide a detailed thermal-mechanical generic report on the behavior of vessel materials for those extreme conditions.

The "Loss of Reactor Coolant and Reactor Coolant Pressure" procedure was reviewed by the NRC staff to determine its conformance with the B&W guidelines. Comments generated as a result of this review were incorporated in a further revision to the procedure. A member of the NRC staff walked through this emergency procedure in the Davis-Besse control room. The procedure was judged to provide adequate guidance to the operators to cope with a small break LOCA. The instrumentation necessary to diagnose the break, the indications and controls required by the action statements, and the administrative controls which prevent unacceptable limits from being exceeded are readily available to the operators. We conclude that the operators should be able to use this procedure to bring the plant to a safe shutdown condition in the event of a small break accident.

An audit of 9 of the 25 licensed reactor operators and senior reactor operators was conducted by the NRC staff to determine the operators' understanding of the small break accident, including how they are required to diagnose and respond to it. The DB-1 staff has conducted special training sessions for the operators on the concept of and use of Emergency Procedure 1202.06. The operators were found to have sufficient knowledge of the small break phenomenon and the general requirements of the emergency procedure, although some deficiences were identified which were primarily due to the operators' lack of familiarity with the recently revised procedure. All operators will receive additional training on EP 1202.06 and a facility administered audit prior to assuming licensed duties during power operation.

The audit of the operators also included questioning about the TMI-2 accident and the resulting design changes made at DB-1. The discussions covered the initiating events of the incident, the response of the plant to the simultaneous loss of feedwater and small break LOCA (PORV stuck open), and operator actions that were taken during the course of the incident. In addition, similarities and differences between the TMI-2 accident and the DB-1 incident of September 24, 1977 were discussed. We found their level of understanding sufficient to be able to respond to a similar situation if it happened at DB-1. We also conclude that they have adequate knowledge of subcooling and saturated conditions and are able to recognize each condition in the primary coolant system by several methods. The AFW system was also discussed during the audit to determine the operators' ability to assure proper starting and operation of the system during normal conditions, as well as during adverse conditions such as loss of offsite power or loss of main feedwater. The long-term operation of the system was examined to evaluate the operators' ability to use available manual controls and water supplies. The level of understanding was found to be sufficient to assure proper short- and long-term AFW flow to the steam generators.

The licensed reactor operators and senior reactor operators have received training concerning the TMI-2 accident, small break LOCA recognition, design modifications, and procedure changes. The training included formalized classroom sessions and on-shift review of training material and emergency procedure changes. To determine the effectiveness of this training program, a written exam was administrated to all licensed personnel by the licensee. The exam was reviewed and found acceptable by a member of the NRC staff. Individuals scoring less than 90 percent on the exam will receive additional training and will not assume licensed duties until a score of at least 90 percent is attained on an equivalent, but different exam. The NRC staff conducted audits to evaluate the effectiveness of the training program. The results were judged satisfactory with some deficiencies noted to the DB-1 staff. The DB-1 staff will use the results of these audits as well as any generic weaknesses discovered on the written exams in their development of future training and requalification programs. The NRC staff will review all results and records as part of the normal inspection function of the DB-1 requalification program. We conclude that there is adequate assurance that the operators at DB-1 have, and will continue to receive, a sufficient level of training concerning the TMI-2 accident.

Based on the above evaluation, we conclude that the licensee has complied with the requirements of Item (d) of the Order.

Item (e)

The Order requires that:

> "All licensed reactor operators and senior reactor operators will have completed the Three Mile Island Unit No. 2 simulator training at B&W."

The licensee has confirmed that all reactor operators and senior reactor operators have completed the TMI-2 simulator training at B&W as required by the Order. This training consisted of a class discussion of the TMI-2 event and a demonstration of the event on the simulator and how it should have been controlled. The class discussion was about one hour long and the remainder of the four hour session was conducted on the simulator. The TMI-2 event, including operational errors, was demonstrated to each operator. The event was again initiated and the operators were given "hands-on" experience in successfully regaining control of the plant by several methods. Other transients, which resulted in depressurization and saturation conditions, were presented to the operators, in which they maneuvered the plant to a stable, subcooled condition.

The licensee has submitted copies of procedures that were revised as a result of this Order and actions the licensee has taken to preclude the occurrence of an incident similar to that which occurred at TMI-2.* The procedures reviewed by the staff include:

*[As noted on page 16 of this Safety Evaluation, additional and more detailed analyses of loss-of-feedwater transients and other anticipated transients will be done, which could affect these procedures in the long-term.]

EP 1202.01	Load Rejection
EP 1202.02	Station Blackout
EP 1202.03	RCS Overpressure Anticipatory Manual Trip
EP 1202.04	Reactor-Turbine Trip
EP 1202.06	Loss of Reactor Coolant and Reactor Coolant Pressure
EP 1202.14	Loss of RC Flow/RCP Trip
EP 1202.22	High Condenser Pressure
EP 1202.24	Steam Supply System Rupture
EP 1202.26	Loss of Steam Generator Feed
AB 1203.04	Depressurization of the RCS with Safety Grade Equipment
AB 1203.02	Loss of All AC Power
AP 3003.41-.44	High Pressure Injection High Flow Alarm
AP 3003.49-.50	Low Pressure Injection High Flow Alarm
AP 3003.51-.54	High Pressure Injection Low Flow Alarm
AP 3003.59-.60	Low Pressure Injection Low Flow Alarm
SP 1105.16	Steam and Feedwater Rupture Control System Operating Procedure
SP 1106.06	Auxiliary Feedwater System
ST 5071.01	Auxiliary Feedwater System Monthly Test
Special Order No. 20	Additional Guidance for Checking Critical Parameters for Emergency Procedures

The licensee's revised procedures provide additional guidance for the operators when coping with emergency plant conditions. Where appropriate, operators are directed to recheck certain critical plant parameters. Operators are also directed to check alternate instrument channels to confirm readings and reduce the possibility of reliance on faulty or misleading indications.

NRC staff comments on the licensee's procedures have been incorporated into the revised documents. These revisions have been reviewed by the staff and determined to be acceptable. The staff walked through the following procedures with the control room operators: EP 1202.06 ("Loss of Reactor Coolant and Reactor Coolant Pressure"), EP 1202.14 ("Loss of RC Flow/ RCP Trip"), EP 1202.26 ("Loss of Steam Generator Feed"), and SP 1106.06 ("Auxiliary Feedwater System"). Based on this walk through and interviews with the operators, (see the discussion of the NRC staff audit of operators under Item (d)), we conclude that the procedures are functionally adequate and the operator training on their use is satisfactory.

Based on the above evaluation, we conclude that the licensee is in compliance with Item (e) of the Order.

Item (f)

The Order requires that the licensee:

> "Submit a reevaluation of the TECO analysis of the need for automatic or administrative control of steam generator level setpoints during auxiliary feedwater system operation, previously submitted by TECO letter of December 22, 1978, in light of the Three Mile Island No. 2 incident."

By letter, (Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated May 19, 1979), the licensee provided additional discussion of the steam generator dual level setpoint. The need for this feature is to reduce the potential for loss of pressurizer level indication as a result of overcooling of the primary system for non-LOCA events. The results of a natural circulation test conducted at DB-1 and B&W analyses demonstrate that DB-1 can be operated at a low steam generator level (35 inches on the startup range instrumentation). The high level setpoint (96 inches indicated on the startup range instrumentation) is required since previous small break analyses assumed that auxiliary feedwater was controlled to a steam generator level of 96 inches. Pending incorporation of permanent design modifications to provide the automatic dual setpoint steam generator level control, emergency procedures instruct the operator to manually control the steam generator level at 35 inches for all events requiring AFW unless an SFAS level 2* signal occurs. When the SFAS level 2 signal occurs, the operator is instructed to control the steam generator level at 96 inches by placing the AFW mode selector switch in the auto essential position. This manual provision required no previous change to the design of the AFW control system. The future circuitry modification, to automatically control to 35 inches, will be reviewed by the staff during the long term. TECO has cited Reference 9 to demonstrate that no unreviewed safety issues or detrimental accident consequences would result if the operator failed to manually control the steam generator level at 35 inches. The staff reviewed the information contained in this reference and concluded that additional information was required to verify that the effects of manually controlling the steam generator level at 35 inches is adequate for the DB-1 FSAR Chapter 15 transient and

*[SFAS level 2 - An SFAS level 2 signal is developed when reactor coolant system pressure drops to 1600 psig or containment vessel pressure increases to 4 psig.]

accident analyses, and the more recent B&W small break analyses (Reference 1). By letter, (Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated June 15, 1979), the licensee stated that the control of the steam generator level at 35 inches has no adverse effect on the DB-1 FSAR analyses, since the peak reactor temperature and pressure following the most severe transients (loss of feedwater, feedwater line breaks, loss of offsite power) occur prior to initiation of the AFW. The results of natural circulation testing conducted at DB-1 support the effectiveness of the 35 inch steam generator control level to maintain natural circulation and remove decay heat for: (1) transients that result in loss of forced circulation (loss of offsite power) and (2) for small breaks (less than 0.01 ft.^{2}) that depressurize slow enough that it is possible to manually control the steam generator level prior to actuation of the SFAS level 2 signal. For small breaks larger than 0.01 ft.^{2}, reduction of the reactor coolant system pressure to SFAS level 2 occurs prior to the steam generator level decreasing to 96 inches. With the steam generator level controlled at 35 inches, the effectiveness of natural circulation is such that there is no small break size that will result in repressurization of the primary system without an SFAS level 2 actuation. The staff has reviewed the information provided by TECO in the referenced documents and concludes that dual level setpoints, with manual control of the steam generator level at 35 inches, are acceptable. Also, the NRC staff has verified that this manual control capability has been previously demonstrated.

The licensee has submitted revised procedures, which the staff has reviewed, that provide requirements for steam generator level control. These procedures include: EP 1202.06 ("Loss of Reactor Coolant and Reactor Coolant Pressure"), EP 1202.14 ("Loss of RC Flow/RCP Trip") and EP 1202.26 ("Loss of Steam Generator Feed"). The NRC staff has verified that these procedures instruct the operator to confirm that the AFW mode selector switches are in the auto-essential position and maintaining steam generator level at 96 inches on the startup range indication in the event SFAS level 2 condition is present.

If a SFAS level 2 condition is not present and an AFW system demand event occurs, steam generator levels will automatically control at 96 inches (since the AFW mode selector switches are in the auto-essential position). The operator is directed to take manual control of steam generator level and maintain level at 35 inches on the startup range indication. If an SFAS Level 2 condition subsequently develops, the operator must return the AFW mode selector switches to the auto-essential position to allow automatic level control at 96 inches. Therefore, the emergency procedures are written to permit manual control of steam generator level after an automatic initiation of AFW only if an SFAS level 2 condition is <u>not</u> present.

If a SFAS level 2 condition is present (or develops), the operator is directed to leave (or return) the AFW mode selector switches in the auto-essential position. In addition, a warning plate has been installed adjacent to the mode selector switch for each AFW train, reminding the operator of the requirement to maintain the switch in the auto-essential position mode if an SFAS level 2 condition is present. The NRC staff has verified the installation of this warning plate. Also, during the audit the NRC staff confirmed that the control room operators are aware of the requirements outlined in the revised procedures and understand the purpose of the warning plate.

Based on the above evaluation, we conclude that the licensee has complied with the requirements of Item (f) of the Order.

Item (g)

The Order requires that the licensee:

> "Submit a review of the previous TECO evaluation of the September 24, 1977 event involving equipment problems and depressurization of the primary system at Davis-Besse 1 in light of the Three Mile Island Unit No. 2 incident."

By letter (Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated May 18, 1979), the licensee submitted additional discussion of the September 24, 1977 event.

This event was similar in several important areas to the TMI-2 accident. The initiating malfunction was a loss of main feedwater (the same as TMI-2); however, the ensuing trainsient was much less severe than TMI-2 for several significant reasons. The following discussion compares The DB-1 event to the accident at TMI-2. The bases for this comparison are the six human, design and mechanical failures described in IE Bulletin 79-05A (April 5, 1979) which resulted in core damage and radiation releases at the TMI-2 nuclear plant.

1. <u>At the time of the initiating event, loss of feedwater, (at TMI-2) both of the auxiliary feedwater trains were valved out of service</u>.

   The DB-1 loss of feedwater (LOFW) event initiated both trains of AFW. However, only one train fed its associated steam generator (SG) due to a malfunction of a turbine governor which kept one of the two AFW pump turbines at a speed insufficient to pump water into its associated SG.

   As a result of the DB-1 event, the modifications that have been made include: (1) the AFW pump turbine governors were modified to prevent binding malfunctions; (2) springs were installed in the AFW governor to prevent closure of the governor valve due to vibration; (3) the AFW governor control circuitry relays were replaced (see additional AFW discussions in Item (a)).

2. <u>The pressurizer power-operated relief valve (PORV), which opened during the initial pressure surge (at TMI-2), failed to close when pressure decreased below the actuation level</u>.

   During the DB-1 LOFW, the PORV also failed to close, causing loss of coolant and some voiding in the reactor coolant system (RCS). However, the operators recognized the open PORV about 20 minutes into the event (compared with 2 1/2 hours at TMI-2) and responded by closing the PORV block valve and reinitiating high pressure injection (HPI) flow.

The DB-1 unit has been modified to provide the operator with a better status of the position of the PORV. The emergency procedures were also revised and now require the operator to verify that no leak exists at the top of the pressurizer by monitoring the saturation curve and quench tank pressure and level.

3. Following rapid depressurization of the pressurizer (at TMI-2), the pressurizer level indication may have led to erroneous inferences of high level in the RCS. This erroneous high level indication apparently led the operators to prematurely terminate HPI, even through voids existed in the RCS.

For the DB-1 LOFW event, the operator also initially terminated HPI due to a high pressurizer level indication; however, the operator recognized the open PORV at 20 minutes and reinitiated HPI at 49 minutes (after failing to control pressurizer level with a second makeup pump).

DB-1 procedures have been revised and now require that for all cases in which HPI is initiated, maximum HPI flow is to be maintained unless one of two criteria is met. These criteria are addressed in Item (d).

4. Because the containment does not isolate on HPI initiation (at TMI-2), the highly radioactive water from the relief valve discharge was pumped out of containment by the automatic initiation of a transfer pump. This water entered the radioactive waste treatment system in the auxiliary building

<u>where some of it overflowed to the floor. Outgassing from this water and discharge through the auxiliary building ventilation system and filters was the principal source of the offsite release of radioactive noble gases</u>.

Containment isolation at DB-1 occurs at either 1600 psig RCS pressure (HPI initiation) or 4 psig containment vessel pressure. During the DB-1 event, containment isolation signals occurred and the sump was not pumped outside containment as at TMI-2.

5. <u>Subsequently, the HPI system was intermittently operated (at TMI-2) attempting to control RCS inventory losses through the PORV, apparently based on pressurizer level indication. Due to the presence of steam and/or noncondensible voids elsewhere in the RCS, this led to a further reduction in primary coolant inventory</u>.

During the DB-1 event, the operator initially tried to control the pressurizer level decrease with a second make-up pump after closing the PORV block valve. However, after the pressurizer level decreased further he restarted a HPI pump. When the pressurizer level was recovered, he terminated the HPI flow. At this time plant parameters were under control and the plant was brought to a stabilized condition.

As indicated in Part 3 above, DB-1 procedures have been revised to require that for all cases in which HPI is initiated, maximum HPI flow is to be maintained unless one of two criteria is met. These criteria are addressed in Item (d).

6. <u>Tripping of reactor coolant pumps during the course of the transient (at TMI-2), to protect against pump damage due to pump vibration, led to fuel damage since voids in the RCS prevented natural circulation</u>.

   During the DB-1 incident, two RCP's were tripped to reduce system heat input into the RCS. One RCP per loop was maintained in operation throughout the incident.

   The DB-1 emergency operating procedures now require keeping at least one RCP per loop running in the event of a small LOCA.

To summarize Item (g) of the Order, the staff views the September 24, 1977 event at DB-1 to have been similar to the TMI-2 event in several important aspects. However, significant differences in plant status and operator response contributed to produce a much less severe transient. The staff concludes that satisfactory improvements in both design and emergency procedures have been made since the DB-1 event and, that, the licensee has complied with the requirement of Item (g) of the Order.

<u>CONCLUSION</u>

We conclude that the actions described above fulfill the requirements of our Order of May 16, 1979 in regard to Paragraph (1) of Section IV. The licensee having met the requirements of Paragraph (1) may restart DB-1 as provided by Paragraph (2). Paragraph (3) of Section IV of the Order remains in force until the long term modifications set forth in Section II of the Order are completed and approved by the NRC.

REFERENCES

1. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting report entitled, "Evaluation of Transient Behavior and Small Reactor Cooalnt System Breaks in the 177 Fuel Assembly Plant," dated May 7, 1979.

2. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting revised Appendix 1, "Natural Circulation in B&W Operating Plants (Revision 1)," dated May 8, 1979.

3. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting additional information regarding Appendix 2, "Steam Generator Tube Thermal Stress Evaluation," to report identified in Item 1 above, dated May 10, 1979.

4. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing an analysis for "Small Break in the Pressurizer (PORV) with no Auxiliary Feedwater and Single Failure of the ECCS," identified as Supplements 1 and 2 to Section 6.0 of report in Item 1, dated May 12, 1979.

5. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing Supplement 3 to Section 6 of report in Item 1, dated May 24, 1979.

6. Letter from Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated May 22, 1979, providing Volume III to Reference 1 for the raised loop plant.

7. Letter from Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated May 23, 1979.

8. Letter from Lowell E. Roe (TECO-Serial No. 517) to Harold R. Denton (ONRR) dated June 15, 1979.

9. Letter from Lowell E. Roe (TECO) to Mr. Robert W. Reid (NRC) dated December 22, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

TOLEDO EDISON AND

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO. 50-346

NOTICE OF AUTHORIZATION TO RESUME OPERATION

The United States Nuclear Regulatory Commission issued an Order on May 16, 1979 (44 F.R. 29767, May 22, 1979), to The Toledo Edison and The Cleveland Electric Illuminating Company (TECO or The Licensee), holders of Facility Operating License No. NPF-3, for the Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse), confirming that the licensee accomplish a series of actions, both immediate and long-term, to increase the capability and reliability of Davis-Besse to respond to various transient events. In addition, the Order confirmed that the licensee would maintain the plant in a shutdown condition until the following actions had been satisfactorily completed:

(a) Review all aspects of the safety grade auxiliary feedwater system to further upgrade components for added reliability and performance: Present modifications will include the addition of dynamic braking on the auxiliary feedpump turbine speed changer and provision of means for control room verification of the auxiliary feedwater flow to the steam generators. This means of verification will be provided for one steam generator prior to startup from the present maintenance outage and for the other steam generator as soon as vendor-supplied equipment is available (estimated date is June 1, 1979). In addition, the licensees will review and verify the adequacy of the auxiliary feedwater system capacity.

(b) Revise operating procedures as necessary to eliminate the option of using the Integrated Control System as a backup means for controlling auxiliary feedwater flow.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) All licensed reactor operators and senior reactor operators will have completed the Three Mile Island Unit No. 2 simulator training at B&W.

(f) Submit a reevaluation of the TECO analysis of the need for automatic or administrative control of steam generator level setpoints during auxiliary feedwater system operation, previously submitted by TECO letter of December 22, 1978, in light of the Three Mile Island Unit No. 2 incident.

(g) Submit a review of the previous TECO evaluation of the September 24, 1977 event involving equipment problems and depressurization of the primary system at Davis-Besse 1 in light of the Three Mile Island Unit No. 2 incident.

By letters dated April 27 and May 4, 1979 and supplemented by sixteen letters dated May 11, 18, 19, 22 (2), 23 (2), 26 (2), 29, and June 15 (2), 18, 21, 23, and 25, 1979, the licensee has documented the actions taken in response to the May 16 Order. Notice is hereby given that the Director of Nuclear Reactor Regulation (the Director) has reviewed this submittal and has concluded that the licensee has satisfactorily completed the actions prescribed in items (a) through (g) of paragraph (1) of Section IV of the Order, that the specified analyses are acceptable and the specified implementing procedures are appropriate. Accordingly, by letter dated July 6, 1979, the Director has authorized the licensee to resume operation of Davis-Besse. The bases for the Director's conclusions are more fully set forth in a Safety Evaluation dated July 6, 1979.

Copies of (1) the licensee's letters dated April 27 and May 4, 1979 and sixteen letters dated May 11, 18, 19, 22 (2), 23 (2), 26 (2), 29, and June 15 (2), 18, 21, 23, and 25, 1979, (2) the Director's letter dated July 6, 1979 and (3) the Safety Evaluation dated July 6, 1979, are available for inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. 20555, and are being placed in the Commission's local public document room in The IDA Rupp Public Library, 310 Madison Street, Port Clinton, Ohio 43452. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Dated at Bethesda, Maryland
this 6th day of July 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket No.: 50-302

July 6, 1979

Mr. W. P. Stewart
Manager, Nuclear Operations
Florida Power Corporation
P. O. Box 14042
Mail Stop C-4
St. Petersburg, Florida 33733

Dear Mr. Stewart:

By Order of May 16, 1979, the Commission confirmed your undertaking a series of actions, both immediate and long-term, to increase the capability and reliability of the Crystal River, Unit No. 3 to respond to various transient events. In addition, the Order confirmed that you would maintain the plant in a shutdown condition until the following actions had been satisfactorily completed:

"(a) Upgrade the timeliness and reliability of delivery from the Emergency Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of May 1, 1979."

"(b) Develop and implement operating procedures for initiating and controlling emergency feedwater independent of Integrated Control System control."

"(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip."

"(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action."

"(e) All licensed reactor operators and senior reactor operators will have completed the TMI-2 simulator training at B&W."

By your letter dated May 1, 1979 and supplemented by five letters, dated May 16, June 12, 15, 22, and 29, 1979, you have documented the actions taken in response to the May 16 Order. We have reviewed this submittal, and are satisfied that, with respect to Crystal River, Unit 3, you have satisfactorily completed the actions prescribed in items (a) through (e) of paragraph (1) of Section IV of the Order, the specified analyses are acceptable, and the specified implementing procedures are appropriate. The bases for these conclusions are set forth in the enclosed Safety Evaluation.

As noted on page 12 of the Safety Evaluation, you are required to conduct a test during power ascension (<15% rated power) to demonstrate the capability to provide and control emergency feedwater flow to both steam generators independent of the integrated control system.

Appropriate Technical Specifications for Limiting Conditions for Operation and for surveillance requirements should be developed as soon as practicable and provided to the staff within seven days with regard to the design and procedural changes which have been completed in compliance with the provisions of the May 16, 1979 Commission Order. The revised Technical Specifications should cover:

(1) Addition of flow rate indication for the emergency feedwater system;

(2) Addition of the auto start circuitry for the emergency feedwater pumps;

(3) Addition of the anticipatory reactor trips; and

(4) Changes in set points for high pressure reactor trip and PORV actuation.

Within 30 days of receipt of this letter, you should provide us with your schedule for completion of the long-term modifications described in Section II of the May 16 Order.

My finding of satisfactory compliance with the requirements of items (a) through (e) of paragraph (1) of Section IV of the Order will permit resumption of operation in accordance with the terms of the Commission's Order; it in no way affects your duty to continue in effect all of the above provisions of the Order pending your submission and approval by the Commission of the Technical Specification changes necessary for each of the required modifications.

Sincerely,

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Safety Evaluation
2. Notice

cc w/encls:
See next page

Florida Power Corporation

cc w/enclosure(s):
Mr. S. A. Brandimore
Vice President and General Counsel
P. O. Box 14042
St. Petersburg, Florida 33733

Mr. Wilbur Langely, Chairman
Board of County Commissioners
Citrus County
Iverness, Florida 36250

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
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EVALUATION OF LICENSEE'S COMPLIANCE

WITH THE NRC ORDER DATED MAY 16, 1979

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING STATION

DOCKET NO. 50-302

July 2, 1979

INTRODUCTION

By Order dated May 16, 1979, (the Order) the Florida Power Corporation (licensee or FPC) was directed by the NRC to take certain actions with respect to Crystal River Unit No. 3 (CR-3). Prior to this Order, and as a result of a preliminary review of the Three Mile Island Unit No. 2 (TMI-2) accident, the NRC staff initially identified several human errors that contributed significantly to the severity of the event. All holders of operating licenses were instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's Office of Inspection and Enforcement (IE). Subsequently, an additional bulletin was issued by IE which instructed holders of operating licenses for Babcock & Wilcox (B&W) designed reactors to take further actions, including immediate changes to decrease the reactor high pressure reactor trip point and increase the pressurizer power-operated relief valve (PORV) setting.*

The NRC staff identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors. Those were identified as items (a) through (e) on page 1-7 of the "Office of Nuclear Reactor Regulation Status Report to the Commission" dated April 25, 1979. After a series of discussions between the NRC staff and the licensee concerning possible design modifications and changes in operating procedures, the licensee agreed, in a letter dated May 1, 1979, to perform promptly certain actions. The Commission found that operation of the plant should not be resumed until actions described in paragraphs (a) through (e) of paragraph (1) of Section IV of the Order were satisfactorily completed.

*[IE Bulletins Nos. 79-05 (April 1, 1979), 79-05A (April 5, 1979), and 79-05B (April 21, 1979) apply to all B&W facilities.]

Our evaluation of the licensee's compliance with items (a) through (e) of paragraph (1) of Section IV of the Order is given below. In performing this evaluation we have utilized additional information provided by the licensee on May 16, and June 12, 15, 22 and 29, 1979, and numerous discussions with the licensee's staff. Confirmation of design and procedural changes was made by members of the NRC staff at the Crystal River site. An audit of the Crystal River reactor operators was also performed by the NRC staff to assure that the design and procedural changes were understood and were being correctly implemented by the operators.

EVALUATION

Item (a)

It was ordered that the licensee take the following action:

> "Upgrade the timeliness and reliability of delivery from the Emergency Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of May 1, 1979."

The Crystal River emergency feedwater (EFW) system design has one turbine-driven pump that is automatically actuated and controlled independent of offsite power, and one motor-driven EFW pump that is automatically started if offsite power is available, but must be manually started on a vital AC bus if offsite power is lost. By reference above to Enclosure (1) of the licensee's letter of May 1, 1979, it was ordered that the licensee:

1. "Review procedures, revise as necessary and conduct training to ensure timely and proper starting of motor-driven emergency

feedwater (EFW) pump from engineered safeguards bus A upon loss of offsite power."

The licensee has revised EP-101 ("Unit Blackout") to provide the operators with a procedure for loading the motor-driven EFW pump on engineered safeguards bus 3A. This procedure will be used only if the following three conditions are met: (1) loss of offsite power, (2) the turbine-driven pump is not functioning, and (3) EFW is required. The procedure directs the operators to strip (remove) the following loads from bus 3A prior to loading the motor-driven EFW pump on the bus: (1) the decay heat removal pump, (2) building spray pump, (3) closed loop cooling pump, and (4) raw water pump. The loads stripped from the bus are not required for shutdown cooling during the period that EFW is used for decay heat removal. However, a redundant decay heat removal train would be available on the other emergency bus. When EFW is no longer needed, the motor-driven EFW pump would be removed from the bus and the other loads restored to the emergency bus.

The NRC staff performed an audit at the site and verified that the operators were knowledgeable in the steps of this procedure.

The NRC staff concludes that the licensee has adequate procedures and the operators are properly trained to start the motor-driven EFW pump from the diesel powered engineered safeguards bus 3A upon a loss of offsite power, and therefore, meets the requirements of this part of the Order.

It was also ordered that:

2. "To assure that EFW will be aligned in a timely manner to inject on all EFW demand events when in the surveillance test mode, procedures will be implemented and training conducted to provide an operator at the necessary valves in communication with the control room during the surveillance mode to carry out the valve alignment changes upon EFW demand events."

Surveillance Procedure SP-349 ("Emergency Feedwater System Operability Demonstration") directs the operators to close the EFW pumps' discharge valves to perform the test and then provides direction to reopen the valves to their normal operating positions following the test. These discharge valves are motor-operated valves that are controlled from the main control room; therefore, there is no need to station an operator at the valve locations during the surveillance testing. SP-349 requires that an operator determine that the EFW system valves are properly aligned and a second operator verifies the valve positions following the test. In addition to the independent verification of valve lineups required following surveillance testing, independent verification is also required upon completion of maintenance on the system. A valve lineup check list for the EFW system is included in SP-300 ("Operating Daily Surveillance Log").

The NRC staff has conducted an audit to verify that the operators are aware of the valve lineup requirements.

The NRC staff concludes the licensee has developed adequate procedures and has properly trained operators to verify correct valve alignments in the EFW system, and, therefore, is in compliance with this part of the Order.

It was ordered that:

3. "Emergency feedwater bypass valves are normally in the open position. Procedures have been developed and implemented to require the operator to take control of these valves upon failure of the ICS steam generator level control. If the ICS level control does not fail the operator will close the bypass valves. Those valves in the EFW system not locked in position are verified to be in the proper position on a daily basis. Training will be conducted on these revised procedures prior to June 1, 1979."

Emergency feedwater flow is normally controlled by the integrated control system (ICS) to maintain the required steam generator levels by actuation of the air-operated feedwater startup valves. An alternate path, independent of the ICS, is provided through the motor-operated EFW bypass valves. The bypass valves are normally maintained in the open position; however, following EFW activation, the operators must close the bypass valves and monitor steam generator levels and EFW flow to determine if the ICS is functioning properly. If the ICS fails to maintain the proper steam generator levels, the operator is directed to control the level by throttling flow with the bypass valves.

The licensee has modified the following emergency procedures to provide this guidance to the operators: EP-101 ("Unit Blackout"), EP-103 ("Loss of RC Flow/RC Pump Trip"), EP-106 ("Loss of Reactor Coolant or Reactor Coolant Pressure"), and EP-108 ("Loss of Steam Generator Feed").

The licensee has installed an ultrasonic flowrate meter system to provide control room indication of emergency feedwater flowrate in gallons per minute (gpm) to each steam generator. Each system consists of the ultrasonic flow transducer, mounted on the EFW piping, and the associated flow display computer, mounted locally. Flowrate indicators are also located in the control room on the main control board.

In addition to the directions for operator control of EFW flow if required, the licensee has provided for a daily verification of valve lineup in the EFW system in SP-300 ("Operating Daily Surveillance Log").

The NRC staff has conducted an audit at the site and verified that the operators are trained in these procedures. The NRC staff concludes that the licensee has provided adequate procedures and operator training to control the EFW system independent of the ICS, and is thus in compliance with this part of the Order.

It was also ordered that:

4. "The EFW pumps will be verified operable in accordance with the CR #3 Technical Specifications and Surveillance Procedures."

The Technical Specifications for CR-3 require a monthly test of the turbine-driven EFW pump to demonstrate its operability. The surveillance procedure requires running both of the EFW pumps with their discharge valves closed, with flow through the recirculation line, and measuring the discharge pressure of the pumps. We have reviewed the test procedure and find it acceptable. Satisfactory results of this monthly surveillance test is an acceptable basis for demonstrating the operability of the EFW pumps and, therefore, we conclude that the licensee is in compliance with this part of the Order.

The licensee was also ordered to:

5. "Review and revise, as necessary, the procedures and training for providing alternate sources of water to the suction of the EFW pumps."

Emergency Procedure EP-108 ("Loss of Steam Generator Feed") provides adequate direction to the operators for providing alternate sources of water to the suction of the EFW pumps. The primary source of water to the EFW pumps is the condensate storage tank (CST), which has a capacity of 150,000 gallons. The operator is alerted by a level alarm on the CST when the level drops to 89,000 gallons. The condenser hotwell, which has a capacity of 200,000 gallons, is the alternate source of water to the EFW pumps. The procedure directs the operators to open the motor-operated valves that will connect the hotwell to the suction of the EFW pumps and then to close the motor-operated suction valves from the CST. The procedure also contains instructions on valve operation to provide a third source of demineralized water from the water treatment system if needed.

The NRC staff, at the site, has verified that the control room operators are properly trained to carry out these procedures. The NRC staff concludes that the licensee has complied with the requirements to review and revise procedures and operator training for providing alternate sources of water for the EFW system, and, is thus in compliance with this portion of the Order.

The licensee was ordered to:

5. "Remove the interlock which prevents the turbine-driven emergency feedwater pump operation when the motor-driven emergency feedwater pump is running."

The licensee has removed the interlock. The turbine-driven EFW pump will start, if required, regardless of the motor-driven EFW pump status.

Based on the above design modification, we conclude that the licensee has complied with this portion of the Order.

It was also ordered that:

7. "In the event emergency feedwater is necessary and offsite power is available, an auto start signal will be provided to the motor-driven emergency feedwater pump."

The licensee has installed circuitry to provide automatic starting of the motor-driven EFW pump if offsite power is available. The auto start signals include either of the following:

(1) coincident loss of both main feedwater pumps, sensed by the loss of control oil pressure; or

(2) coincident low-low steam generator level in both steam generators, detected by existing and new equipment in the ICS.

Provisions have been included to manually bypass the loss of main feedwater pumps signal to allow for startup and/or shutdown. The bypass switch is keylocked, with annunciation and administrative control. The steam generator low-low level signal is not bypassed.

In addition, the licensee has modified the turbine-driven EFW pump start circuitry to include the same set of signals. Previously, this pump automatically started only on loss of both main feedwater pumps. Based on the above design modifications, we conclude that the licensee has complied with this portion of the Order.

It was also ordered that:

8. "Design review and modification, as necessary, will be conducted to provide control room annunciation for auto start conditions of the EFW system."

The licensee has provided control room annunciation to alert the operators that the EFW pumps (motor-driven and/or turbine-driven) have started when required or failed to start when required.

The conditions which initiate the above alarms include the same signals as discussed in Part 7 above:

(1) loss of main feedwater; or

(2) low-low level in both steam generators.

These signals are combined with the pump status (start or fail to start) to provide the annunciation. Based on the above modifications, we conclude that the licensee has complied with this portion of the Order.

It was also ordered:

9. "Verification has been made that the air-operated level control valves (a) fail to the 50% open position upon loss of power to the electrical/pressure converter, and (b) fail to the as is position upon loss of instrument air and electrical power to the air lock. At full load these valves are in the full (100%) open positions and at low power levels (below 15%) they are partially open controlling flow. If these valves were to fail closed, feedwater flow would be controlled using the EFW bypass valves as described in Item 3 above."

The licensee has completed its verification tests of the failure mode of the air-operated level control valves. The results show that one air-operated level control valve fails to a 54% open position and the other fails to a 47% open position upon loss of electrical power to the electrical/pressure converter. These failure positions are within acceptable tolerance of the 50% open position specified in the Order. On a test for loss of instrument air, both air-operated level control valves failed as is, i.e., remained at approximately the 50% open position during the test. The EFW bypass valves are motor-operated regulating valves which are operated independently from the ICS as discussed in part 3 above. If the air-operated level control valves remain closed or ICS fails, EFW flow would be manually controlled using the EFW bypass valves. We conclude that the licensee has satisfied this portion of the Order.

Conclusion:

Based upon our evaluation of Parts 1 through 9 above, we conclude that the licensee has upgraded the timeliness and reliability of delivery from the EFW system by carrying out the actions identified in Enclosure 1 of the licensee's letter of May 1, 1979, and is, therefore, in compliance with Item (a) of the Order.

Item (b)

The licensee was ordered to:

> "Develop and implement operating procedures for initiating and controlling emergency feedwater independent of Integrated Control System Control."

The NRC staff has reviewed the revised procedures for the EFW system to assure that there is sufficient guidance for the operators to actuate the system if automatic initiation fails and to control steam generator levels at the required values. The The NRC staff review of the procedures and the operator training focused on whether the operators were directed to observe the proper instruments and whether operators were given specific values of parameters, such as steam generator level, to maintain by operating the control valves. The review also determined that the validity of the instrument readings of certain key parameters, such as steam generator level, would be confirmed. The modifications to the procedures to satisfy these determinations were verified by the NRC staff. (See further discussion of these procedures in part 3 of Item (a)).

We will require the licensee to perform a test during power ascension (less than 15% rated power) to demonstrate the capability to provide and control EFW flow to both steam generators. The primary objective is to verify that the operators can initiate EFW and control steam generator levels independent of the ICS. A member of the NRC staff will witness the test and verify acceptability prior to authorizing the licensee to proceed to full power operation.

The NRC staff audited a sample of Crystal River operators to determine if they were familiar with the revised procedures and could implement them correctly. Based on the NRC audit, we conclude that the revised procedures and operator training are satisfactory, and, therefore, the licensee is in compliance with Item (b) of the Order.

Item (c)

The Order required that the licensee:

> "Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or on turbine trip."

The CR-3 original design did not have a direct reactor trip from a malfunction in the secondary system (loss of main feedwater and/or turbine trip). To obtain an earlier reactor trip (rather than delaying the trip until an operator took action or until a primary system parameter exceeded its trip setpoint), the licensee committed to install a hard-wired, control-grade reactor trip on the loss of all main feedwater and/or on turbine trip (letter from B. L. Griffin (FPC) to H. Denton (NRC) dated May 1, 1979). The purpose of this anticipatory trip is to minimize the potential for opening of the PORV and/or the safety valves on the pressurizer.

The licensee has added control-grade circuitry which is designed to provide an automatic reactor trip when either the main turbine trips or all main feedwater is lost.

The main turbine/generator trip is sensed by an existing pressure switch in the turbine electro-hydraulic control system. On a turbine trip, the pressure switch energizes a normally deenergized relay in the ICS. A contact from this relay is arranged in a normally energized circuit containing two parallel reactor trip actuation relays. Deenergizing both of these relays provides an output to energize the 125 DC volt shunt trip coils of the two reactor trip breakers. Energizing both reactor trip breakers trips the reactor.

The loss of main feedwater is sensed by either of two signals: loss of the main feedwater pumps or low-low level in both steam generators (the same signals which start EFW). The signals are generated separately for each feedwater path. Any one of these signals will energize a relay in the ICS (one relay for each feedwater path). The contacts from these relays are arranged in the same circuity as the reactor trip actuation relays such that any coincidence of signals from the two feedwater paths will deenergize the relays, in the same manner as the turbine trip, causing a reactor trip.

Provisions have been included to automatically bypass and reinstate the loss of main feedwater pump and turbine trip signals at less than 10% power to allow for normal startup and shutdown of equipment without tripping the reactor. Operator verification of the bypass removal is required by procedure during power ascention.

The licensee has analyzed this additional circuitry with respect to its independence from the existing reactor trip system. They have stated that the shunt coil is part of the existing AC reactor trip breaker. Each shunt coil is powered by a separate Class IE 125 volt supply and operates independently from the 120 volt AC undervoltage trip coil of the same AC reactor trip breaker, which receives a safety-grade reactor protection system trip signal.

An NRC inspector has confirmed that the checkout tests for the circuitry were completed successfully. In addition, the licensee has committed to perform a monthly test on the added circuitry in order to demonstrate its ability to open the AC reactor trip circuit breakers.

Based on our review of the implementation of the trip circuitry with respect to its independence from the existing trip circuitry, we conclude that this addition will not degrade the existing reactor protection system design.

Based on the licensee's design modifications and commitment to perform a monthly test on the new circuitry, we conclude that there is reasonable assurance that the system will perform its intended function.

Based on the above evaluation, we conclude that the licensee is in compliance with the requirements of Item (c) of the Order.

Item (d)

This item in the Order required the licensee to:

> "Complete analyses for potential small breaks and develop and implement operating instructions to define operator action."

By letter dated May 1, 1979, the licensee committed to providing the analyses and operating procedures of this requirement.

B&W, the reactor vendor for the Crystal River plant, submitted analyses entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" and supplements to these analyses (References 1 through 6). The major parameters used in this generic study bound the Crystal River plant. The staff evaluation of the B&W generic study has been completed and the results of the evaluation will be issued as a NUREG report in July 1979.

A principal finding of our generic review is a reconfirmation that loss-of-coolant accident (LOCA) analyses of breaks at the lower end of the small break spectrum (smaller than 0.04 sq. ft.) demonstrate that a combination of heat removal by the steam generators and the high pressure injection (HPI) system combined with operator action ensure adequate core cooling. The EFW system, used to remove heat through the steam generators, has been modified to enhance its reliability as discussed in Item (a). The HPI system is capable of providing emergency core cooling up to the safety valve pressure setpoint. The ability to remove heat via the steam generators has always been recognized to be an important consideration when analyzing very small breaks. Separate sensitivity analyses (for breaks smaller than 0.01 sq. ft.) were performed assuming permanent loss of all feedwater (with operator initiation of the HPI system at 20 minutes), and loss of feedwater for only the first 20 minutes of the accident. Uncovering of the reactor core was not predicted for these events. The calculated peak cladding temperature was less than 800°F, well below the 10 CFR 50.46 requirement of 2200°F. These results are applicable to Crystal River considering the ability to manually start the redundant EFW pumps and HPI pumps from the control room, assuming failure of automatic EFW actuation.

Another aspect of the study was the assessment of recent design changes on the lift frequency of the pressurizer PORV and safety valves. The design changes included: (1) a change in the setpoint of the PORV from 2255 psig to 2450 psig; (2) change in the high pressure reactor trip setpoint from 2355 psig to 2300 psig; and (3) the installation of an anticipatory reactor trip on turbine trip and/or loss of all main feedwater. In the past, during the turbine trip or loss of feedwater transients, the PORV lifted. With the design changes, the initial pressure increase of these transients do not result in lifting of this valve. However, the consequent depressurization could initiate HPI which could repressurize the system and lift the PORV valve. It is expected that the operator would terminate HPI before the PORV or safety valves lift, since the 50°F subcooling criteria would be satisfied at pressures below the PORV setpoint.* Also, lifting of both the PORV and safety valves might occur in the cases of control rod withdrawal or inadvertent boron dilution transients, using the normally conservative assumptions found in the Final Safety Analysis Report Chapter 15 safety analyses. The above

*(The 50°F subcooling criteria is discussed on page 20 of this evaluation.)

design changes do not effect the lift frequency of the valves for these Chapter 15 safety analyses.

Based on our review of the small break analyses presented by B&W, the staff has determined that a loss of all main feedwater with (1) an isolated PORV, but safety valves opening and closing as designed, or (2) a stuck open PORV does not result in uncovering the reactor core, provided either EFW or HPI (2 pumps) is initiated within 20 minutes. Based on the consequences calculated for small break LOCAs and loss of all main feedwater events, and taking into account expected reliability of the EFW and HPI systems, we conclude that the licensee has complied with the analyses portion of Item (d) of the Order.

To support long-term operation of the facility, requirements will be developed for additional and more detailed analyses of loss of feedwater and other anticipated transients. More detailed analyses of small break LOCA events are also needed for this purpose. Accordingly, the licensee will be required to provide the analyses discussed in Sections 8.4.1 and 8.4.2 of the recent NRC "Staff Report of the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company" (NUREG 0560). Further details on these analyses and their applicability to other PWRs and BWRs will be specified by the staff in the near future. In addition, to assist the staff in developing more detailed guidance on design requirements of PORV and safety valve reliability during anticipated transients, as discussed in Section 8.4.6 of the NUREG 0560, the licensee will be required to provide analyses of the lift frequency and mechanical reliability of the pressurizer PORV and safety valves of the Crystal River facility.

The B&W analyses show that some operator actions, both immediate and follow-up, are required under certain circumstances for a small break accident. Immediate operator actions are defined as those actions, committed to memory by the operators, which must be carried out as soon as the problem is diagnosed. Follow-up actions require operators to consult and follow the steps in written and approved procedures. These procedures must always be readily available in the control room for the operators' use.

Guidelines were developed by B&W to assist the operating B&W facilities in the development of emergency procedures for the small break accident. "Operating Guidelines for Small Breaks" were issued by B&W on May 5, 1979, and reviewed by the NRC staff. These guidelines were revised on May 15, 1979, to include revisions recommended by the staff (Reference 7). In response to these guidelines, the licensee made substantial revisions to EP-106 ("Loss of Reactor Coolant/RC System Pressure"), and EP-103 ("Loss of RC Flow/RC Pump Trip"). These emergency procedures define required operator actions in response to a spectrum of break sizes for a LOCA in conjunction with various equipment availability and failures.

EP-106 ("Loss of Reactor Coolant/RC System Pressure") is divided into three sections. The first section deals with a leak or rupture that is within the capability of one makeup pump.* In this case, the operators proceed with an orderly plant shutdown, if the leak is in excess of the Technical Specification limits.

The second section of EP-106 defines required operator actions for a small break that is within the capability of the HPI system to maintain RCS pressure and

*[At CR-3 the HPI pumps are used for makeup pumps.]

pressurizer level. This assumes that the initial break was of a size sufficient to cause a depressurization with a resulting reactor trip and HPI actuation. This part of the procedure provides the operators with the guidance necessary to achieve a safe hot shutdown condition for a variety of degraded conditions. If all feedwater is lost, a heat removal path is established by the HPI system through the break and the pressurizer PORV or the safety valves. Once feedwater is reestablished, the steam generators can be used as a heat sink. If the reactor coolant pumps are not available, the operator is directed to EP-103 ("Loss of Reactor Flow/RCP Trip") which defines the actions necessary to establish natural circulation. Additional guidance is provided in EP-103 if natural circulation is not immediately achieved. This includes "bumping" reactor coolant pumps or if they cannot be operated, using the PORV to control RCS pressure until either forced flow or natural circulation can be achieved. If natural circulation has been established and plant conditions are stable, the operator is directed to AP-113 ("Reactor Cooldown by Natural Circulation"). If forced circulation is established, the normal plant cooldown procedure (OP-209, "Plant Cooldown") is used in conjunction with EP-106.

The third section of EP-106 deals with a large pipe rupture in which the system depressurizes to the point of low pressure injection (LPI). The system response is not dependent upon the availablity of reactor coolant pumps or feedwater and, therefore, no other procedures need be referenced.

For all cases in which HPI is manually or automatically initiated, the operators are specifically instructed to maintain maximum HPI flow unless one of the following criteria is met:

(1) the LPI system is in operation and providing cooling at a rate in excess of 1000 gpm and the situation has been stable for 20 minutes; or

(2) all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If the 50 degrees subcooling cannot be maintained after HPI cutoff, HPI shall be reactuated.

A requirement to determine and maintain 50°F subcooling has been incorporated in all other procedures in which HPI has been manually or automatically initiated. These procedures include, EP-105 ("Steam Supply System Rupture"), EP-108 ("Loss of Steam Generator Feed"), EP-101 ("Unit Blackout"), and EP-103 ("Loss of RC Flow/RC Pump Trip"). Each of these procedures, in addition to EP-106 ("Loss of Reactor Coolant/RC System Pressure") procedure, provide additional instructions to the operators in the event of faulty or misleading indications. A subsequent action statement directs the operators to check alternate instrumentation channels to confirm key parameter readings. The Crystal River staff has made revisions to all of their emergency procedures to include this confirmation. The CR-3 incore thermocouples will be hard-wired to a dedicated monitoring system which is programmed to alarm at high temperature. In addition, the operators will be able to check all input readings and/or get a printout of the status of each thermocouple with this system. A process computer in the control room is also available to provide this indication.

If feedwater is not initially available following a transient or accident, core cooling is maintained by flow from two HPI pumps and relief through the PORV, which is opened by the operator. B&W has performed studies that show density differences between the downcomer and reactor core will cause recirculation flow between the core exit and downcomer via the vent valves. Mixing of the hot core exit water with the cold HPI water will provide sufficiently warm vessel temperatures to preclude any significant thermal shock effects to the vessel. Under these conditions, with no circulation of water from the steam generators, the cold leg resistance temperature detectors (RTD) may not provide a satisfactory indication of the vessel temperature. B&W has recommended using the core exit thermocouples as a measure of vessel temperature, based on B&W analyses that conservatively show that the vent valves will open at temperature differences between the core exit and downcomer of less than 150°F. They have also proposed a more appropriate pressure-temperature limit curve for the vessel that reflects allowable stresses under these faulted conditions (no feedwater).

The NRC staff has reviewed these guidelines and finds them acceptable based on the expected recirculation through the vent valves and the vessel stress limits used. The licensee has incorporated these revised guidelines into Emergency Procedure EP-108 ("Loss of Steam Generator Feed"). Subsequent restoration of EFW would depressurize the reactor coolant system to below 600 psig where pressure vessel integrity is assured for any reasonable thermal transients that might subsequently occur. We conclude that further reliability analyses are needed as part of the long-term requirements of the Order to confirm that EFW can be restored (if lost) in a reasonable period of time. B&W has agreed to provide a detailed thermal-mechanical report on the behavior of vessel materials for these extreme conditions, to be applicable generically to the Oconee class of plants, which includes Crystal River.

The Crystal River Unit 3 main control board has an annunciator which alarms when the PORV solenoid is energized (to open the valve). In addition, there are 3 indicating lights which are actuated by 2 selector switches of the valve control circuitry. The green light is lit when the "AUTO-OPEN" selector switch is in the "AUTO" position. In this position, the pressure signal will provide the open and close control of this valve. The red light is lit when the same switch is in the "OPEN" position. In this position, the selector switch will control the valve (to the open position). The amber light is lit when the "NORM-LO" selector switch is in the "LO" position. In this position, the low pressure protection circuit is operable and can open the valve for this mode of operation.

EP-106 ("Loss of Reactor Coolant/RC System Pressure") was reviewed by the NRC staff to determine its conformance with the B&W guidelines. Comments generated as a result of this review were incorporated in a further revision to the procedure. A member of the NRC staff walked through this emergency procedure in the Crystal River control room. The procedure was judged to provide adequate guidance to the operators to cope with a small break LOCA. The instrumentation necessary to diagnose the break, the indications and controls required by the action statements, and the administrative controls which prevent acceptable limits from being exceeded are readily available to the operators. We conclude that the operators should be able to use this procedure to bring the plant to a safe shutdown condition in the event of a small break accident.

An audit of 8 of the 23 licensed operators assigned to shift duty was conducted by the NRC staff to determine the operators' understanding of the small break accident, including how they are required to diagnose and respond to it. The Crystal River staff has conducted special training sessions for the operators on the concept and use of EP-106 and other emergency procedures related to the small break accident. The audit revealed several deficiencies in the knowledge of the small break phenomenon and the requirements of the procedure. Additionally, there were deficiencies in the knowledge of the details of the recent design modifications made to the Crystal River plant. These deficiencies were primarily the result of design modifications and procedure revisions not finalized at that time. As a result of the audit, each licensed individual received additional training by the plant training organization and by the General Physics Corporation (GPC). This additional training has been completed and verified by the NRC staff. A subsequent reaudit of 10 licensed individuals by the NRC revealed satisfactory results.

The audit of the operators also included questioning about the TMI-2 incident and the resulting impact on the Crystal River plant. The discussions covered the initiating events of the incident, the response of the plant to the simultaneous loss of feedwater and small break LOCA (PORV stuck open), and the operational actions that were taken during the course of the incident. We identified a deficiency in interpreting the initial sequence of the TMI-2 incident on the part of several of the operators. Additional training has been conducted in this area by the plant staff and their consultant GPC and has been verified by the NRC staff.

In summary, we found their level of understanding sufficient to be able to respond to a similar situation if it happened at Crystal River. We also conclude they have adequate knowledge of subcooling and saturated conditions and are able to recognize each in the primary coolant system by several methods. The EFW system was also discussed during the audit to determine the operators' ability to assure proper starting and operation of the system during normal conditions, as well as during adverse conditions such as loss of offsite power or loss of normal feedwater. The long-term operation of the system was examined to evaluate the operators' ability to use available manual controls and water supplies. The level of understanding was found to be sufficient to assure proper short-and long-term EFW flow to the steam generators.

In addition to the oral audit conducted by the NRC, the licensee administered a written examination to all licensed personnel. Individuals scoring less than 90 percent on the exam will receive additional training and will not assume licensed duties until a score of at least 90 percent is attained on an equivalent, but different exam. The written exam and the grading were audited by the NRC staff and judged to be acceptable. The staff will also review all subsequent results and records as part of the normal inspection function of the Crystal River re-qualification program. We conclude that there is adequate assurance that the operators at Crystal River have and will continue to receive a high level of training concerning the TMI-2 accident and the consequent impact on their unit.

Based on the foregoing evaluation, we conclude that the licensee has complied with the requirements of Item (d) of the Order.

Item (e)

The Order required that:

> "All licensed reactor operators and senior reactor operators will have completed the TMI-2 simulator training at B&W."

The licensee has confirmed that all reactor operators and senior reactor operators have completed the TMI-2 simulator training at B&W as required by the Order. This training consisted of a class discussion of the TMI-2 event and a demonstration of the event on the simulator as it occurred and how it should have been controlled. The class discussion was about one hour long and the remainder of the four hour session was conducted on the simulator. The TMI-2 event, including operational errors, was demonstrated to each operator. The event was again initiated and the operators were given "hands-on" experience in successfully regaining control of the plant by several methods. Other transients which resulted in depressurization and saturation conditions were presented to the operators in which they maneuvered the plant to a stable, subcooled condition.

Based on the foregoing evaluation, we conclude that the licensee has complied with the requirements of Item (e) of the Order.

CONCLUSION

We conclude that the actions descrilbed above fulfill the requirements of our Order of May 16, 1979, in regard to Paragraph (1) of Section IV. The licensee having met the requirements of Paragraph (1) may restart Crystal River as provided by Paragraph (2). Paragraph (3) of Section IV of the Order remains in force until the long-term actions set forth in Section II of the Order are completed and approved by the NRC.

REFERENCES

1. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting report entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," dated May 7, 1979.

2. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting revised Appendix 1, "Natural Circulation in B&W Operating Plants (Revision 1)," dated May 8, 1979.

3. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting additional information regarding Appendix 2, "Steam Generator Tube Thermal Stress Evaluation," to report identified in Item 1 above, dated May 10, 1979.

4. Letter from J. H. Taylor (B&W) to T. M. Novak (NRC) providing background information on reactor coolant pump operation, dated May 10, 1979.

5. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing an analysis for "Small Break in the Pressurizer (PORV) with no Auxiliary Feedwater and Single Failure of the ECCS" identified as Supplements 1 and 2 to Section 6.0 of report in Item 1, dated May 12, 1979.

6. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing Supplement 3 to Section 6 of report in Item 1, dated May 24, 1979.

7. Letter from J. H. Taylor (B&W) to Z. R. Rosztoczy (NRC) transmitting revised "Operating Guidelines for Small Breaks," dated May 16, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

FLORIDA POWER CORPORATION

DOCKET NO. 50-302

NOTICE OF AUTHORIZATION TO RESUME OPERATION

The United States Nuclear Regulatory Commission issued an Order (the Order) on May 16, 1979 (44 F.R. 29765, May 22, 1979), to the Florida Power Corporation (FPC or licensee), holder of Facility Operating License No. DPR-72, for the Crystal River Unit No. 3 Nuclear Generating Plant (the facility or Crystal River Unit 3), confirming that the licensee accomplish a series of actions, both immediate and long-term, to increase the capability and reliability of the facility to respond to various transient events. In addition, the Order confirmed that the licensee would maintain the plant in a shutdown condition until the following actions had been satisfactorily completed:

"(a) Upgrade the timeliness and reliability of delivery from the Emergency Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of May 1, 1979."

"(b) Develop and implement operating procedures for initiating and controlling emergency feedwater independent of Integrated Control System control."

"(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip."

"(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action."

"(e) All licensed reactor operators and senior reactor operators will have completed the TMI-2 simulator training at B&W."

By letter dated May 1, 1979 and supplemented by five letters dated May 16, June 12, 15, 22, and 29, 1979, FPC has documented the actions taken in response to the May 16 Order. Notice is hereby given that the Director of

Nuclear Reactor Regulation (the Director) has reviewed this submittal and has concluded that the licensee has satisfactorily completed the actions prescribed in items (a) through (e) of paragraph (1) of Section IV of the Order, that the specified analyses are acceptable and the specified implementing procedures are appropriate. Accordingly, by letter dated July 6, 1979, the Director has authorized the licensee to resume operation of Crystal River Unit 3. The bases for the Director's conclusions are more fully set forth in a Safety Evaluation dated July 2, 1979.

Copies of (1) the licensee's letter dated May 1, 1979 and five letters dated May 16, June 12, 15, 22, and 29, 1979, (2) the Director's letter dated July 6, 1979 and (3) the Safety Evaluation dated July 2, 1979, are available for inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. 20555, and are being placed in the Commission's local public document room in the Crystal River Public Library, 668 N. W. First Avenue, Crystal River, Florida 32629. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Dated at Bethesda, Maryland
this 6th day of July 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 18, 1979

Dockets Nos.: 50-269
50-270
and 50-287

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

By Order of May 7, 1979, the Commission confirmed your undertaking a series of actions, both immediate and long term, to increase the capability and reliability of Oconee Units Nos. 1, 2 and 3 to respond to various transient events. In addition, the Order confirmed that you would shut down Oconee Unit No. 3 on April 28, 1979, an additional Oconee unit on May 12, 1979, and remaining unit on May 19, 1979, unless, with respect to the latter two units, the following actions had been accomplished prior to the prescribed dates:

(a) Install automatic starting of the interconnected emergency feedwater system so that all three pumps will receive a start signal from any affected unit, and test the system for stability. The emergency feedwater pump discharge flow will be connected to the interconnection headers such that each or all of the emergency feedwater pumps can supply water to any unit. Until these modifications and tests are completed, operating personnel will be stationed at each emergency feedwater pump with a direct communication link to that unit's control room. In addition, the following procedural changes, put into effect on April 25, 1979 to enhance the reliability of the emergency feedwater system, will remain in force:

(1) The discharges of these pumps have been tied together by alignment of manual valves such that each and all of the pumps can supply emergency feedwater to any Oconee unit requiring it.

(2) Administrative controls have been established so that in the event of loss of both main feedwater pumps on an affected unit, that unit's emergency feedwater pump will start automatically, backed up by remote manual start from the control room. If the pump fails to start automatically, the operator stationed at that pump will start the pump locally, and has been trained to do so. In addition, the other two available emergency feedwater pumps will be started remotely from their unit's control room or locally if required to provide two more sources of feedwater to the affected unit.

(3) Emergency feedwater flow to the steam generators will be assured by the control room operator who has been trained to maintain the necessary level.

(b) Develop and implement operating procedures for initiating and controlling emergency feedwater independent of Integrated Control System control.

(c) Implement a hard-wired control-grade reactor trip on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) All licensed reactor operators and senior reactor operators assigned to the Oconee control rooms will have completed the TMI-2 simulator training at B&W.

By submittal of May 7, 1979, as supplemented by two letters dated May 9, 1979, you have documented the actions taken in response to the May 7 Order. I have reviewed this submittal, and am satisfied that, with respect to Oconee Units Nos. 1, 2 and 3, you have satisfactorily completed the actions prescribed in items (a) through (e) of paragraph (1) of Section IV of the Order, the specified analyses are acceptable, and the specified implementing procedures are appropriate. The bases for these conclusions are set forth in the enclosed Safety Evaluation.

As noted on page 5 of the Safety Evaluation you have performed flow stability tests with two units operating (Unit No. 3 having been shut down previously for refueling) and two emergency feedwater pumps operating. The staff has concluded that the tests, which did not result in any flow instability, are satisfactory and that you have complied with the requirement of the Order that a stability test be performed. However, we will require Duke Power Company to demonstrate acceptable flow rates and flow stability with only two operating steam driven emergency feedwater pumps when all three nuclear units are operational. The test plan must be reviewed by us before the tests are performed.

Appropriate Technical Specifications for Limiting Conditions for Operation and fur Surveillance requirements should be developed as soon as practicable and provided to the staff within seven days with regard to the design and procedural changes which have been completed in compliance with the provisions of the May 7, 1979 Commission Order. The revised Technical Specifications should cover:

(1) Changes to the Emergency Feedwater System;

(2) Plant alignment changes made to ensure control of emergency feedwater independent of the Integrated Control System;

(3) Addition of the Anticipatory Reactor Trip; and

(4) Changes in set points for high pressure reactor trip and PORV actuation.

Within 30 days of receipt of this letter, you should provide us with your schedule for completion of the long-term modifications described in Section II of the May 7, Order.

My finding of satisfactory compliance with the requirements of items (a) through (e) of paragraph (1) of Section IV of the Order will permit resumption or continuation of operation in accordance with the terms of the Commission's Order; it in no way affects your duty to continue in effect all of the above provisions of the Order pending your submission and approval by the Commission of the Technical Specification changes necessary for each of the required modifications.

Sincerely,

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor
Regulation

Enclosures:
1. Safety Evaluation
2. Notice

cc w/enclosures: See next page

cc w/enclosure(s):
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Oconee Public Library
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

EVALUATION OF LICENSEE'S COMPLIANCE

WITH THE NRC ORDER DATED MAY 7, 1979

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

DOCKETS NOS. 50-269, 270 AND 287

INTRODUCTION

By order dated May 7, 1979 (the Order), the Duke Power Company (DPC or the licensee) was ordered by the NRC to take certain actions with respect to Oconee Nuclear Station, Units 1, 2 and 3. Prior to this Order and as a result of a preliminary review of the Three Mile Island Unit No. 2 accident, the NRC staff initially identified several human errors that contributed significantly to the severity of the event. All holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's Office of Inspection and Enforcement (IE). Subsequently, an additional bulletin was issued by IE which instructed holders of operating licenses for Babcock & Wilcox (B&W) designed reactors to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer pilot-operated relief valve setting.

The NRC staff identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors. Those were identified as items (a) through (e) on page 1-7 of the Office of Nuclear Reactor Regulation Status Report to the Commission on April 25, 1979. After a series of discussions between the NRC staff and the licensee concerning possible design modifications and changes in operating procedures, the licensee agreed in letters dated April 25, 26, and May 4, 1979 to perform promptly certain actions. The Commission found that operation of all units should not be resumed or continued on an indefinite basis until actions described in paragraphs (a) through (e) of paragraph (1) of Section IV of the Order were satisfactorily completed.

Our evaluation of the licensee's compliance with items (a) through (e) of paragraph (1) of Section IV of the Order is given below. In performing this evaluation we have utilized additional information provided by the licensee on May 3, 8, 10, and 16, 1979 and numerous discussions with the licensee's staff. We have utilized confirmation of design and procedure changes by the NRC resident inspector at the Oconee site and an audit by the NRC staff of the training of the Oconee reactor operators to assure that the design and procedure changes are understood by the operators and that the revised procedures are being correctly implemented by the operators.

EVALUATION

Item (a)

The original Oconee design had a single emergency feedwater (EFW) pump for each unit that was actuated automatically when the main feedwater was lost on that unit. There were provisions for manually interconnecting the discharge of the EFW pumps so that they could service all three units. In letters from W. Parker (DPC) to H. Denton (NRC) dated April 25, 1979 and W. Lee (DPC) to H. Denton (NRC) dated April 26, 1979, the Duke Power Company committed to installing an automatic starting feature of the interconnected emergency feedwater system so that all three EFW pumps will receive a start signal from any affected unit, and to performing a stability test of the system when more than one unit was using EFW. A trained operator stationed at each EFW pump to actuate it locally, if required, and a trained operator in the control room to maintain required steam generator water levels were also included in the licensee's commitments.

In particular, it was ordered that the licensee shall take the following actions with respect to Oconee 1, 2 and 3:

> "Install automatic starting of the interconnected emergency feedwater system so that all three pumps will receive a start signal from any affected unit..." "The emergency feedwater pump discharge flow will be connected to the interconnection headers such that each or all of the EFW pumps can supply water to any unit." "The discharges of these pumps have been tied together by alignment of manual valves such that each and all of the pumps can supply emergency feedwater to any Oconee Unit requiring it."

The licensee has aligned his EFW system so that all three EFW pumps feed a common header and there are separate lines from this common header which deliver EFW through a control valve to each of the two steam generators in the three units. This alignment, which has been verified by an NRC inspector at the site, was accomplished by isolating other alternate flow paths in the feedwater system with existing manual valves.

We have reviewed the piping and instrumentation drawings and have determined that no active failure to a mechanical component such as a pump or valve would preclude obtaining EFW flow in any unit. The licensee has performed flow tests with two EFW pumps operating and providing flow to the four steam generators associated with two units. The minimum flow under this condition was greater than 720 gpm per unit which is acceptable. We have reviewed the modification to determine the minimum flow going to a unit if only one steam generator was functioning. The test to determine the maximum flow to one steam generator was not performed, but the licensee stated that the results of his analysis show that a minimum of 720 gpm EFW flow will be delivered to any unit with only one steam generator functioning. The licensee also stated that the flow characteristics of the EFW system used in the analysis were confirmed in the flow tests. In one of these flow tests with only one steam generator functioning in one of the units, the flow to that steam generator with about a 60% valve opening was approximately 500 gpm. We expect higher flows would be obtained by operater action. This 500 gpm flow rate with one steam generator available is acceptable based on automatic initiation of EFW.

Based on the above evaluation, the NRC staff concludes that the licensee has complied with the requirement for an interconnected EFW system such that each or all of the EFW pumps can supply water to any unit.

The licensee has added new circuitry which is designed to automatically start all three of the existing turbine-driven EFW pumps on loss of all main feedwater to any of the three units and to open the EFW regulating valves in the affected unit(s). The loss of main feedwater supply to each unit is sensed by a normally energized auxiliary relay. The relay is deenergized by either low pressure in the main feedwater discharge header or by both main feedwater pumps tripped in the affected unit. The relay actuates each of the three EFW pumps and the solenoids which control the EFW regulating valves to the two steam generators in the affected unit. Loss of the 115 VAC power will therefore actuate the three turbine-driven EFW pumps and automatically open the EFW regulating valves to the preselected setting and deliver EFW to the steam generators. This actuation is independent of the Integrated Control System (ICS).

The licensee has performed tests to demonstrate that these design modifications will automatically start all three EFW pumps and open the EFW regulating valves in the affected unit on loss of all main feedwater to any of the three units. The NRC resident inspector has verified the test results and we have concluded that the tests are satisfactory.

Based on the above, the NRC staff concludes that the licensee has complied with the requirement for automatic starting of the interconnected EFW system so that all three pumps will receive a start signal from any affected unit.

It was also ordered that the licensee "....test the system for stability." The licensee performed flow stability tests with two units operating and two EFW pumps available. The EFW flow to one of the steam generators was stopped by closing a manual valve in the discharge line and the reactor operators inside the control room manually adjusted the flow to the other three steam generators to prescribed flow rates. The manual control of the emergency feedwater flow did not result in any flow instability or operator problems. We conclude that these tests are satisfactory and that the licensee has complied with the requirement to perform a stability test. However, we require the licensee to perform additional startup tests on the EFW system when all three units are operational to demonstrate acceptable flow rates and flow stability with only two operating EFW pumps. The test plan must be reviewed by the NRC staff prior to the tests, which will be witnessed by an NRC inspector.

It was also ordered that:

> "Administrative controls have been established so that in the event of loss of both main feedwater pumps on an affected unit, that unit's EFW pump will start automatically, backed up by remote manual start from the control room. If the pump fails to start automatically, the operator stationed at that pump will start the pump locally, and has been trained to do so.
>
> In addition, the other two available EFW pumps will be started remotely from their unit's control room or locally if required to provide two more sources of feedwater to the affected unit."

The licensee has stationed an operator at each of the EFW pumps to start the pumps locally if required. The procedures associated with the EFW system (discussed in item (b)) have provisions for verifying that all EFW pumps start automatically with local initiation if required. The operators stationed locally at the pumps have been trained to start the pumps and have procedures for accomplishing this task. The implementation of this requirement has been verified by an NRC inspector. The actuation of the EFW pump locally is fairly simple. Depending on the fault, it may require bleeding off air to open the steam admission valve to the turbine (these are turbine driven pumps) before opening the governor valves with a local controller. Members of the NRC staff at the site have estimated it would take less than 5 minutes to actuate the pumps locally even allowing for several false starts. The operator stationed locally is also provided with piping and instrumentation diagrams (P&IDs) of the EFW pumps, descriptions of the governor valves, and auxiliaries such as the lube oil and cooling water. If necessary, the local operator can take remedial actions if subsequent problems with pump operation arise.

Based on our inspection at the site, we conclude that the licensee has complied with the requirement to have a trained operator stationed at each EFW pump and has administrative procedures covering his actions if required.

It was also ordered that:

> "Emergency feedwater flow to the steam generators will be assured by the control room operator who has been trained to maintain the necessary level."

The licensee has modified his procedures on the EFW system as discussed in item (b) of this evaluation. These procedures include verification of flow and manual control of steam generator water level. The regulating valves in the individual EFW discharge lines open automatically upon signal of need from the affected unit(s); it is only later that the valves are controlled by the operator. The licensee has installed flow orifices in each of the EFW discharge lines with indicators in the control room.

The NRC staff at the site has verified that control room operators is properly trained to carry out these procedures. We conclude that the licensee has complied with the requirement that a trained control room operator shall maintain the necessary steam generator water level.

Based on the above, we conclude that the licensee has complied with the requirements of item (a) of paragraph (1) of the NRC Order, with the provision for additional flow tests when all three units are operational.

Item (b)

By letter from W. Lee (DPC) to H. Denton (NRC) dated April 26, 1979, the licensee committed to develop and implement operating procedures for initiating and controlling EFW independent of ICS control. With the installation of the modified EFW system, discussed in item (a), the licensee has bypassed the previous piping and valve alignments that were controlled by the ICS. As a result, the present EFW system is totally separate from the ICS. The order requires the licensee to:

"Develop and implement operating procedures for initiating and controlling EFW independent of Integrated Control System control."

The licensee has revised his emergency procedures related to the use of the EFW system to reflect the separation of the EFW system from the ICS. The key operator actions resulting from the system modification are to verify that all EFW pumps are actuated either automatically or manually and to maintain the steam generator water level at prescribed values which depend on whether the reactor coolant pumps are operating. These procedures would be implemented by the operator whenever there is a loss of all main feedwater caused by feedwater system problems or loss of offsite power. For all other events, the steam generator water level would be controlled by the ICS using the startup valves and the main feedwater pumps.

We have reviewed the revised procedures for the EFW system to assure that there is sufficient guidance to the operator to activate the system if the automatic initiation fails and to manually control the steam generator water level to specified values.

The review of the procedures included consideration of verifying readings of certain key parameters by using alternate instrumentation and specification of parameter values that must be controlled by the operator. Our comments on the procedures were incorporated by the licensee and verified by the NRC resident inspector. The licensee has committed to provide double verification of the restoration of equipment following surveillance tests or maintenance on the EFW system.

NRC staff at the Oconee site walked through the EFW procedures with Oconee operators to evaluate whether the procedures were functionally adequate. In addition, the NRC staff audited a sample of Oconee operators to determine if they were familiar with the revised procedures and would implement them correctly. Based on the NRC staff audit, we conclude that the revised procedures and operator training are satisfactory.

The procedures reviewed addressed the following emergency conditions:

1. Loss of Main Feedwater Pumps
2. Loss of Main Feedwater Pumps and Emergency Feedwater
3. Loss of Station Power and/or Loss of Instrument Air
4. Loss of Reactor Coolant Flow both with and without Station power and instrument air
5. Steam line break inside the reactor building both with and without Station power and instrument air.
6. Steam line break outside the reactor building both with and without Station power and instrument air.

Based on our review and verification, we find that the licensee has complied with the requirements of item (b) of paragraph (1) of the Order.

<u>Item (c)</u>

The original Oconee design did not have any direct reactor trips that would be initiated by a malfunction in the secondary system (feedwater and steam). To obtain an earlier reactor trip (rather than delaying the trip until an operator took action or when a primary system parameter exceeded its trip setting) the licensee committed to install a hard-wired control-grade reactor trip on loss of all main feedwater and/or turbine trip. (Letter from W. Lee (DPC) to H. Denton (NRC) dated April 26, 1979.) The Order requires that:

> "Implement a hard-wired control-grade reactor trip on loss of main feedwater and/or turbine trip."

The purpose of this anticipatory trip is to minimize the potential for opening the Pilot Operated Relief Valves (PORVs) and/or the safety valves on the pressurizer. The licensee has estimated that the new anticipatory trip will result in a reactor trip 3 to 10 seconds earlier for loss of all main feedwater and turbine trip events, and the staff generally agrees.

The licensee has added new nonsafety-grade circuitry to Units 1, 2, and 3 which is designed to provide an automatic reactor trip when either the main turbine trips or all the main feedwater is lost.

The main turbine trip is sensed by an existing normally deenergized auxiliary relay in the main turbine Electro-Hydraulic control system. The relay is energized by a Class IE dc power supply upon any turbine trip signal on the main turbine trip bus. The relay provides two contact closures to energize two dc shunt coils (one in each of the two reactor trip ac circuit breakers) to open each of the breakers and trip the reactor. The shunt coil power supply is also from the Class IE dc source. Both ac circuit breakers must be opened to cause reactor trip. Provisions have been included to bypass the turbine trip signal to the breaker (via a main control room switch) for startup and/or low power levels.

The total main feedwater loss is sensed by a normally energized auxiliary relay in the EFW actuation circuitry. The relay is deenergized by either a low pressure signal in the main feedwater discharge header or indication that both main feedwater pumps are tripped (indication is provided by feedwater pump turbine steam stop valve limit switches.) The relay provides a contact closure to energize each of the same two dc shunt coils energized in the event of a turbine trip to open the ac circuit breaker and trip the reactor. A loss of the 115 VAC Class IE power to this relay will cause a reactor trip and initiate EFW.

The turbine trip and main feedwater trip circuits are wired in parallel to each coil such that either signal will cause the ac circuit breakers to open.

The licensee has analyzed this additional circuitry with respect to its independence from the existing reactor trip system. They have stated that the shunt coil is part of the existing ac reactor trip breaker. This shunt coil is powered by a Class IE 125 VDC supply. It is not classified as part of the existing reactor trip system. However, it is separate and operates independently from the 120 VAC undervoltage trip coil of the same ac breaker. The reactor trip safety-grade signal deenergizes the 120 VAC undervoltage coil to produce a trip of the same ac breaker. The new cabling associated with the added circuitry is located in the cable room/control room area and is routed with the non-Class IE cabling throughout.

The licensee has committed to perform a monthly test on the added circuitry in order to demonstrate its ability to open the ac circuit breakers.

Based on our review of the implementation of the trip circuitry with respect to its independence from the existing trip circuitry, we conclude that this addition will not degrade the existing reactor protection system design.

Based on the licensee's design and commitment to perform a monthly test on the new circuitry, we conclude that there is reasonable assurance that the system will perform its function. The resident IE inspector has confirmed that the checkout tests for this circuitry were completed successfully.

On the basis of the above, we conclude that the new trip complies with the requirements of item (c) of paragraph (1) of the Order.

Item (d)

This item in the Order requires the licensee to:

> "Complete analyses for potential small breaks and implement operating instructions to define operator action."

By letter from W. Lee (DPC) to H. Denton (NRC) dated April 26, 1979, the licensee committed to providing the analyses and operating procedures of this requirement.

Babcock & Wilcox, the reactor vendor for the Oconee plants, submitted an analysis entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" and supplements to this analysis (References 1 through 5). The major parameters used in this generic study conservatively bound the Oconee plants. EG&G on NRC's request also performed two reactor coolant system small break calculations for the Oconee plant. The EG&G calculations were consistent with B&W results. The staff evaluation of B&W generic study has been completed and the results of the evaluation will be issued as a NUREG report prior to June 1, 1979.

A principal finding of our generic review is a reconfirmation that Loss of Coolant Accident (LOCA) analyses of breaks at the lower end of the small break spectrum (smaller than 0.04 sq. ft.) demonstrate that a combination of heat removal by the steam generators, high pressure injection system and operator action ensure adequate core cooling. The auxiliary feedwater system used to remove heat through the steam generators has been modified to enhance its reliability as discussed in item (a). The high pressure injection system is capable of providing emergency core cooling even at the safety valve pressure set point. Reactor core uncovery is not predicted for these events. The calculated peak cladding temperature was less than 800°F, well below the 10 CFR 50.46 requirement of 2200°F. The ability to remove heat via the steam generators has always been recognized to be an important consideration when analyzing small breaks. Sensitivity analyses were performed with acceptable results assuming permanent loss of all feedwater (with operator initiation of the high pressure injection system at 20 minutes) and loss of feedwater for only the first 20 minutes of the accident. These are acceptable results considering the ability to locally start the EFW pumps in five minutes as discussed under item (a) of this evaluation, assuming failure of automatic EFW actuation.

Another aspect of the studies was the assessment of recent design changes on the lift frequency of pressurizer safety and relief valves. The design changes included change in the setpoint of the pressurizer relief valve from 2255 psi to 2450 psi, change in the high pressure reactor trip setpoint from 2355 psi to 2300 psi and the installation of anticipatory reactor trips on turbine trip and on loss of feedwater. In the past, during turbine trip and loss of feedwater transients the pressurizer relief valves were lifted. With the new design these transients do not result in lifting of the relief valve. However, lifting of both relief and safety valves might occur in case of rod withdrawal and boron dilution transients, using the normally conservative assumptions found in the Chapter 15 safety analyses. The above design changes did not effect the lift frequency of the safety valves for these events.

Based on our review of the small break analyses presented by B&W, the staff has determined that a loss of all main feedwater with (a) an isolated PORV, or (b) a stuck open PORV consequentially does not result in core uncovery, provided either EFW or 2 HPI pumps are initiated within 20 minutes. Based on the acceptable consequences calculated for small break LOCAs and loss of all main feedwater events and the expected reliability of the EFW and high pressure injection systems, we conclude that the licensee has complied with the analysis portion of paragraph (1)(d) of the Order.

To support longer term operation of these facilities, requirements will be developed for additional and more detailed analyses of loss of feedwater and other anticipated operational transients. More detailed analysis of small break LOCA events are also needed for this purpose. Accordingly, the licensee will be required to provide the analyses discussed in Sections 8.4.1 and 8.4.2 of the recent NRC Staff Report of the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company (NUREG 0560). Further details on these analyses and their applicability to other PWRs and BWRs will be specified by the staff in the near future. In addition, to assist the staff in developing more detailed guidance on design requirements of relief and safety valve reliability during anticipated transients, as discussed in Section 8.4.6 of the NUREG report, the licensee will be required to provide analyses of the mechanical reliability of the pressurizer relief and safety valves of the Oconee facilities.

The B&W analyses show that some operator action, both immediate and followup, is required under certain circumstances for a small break accident. Immediate operator action is defined as those actions, committed to memory by the operators, which are necessary to take as soon as the problem is diagnosed. In order to carry out followup actions the operators must consult and follow instructions in written and approved procedures. These procedures must always be readily available in the control room for the operators' use. Guidelines were developed by B&W in order to assist the utility staff of the operating B&W facilities to develop emergency procedures for the small break accident.

The "Operating Guidelines" for Small Breaks were issued by B&W on May 5, 1979, and reviewed by the NRC staff. Revisions recommended by the staff were incorporated in the guidelines. In response to these guidelines, the DPC staff at the Oconee Nuclear Station made substantial revisions to EP/0/A/1800/4, Loss of Reactor Coolant. This emergency procedure defines the required operator action in response to a spectrum of break sizes for a LOCA in conjunction with various equipment availability and failures. The procedure is divided into eight sections beginning with excessive reactor coolant system leakage without a reactor trip and concluding with a rupture in excess of the capability of three high pressure injection pumps. The latter case is the larger break accident in which the system depressurizes to the point of low pressure injection.

Six cases of small break accidents are considered in the procedure. The first one assumes that feedwater to the steam generators and the reactor coolant pumps is available but the reactor is not automatically tripped. The second case increases the break size to cause an automatic trip of the reactor. In both cases, the required operator actions are generally the same and a safe, cold shutdown of the plant is accomplished with normal cooldown procedures.

The other four small break procedures provide guidance to the operators for dealing with degraded conditions such as loss of feedwater and/or loss of reactor coolant pumps. If feedwater is lost, a heat removal path is established from the high pressure injection system through the break and pressurizer PORV or the safety valves. Once feedwater is reestablished, the steam generators can be used as a heat sink.

If the reactor coolant pumps are not available, the operators are directed to establish and verify natural circulation. Additional guidance is provided if natural circulation is not immediately achieved.

For all cases in which HPI is manually or automatically initiated, the operators are specifically instructed to maintain maximum high pressure injection flow unless two criteria are met. These criteria are:

1. Both LPI pumps are in operation and flowing at a rate in excess of 1000 gpm and the situation has been stable for 20 minutes, or

2. All hot and cold leg temperatures are at least 50 degrees below the saturation temperatures for the existing reactor coolant system pressure. If the 50 degrees subcooling cannot be maintained after high pressure injection cutoff, the HPI shall be reactuated.

The requirement to determine and maintain 50°F subcooling has been incorporated in all other procedures in which HPI has been manually or automatically initiated. These procedures include: Steam Supply System Rupture, Steam Generator Tube Rupture, Loss of Reactor Coolant Flow and Loss of Steam Generator Feedwater. Each of these procedures, in addition to the Loss of Reactor Coolant procedure, provide additional instructions to the operators in the event of faulty or misleading indications. A subsequent action statement directs the operators to check alternate instrumentation channels to confirm the key parameter readings.

The Loss of Reactor Coolant procedure was reviewed by the NRC staff to determine its conformance with the B&W guidelines. Comments generated as a result of this review were incorporated in a further revision to the procedure. Two members of the NRC staff "walked through" the latest revision to this emergency procedure in the Oconee control rooms. The procedure was judged to provide adequate guidance to the operators to cope with a small break LOCA. The instrumentation necessary to diagnose the break, the indications and controls required by the action statements, and the administrative controls which prevent unacceptable limits from being exceeded are readily available to the operators. We conclude that the operators should be able to use this procedure to bring the plant to a safe shutdown condition in the event of a small break accident.

An audit of fourteen licensed operators and senior operators was conducted by the NRC staff to determine their understanding of the small break accident including how the operators are required to diagnose and respond to it. The Oconee staff has conducted special training sessions for the operators on the concept of and use of the emergency procedures, EP 1800/4. We found the operators had sufficient knowledge of the small break phenomenon and the general requirements of the emergency procedure. Each licensed individual has since received additional training on the approved procedure prior to assuming his shift duties.

The audit of the operators also included questioning about the TMI-2 incident and the resulting design changes made at Oconee. Our discussions with them covered the initiating events of the incident, the response of the plant to the simultaneous loss of feedwater and small break LOCA (PORV stuck open), and the operational errors that were apparently made during the course of the incident. We found their level of understanding sufficient to be able to respond to a similar situation if it happened at Oconee. We can also conclude they have adequate knowledge of thermodynamic processes of subcooling and saturated conditions and are able to recognize each in the primary coolant system.

Based on the foregoing evaluation, we conclude that the licensee has complied with the requirements of item (d) of paragraph (1) of the Order.

Item (e)

By letter from W. Parker (DPC) to J. O'Reilly (NRC) dated May 4, 1979, the licensee has confirmed that all reactor operators and senior reactor operators assigned to the Oconee control rooms have completed the TMI-2 training at B&W as required by the Order. This training consisted of a class discussion of the TMI-2 event and a demonstration of the event on the simulator as it occurred and how it should have been controlled. The class discussion was about one hour long and the remainder of the four to six hour session was conducted on the simulator. The TMI-2 event, including operational errors, was demonstrated to each operator. The event was again initiated and the operators were given "hands-on" experience in successfully regaining control of the plant by several methods. Other transients which resulted in depressurization and saturation conditions were presented to the operators in which they maneuvered the plant to a stable, subcooled condition.

The licensed operators and senior operators have received in excess of 23 hours of training concerning the TMI-2 accident and followup actions. Nearly 30 percent of the licensed individuals on shift duty at the 3 units were interviewed by the NRC. The results were judged to be satisfactory with some generic deficiencies noted to their management. In order to correct these deficiencies, Duke Power Company has committed (letter from W. O. Parker (DPC) to H. Denton (NRC) dated May 16, 1979) to a written examination by their training services group. An individual must receive a grade of 90% before he will be utilized at the control board of an operating unit. For long term verification of the effectiveness of the training, Duke Power Company has contracted with B&W and General Physics Corporation to independently perform audits on the operators. The NRC staff will review all results and recommendations as part of our normal inspection function of their requalification program. We conclude that there is adequate assurance that the operators at Oconee have and will continue to receive a high level of training concerning the TMI-2 accident and the consequent impact at their station.

Based on our interviews with a sample of the licensed operators and Duke Power Company's commitment to examine all of their operators, we conclude that the licensee has complied with the requirements of item (e) paragraph (1) of the Order.

CONCLUSION

We conclude that the actions described above fulfill the requirements of our Order of May 7, 1979 in regard to Paragraph (1) of Section IV. The licensee having met the requirements of Paragraph (1) need not shut down Unit 1 as described in Paragraph (4) and may restart Unit 2 and Unit 3 as provided by Paragraphs (2) and (3). Paragraph (2) of Section IV of the Order is related specifically to Oconee Unit No. 3, which is currently shutdown for a reload. Unit 3 is undergoing a reload review and cannot restart until NRC issues a license amendment related to the reload review in addition to meeting the requirements of Paragraph (2) of Section IV of the Order. Paragraph (5) of Section IV of the Order remains in force until the long term modifications set forth in Section II of the Order are completed and approved by the NRC.

Dated: May 18, 1979

REFERENCES

1. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting report entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," dated May 1979.

2. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting revised Appendix 1, "Natural Circulation in B&W Operating Plants (Revision 1)," dated May 8, 1979.

3. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting additional information regarding Appendix 2, "Steam Generator Tube Thermal Stress Evaluation," to report identified in Item 2 above, dated May 10, 1979.

4. Letter from J. H. Taylor (B&W) to T. M. Novak, providing background information on reactor coolant pump operation, dated May 10, 1979.

5. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing an analysis for "Small Break in the Pressurizer (PROV) with no Auxiliary Feedwater and Single Failure of the ECCS" identified as Supplements 1 and 2 to Section 6.0 of report in Item 2, dated May 12, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DUKE POWER COMPANY

DOCKETS NOS. 50-269, 50-270, AND 50-287

NOTICE OF AUTHORIZATION TO RESUME OPERATION

The United States Nuclear Regulatory Commission issued an Order on May 7, 1979 (44 F.R. 27776, May 11, 1979), to Duke Power Company (the licensee), holder of Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for Oconee Nuclear Station, Units Nos. 1, 2 and 3, confirming that the licensee accomplish a series of actions, both immediate and long term, to increase the capability and reliability of Oconee Units Nos. 1, 2 and 3 to respond to various transient events. In addition, the Order confirmed that the licensee would shut down Oconee 3 on April 28, 1979, an additional Oconee unit on May 12, 1979, and remaining unit on May 19, 1979, unless, with respect to the latter two units, the following actions had been accomplished prior to the prescribed dates:

(a) Install automatic starting of the interconnected emergency feedwater system so that all three pumps will receive a start signal from any affected unit, and test the system for stability. The emergency feedwater pump discharge flow will be connected to the interconnection headers such that each or all of the emergency feedwater pumps can supply water to any unit. Until these modifications and tests are completed, operating personnel will be stationed at each emergency feedwater pump with a direct communication link to that unit's control room. In addition, the following procedural changes, put into effect on April 25, 1979 to enhance the reliability of the emergency feedwater system, will remain in force:

(1) The discharges of these pumps have been tied together by alignment of manual valves such that each and all of the pumps can supply emergency feedwater to any Oconee Unit requiring it.

Copies of (1) the licensee's letter dated May 7, 1979, and two letters dated May 9, 1979, (2) the Director's letter dated May 18, 1979 and (3) the Safety Evaluation dated May 18, 1979 are available for inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555, and are being placed in the Commission's local public document room at the Oconee County Library, 201 South Spring, Walhalla, South Carolina 29691. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

FOR THE NUCLEAR REACTOR REGULATION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Dated at Bethesda, Maryland
this 18th day of May 1979.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket Nos.: 50-269, 50-270
and 50-287

OCT 16 1979

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Duke Power Company
P.O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

SUBJECT: EMERGENCY FEEDWATER FLOW RATE AND STABILITY TEST FOR OCONEE 1, 2, & 3

In your letter dated August 22, 1979, you requested exemption from the flow rate and flow stability test that we required in our May 18, 1979 evaluation of your compliance with the NRC Order of May 7, 1979.

We have reviewed your request and conclude that the flow test we required in our May 18, 1979 evaluation will not be necessary provided all motor-operated pumps are available prior to three unit operation. The enclosed evaluation describes the details of our review and provides the basis for our conclusion.

As stated on page two of the enclosed evaluation, the addition of the two motor-driven pumps to each unit requires that new analyses be performed regarding a main steam line break inside containment since the peak containment pressure may be affected due to the emergency feedwater flow which is dependent on manual actions to isolate flow to the affected steam generator. In performing the analyses, you must consider the run out flow from the turbine-driven pump and one motor-driven pump. Please provide us a date by which we can expect to receive the revised analyses.

If you have any additional questions, please do not hesitate to call me.

Sincerely,

Original signed by:

Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors

Enclosure:
Supplement 1 to Evaluation of
Licensee's Compliance with the
NRC Order dated May 7, 1979

cc: See attached distribution list

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ENCLOSURE

SUPPLEMENT 1 TO "EVALUATION OF LICENSEE'S COMPLIANCE WITH THE NRC ORDER DATED MAY 7, 1979 - DUKE POWER COMPANY - OCONEE NUCLEAR STATION, UNITS 1, 2, & 3 - DOCKET NOS. 50-269, 270, AND 287" DATED MAY 18, 1979

DISCUSSION

Our staff evaluation of Duke Power Company's compliance with the Commission Order of May 7, 1979, required that the licensee demonstrate acceptable flow rates and flow stability with only two operating steam-driven emergency feedwater (EFW) pumps when all three units were operational. This test was required since the EFW system for Oconee, existing at that time, consisted of only three turbine-driven pumps which would normally be cross-connected to feed all three units. The failure of one pump would necessitate that two turbine-driven pumps be capable of supplying adequate EFW to all three units. To ensure adequate flow to all three units, we required a test to show that two pumps could meet the EFW needs for all three units.

However, since this requirement was transmitted on May 18, 1979, there has been no instance in which all three units were operating. Duke Power Company is presently modifying the EFW design at Oconee such that each unit will have a turbine-driven pump and two motor-driven pumps. Each three-pump system will be dedicated to one reactor unit and will operate with the systems not cross-connected. The modifications are complete for Unit 3 and essentially complete for Unit 1. The modifications to Unit 2 are partially complete and should be finished during the present scheduled shutdown. Duke Power Company states that, with the completion of installation of the motor-driven pumps for each unit, the operation of all three units with the cross-connected steam-driven EFW pumps will no longer be an established mode of operation, thereby making the test described above unnecessary.

The modified EFW system will include two motor-driven pumps (500 gpm) and one turbine-driven pump (1080 gpm) for each unit. All three pumps will automatically start on loss of main feedwater pumps or low feedwater pressure. A single failure of any pump could reduce the available feedwater supply to a minimum of 1000 gpm (assuming failure of a turbine-driven pump) to one unit when all three units are operating. A single failure of one pump for the original three turbine-driven pump design would reduce the available feedwater to each unit to 720 gpm, assuming no flow instabilities. Since the two motor-driven pumps will provide additional feedwater capability assuming a single failure, the emergency feedwater system will no longer be required to be cross-connected between units. Cross-connected EFW system operation was the basis for the flow testing requirements with two turbine-driven EFW pumps supplying three units.

The addition of the two motor-driven pumps to each unit requires that new analyses be performed regarding a main steam line break inside containment since the peak containment pressure may be affected due to the emergency feedwater flow which is dependent on manual actions to isolate flow to the affected steam generator. The licensee must perform an analysis which considers the run out flow from the turbine-driven and one motor-driven EFW pumps.

CONCLUSION

Based on our review of the proposed modifications as described above, and with the resolution of the main steam line break concern, we conclude that the addition of the motor-driven pumps will be an improvement relative to the existing "turbine only" configuration and that the flow testing requirements for two turbine-driven pumps supplying three units are not necessary. Simultaneous three unit operation will not occur prior to the installation and testing of the motor-driven pumps. These tests will be witnessed by the IE site inspector.

Dated this 10th day of October 1979
Bethesda, Maryland

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 27, 1979

Docket No. 50-312

Mr. J. J. Mattimoe
Assistant General Manager and
Chief Engineer
Sacramento Municipal Utlity District
6201 S Street
P. O. Box 15830
Sacramento, California 95813

Dear Mr. Mattimoe:

By Order of May 7, 1979, the Commission confirmed your undertaking a series of actions, both immediate and long term, to increase the capability and reliability of the Rancho Seco Nuclear Generating Station to respond to various transient events. In addition, the Order confirmed that you would shut down Rancho Seco on April 28, 1979, and maintain the plant in a shutdown condition until the following actions had been satisfactorily completed:

(a) Upgrade the timeliness and reliability of delivery from the Auxiliary Feedwater System by carrying out actions as identified in Enclosure 1 of your letter of April 27, 1979.

(b) Develop and implement operating procedures for initiating and controlling auxiliary feedwater independent of Integrated Control System control.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) Provide for one Senior Licensed Operator assigned to the control room who has had Three Mile Island Unit No. 2 (TMI-2) training on the B&W simulator.

By submittal of May 14, 1979, as supplemented by seven letters dated May 22, 24, 29, 30(3) and June 6, 1979, you have documented the actions taken in response to the May 7 Order. We have reviewed this submittal, and are satisfied that, with respect to Rancho Seco, you have satisfactorily completed the actions prescribed in items (a) through (e) of paragraph (1) of Section IV of the Order, the specified analyses are acceptable, and the specified implementing procedures are appropriate. The bases for these conclusions are set forth in the enclosed Safety Evaluation.

As noted on page 13 of the Safety Evaluation, you will be required to conduct a test during power operation to demonstrate operator capability to assume manual control of the Auxiliary Feedwater System independent of the Integrated Control System.

Appropriate Technical Specifications for Limiting Conditions for Operation and for surveillance requirements should be developed as soon as practicable and provided to the staff within seven days with regard to the design and procedural changes which have been completed in compliance with the provisions of the May 7, 1979 Commission Order. The revised Technical Specifications should cover:

(1) Addition of flow indication to the Auxiliary Feedwater System;

(2) Addition of the Anticipatory Reactor Trips; and

(3) Changes in set points for high pressure reactor trip and PORV actuation.

Within 30 days of receipt of this letter, you should provide us with your schedule for completion of the long term modifications described in Section II of the May 7 Order, and you should submit for staff review the model used in the analysis for potential small breaks referenced in your letter of May 14, 1979.

My finding of satisfactory compliance with the requirements of items (a) through (e) of paragraph (1) of Section IV of the Order will permit resumption of operation in accordance with the terms of the Commission's Order; it in no way affects your duty to continue in effect all of the above provisions of the Order pending your submission and approval by the Commission of the Technical Specification changes necessary for each of the required modifications.

Sincerely,

Roger J. Mattson
for Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Safety Evaluation
2. Notice

cc w/enclosures: See next page

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<u>EVALUATION OF LICENSEE'S COMPLIANCE</u>

<u>WITH THE NRC ORDER DATED MAY 7, 1979</u>

<u>SACRAMENTO MUNICIPAL UTILITY DISTRICT</u>

<u>RANCHO SECO NUCLEAR GENERATING STATION</u>

<u>DOCKET NO. 50-312</u>

June 27, 1979

INTRODUCTION

By Order dated May 7, 1979, (the Order) the Sacramento Municipal Utility District (SMUD or licensee) was directed by the NRC to take certain actions with respect to Rancho Seco Nuclear Generating Station. Prior to this Order and as a result of a preliminary review of the Three Mile Island Unit No. 2 (TMI-2) accident, the NRC staff initially identified several human errors that contributed significantly to the severity of the event. All holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's Office of Inspection and Enforcement (IE). Subsequently, an additional bulletin was issued by IE which instructed holders of operating licenses for B&W designed reactors to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer power-operated relief valve (PORV) setting.*

The NRC staff identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors. Those were identified as items (a) through (e) on page 1-7 of the "Office of Nuclear Reactor Regulation Status Report to the Commission" dated April 25, 1979. After a series of discussions between the NRC staff and the licensee concerning possible design modifications and changes in operating procedures, the licensee agreed, in a letter dated April 27, 1979, to perform promptly certain actions. The Commission found that operation of the plant

*[IE Bulletins Nos. 79-05 (April 1, 1979), 79-05A (April 5, 1979), and 79-05B (April 21, 1979) apply to all B&W facilities.]

should not be resumed until actions described in paragraphs (a) through (e) of paragraph (1) of Section IV of the Order were satisfactorily completed.

Our evaluation of the licensee's compliance with items (a) through (e) of paragraph (1) of Section IV of the Order is given below. In performing this evaluation we have utilized additional information provided by the licensee on May 14, 22, 24, 29, 30, and June 6, 1979, and numerous discussions with the licensee's staff. Confirmation of design and procedure changes was made by members of the NRC staff at the Rancho Seco site. An audit of the Rancho Seco reactor operators was also performed by the NRC staff to assure that the design and procedure changes were understood and were being correctly implemented by the operators.

EVALUATION

Item a

It was ordered that the licensee take the following action:

> "Upgrade the timeliness and reliability of delivery from the Auxiliary Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of April 27, 1979."

The Rancho Seco auxiliary feedwater (AFW) design has one turbine/motor tandem drive pump (P-318) that is automatically actuated and controlled independent of offsite power, and one motor-driven AFW pump (P-319) that is automatically started, but must be manually transferred to a vital AC bus if offsite power is lost. The turbine/motor driven pump will be manually started, according to procedure, from a vital AC bus if the turbine drive fails. By reference above to Enclosure (1) of the licensee's letter of April 27, 1979, it was ordered that the licensee:

> "1. Review procedures, revise as necessary and conduct training to ensure timely and proper starting of motor driven auxiliary feedwater (AFW) pump(s) from vital AC buses upon loss of offsite power."

The licensee has developed Section 7.5 of Operating Procedure A.51 ("Auxiliary Feedwater System") to provide specific direction for the operator on the steps required to load motor driven pump P-319 on nuclear service bus 4A and to secure the steam to the turbine on the dual-drive pump P-318, in the event of inoperability of the steam drive, and load the motor drive on nuclear service bus 4B. Bypass keys are required to complete the connection of the auxiliary feedwater pump motors to the diesel powered buses (nuclear service buses 4A and 4B); these keys are available in the office adjacent to the control room. Emergency Procedure D.1 ("Load Rejection") directs the operator to use Operating Procedure A.51 if main feed pump operation cannot be maintained. The NRC staff verified that the operators are knowledgeable in the procedure for loading the AFW pumps on the vital AC buses. The NRC staff concludes that the licensee has adequate procedures and the operators are trained to start the AFW system from diesel powered buses upon loss of offsite power or load rejection and therefore, is in compliance with this part of the Order.

It was also ordered that:

"2. To assure that AFW will be aligned in a timely manner to inject on all AFW demand events when in the surveillance test mode, procedures will be implemented and training conducted to provide an operator at the necessary valves in phone communications with the control room during the surveillance mode to carry out the valve alignment changes upon AFW demand events."

Surveillance Procedures SP 210.01A and SP 210.01B are used for the quarterly surveillance and inservice testing of auxiliary feed pumps P-318 and P-319, respectively. These procedures have been revised to include the following statement; "Station an operator at FWS-055, auxiliary feedwater system full flow recirculation valve in continuous communication with the control room until FWS-055 is secured closed at the completion of this test." In addition to the above procedure revisions, the licensee has added FWS-492 (bypass valve for FWS-055) to the "Locked Valve List" (SP 214.03). The licensee has also incorporated independent verification of valve lineups following surveillance testing and/or maintenance of the AFW system.

The NRC staff has reviewed SP 210.01A and SP 210.01B to verify that the procedures contain specific directions to return each valve that was operated during the conduct of the surveillance test to its proper position. The local operator has to close a valve (FWS-Ø55) when so instructed by the control room operator or if he loses communication with the control room. The NRC staff has verified that the operators are familiar with this test procedure. We conclude that the licensee has adequate procedures to assure that AFW will be aligned in a timely manner to inject on all AFW demand events when in the surveillance test mode and therefore, is in compliance with this part of the Order.

It was ordered that:

> "3. Procedures will be developed and implemented and training conducted to provide for control of steam generator level by use of safety grade AFW bypass valves in the event that ICS steam generator level control fails."

The licensee has developed Emergency Procedure D.14 ("Loss of Steam Generator Feed") that describes the symptoms that would result from a loss of main feedwater control that may have been caused by an integrated control system (ICS) failure. The procedure has been reviewed by the NRC staff. The operator is directed to restore feedwater to the steam generators by one of three methods. The preferred method is described in Section 7.7 of Operating Procedure A.51 ("Auxiliary Feedwater System"). Section 7.7 directs the operator to: close the ICS controlled AFW control valves; start the AFW pumps; and maintain the steam generator levels, specified in the procedure, by manually operating the motor driven AFW bypass valves from the control room. In this mode the pumps and valves will operate independent of the ICS. The operator is provided with AFW flow rate and steam generator level indications in the control room for each steam generator.

Since the AFW bypass valves will fully open on a safety features actuation signal (SFAS)*, the operator is provided with instructions on how to take manual control of the valves after a SFAS. NRC staff has conducted an audit of the operator training and verified that the operators have been trained to carry out those procedures.

The NRC staff concludes that the licensee has developed adequate procedures and operator training to control AFW flow to the steam generators to specified values independent of the ICS, should a failure of the ICS occur, and therefore, is in compliance with this part of the Order.

It was also ordered:

> "4. Verification that Technical Specification requirements of AFW capacity are in accordance with the accident analysis will be conducted. Pump capacity with mini flow in service will also be verified."

The licensee has conducted the verification that Technical Specification requirements of AFW capacity are in accordance with the accident analysis for the Rancho Seco Nuclear Station. The Technical Specification states, as a

*[The safety features actuation system (SFAS) monitors variables to detect loss of reactor coolant system boundary integrity. Upon detection of "out-of-limit" conditions of these variables, it initiates emergency core cooling (ECC) which consists of high pressure injection (HPI) and low pressure injection (LPI), Reactor Building cooling and isolation, and Reactor Building spray systems. Additionally, it starts diesel generators GEA and GEB, which are in standby redundance with the nuclear service buses 4A and 4B.]

limiting condition for operation, capability to supply feedwater at a process flow rate corresponding to a decay heat level of 4.5 percent of full reactor power from at least one of the following means:

(a) a condensate pump and a main feed pump, or

(b) a condensate pump, or

(c) an auxiliary feedwater pump.

A letter from Babcock & Wilcox to the licensee, dated May 16, 1979, states that it has performed an analysis of the required AFW flow rate for the Rancho Seco Plant which shows that a decay heat level of 4.5 percent of rated power, plus the heat input from the RCPs, will require a total flow rate to either or both steam generators of approximately 760 gpm.

Each of the two AFW pumps are sized to deliver 780 gpm to steam generators with 60 gpm mini flow in service. This pump capacity exceeds the minimum required AFW flow rate in the Rancho Seco safety analysis and Technical Specifications. AFW pump capacity, with mini flow in service, has been verified by performing the quarterly "AFW System Surveillance Test" and the "Auxiliary Feedwater Flow Indicator Functional Test" (STP 612). The results of these tests demonstrated that each of the two AFW pumps has the capability to deliver a minimum of 780 gpm into the steam generators, with mini flow in service. The licensee will reconfirm the minimum AFW flow rate to the steam generators in a test immediately following startup.

Based on our review of the AFW flow rate test results, performed to date, we conclude that the licensee is in compliance with this part of the Order.

It was also ordered that:

> "5. Modifications will be made to provide verification in the control room of AFW flow to each steam generator."

To verify that AFW is being pumped to the steam generators, the licensee has installed Clampitron Flowmeters on both of the AFW injection flow paths, downstream of the AFW control valves, so that the actual flow rate to each steam generator will be measured. The Clampitron Flowmeters consists of transducers, attached to the AFW piping, connected to a flow display computer. On command from the flow display computer, the transducer transmits an ultrasonic beam through the water inside the pipe and the velocity of the beam, as affected by AFW flow, is analyzed by the flow display computer, which calculates the AFW flow rate in gpm. The AFW flow rate is displayed in the control room. A calibration test (STP-612) was conducted by the licensee to functionally test the performance of the flowmeters. Performance of this test demonstrated that the indicated flow rate agreed with the calculated flow rate within the ±20% acceptance criteria specified in the procedure.

Based on our review of this design modification and test results, we conclude that the licensee is in compliance with this part of the Order.

It was also ordered that the licensee:

> "6. Review and revise, as necessary, the procedures and training for providing alternate sources of water to the suction of the AFW pumps."

Control room alarms are available to alert the operator to perform the manual transfer of the AFW supply source from the condensate storage tank (CST) to the plant reservoir. The CST is designed to seismic Category I criteria. The licensee has reviewed and revised his Emergency Procedures D.10 ("Loss of Reactor Coolant Flow/RCP Trip"), D.14 ("Loss of Steam Generator Feed"), and Operating Procedure A.51 ("Auxiliary Feedwater System") to provide guidance for the operator to obtain an alternate source of water for the suction of the AFW pumps. The revised procedures require the operator to break condenser vacuum when the level reaches a level alarm point of approximately 29 feet and to shift the AFW pump suction to the plant reservoir when the CST level is down to a second alarm point of approximately 3 feet from the bottom of the tank. The capacity of the CST is large enough to provide cooling for about 24 hours before this transfer is required. The shifting to an alternate source of AFW pump suction is accomplished by manually operating four isolation valves at a local valve station. The operator has about 40 minutes to effect the transfer. The NRC staff has reviewed the revised Emergency Procedures D.10 and D.14 and Operating Procedure A.51 and concludes that these procedures provide sufficient guidance to the operator for a timely shifting to an alternate water source for the AFW pumps, before the CST is emptied.

The NRC staff has verified that the control room operators are properly trained to carry out these procedures. We conclude that the licensee has complied with the requirements of this part of the Order.

It was also ordered that:

> "7. Design review and modification, as necessary, will be conducted to provide control room annunciation for all auto start conditions of the AFW system."

The licensee has provided indication for all auto start conditions of the AFW system on an annunciation panel inside the control room. The conditions which will actuate the annunciator are:

(a) loss of all reactor coolant pumps, or
(b) low discharge pressure (850 psig) on both main feedwater pumps, or
(c) manual start of the motor driven AFW pump.

A safety features actuation signal, which will also automatically start AFW, had already been annunciated in the control room before the current modifications. Based on our review of this design modification, we conclude that the licensee is in compliance with this part of the Order.

It was ordered that:

"8. Procedures will be developed and implemented and training conducted to provide guidance for timely operator verification of any automatic initiation of AFW."

The conditions that will automatically initiate auxiliary feedwater are adequately described in Operating Procedure A.51, ("Auxiliary Feedwater System"). The operators are directed, as an immediate action, to verify that the AFW flow has automatically started on loss of both main feedwater pumps in Emergency Procedure D.14 ("Loss of Steam Generator Feed") and on loss of all reactor coolant pumps in Emergency Procedure D.10 ("Loss of Reactor Coolant Flow/RCP Trip"). Both procedures require the following immediate actions by the operator: verify that the auxiliary feedwater pumps have automatically started; that there is flow to the steam generators; and that the proper steam generator levels are being maintained. The NRC staff has performed an audit and verified that the operators are trained in these procedures.

Based on review of these procedures, we conclude that the licensee has provided guidance for timely operator verification of any automatic initiation of AFW and therefore, is in compliance with this part of the Order.

It was also ordered:

"9. Verification will be made that the air operated level control valves (a) Fail to the 50% open position upon loss of electrical power to the electrical to pressure converter, and (b) Fail to the 100% open position upon loss of service air. The AFW bypass valves are safety grade."

The licensee has completed its verification test for the failure mode of the air operated level control valves. The test results show that both air operated level control valves fail to the 100% open position on loss of air pressure at the valve operators. On tests for loss of control signal to the electric to pressure converters, one level control valve failed to the 50% open position and the other one failed to the 60% open position, which are acceptable. The AFW bypass valves are safety grade, motor-operated valves which are operated independently from the ICS as discussed in Part 3 above. Based on our review of the test results on the air operated level control valves and the safety grade design of the bypass valves, we conclude that the licensee is in compliance with this part of the Order.

Based upon our evaluation, we conclude that the licensee has upgraded the timeliness and reliability of delivery from the AFW system by carrying out the actions identified in Enclosure 1 of the licensee's letter of April 27, 1979, and therefore, is in compliance with Item (a) of the Order.

Item (b)

It was ordered that the licensee:

"Develop and implement operating procedures for initiating and controlling auxiliary feedwater independent of Integrated Control System (ICS) control."

We have reviewed the revised procedures for the AFW system to assure that there is sufficient guidance for the operator to actuate the system if the automatic initiation failed, and to control steam generator levels at the required values. The review of the procedures focused on verifying that the operator is directed to observe the proper instruments and that the operator is directed to maintain specific values of parameters by manual control, such as steam generator levels. The review also determined that the operator should confirm the validity of the instrument readings of certain key parameters, such as steam generator levels. The necessary modifications to the procedures to satisfy these requirements were presented to the licensee, and the NRC staff has verified that the modifications have been incorporated in the procedures. (See further discussion of these procedures in part 3 of Item (a.)

The licensee will conduct a startup test at low power (<15%) to demonstrate the capability to provide and control flow to the steam generators using the AFW bypass valves.

During the visit to the site, the NRC staff walked through the AFW procedures with the operators to evaluate whether the procedures were functionally adequate. In addition, the NRC staff audited a sample of Rancho Seco operators to determine if they were familiar with the revised procedures and could implement them correctly. Based on the NRC staff audit, we conclude that the revised procedures and operator training are satisfactory and therefore, the licensee is in compliance with Item (b) of the Order.

Item (c)

The original Rancho Seco design did not have any direct reactor trips that would be initiated by a malfunction in the secondary system. To obtain an anticipatory reactor trip (rather than delaying the trip until a primary system parameter exceeded its trip setting) the licensee committed to install a hard-wired, control-grade, reactor trip on loss of all main feedwater and/or turbine trip. The Order requires that the licensee:

> "Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip."

The licensee has added control-grade circuitry to Rancho Seco, which is designed to provide an automatic reactor trip when either the main turbine trips or all main feedwater is lost. The purpose of the anticipatory trip is to minimize the potential for opening of the power-operated relief valve (PORV) and/or the safety valves on the pressurizer. The licensee has indicated that this new circuitry meets this objective by providing a reactor trip during the incipient stage of the related transients (turbine trip and/or loss of main feedwater).

The main turbine trip is sensed by an existing, normally deenergized relay in the main turbine/generator protection system. The relay is energized by the protective trips of the turbine and/or generator. Power is supplied by an onsite battery source.

The loss of all main feedwater is sensed by two newly installed pressure switches (one in each of the two main feedwater pump discharge lines). The pressure switches actuate (close) on low pressure in the header. Power is supplied by the same onsite battery source. In order to prevent an inadvertent reactor trip during startup or shutdown, the loss of all main feedwater trip input is cut-out of the circuitry by a keylock switch. The key for this switch is maintained in the custody of the shift supervisor and is located in the control room. When the switch is placed in the "cut-out" position, it is annunciated on the main control board. The operating procedures specify when the switch is placed in the "normal" or "cut-out" position.

Either signal (turbine trip or loss of all main feedwater) will actuate a reactor trip relay, which in turn provides an input to both of the shunt coils of the AC reactor trip breakers. Energizing both of the shunt coils causes a reactor trip.

The licensee has analyzed this additional circuitry with respect to its independence from the existing reactor trip system. They have stated that the shunt coil is part of the existing AC reactor trip breaker. Each shunt coil is powered by a separate Class IE 125 VDC supply and operates independently from the 120 VAC undervoltage trip coil which receives the safety-grade reactor trip signal.

An NRC inspector has confirmed that the check-out tests for this circuitry have been completed successfully. In addition, the licensee has committed to perform a monthly periodic test on the added circuitry in order to demonstrate its ability to open the AC reactor trip breakers via the shunt coil.

Based on our review of the implementation of the trip circuitry, with respect to its independence from the existing reactor trip circuitry, we conclude that this addition will not degrade the existing reactor protection system design.

Based on the licensee's design modifications and commitment to perform a monthly test on the new circuitry, we conclude that there is reasonable assurance that the system will perform its function.

On the basis of the evaluation above, we conclude that the licensee has complied with the requirements of Item (c) of the Order.

Item (d)

This item in the Order requires the licensee to:

> "Complete analyses for potential small breaks and develop and implement operating instructions to define operator action."

In the licensee's letter of April 27, 1979, the licensee committed to providing the analyses and operating procedures of this requirement.

Babcock and Wilcox, the reactor vendor for the Rancho Seco plant, submitted analyses entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" and supplements to these analyses (References 1 through 6). The major parameters used in this generic study bound the Rancho Seco plant. The staff evaluation of the B&W generic study has been completed and the results of the evaluation will be issued as a NUREG report in June 1979.

A principal finding of our generic review is a reconfirmation that Loss-of-Coolant Accident (LOCA) analyses of breaks at the lower end of the small break spectrum (smaller than 0.04 sq. ft.) demonstrate that a combination of heat removal by the steam generators, the high pressure injection system and operator action ensure adequate core cooling. The AFW system used to remove heat through the steam generators has been modified to enhance its reliability as discussed in item (a). The high pressure injection system is capable of providing emergency core cooling even at the safety valve pressure setpoint. The ability to remove heat via the steam generators has always been recognized to be an important consideration when analyzing very small breaks. Separate sensitivity analyses were performed assuming permanent loss of all feedwater (with operator initiation of the high pressure injection system at 20 minutes) and loss of feedwater for only the first 20 minutes of the accident for breaks of 0.01 sq. ft. Reactor core uncovery is not predicted for these events. The calculated peak cladding temperature was less than 800°F, well below the 10 CFR 50.46 requirement of 2200°F. These results are applicable to Rancho Seco considering the ability to manually start the redundant AFW pumps from the control room, assuming failure of automatic AFW actuation.

Another aspect of the study was the assessment of recent design changes on the lift frequency of pressurizer safety and relief valves. The design changes included: a change in the setpoint of the pressurizer power-operated relief valve (PORV) from 2255 psi to 2450 psi; change in the high pressure reactor trip setpoint from 2355 psi to 2300 psi; and the installation of an anticipatory reactor trip on turbine trip and/or on loss of all main feedwater. In the past, during the turbine trip or loss of feedwater transients, the PORV lifted. With the design changes the initial pressure increase of these transients do not result in lifting of this valve. However, the consequent depressurization could initiate safety injection which in turn could repressurize the system and lift the relief valve. It is expected that the operator would terminate HPI before the relief valve or safety valves lift, since the 50°F subcooling criteria would be satisfied at pressures below the PORV setpoint. Also, lifting of both the PORV and safety valves might occur in the case of control rod withdrawal or inadvertent boron dilution transients, using the normally conservative assumptions found in the Chapter 15 safety analyses. The above design changes do not effect the lift frequency of the valves for these Chapter 15 safety analyses.

Based on our review of the small break analyses presented by B&W, the staff has determined that a loss of all main feedwater with (a) an isolated PORV, but safety valves opening and closing as designed, or (b) a stuck open PORV does not result in core uncovery, provided either AFW or 2 HPI pumps is initiated within 20 minutes. Based on the acceptable consequences calculated for small break LOCAs and loss of all main feedwater events coupled with the expected reliability of the AFW and HPI systems, we conclude that the licensee has complied with the analyses portion of Item (d) of the Order.

To support longer term operation of the facility, requirements will be developed for additional and more detailed analyses of loss of feedwater and other anticipated transients. More detailed analyses of small break LOCA events are also needed for this purpose. Accordingly, the licensee will be required to provide the analyses discussed in Section 8.4.1 and 8.4.2 of the recent NRC "Staff Report of the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company" (NUREG 0560). Further details on these analyses and their applicability to other PWRs and BWRs will be specified by the staff in the near future. In addition, to assist the staff in developing more detailed guidance on design requirements of relief and safety valve reliability during anticipated transients, as discussed in Section 8.4.6 of the NUREG report, the licensee will be required to provide analyses of the lift frequency and mechanical reliability of the pressurizer relief and safety valves of the Rancho Seco facility.

The B&W analyses show that some operator action, both immediate and followup, is required under certain circumstances for a small break accident. Immediate operator action is defined as those actions committed to memory by the operators which must be carried out as soon as the problem is diagnosed. Follow-up actions require operators to consult and follow the steps in written and approved procedures. These procedures must always be readily available in the control room for the operators' use. Guidelines were developed by B&W to assist the operating B&W facilities in the development of emergency procedures for the small break accident.

The "Operating Guidelines for Small Breaks" were issued by B&W on May 5, 1979 and reviewed by the NRC staff. Revisions recommended by the staff were incorporated in the guidelines. In response to these guidelines, the staff at Rancho Seco made substantial revisions to Emergency Procedure D.5 ("Loss of Reactor Coolant/ Reactor Coolant System Pressure") and Operating Procedure B.4 ("Plant Shutdown and Cooldown"). These procedures define the required operator action in response to a spectrum of break sizes for a loss-of-coolant accident in conjunction with various equipment availability and failures.

Emergency Procedure D.5 (EP D.5) is divided into three sections. The first section deals with a small leak within the capability of a makeup pump. In this case, the operators proceed with an orderly plant shutdown unless pressurizer or makeup tank levels fall below prescribed limits. If these limits are exceeded the reactor is manually tripped and high pressure injection is initiated.

The second section of EP D.5 defines the required operator action for a small break not within the capability of a makeup pump. This section provides the operator with the guidance necessary to achieve a safe hot shutdown condition for a variety of degraded conditions. If all feedwater is lost, a heat removal path is established by the high pressure injection system through the break and the pressurizer power-operated relief valve or the safety valves. Once feedwater is reestablished, the steam generators can be used as a heat sink. If the reactor coolant pumps are not available, the operator is directed to Operating Procedure B.4 (OP B.4) which defines the actions necessary to cool down the plant by natural circulation. Additional guidance is provided in OP B.4 if natural circulation is not immediately achieved.

The third section of EP D.5 defines the actions necessary in the event of a large rupture. In this case the system depressurizes to the point of low pressure injection.

For all cases in which high pressure injection is manually or automatically initiated, the operators are specifically instructed in EP D.5 to maintain maximum HPI flow unless one of the following criteria are met:

(1) The LPI system is in operation and providing cooling at a rate in excess of 1000 gpm and the situation has been stable for 20 minutes, or

(2) All hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If the 50 degrees subcooling cannot be maintained after HPI cutoff, HPI shall be reactuated.

A requirement to determine and maintain 50°F subcooling has been incorporated in all other procedures in which HPI has been manually or automatically initiated. These procedures include, "Steam Supply System Rupture," and "Loss of Steam Generator Feedwater." Each of these procedures, in addition to the "Loss of Reactor Coolant/Reactor Coolant System Pressure" procedure, provide additional instructions to the operators in the event of faulty or misleading indications.

A subsequent action statement directs the operators to check alternate instrumentation channels to confirm key parameter readings. The Rancho Seco staff has made revisions to all of their emergency procedures to include this requirement.

If feedwater is not initially available following a transient or accident, core cooling is maintained by flow from two HPI pumps and relief through the PORV, which is opened by the operator. B&W has performed studies that show that density differences between the downcomer and reactor core will cause recirculation flow between the core exit and downcomer via the vent valves. Mixing of the hot core exit water with the cold HPI water will provide sufficiently warm vessel temperatures to preclude any significant thermal shock effects to the vessel. Under these conditions with no circulation of water from the steam generators, the cold leg thermocouple (located upstream of the reactor coolant pump) does not provide a satisfactory indication of the vessel temperature. B&W has recommended using the core exit thermocouples as a measure of vessel temperature, based on B&W analyses that conservatively show that the vent valves will open at temperature differences between the core exit and downcomer of less than 150°F. They have also proposed a more appropriate pressure-temperature limit curve for the vessel that reflects allowable stresses under these faulted conditions (no feedwater).

The NRC staff has reviewed these guidelines and finds them acceptable because of the expected recirculation through the vent valves and the vessel stress limits used. The licensee has incorporated these revised guidelines in his procedures for loss of all feedwater. Subsequent restoration of AFW would depressurize the reactor coolant system to below 600 psi where pressure vessel integrity is assured for any reasonable thermal transients that might subsequently occur. We conclude that further reliability analyses are needed as part of the long-term requirements of the Order to confirm that AFW can be restored (if lost) in a reasonable period of time. B&W has agreed to provide a detailed thermal-mechanical report on the behavior of vessel materials for these extreme conditions, to be applicable generically to the Oconee class of plants, which includes Rancho Seco.

The "Loss of Reactor Coolant/Reactor Coolant System Pressure" procedure was reviewed by the NRC staff to determine its conformance with the B&W guidelines. Comments generated as a result of this review were incorporated in a further revision to the procedure. A member of the NRC staff walked through this emergency procedure in the Rancho Seco control room. The procedure was judged to provide adequate guidance to the operators to cope with a small break loss-of-coolant accident. The instrumentation necessary to diagnose the break, the indications and controls required by the action statements, and the administrative controls which prevent unacceptable limits from being exceeded are readily available to the operators. We conclude that the operators should be able to use this procedure to bring the plant to a safe shutdown condition in the event of a small break accident.

An audit of seven of 14 licensed operators and senior operators assigned to shift duty (22 total licensed personnel) was conducted by the NRC staff to determine the operators' understanding of the small break accident, including how they are required to diagnose and respond to it. The Rancho Seco staff has conducted special training sessions for the operators on the concept and use of EP D.5. The audit revealed that, except for one deficiency, the operators had sufficient knowledge of the small break phenomenon and the requirements of the procedure. This deficiency, verification of natural circulation, was brought to the attention of the plant staff. Each licensed individual received additional training in this area by the plant training organization and General Physics Corporation. They also received training on the revisions made to EP D.5 as a result of the NRC review. This additional training has been completed and verified by the NRC staff.

The audit of the operators also included questioning about the TMI-2 incident and the resulting design changes made at Rancho Seco. The discussions covered the initiating events of the incident, the response of the plant to the simultaneous loss of feedwater and small break LOCA (PORV stuck open), and the operational actions that were taken during the course of the incident. We identified a deficiency in interpreting the initial sequence of the TMI-2 incident on the part of several of the operators. Additional training has been conducted in this area by the plant staff and their consultant and verified by the NRC staff.

Otherwise, we found their level of understanding sufficient to be able to respond to a similar situation if it happened at Rancho Seco. We also concluded they have adequate knowledge of subcooling and saturated conditions and are able to recognize each in the primary coolant system by various methods. The AFW system was also discussed during the audit to determine the operators' ability to assure proper starting and operation of the system during normal conditions, as well as during adverse conditions such as loss of offsite power or loss of normal feedwater. The long term operation of the system was examined to evaluate the operators' ability to use available manual controls and water supplies. The level of understanding was found to be sufficient to assure proper short and long term AFW flow to the steam generators.

In addition to the oral audit conducted by the NRC, the licensee administered a written examination to all licensed personnel. Individuals scoring less than 90 percent on the exam will receive additional training and will not assume licensed duties until a score of at least 90 percent is attained on an equivalent, but different exam. The written exam and the grading was audited by the NRC staff and judged to be satisfactory. The staff will also review all subsequent results and records as part of the normal inspection function of the Rancho Seco requalification program. We conclude that there is adequate assurance that the operators at Rancho Seco have and will continue to receive a high level of training concerning the TMI-2 accident and the consequent impact at their station.

Based on the foregoing evaluation, we conclude that the licensee has complied with the requirements of Item (d) of the Order.

Item (e)

The Order requires that the licensee:

> "Provide for one senior licensed operator assigned to the control room who has had TMI-2 training on the B&W simulator."

The licensee has confirmed that this item of the Order has been completed and has further committed that all reactor operators and senior reactor operators will have completed the TMI-2 simulator training at B&W by June 21, 1979.* This training consists of a class discussion of the TMI-2 event followed by a demonstration of the event on the simulator as it occurred and the proper actions that should be taken to control the accident. The class discussion is about four hours long and the remainder of the session is conducted on the simulator. The TMI-2 event, including operational errors, is demonstrated to each operator. The event is again initiated and the operators are given "hands-on" experience in successfully regaining control of the plant by several methods. Other transients which result in depressurization and saturation conditions are presented to the operators and they must maneuver the plant to a stable, subcooled condition.

Based on the above actions by the licensee, we conclude that the licensee is in compliance with Item (e) of the Order.

Conclusion

We conclude that the actions described above fulfill the requirements of our Order of May 7, 1979 in regard to Paragraph (1) of Section IV. The licensee having met the requirements of Paragraph (1) may restart Rancho Seco as provided by Paragraph (2). Paragraph (3) of Section IV of the Order remains in force until the long term modifications set forth in Section II of the Order are completed and approved by the NRC.

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* This action has been completed and satisfies the long-term portion of the Order in this regard.

REFERENCES

1. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting report entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," dated May 7, 1979.

2. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting revised Appendix 1, "Natural Circulation in B&W Operating Plants (Revision 1)," dated May 8, 1979.

3. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC) transmitting additional information regarding Appendix 2, "Steam Generator Tube Thermal Stress Evaluation," to report identified in Item 2 above, dated May 10, 1979.

4. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing an analysis for "Small Break in the Pressurizer (PORV) with no Auxiliary Feedwater and Single Failure of the ECCS," identified as Supplements 1 and 2 to Section 6.0 of report in Item 2, dated May 12, 1979.

5. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing an analysis for "Small Break in the Pressurizer (PORV) with no Auxiliary Feedwater and Single Failure of the ECCS" identified as Supplements 1 and 2 to Section 6.0 of report in Item 2, dated May 12, 1979.

6. Letter from J. H. Taylor (B&W) to R. J. Mattson (NRC), providing Supplement 3 to Section 6 of report in Item 2, dated May 24, 1979.

UNITED STATES NUCLEAR REGULATORY COMMISSION

SACRAMENTO MUNICIPAL UTILITY DISTRICT

DOCKET NO. 50-312

NOTICE OF AUTHORIZATION TO RESUME OPERATION

The United States Nuclear Regulatory Commission issued an Order on May 7, 1979 (44 F.R. 27779, May 11, 1979), to Sacramento Municipal Utility District (the licensee), holder of Facility Operating License No. DPR-54, for the Rancho Seco Nuclear Generating Station (Rancho Seco), confirming that the licensee accomplish a series of actions, both immediate and long term, to increase the capability and reliability of Rancho Seco to respond to various transient events. In addition, the Order confirmed that the licensee would shut down Rancho Seco on April 28, 1979, and maintain the plant in a shutdown condition until the following actions had been satisfactorily completed:

(a) Upgrade the timeliness and reliability of delivery from the Auxililary Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of April 27, 1979.

(b) Develop and implement operating procedures for initiating and controlling auxiliary feedwater independent of Integrated Control System control.

(c) Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) Provide for one Senior Licensed Operator assigned to the control room who has had Three Mile Island Unit No. 2 (TMI-2) training on the B&W simulator.

By submittal of May 14, 1979, seven letters dated May 22, 24, 29, 30(3) and June 6, 1979, the licensee has documented the actions taken in response to the May 7 Order. Notice is hereby given that the Director of Nuclear Reactor Regulation (the Director) has reviewed this submittal and has concluded that the licensee has satisfactorily completed the actions prescribed in items (a) through (e) of paragraph (1) of Section IV of the Order, that the specified analyses are acceptable and the specified implementing procedures are appropriate. Accordingly, by letter dated June 27, 1979, the Director has authorized the licensee to resume operation of Rancho Seco. The bases for the Director's conclusions are more fully set forth in a Safety Evaluation dated June 27, 1979.

Copies of (1) the licensee's letters dated May 14, 1979, and seven letters dated May 22, 24, 29, 30(3) and June 6, 1979, (2) the Director's letter dated June 27, 1979 and (3) the Safety Evaluation dated June 27, 1979, are available for inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. 20555, and are being placed in the Commission's local public document room in the Business and Municipal Department, Sacramento City-County Library, 828 I Street, Sacramento, California 95814. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Dated at Bethesda, Maryland,
this 27th day of June 1979.

E

APPENDIX E

LETTERS ISSUING AUXILIARY FEEDWATER SYSTEM REQUIREMENTS

The NRC issued letters to licensees of pressurized water reactors and boiling water reactors advising of requirements for auxiliary feedwater systems. Some of the requirements were generic according to reactor vendor (Combustion Engineering or Westinghouse) and others were plant-specific.

Copies are contained in this appendix as follows:

Arkansas 2 (C-E)	11/06/79	Indian Point 2 (W)	11/07/79
Calvert Cliffs 1&2 (C-E)	11/07/79	Indian Point 3 (W)	11/07/79
Ft. Calhoun 1 (C-E)	10/22/79	Kewaunee (W)	9/21/79
Maine Yankee (C-E)	10/18/79	North Anna 1 (W)	9/28/79
Millstone 2 (C-E)	10/22/79	Point Beach 1&2 (W)	9/21/79
Palisades (C-E)	10/30/79	Prairie Island 1&2 (W)	10/16/79
St. Lucie 1 (C-E)	10/17/79	Salem 1 (W)	9/21/79
Beaver Valley 1 (W)	10/11/79	San Onofre 1 (W)	11/15/79
D. C. Cook 1&2 (W)	10/30/79	Surry (W)	9/25/79
Farley 1 (W)	10/13/79	Trojan (W)	10/03/79
Ginna 1 (W)	10/22/79	Turkey Point 3&4 (W)	10/16/79
H. B. Robinson	9/21/79	Yankee Rowe 1 (W)	11/09/79
Haddam Neck (W)	10/11/79	Zion 1&2 (W)	9/18/79

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 6, 1979

Docket No. 50-368

Mr. William Cavanaugh, III
Executive Director of Generation
and Construction
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Cavanaugh:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT ARKANSAS NUCLEAR ONE UNIT 2

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Combustion Engineering-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Combustion Engineering report CEN-114-P (Amendment 1-P)

entitled, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/enclosures:
See next page

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ENCLOSURE 1

X.1 (CE) ARKANSAS 2

EMERGENCY FEEDWATER SYSTEM

X.1.1 System Description

X.1.1.1 Configuration - Overall Design

The emergency feedwater system (EFWS) as shown in Figure 1 consists of primary and secondary sources of water and two emergency feedwater pumps, which feed either or both steam generators (SG). The primary source of EFWS water is a non-seismic Category I condensate storage tank with a 200,000 gallon capacity, 160,000 gallons of which are dedicated to the EFWS by the Technical Specifications. There is also available 100,000 gallons of water from a swing condensate storage tank which is shared by both Unit 2 and Unit 1.

The secondary source of water is the plant service water system, a seismic Category I system, whose water source is either the emergency cooling pond or the Dardanell Reservoir. The service water system is seismic Category I up to and including the suction piping from the emergency cooling pond (the ultimate heat sink). Three service water pumps can draw water from the reservoir (normal mode) or from the emergency cooling pond. The reservoir can also act as a long term water source.

The primary condensate storage tank is normally lined up to supply water to both EFW pumps through manually operated locked open isolation valves. Train A includes the motor-driven pump and Train B includes the turbine-driven pump, each having 100% capacity, and designed to deliver 575 gpm @ 1390 psig. The discharge lines from each pump are cross-connected through two normally closed (NC) manual isolation valves. Upon low suction pressure to the operating pump(s), the suction to the pump is automatically aligned to the secondary water source.

Under the worst transient conditions the licensee estimates that, without the EFW flow, the SG would boil dry in 14 minutes, following loss of main feedwater with reactor trip.

X.1.1.2 Components

Except for the condensate storage tanks, the EFWS components (pumps, valves, and valve operators) and piping are safety grade, seismic Category I and tornado missile protected. The power supplies and instrumentation are Class 1E. Each EFW pump is located in a separate room. Because the pumps are located below the probable maximum flood level, these rooms are watertight with watertight doors to prevent flooding. There are two room coolers in the steam-driven pump room and one room cooler in the motor-driven pump room. The lubricating system for each EFW pump is air-cooled by vanes on its pump shaft.

X.1.1.3 Power Sources

Except for the turbine pump steam admission valves immediately downstream of steam generators A and B, respectively, Train B of the system (turbine-driven pump) obtains control power and power for operating valves from the Division II bus, a Class 1E DC source. The turbine pump steam admission valves receive power from Division I and II Class 1E AC sources, respectively. However, they are locked open with power removed during operation.

Train A (motor-driven pump) is powered by the Division I bus, a Class IE AC source. Except for the two ball valves used to isolate the steam generators which are DC powered, all the valves in Train A are AC powered. The onsite emergency power system consists of two divisions, each being supplied by an independent diesel generator and corresponding DC battery system. Both the diesel generators and battery systems are located in separate seismic Category I rooms.

X.1.1.4 Instrumentation and Controls

X.1.4.1 Controls

Steam generator level is controlled automatically by the engineered safety features actuation system (ESFAS) and can be controlled manually from the control room. Steam generator level indication and alarm are available to the operator in the control room. EFWS flow to the steam generator is automatically terminated when the level reaches a high point, and low steam generator level will automatically reestablish emergency feedwater flow. This on-off type of flow control is accomplished by opening or closing the ball valves located at the inlet to the steam generators.

X.1.1.4.2 Information Available to Operator

System information available to the operator in the control room to assess the performance of the emergency feedwater system is as follows:

- Position indicating lights for each electrical and pneumatic operated valve.
- Steam generator level
- Steam generator pressure
- EFWS flow indication in each of the four water paths to the steam generators.

X.1.1.4.3 Initiating Signals for Automatic Operation

The EFW pumps and flow path control valves are automatically actuated by the ESFAS whenever any of the following two out of four coincident logic conditions exist:

1. Steam generator (A & B) low level
2. Steam generator (A & B) low pressure
3. Steam generator differential pressure-high (SG-A>SG-B)
4. Steam generator differential pressure-high (SG-B>SG-A)

Main steam line break isolation is accomplished automatically whereas a main feedwater line break is manually isolated.

If steam generator isolation is required, as in the case of a postulated main steam line or feedwater line break, the ESFAS will open only the EFW valves leading to the intact steam generator. A combination of measured variables (level and pressure) for each steam generator are used to determine which steam generator is intact.

X.1.1.5 Testing and Technical Specifications

X.1.1.6 The EFWS is periodically tested and has Limiting Conditions of Operation in accordance with the Technical Specifications as follows:

EMERGENCY FEEDWATER SYSTEM

Two emergency feedwater pumps and associated flow paths shall be OPERABLE with:

a. One motor driven pump capable of being powered from an OPERABLE emergency bus, and
b. One turbine driven pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one emergency feedwater pump inoperable, restore the inoperable pump to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

Each emergency feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the turbine driven pump develops a discharge pressure of $\geq$1200 psig at a flow of $\geq$560 gpm when the secondary steam supply pressure is greater than 865 psig and the pump speed is $\leq$3600 rpm. The provisions of Specification 4.0.4 are not applicable.

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on MSIS or ESFAS test signals.

2. Verifying that the motor driven pump starts automatically upon receipt of an ESFAS test signal.

3. Verifying that the turbine driven pump steam supply MOV opens automatically upon receipt of an ESFAS test signal.

X.1.2 Reliability Evaluation

X.1.2.1 Dominant Failure Modes

X.1.2.1.1 Loss of Main feedwater (LOFW)

No single failure was identified which would make both feedwater trains unavailable. Thus the dominant failure modes were combinations of two independent failures, each failing one subsystem.

X.1.2.1.2 LOFW With Loss of Offsite AC Power

The dominant failure modes are the same as those identified above in the case of loss of main feedwater only.

X.1.2.1.3 LOFW with Only DC Power Available

The dominant failure modes for this event are failure of the turbine driven pump subsystem due to test and maintenance outages, hardware failure, or human error.

Since the motor driven EFW pump would not be available upon loss of all AC power, auxiliary feedwater flow would be dependent on the single turbine driven pump subsystem. Single valve or pump failure, or a manual valve being left in the closed position, or the subsystem being out due to test and maintenance are all significant contributors to the unavailability of the EFWS during this event.

X.1.2.1.4 Potential Interactions

None

X.1.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW* system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant-specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.1.3.1 Short-Term

1. Recommendation GS 6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verity that the valves are properly aligned.

   The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam

* The term AFW system as used in these recommendations applies to the ANO-2 EFW system.

   generators. The flow test should be conducted with AFW system valves in their normal alignment.

2. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
   - Testability of the initiation signals and circuits shall be a feature of the design.
   - The initiation signals and circuits should be powered from the emergency buses.
   - Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
   - The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

3. Recommendation - The Surveillance Requirements section of the Technical Specications should add pressure and flow acceptance criteria for the periodic (31-day) testing of the motor driven pumps.

X.1.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendaton - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within the design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

"Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system

train, and there is only one remaining AFW train available for operation should propose the Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.1.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

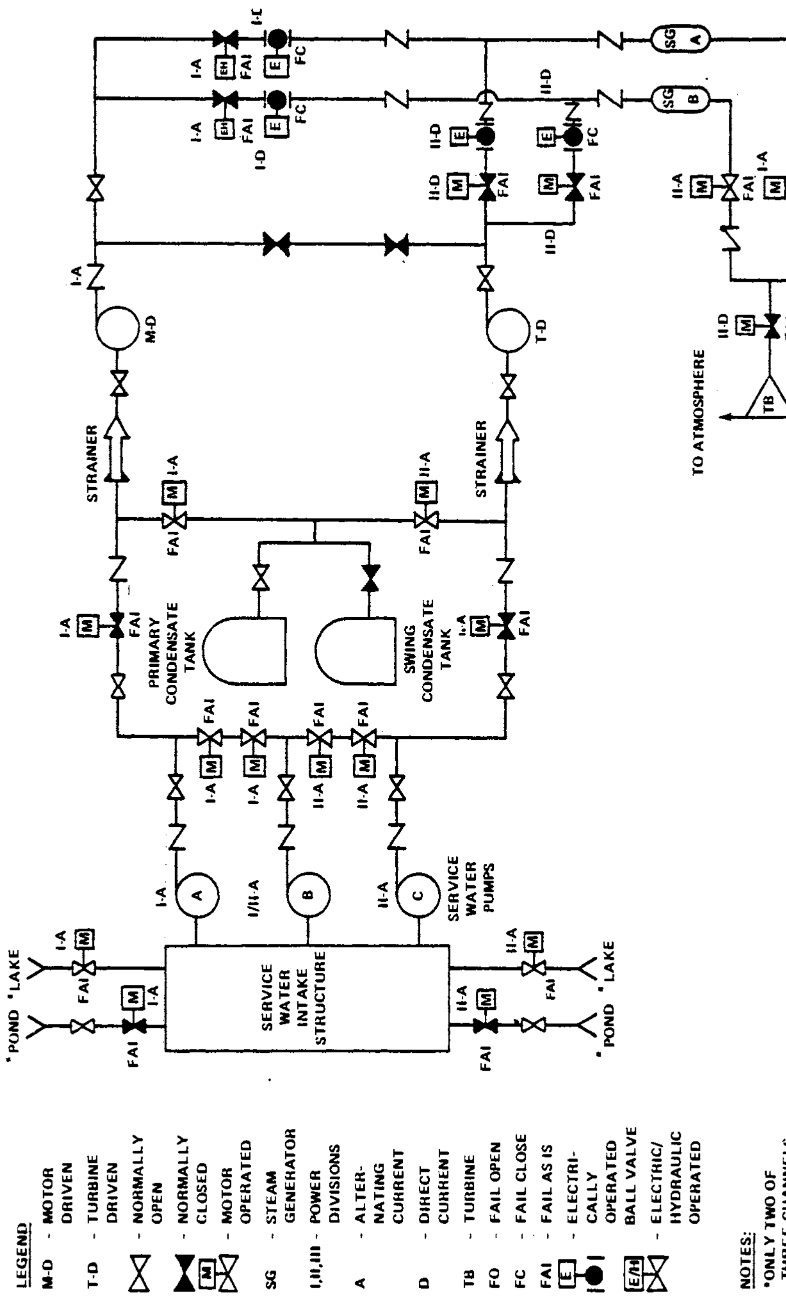
1. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

2. Recommendation - The Arkansas Unit 2 AFW system design does not meet the high energy line break criteria in SRP-10.4.9 and Branch Technical Position 10-1; namely, that the AFW system should maintain the capability to supply the required AFW flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines concurrent with a single active failure.

   The licensee should evaluate the postulated pipe breaks stated above and (1) determine any AFW system design changes or procedures necessary to detect and isolate the break and direct the required feedwater flow to the steam generator(s) before they boil dry or (2)

describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.

3. Recommendation - Concern was expressed to the licensee about the capability of the design to isolate a break occurring downstream of the steam admission valve to the turbine-driven pump during AFWS operation concurrent with a single active failure of the DC emergency Division II. Assuming that without DC, the corresponding diesel generator will not be able to start, the break could not be isolated because of the loss of DC and AC power in Division II. The licensee advised that analysis has been performed showing that there is sufficient residual magnetism to flash the diesel generator field and consequently the Division II diesel-generator can be brought up to speed and voltage without the need of DC from the emergency batteries. Thus, the break could be isolated if the failure of the DC emergency Division II does not result also in the loss of AC in the same division. The licensee should submit for staff review the analysis with regard to starting the diesel generator without DC emergency power available.



EMERGENCY FEEDWATER SYSTEM

Arkansas 2
Figure 1

ENCLOSURE 2

BASIS FOR AUXILIARY FEEDWATER SYSTEM FLOW REQUIREMENTS

As a result of recent staff reviews of operating plant auxiliary feedwater systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of offsite and onsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above.

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:
   - Maximum RCS pressure (PORV or safety valve actuation)
   - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature
   - RCS cooling rate limit to avoid excessive coolant shrinkage
   - Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cooldown the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant conditions considered in 1.a. above including:
   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.
   b. Time delay from initiating event to reactor trip.
   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).
   d. Minimum steam generator water level when initiating event occurs.
   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.
   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.
   g. Minimum number of steam generators that must receive AFW flow, e.g., 1 of 2, 2 of 4?
   h. RC flow condition - continued operation of RC pumps or natural circulation.
   i. Maximum AFW inlet temperature.
   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accomodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.
   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.
   l. Operating condition of steam generator normal blowdown following initiating event.
   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.
   n. Time at hot standby and time to cooldown RCS to RHR (or SCS) system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 7, 1979

Docket Nos.: 50-317
50-318

Mr. A. E. Lundvall, Jr.
Vice-President - Supply
Baltimore Gas & Electric Company
P. O. Box 1475
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Dear Mr. Lundvall:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT CALVERT CLIFFS NUCLEAR PLANT UNITS 1 AND 2

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Combustion Engineering-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Combustion Engineering report CEN-114-P (Amendment 1-P)

entitled, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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ENCLOSURE 1

X.2 (C-E) CALVERT CLIFFS UNITS 1 & 2 AUXILIARY FEEDWATER SYSTEM

X.2.1 SYSTEM DESCRIPTION

X.2.1.1 Configuration - Overall Design

Figure 1 is a simplified diagram of the Calvert Cliffs Unit 1 auxiliary feedwater system (AFWS). The Calvert Cliffs Unit 2 AFWS is identical to that of Unit 1. Basically, the AFWS is a manually operated system that includes two steam turbine-driven pumps, each of which can deliver 700 gpm at 1100 psia. Both pumps are located in the auxiliary feedwater pump room.

The normal water supply to each pump is from Condensate Storage Tank No. 12 (CST No. 12), via a common line feeding a branch line to each pump. Flow from each pump discharges into a branch line feeding a common line which in turn branches to each of two steam generators (SG). AFWS flow is controlled by controlling pump speed and by regulating flow through a normally closed (NC) air-operated control valve in the feed line to each SG. Each NC air-operated control valve fails open on loss of air, and each can be bypassed by a loop that includes a normally closed manually operated valve.

AFWS water can be obtained from five sources. The primary water source for both units is CST No. 12 which has a 350,000 gallon capacity, 300,000 gallons of which are dedicated to the AFWS for both units. The licensee stated that this amount of water can cool down both units and will last six to ten hours, depending on the accident or transient that caused the need for AFWS operation. The normal source of AFWS water flows from CST No. 12 through two normally open manually operated valves and a check valve in a common line which branches to the two pumps. This source of water is designed to seismic Category I requirements and is protected against tornado missiles. The other sources of water are neither designed to seismic Category I requirements nor protected against tornado missiles.

The secondary sources of water consist of two-350,000 gallon tanks, CST No. 11 and CST No. 21 for Unit 1 and Unit 2, respectively. Each tank is designed to serve its associated AFWS, without any cross-connection to the other tank, via a single line. This line includes two normally closed manually-operated valves, and is connected to the common header that feeds both auxiliary feedwater pumps. Although none of this water is dedicated for AFWS service, the licensee estimates that it would take about three to five minutes to line-up either tank to its respective AFWS, if required.

Three additional sources of water are: (i) the 350,000 gallon demineralized water tanks; (ii) the two-500,000 gallon pretreated water storage tanks, of which 600,000 gallons are dedicated for fire protection usage; (iii) the well water system. The licensee estimates that it would take approximately fifteen minutes to manually align the 350,000 gallon demineralized water tanks to CST No. 12. The licensee also estimates that it would take approximately thirty minutes to connect the pretreated water storage tanks to the demineralizer system and that it would require approximately one hour if the demineralizer system is bypassed. In either case,

the pretreated water storage tanks would be connected to CST No. 12, in which case the licensee estimates that all of the above tanks would provide for more than ten hours of AFW supply. The well water system has a pumping capability of 966 gpm, and automatically (or manually, if required) replenishes the pretreated water tanks whenever they reach a low level.

The ability to maintain the AFW system function following certain postulated pipe breaks in the main steam, main feedwater and auxiliary feedwater piping systems was evaluated. In the event of feedwater line breaks inside or outside containment or main steam line breaks downstream of the main steam isolation valve (MSIV), acceptable AFWS capability can be retained by feeding the intact steam generator, provided the control valve to the affected steam generator is maintained closed. However, if a steam line break occurs upstream of the MSIV concurrent with a single active failure, or if a steam line break occurs in the common header to the two AFW pump turbines, even without an active failure, potential problems could result in the containment penetration area. In the former case, if the steam inlet motor operated valve (MOV) from the unaffected steam generator to the turbine-driven auxiliary feedwater pump fails to open, loss of AFWS function will result. This AFWS function can be restored by manually opening the bypass valve around the affected MOV, thereby admitting steam to the turbine-drive AFW pumps and restoring AFWS function. This manual action is possible since the bypass valves have operator extensions which extend into the adjacent room. In the latter case, the AFW pump room must be vented and cooled to permit access for isolating the break and manually supplying steam to the AFW pump turbines from the other unit or from the auxiliary steam generator. The licensee estimates that these emergency actions can be accomplished in approximately thirty minutes. Except for the pumps, the AFWS equipment is not qualified for operation in the pipe break environment.

Pipe breaks at two locations in the AFWS were considered: (1) at the steam generator, and (2) in the common discharge header between the pumps and the steam generators (the worst case break). In the former case, manual action (e.g., closing the normally open valve in the affected line) can be taken to assure flow to the unaffected steam generator. The licensee estimates that it would take approximately three minutes to perform the required valve operation(s). In the latter case, however, loss of feedwater function will result and persist until the break itself is repaired.

Depending on the initial plant conditions and the event that causes the need for the AFWS, the licensee estimates that the steam generators would boil dry in approximately thirty minutes if the AFWS is not actuated.

X.2.1.2 Components - Design and Classification

The licensee stated that the components and equipment of the AFWS were designed and classified in accordance with the following table.

Component/Equipment	Environmental Qualification	Design Classification	Seismic Category
Pumps & Turbine	High Energy Pipe Break	Safaty Related	I
Valves/Actuators	Ambient	"	"
Piping	"	"	"
Main Steam System up to MSIV	"	"	"
Condensate Storage Tank No. 12	"	"	"
Condensate Storage Tanks Nos. 11 & 21	"	Non-Safety Related	Non-Seismic
Demineralized Water Tank	"	"	"
Pretreated Water Tank	"	"	"
Deep Well System	"	"	"
Controls and Instrumentation	"	"	"

X.2.1.3 Power Sources

Steam to drive the AFWS turbine-driven pumps is obtained from the steam generators. Each steam generator can supply steam to either or both steam turbine-driven pumps from its main steam line through a normally closed motor operated valve which fails as-is or a normally closed manual bypass valve into a common header. Each AFW pump takes steam from the common header through a normally open manual valve, a check valve, a DC operated normally open stop valve, and an air operated normally closed throttle valve. An alternate source of steam can be obtained from the steam generators of the other unit or from steam generated by the auxiliary steam generator, which uses an oil fired boiler (aux. stm. gen.).* The alternate source of steam is routed through a normally locked closed manual valve connected between the check valve and stop valve on each pump steam supply line.

The two motor operated steam turbine pump inlet valves are powered from separate emergency AC buses. The turbine control valve and the AFWS flow control valve are air operated fail open valves. The turbine stop valves are powered from the DC buses, and fail in the open position. All control and instrumentation power is from emergency buses which can be energized from the diesel generators.

Upon loss of all station AC, local manual action is required to start the system by opening the steam inlet MOV's.

X.2.1.4 Instrumentation and Controls

X.2.1.4.1 Controls

The following controls are located in the Control Room:

1. Hand indicating controllers for
   a. Turbine Control (throttle) Valve
   b. AFWS Regulating Valve
2. Motor Operated Valves - Open/Close
3. Turbine Trip

All controls except the motor operated valve controls are also located at the Remote Shutdown Panel/AFWS Pump Room.

X.2.1.4.2 Information Available to the Operator

The following alarms are located in the control room:

1. Common Alarm Low Pump Suction and Discharge Pressure
2. Condensate Tank Low Level Alarms
   a. Common Alarm Tank 11 & 12
   b. Common Alarm Tank 12 & 21
3. Steam Generator Low Level Alarms

No alarms are located at the Remote Shutdown Panel or the local stations.

The following indicators are located in the control room:

1. AFW Flow Indicator - one per steam generator
2. Steam Generator Level
3. Condensate Storage Tanks' Level Indication
4. Valve Position Indication for
   a. Motor Operated Inlet Steam Valve
   b. Turbine Control Valve
   c. AFW Regulating Valve
   d. Turbine Stop Valve
5. Pump Discharge Pressure
6. Steam Line Pressure
7. Pump Suction Pressure (Common) to be Removed

The following indicators are located at the Remote Shutdown Station:

1. Steam Generator Level
2. CST Level

3. AFW Regulating Valve Position Indicator
4. Pump Discharge Pressure
5. Steam Line Pressure

X.2.1.4.3 Initiating Signals for Automatic Operation

Since the system is a manually initiated system this section is not applicable. Manual AFW initiation is by a semi-dedicated operator in the control room following any reactor trip. The semi-dedicated operator means that the operator has other duties in the control room until that time when the AFWS is needed, then he is dedicated 100% to operate, control and monitor the system.

X.2.1.5 Testing

The pumps are tested on a monthly basis in a recirculating mode of operation for total dynamic head and vibration, and for bearing temperatures at each refueling. All non-manual valves are stroked and timed monthly. The instrumentation at the remote shutdown panel is checked monthly. The normally closed or opened manual valves are not stroked. When the system has been down for maintenance, the normal monthly tests are performed prior to the system being restored to service.

X.2.1.6 Technical Specifications

The following are the technical specifications for the plant.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least two steam turbine driven steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one auxiliary feedwater pump inoperable, restore at least two auxiliary feedwater pumps to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the steam turbine driven pump develops a Total Dynamic Head of $\geq$ 2800 ft. on recirculation flow when the secondary steam supply pressure is greater than 800 psig.

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The No. 12 condensate storage tank (CST) shall be OPERABLE with a minimum contained water volume of 150,000 gallons per unit.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

With the No. 12 condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the No. 21 condensate storage tank as a backup supply to the auxiliary feedwater pumps and restore the No. 12 condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The No. 12 condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The No. 21 condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying that the tank contains a minimum of 150,000 gallons of water and by verifying that the flow path

for taking suction from this tank is OPERABLE with the manual valves in this flow path open whenever the No. 21 condensate storage tank is the supply source for the auxiliary feedwater pumps.

X.2.2 RELIABILITY EVALUATION RESULTS

X.2.2.1 Dominant Failure Modes

Failure modes of the AFWS were assessed for three loss of main feedwater transients. The dominant failure modes for each transient type are discussed below.

- Loss of Main Feedwater (LOFW) with Offsite Power Available

  There are two dominant failure modes of the AFWS for this transient, both of which are related to human errors.

  The first human error is failure of the operator to manually initiate the AFWS. Upon a demand for the AFWS, the operator has approximately 30 minutes to actuate AFWS and prevent steam generators from boiling dry, depending on the cause of the transient. Thus, the human error is the failure to actuate AFWS within this time period.

  The second human error is related to the inadvertent closure of either of two manual valves in the single condensate storage tank supply line to the AFWS pumps. Such an inadvertent closure could result from a number of causes, e.g., personnel error in closing the wrong valve during a test procedure, or an error in failing to reopen the valve after maintenance in adjoining parts of the AFWS. Coupled with this error is the failure of the operator to reopen the valve before damage to the pumps occurs following an AFWS demand. The combination of these errors results in an AFWS failure.

- Loss of MFW with Only Onsite AC Power Available

  This transient is very similar to the transient discussed above. Additional failure modes related to the onsite AC power system were considered; however, these did not have a significant impact. As such the dominant failure modes discussed above are also considered to the applicable for this transient.

- Loss of MFW with Only DC Power Available

  In this transient no AC power, either onsite or offsite, is available. Because of certain AC dependencies, the dominant failure mode is assessed to be the failure of the operator to manually open one of the two steam admission valves to the pump turbine within approximately thirty minutes after the transient. These valves are normally closed motor-operated valves that normally receive power from either the offsite AC power system or the onsite (diesel-generator) AC power system. Since neither of these sources is available in this transient, local manual opening of one of the valves would be required.

X.2.2.2 Principal Dependencies

The principal dependency identified for this AFWS system is that related to human action requirements. For each transient discussed here, human errors are the dominant AFWS failure modes.

Two additional potential dependencies have been noted for the Calvert Cliffs AFWS, both resulting from the physical location of equipment within the plant. These are:

1. Location commonality of AFWS pumps.

   Both AFWS turbine pumps (and some associated valving) are located in a relatively small room sealed with watertight doors. Because of this close proximity of redundant equipment, there exists the potential for total AFWS failure resulting from flooding, missiles, etc., caused by failures within one train or from external causes. (See Recommendations)

2. Location commonality of steam-admission valves.

   Both steam-admission valves for the AFWS pump turbines are located in a common area, the main steam line penetration room. Normal conditions in this area are high temperature and high humidity; thus, there exists some potential for environmentally-caused common mode failures. In addition, because the main steam lines are located just above these valves, the potential environmentally-caused failure of these valves after a steam line break, when AFWS is needed, requires further investigation. (See Recommendations)

X.2.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon as thereafter as is practicable.

X.2.3.1 Short Term

1. Recommendation GS-2 - The licensee presently, by administrative procedure, locks open single valves or multiple valves in series in the AFW system pump suction piping and locks open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspection should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance recommendations of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

- The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW pumps against self-damage before a water flow is initiated, and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at one turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications

at local stations should also be provided if manual initiation and contorl of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation GS-8 - The licensee should install a system to automatically initiate the AFWS. This system need not, in the short-term, be safety-grade; however, it should meet the criteria listed below, which are similar to Item 2.1.7a of NUREG-0578. For the onger term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-1.

   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
   - Testability of the initiating signals and circuits should be a feature of the design.
   - The initiating signals and circuits should be powered from the emergency buses.
   - Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
   - The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
   - The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

6. Recommendation - The licensee should propose modifications to Technical Specifications to require that manual valves that are normally closed or open will be tested periodically.

X.2.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW systems designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.2.3.3 Long-Term

Long-term recommendations for improving the systems are as follows:

1. Recommendation - GL-1 - Licensees with plants having a manual starting AFW system should install a system to automatically initiate the AFW system flow. This system and associated automatic initiation signals

should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation.

2. Recommendation - GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s) should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

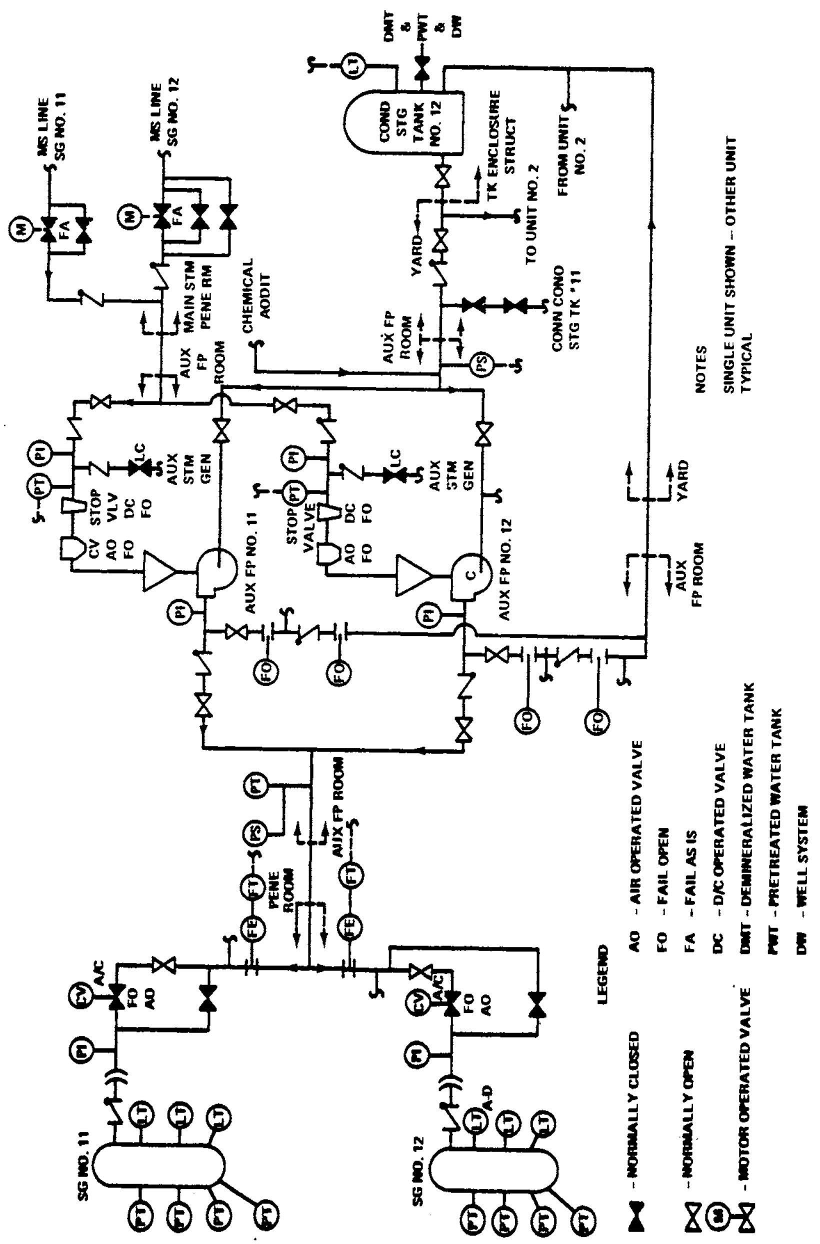
The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

3. Recommendation - GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

4. Recommendation - The motor operated steam inlet valves and other equipment affected by the environmental effects of the main steam and feed line breaks discussed in section 2.1.1 and 2.2.4 should be qualified to the environmental conditions that will be present.

5. Recommendations - The licensee should evaluate the following concerns:
   a) The AFW pump discharge lines and turbine driven AFW steam supply lines combine into different single lines through which all AFW water or steam must flow. (See Figure 1). A pipe break in either of these single flow paths would cause loss of the entire AFW function.
   b) The Calvert Cliffs AFW systems do not meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position 10-1; namely, that the AFW system should maintain the capability to supply the required flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines concurrent with a single active failure.

The licensee should evaluate the postulated pipe breaks stated above and (1) determine any AFW system design changes or procedures necessary to detect and isolate the break and direct the required feedwater flow to the steam generator(s) before they boil dry or (2) describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.



Auxiliary Feedwater System
Calvert Cliffs, Unit 1 & 2
Figure 1

ENCLOSURE 2

BASIS FOR AUXILIARY FEEDWATER SYSTEM FLOW REQUIREMENTS

As a result of recent staff reviews of operating plant auxiliary feedwater systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of offsite and onsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above.

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:
   - Maximum RCS pressure (PORV or safety valve actuation)
   - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperatu
   - RCS cooling rate limit to avoid excessive coolant shrinkage
   - Minimum steam generator level to assure sufficient steam generator heat tra- fer surface to remove decay heat and/or cooldown the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant conditions considered in 1.a. above including:
   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.
   b. Time delay from initiating event to reactor trip.
   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).
   d. Minimum steam generator water level when initiating event occurs.
   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.
   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.
   g. Minimum number of steam generators that must receive AFW flow, e.g., 1 of 2, 2 of 4?
   h. RC flow condition - continued operation of RC pumps or natural circulation.
   i. Maximum AFW inlet temperature.
   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accomodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.
   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.
   l. Operating condition of steam generator normal blowdown following initiating event.
   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.
   n. Time at hot standby and time to cooldown RCS to RHR (or SCS) system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 22, 1979

Docket No. 50-285

Mr. W. C. Jones
Division Manager - Production Operations
Omaha Public Power District
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Jones:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT FORT CALHOUN STATION UNIT NO. 1

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Combustion Engineering-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Combustion Engineering report CEN-114-P (Amendment 1-P)

entitled, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut /for

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

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As stated

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ENCLOSURE 1

X.3 (CE) FT. CALHOUN

AUXILIARY FEEDWATER SYSTEM

X.3.1 System Description

X.3.1.1 Configuration, Overall Design

A simplified diagram of the auxiliary feedwater system (AFWS) for the Ft. Calhoun plant is shown in Figure 1. The AFWS includes a steam turbine-driven pump and a motor-driven pump, each rated 260 gpm @ 2400 ft. head. Each pump is capable of cooling the plant down to the temperature where the shutdown cooling system (SCS) can be used to continue safe plant shutdown. The pumps are located in the seismic Category I auxiliary building, and are protected against internal and external flooding. Piping interconnections are provided to permit either AFW pump to feed directly either or both steam generators through the normal AFW flow path. AFW flow can also be directed to the main feedwater lines upstream of the main feedwater isolation valves.

The primary water supply for the AFWS is the seismic Category I emergency feedwater storage tank (EFST) having a capacity of 63,000 gallons. The EFST is required by Technical Specifications to contain at least 55,000 gallons of water whenever the reactor coolant system (RCS) temperature is above 300°F. The licensee states that this is adequate to maintain hot standby for 8 hours.

The EFST water level is automatically maintained by the condensate system (CS). If the CS is not available, the EFST level will be maintained by either the demineralized water from the water treatment plant or the outside condensate storage tank. In addition, emergency makeup water supply to the EFST may be obtained from the fire main of the fire protection system. EFST water level indicators are provided which will initiate, alarm and annunciate in the main control room on high or low water level.

X.3.1.2 Components - Design Classifications

The AFWS, including instrumentation and control and primary water source, is classified as an engineered safety features system and designed according to seismic Category I and safety grade requirements.

X.3.1.3 Power Sources

The steam turbine driven pump receives steam from either SG from a point upstream of each main steam isolation valve (MSIV) via direct current (DC) power solenoid air operated valves and exhausts directly to the atmosphere. (See Figure 1 for valve(s) normal position and position upon loss of power or air.)

The motor-driven pump receives power from a 4160V vital bus. Upon loss of offsite power, the operator must connect the motor-driven pump train to an emergency diesel generator bus.

X.3.1.4 Instrumentation and Controls

X.3.1.4.1 Controls

The instrumentation and controls within the AFWS have been designed as safety grade and seismic Category I components. The systems' safety function will not be affected by a single failure, since redundancy has been provided. The SG water level is manually controlled by the operator using either one of the DC solenoid air operated valves which are located outside the containment. Manual operation of these valves can be performed locally on loss of compressed air. The pumps (turbine driven and motor driven) can be controlled remotely from the control room or at the auxiliary feedwater control panel.

X.3.1.4.2 Information Available to Operator

The important AFWS information available to the operator includes pump operability (suction flow, discharge, flow), EFST level and temperature. SG flow, SG water level and control valve position indication are also provided in the control room.

X.3.1.4.3 Initiating Signals for Automatic Operations

Both AFW pumps will automatically start on trip of the last operating main feedwater pump. On loss of offsite power, only the turbine driven pump will start automatically; the motor driven pump can be started manually after connecting the motor to an emergency diesel generator bus[1]. AFW flow from the turbine-driven pump will initiate automatically upon loss of all

[1] The licensee is considering the possibility of automating the electric AFW subsystem for the case where offsite AC would be lost.

onsite and offsite AC power. In this event, the steam supply and AFW flow control valves in the turbine pump train open. Also, the turbine pump lube oil is cooled by recirculated AFW flow.

X.3.1.4.4 Testing

The AFWS is tested every 31 days in accordance with technical specification requirements. The system is tested using the pump recirculating line and noting pump pressure and flow. The instrumentation system is checked periodically, in accordance with the technical specifications, each shift, monthly or during refueling outages. AFW flow instrumentation channels for the SGs, flow indicating controls for the AFW pumps, and level indication and level alarm switches are calibrated annually.

In addition to the above periodic testing, the licensee routinely uses the AFWS for shutdown and startup operations. This practice augments the detection of malfunctions in the Ft. Calhoun AFWS periodic surveillance testing.

X.3.1.4.5 Technical Specifications

The Limiting Conditions for Operation stipulate that the reactor coolant system shall not be heated above 300°F unless the following conditions are met:

1. Both auxiliary feedwater pumps are operable. One of the auxiliary feedwater pumps may be inoperable for 24 hours provided that the redundant component shall be tested to demonstrate operability.

2. A minimum of 55,000 gallons of water in the emergency feedwater storage tank and a backup water supply to the emergency feedwater storage tank from the Missouri River by the fire water system.
3. All valves, interlocks and piping associated with the above components required to function during accident conditions are operable.

X.3.2 Reliability Evaluation Results

X.3.2.1 Dominant Failure Modes

The Ft. Calhoun AFWS consists of two subsystems, one includes a motor driven pump and the other a steam turbine driven pump. Either of these two subsystems delivering water to one of the two steam generators provides for adequate decay heat removal given the three loss of main feedwater events considered.

The following failure modes were found to dominate the demand unavailability of the Ft. Calhoun AFWS.

- Loss of Feedwater (LOFW) with Offsite AC Power Available

The dominant failure mode (~ 80% contribution) identified for the Ft. Calhoun AFWS was inadvertent closure of the single, manually operated AFW pump suction valve from the EFST that could make the redundant AFWS subsystems inoperative. Although this valve is located in a security area and is visible and locked open, the licensee plans to further strengthen the administrative checking on this valve and its position status, (i.e., a visual check would be made and logged as part of a routine data logging procedure performed for the turbine and steam plant). This added procedure would result in a check of the valve position status at least several times each day.

The AFWS for Ft. Calhoun is used to supply feedwater to the SG's for routine shutdowns and startups. This routine use is over and above that usage resulting from actual demands and testing demands and serves to further confirm the availability of a flow path through the single locked open pump suction valve. It is considered, however, that even with the above valve status verification procedure in place, this single suction valve remains a major point of vulnerability in the Ft. Calhoun AFWS. This is because all emergency feedwater sources (primary and backup) must pass through this single valve and flow blockage (e.g., disengaged valve gate/disc) could make the AFWS inoperative.

An additional potential vulnerability of the Ft. Calhoun AFWS design was observed; however, this vulnerability was not assessed in detail during this review. This potential vulnerability is associated with the discharge piping cross-connection between the two AFWS subsystems that includes two normally open manual valves (FW 744 and FW 745). This cross-connection was installed by the licensee subsequent to the FSAR review to provide an alternate way to supply AFW flow via the main feedwater system. A single passive failure in this cross-connection would require local operator action to manually close either FW 744 or FW 745 to isolate the two subsystems from one another. The licensee should re-evaluate the position of these valves considering a postulated break in the cross-connection (see short-term recommendation number 6.)

- <u>LOFW with only Onsite AC Power Available</u>

  The Ft. Calhoun vital electrical buses employ two emergency diesel generators (EDG) with load shedding features. The motor-driven pump train of the AFWS can be powered by either EDG unit; however, since it is normally connected to an electrical bus supplied by offsite power, it is shed from the bus on loss of offsite power. As soon as the EDG's pick up their safety loads, the plant operator is required to connect the motor-driven pump to one of the EDGs by switching action in the control room. Assessment of this human dependency and its contribution to the overall AFWS unavailability indicates a small increase relative to the above LOFW transient event ($\leq$ 20%). The single valve in the AFWS suction line remains as the dominant fault contributor.

  <u>LOFW With Only DC Power Available</u>

  In this event, the turbine-driven pump train portion of the Ft. Calhoun AFWS would start automatically. The operator would be expected to provide backup in case the solenoid operated valves (SOVs) in the steam admission line to the turbine-driven pump fail to open. The dominant contributors to AFWS unavailability in this event were:

  -Allowed test and maintenance outage times ($\sim$ 40%)

  -Hardware faults (turbine pumps and manual valves around the turbine pumps) ($\sim$ 50%)

X.3.2.2 <u>Principal Dependencies Identified</u>

1. The single locked open AFW pump suction valve (FW-339) which feeds both AFWS pumps.

2. The potential common mode vulnerability in the cross-connection installed by the licensee due to valves FW 744 and FW 745 being left normally open. Failure in the cross-connection requires local manual actions to correct.

3. The operator being required to connect the motor-driven pump train of AFWS to an EDG bus for the LOFW transient with only onsite AC power available.

X.3.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted GL, and plants specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.3.3.1 Short-Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to

verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for tranferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

- The case in which the primary water supply is not initially available. The procedures for this case should include any operator action required to protect the AFW system pumps against self-damage before water flow is initiated; and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

4. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be a feature of the design.

- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and vaves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capabilty to initiate the AFW system from the control room.

5. The licensee should prepare a procedure that assures that the operator manually connacts the motor-drivan pump train to the bus powered by the emergency diesel generator following loss of offsite power.

6. Since values FW 744 and 745 in one of the AFW pump discharge headers are normally open (see Figure 1), a postulated break in this header would cause loss of the capability to provide AFW flow to both steam generators. The licensee should re-evaluate the position of these valves considering such a postulated pipe break to revise the valve alignment to reduce the impact of such an event on the AFW capability (e.g., close valves FW 744 and FW 745).

X.3.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability ot this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed evironmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578: "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operation,should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.3.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-1 - Licensees with plants having a manual starting AFW system should install a system to automatically initiate the AFW system flow. This system and associated automatic initiation signals

should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation. (Note: This recommendation is applicable to the motor-driven AFW pump subsystem upon the loss of offsite AC power).

2. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

Liensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s) should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure. The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

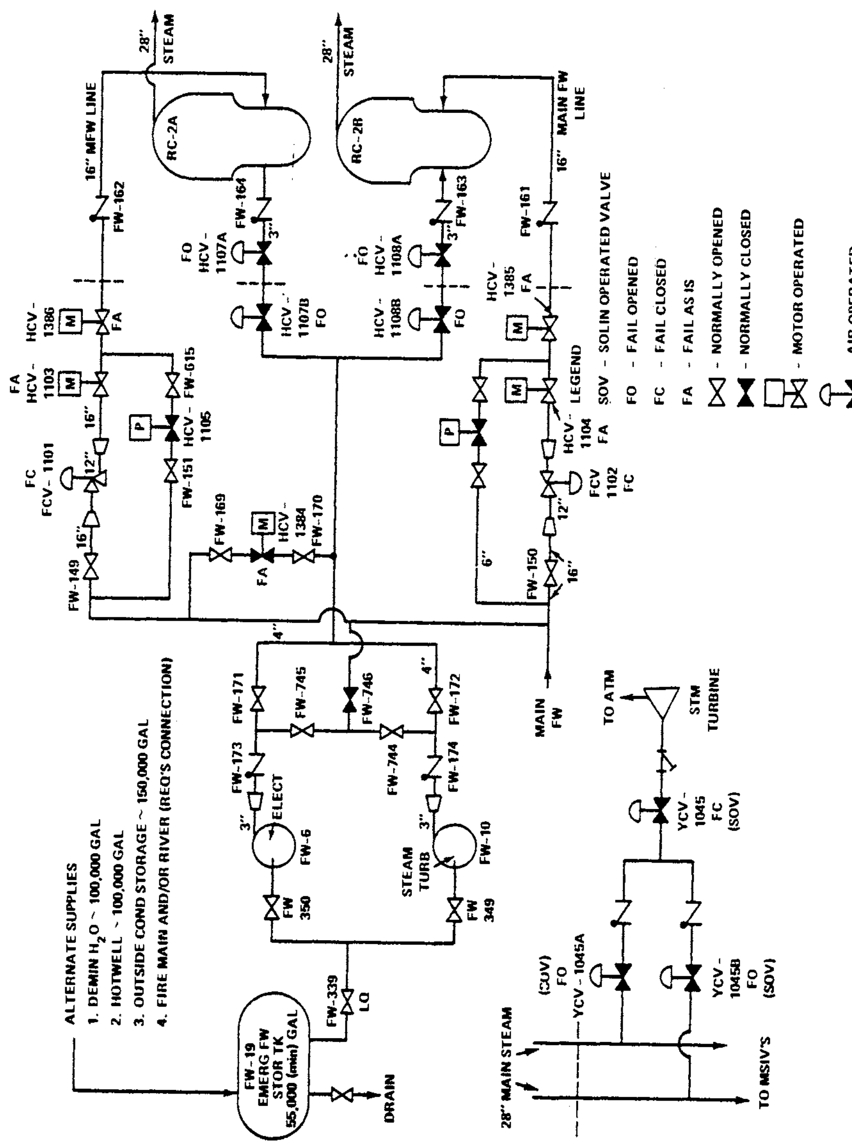
3. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

4. Recommendation - The licensee should evaluate the following concerns:

a. The discharge lines of both AFW pumps combine into a single header through which all AFW water must flow. A pipe break in this single flow path could result in the loss of the entire AFW system function.

b. The Ft. Calhoun AFW system design does not meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position 10-1; namely, that the AFW system should maintain the capability to supply the required AFW flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines concurrent with a single active failure.

The licensee should evaluate the postulated pipe breaks stated above and (1) determine any AFW system design changes or procedures necessary to detect and isolate the break and direct the required feedwater flow to the steam generator(s) before they boil dry or (2) describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.



Auxiliary Feedwater System
Ft. Calhoun
Figure 1

ENCLOSURE 2

BASIS FOR AUXILIARY FEEDWATER SYSTEM FLOW REQUIREMENTS

As a result of recent staff reviews of operating plant auxiliary feedwater systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of offsite and onsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above.

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

   - Maximum RCS pressure (PORV or safety valve actuation)
   - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperatur
   - RCS cooling rate limit to avoid excessive coolant shrinkage
   - Minimum steam generator level to assure sufficient steam generator heat trans fer surface to remove decay heat and/or cooldown the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant conditions considered in 1.a. above including:
   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.
   b. Time delay from initiating event to reactor trip.
   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).
   d. Minimum steam generator water level when initiating event occurs.
   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.
   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.
   g. Minimum number of steam generators that must receive AFW flow, e.g., 1 of 2, 2 of 4?
   h. RC flow condition - continued operation of RC pumps or natural circulation.
   i. Maximum AFW inlet temperature.
   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accomodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.
   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.
   l. Operating condition of steam generator normal blowdown following initiating event.
   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.
   n. Time at hot standby and time to cooldown RCS to RHR (or SCS) system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 18, 1979

Docket No. 50-309

Mr. Robert H. Groce
Licensing Engineer
Maine Yankee Atomic Power Company
20 Turnpike Road
Westborough, MA 01581

Dear Mr. Groce:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT THE MAINE YANKEE ATOMIC POWER STATION

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Combustion Engineering-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Combustion Engineering report CEN-114-P (Amendment 1-P)

entitled, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

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As stated

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ENCLOSURE 1

X.4 (CE) **MAINE YANKEE**

AUXILIARY FEEDWATER SYSTEM

X.4.1 **System Description**

X.4.1.1 **Configuration, Overall Design**

A simplified diagram of the Maine Yankee auxiliary feedwater system (AFWS), is shown in Figure 1. The principal components of the AFWS include two motor-driven pumps and one turbine-driven pump each supplying flow to the three steam generators (SG). Each motor-driven pump and the turbine-driven pump has a rated capacity of 500 gpm @ 1100 psia head. All three pump discharge lines feed a common header such that any one pump can deliver flow to all or any SG. The plant can be cooled to the temperature at which the shutdown cooling system can be used to bring the plant to a safe shutdown by any one pump and one SG. The licensee estimates that the steam generators would boil dry in approximately 30 minutes and 15 minutes as a result of a loss of main feedwater due to the loss of offsite power, and in the event of loss of main feedwater without loss of offsite power, respectively.

The primary water source for the AFWS is the demineralized water storage tank (DWST) having a total storage capacity of 150,000 gallons, 100,000 of which are dedicated to the AFWS. The licensee states that this dedicated inventory is sufficient for about 6 hours of decay heat removal, and that this is ample time to allow for replenishing of the supply from backup sources, if needed.

The seconoary water source is the primary water storage tank with a capacity of 150,000 gallons. This source is normally isolated from the system by locked closed valves. The licensee estimates that manual actuation to open the valves would take five minutes for each valve. A low level in the DWST is alarmed in the control room to alert the operator to connect the secondary source.

Both water sources are backed up by the primary water treatment plant which can supply 300 gpm to the primary or secondary source. The fire protection water system, which has a storage of 3 million gallons, can also be used to provide AFW for an extended period of time. Connection of the fire protection water system to the AFWS is accomplished by a fire hose connection which the licensee estimates takes one-half hour to accomplish.

The two AFW motor-driven pumps are located in the AFW pump house and the AFW turbine driven pump is located in in the main steam and feedwater valve area. Both rooms that contain the AFW pumps are cooled by the normal room ventilation system. The three main steam lines, one from each SG, pass through the main steam and feedwater valve area containing the turbine-driven pump. Thus, a steam

line break in this room could disable the turbine-driven pump. The second motor-driven pump was added as a result of the NRC high energy pipe line break criteria of 1973. All three AFW pumps have a self-contained, self lubricating oil system.

X.4.1.2 Components

The AFW system, including components and piping are safety-grade, seismic Category I and are located in tornado-missile proof buildings. The primary water supply system, including the demineralized water storage tank is seismically designed and is Safety Class 3 up to the regulating valves and Class 2 from the regulating valves to the steam generators. All other water sources are of non-seismic design and non-safety grade.

X.4.1.3 Power Sources

The power supplies and instrumentation are Class 1E. The two motor-driven auxiliary feedwater pumps are supplied from two independent alternating current (AC) power emergency buses. The three pneumatic-operated discharge flow control valves associated with the three steam generators receive control power from the same AC vital instrument bus (Division IV) and fail open. This bus is normally supplied by a corresponding direct current (DC) power emergency bus via an inverter; however, by transfer switch operation, this bus may be connected to the Division I bus.

The steam for the turbine-driven pump is received from the main steam system via a series of valves as shown in Figure 1. Three of these valves are pneumatic diaphragm air-operated valves. One of these air-operated valves fails closed and is designed to close on high containment pressure (5 psig). The other two air-operated valves fail open; one receives its control power from the Division IV vital AC bus and the other from a DC bus.

X.4.1.4: <u>Instrumentation and Controls</u>

X.4.1.4.1 <u>Controls</u>

The auxiliary feedwater system is manually initiated and controlled from the control room upon the loss of the main feedwater system. Steam generator level indication (narrow and wide range) are available to the operator in the control room. The narrow range level channels are designed in accordance with protection system requirements. Although the wide range level channel is not considered to be safety-related, it is powered from the vital AC instrument buses.

X.4.1.4.2 <u>Information Available to Operator</u>

System information available to the operator in the control room to assess the performance of the auxiliary feedwater system is as follows:

- Motor breaker position indicating lights associated with each motor-driven auxiliary feedwater pump.

- Motor amperage for each motor-driven AFW pump.

- Control signal indication (0-100%) to each air operated auxiliary feedwater flow control valve.

- Auxiliary feedwater flow path discharge pressure indicating light.

- Demineralized water storage tank water level indication.

- Steam generator levels.

X.4.1.4.3 <u>Initiating Signals for Automatic Operation</u>

The auxiliary feedwater system is initiated manually from the control room.

X.4.1.5 <u>Testing</u>

The power supplies are tested monthly. Diesel generators are started and connected to the bus and operated for 2 hours at full load. The AFW pumps, the system valves, and instrumentation are tested quarterly in accordance with Technical Specifications. Full flow testing is performed upon startup and shutdown.

X.4.1.6 <u>Technical Specification</u>

The following aspects of the Maine Yankee Technical Specifications, including Limiting Conditions of Operation, are applicable to the AFWS.

1. The following conditions must be met for a steam generator to be considered operable for decay heat removal.

   A. The reactor coolant system must be closed and pressurized to 100 psi above saturation pressure.

   B. The steam generator must have both the cold and hot leg stop valves fully open.

   C. The steam generator water level must be above the top of the tube bundle.

   D. An inventory of over 100,000 gallons of primary grade feedwater must be available.

   E. A feed pump must be operable.

2. The reactor shall not be maintained in a power operating condition unless the following conditions are met to assure post shutdown heat removal capability.

   A. Two steam generator feed pumps are operable.

   B. An inventory of over 100,000 gallons of primary grade feedwater is available.

   Exception: If either steam generator auxiliary feed pump becomes inoperable continued power operation is permitted for a maximum of 7 days. In this stuation, the operable feed pump is to be tested once a day.

X.4.2 <u>Reliability Evaluation</u>

X.4.2.1 <u>Dominant Failure Modes</u>

The following failure modes were found to dominate the demand unavailability of the Maine Yankee AFWS:

- <u>Loss of Feedwater (LOFW) With Offsite Power Available</u>
  Failure of the operator to start the AFW system was assessed to be the dominant failure mode for this transient. To start the system the operator must start one of the 3 pumps and open one of the 3 air operated flow control valves. The licensee estimates that it will take approximately 15 minutes to boil dry the SG for this transient followed by reactor trip.

X.4.2.1.2 LOFW With Only Onsite Power Available

Failure of the operator to start the AFW system, as in the previous case, was assessed to be the dominant failure mode for this transient. The same system startup procedure as above is followed. The licensee estimates that it will take approximately 30 minutes to boil dry the SG for this transient followed by a reactor trip.

X.4.2.1.3 LOFW With Only DC Power Available

The dominant fault contributors for this event are failure of the turbine-driven AFW train due to hardware malfunctions or maintenance errors or failure of the operator to start the system. Another potential fault contributor for this event is the CIS isolation valves. On total loss of AC, the CIS isolation valve, which fails closed, is held open by an air accumulator supplied by an AC powered air compressor. If sufficient air leaks through the seals, the valve could fail closed, thereby stopping steam flow to the AFWS turbine and in turn stopping AFW flow. It is presently unknown whether the operator can manually open this valve locally.

X.4.2.2 Potential Interactions

Potential interactions between systems that could affect AFWS operation include:

- Interaction with the Containment Isolation System - Upon a 5 psig containment pressure or an air or power failure to the air operated CIS isolation valve will cause the CIS isolation valve

to close. Such closing blocks steam flow to the turbine-driven AFW pump, thereby rendering the steam turbine drive AFWS pump ineffective for those situations which necessitate the use of said pump.

- Interactions With the Power Supply System - All AFWS flow control valves derive their control power from the same vital instrument bus. Upon the loss of power on this bus, these valves would fail open (safe); however, since the actual response of these valves is unknown under a degraded bus condition, there is a potential for adverse valve response.

- Piping System Interactions - Since all three trains feed a common header, a break in this line could cause the loss of the entire system.

X.4.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW systems reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL and plant-specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.4.3.1 Short Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be incorporated into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as as follows:

   Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

   The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

4. Recommendation GS-8 - The licensee should install a system to automatically initiate AFW system flow. For the short-term, this system need not be safety-grade; however, it should meet

the criteria listed below, which are similar to Item 2.1.7a of NUREG-0578. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-1.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits should be a feature of the design.
- The initiating signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

5. <u>Recommendation</u> - The licensee should propose a revision to the Technical Specification to require periodic AFWS operability testing on a monthly frequency rather than quarterly in conformance with current Standard Technical Specifications.

6. <u>Recommendation</u> - A pneumatic-operated valve in the steam supply line to the turbine-driven AFW pump, and the three pneumatic-operated AFW flow control valves derive their power from the same AC vital instrument bus. Although these valves are designed to fail open upon the loss of air or power, thereby assuring auxiliary feedwater flow to the steam generators upon such losses, it cannot be concluded that all failures will result in opening the valves. The consequences of voltage degradation should be analyzed as well as other failures (e.g., restricted air flow) to assure that such events would not incapacitate the auxiliary feedwater system. Establish suitable emergency procedures to assure AFWS function for such events. (See Long-Term Recommendation Number 3.)

7. The licensee should verify that the air accumulator will hold the containment isolation valve in the turbine driven pump steam supply line open for at least two hours following loss of all AC power.

X.4.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary flow to each steam generator should be provided in the control room.

   "The auxiliary feedwater flow instrument channels should be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.4.3.3 Long Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-1 - Licensees with plants having a manual starting AFW system, should install a system to automatically initiate the AFW system flow. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation.

2. Recommendation - GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

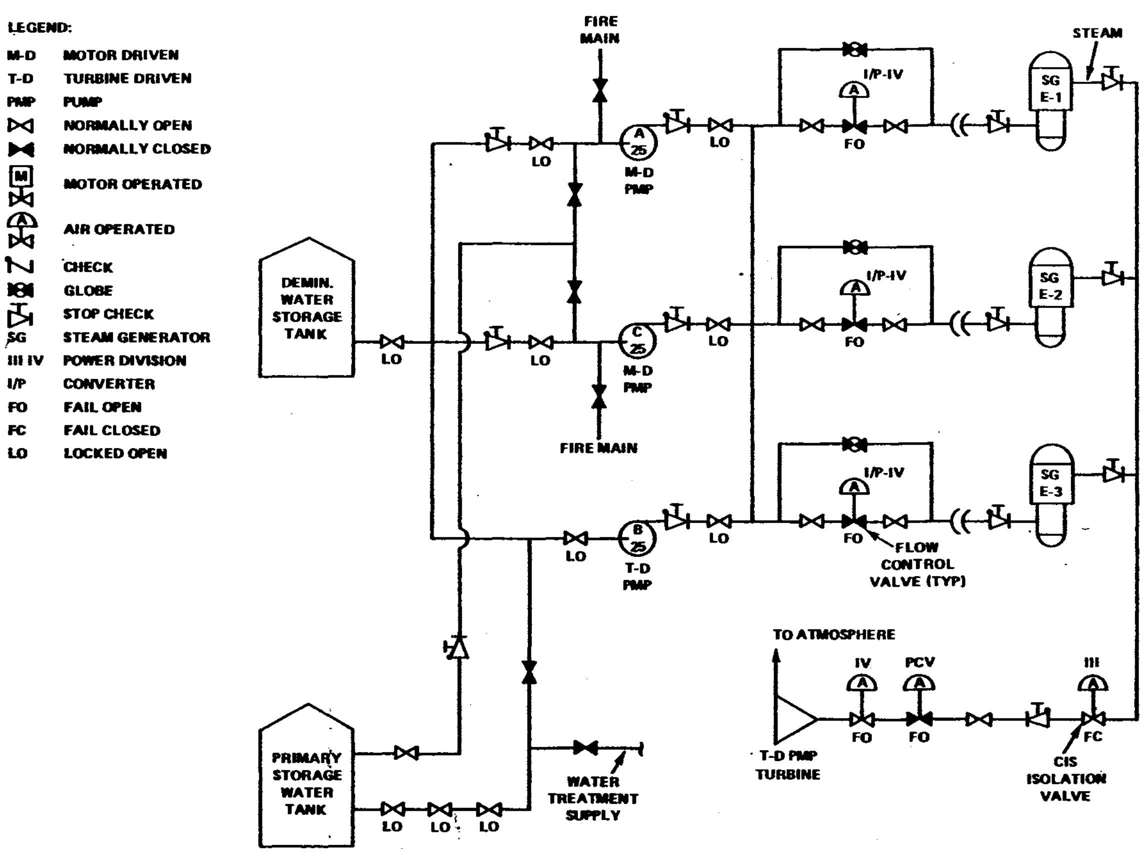
   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s) should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure. The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

3. Recommendation - Modify the AFWS design to eliminate the potential for adverse response of the three AFW flow control valves and one of the steam admission valves to the turbine pump due to degradation of power of the Division IV vital bus, e.g., provide service to these valves from different Division.

4. Recommendation - The licensee should evaluate the following concerns:

   a. A pipe break in the auxiliary feedwater system common discharge header could result in the loss of auxiliary feedwater system function even without a postulated single active failure. The licensee indicated that in such an event the auxiliary feedwater can be manually routed through the main feedwater lines to the steam generators.

   b. In the event of a steam or feedwater line break (main or auxiliary) the isolation of the auxiliary feedwater flow paths to the affected steam generator is accomplished manually. The licensee should evaluate the postulated pipe breaks stated above and (1) determine any AFW system design changes or procedures necessary to detect and isolate the break and direct the required feedwater flow to the steam generator(s) before they boil dry or (2) describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.



Auxiliary Feedwater System
Maine Yankee
Figure 1

ENCLOSURE 2

BASIS FOR AUXILIARY FEEDWATER SYSTEM FLOW REQUIREMENTS

As a result of recent staff reviews of operating plant auxiliary feedwater systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of offsite and onsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above.

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:
   - Maximum RCS pressure (PORV or safety valve actuation)
   - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperatur
   - RCS cooling rate limit to avoid excessive coolant shrinkage
   - Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cooldown the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant conditions considered in 1.a. above including:
   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.
   b. Time delay from initiating event to reactor trip.
   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).
   d. Minimum steam generator water level when initiating event occurs.
   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.
   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.
   g. Minimum number of steam generators that must receive AFW flow, e.g., 1 of 2, 2 of 4?
   h. RC flow condition - continued operation of RC pumps or natural circulation.
   i. Maximum AFW inlet temperature.
   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accomodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.
   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.
   l. Operating condition of steam generator normal blowdown following initiating event.
   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.
   n. Time at hot standby and time to cooldown RCS to RHR (or SCS) system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 22, 1979

Docket No. 50-336

Mr. W. G. Counsil, Vice-President
Nuclear Engineering & Operations
Northeast Nuclear Energy Company
P. O. Box 270
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Dear Mr. Counsil:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT MILLSTONE NUCLEAR POWER STATION UNIT 2

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Combustion Engineering-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Combustion Engineering report CEN-114-P (Amendment 1-P) entitled, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Robert L. Ferguson/for

Darrell G. Eisenhut, Acting Director
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ENCLOSURE 1

X.5 (CE) MILLSTONE 2

AUXILIARY FEEDWATER SYSTEM

X.5.1 System Description

X.5.1.1 Configuration and Overall Design

The auxiliary feedwater system (AFWS) is designed to supply water to the steam generators (SG) for reactor coolant system decay heat removal when the main feedwater system is not available. It is also used for plant startups and shutdowns below the power level where the main feedwater system is not required.

The AFWS is shown in simplified form on Figure 1. The system consists of a steam turbine-driven pump having a 600 gpm capacity, and two motor-driven pumps each having a 300 gpm capacity. The steam supply to the turbine is obtained from a common line connected to lines coming from each of two steam generators. The AFWS is normally aligned as indicated on Figure 1, the motor-driven pumps supplying No. 1 SG and the turbine driven pump supplying No. 2 SG.

A condensate storage tank (CST) of 250,000 gallons capacity is the primary source of water for the AFWS, and the primary water storage tank (PWST) of 150,000 gallons capacity is the secondary source. Another back-up source consists of two 250,000 gallon fire storage water tanks. In addition, a connection to the city water supply exists which can be used to provide AFW for an extended period of time, if required.

The AFWS is manually actuated from the control room. The pumps and appropriate valves can be controlled from the control room and from the remote shutdown panel.

X.5.1.2 Component Design Classification

The pumps, motors and piping associated with the AFWS are designed to seismic Category I requirements. The CST is not designed to seismic Category I requirements; however, a seismic Category I missile barrier surrounds the CST. This barrier will contain the water in the event of a CST tank failure.

X.5.1.3 Power Sources

The motor-driven pumps are supplied from separate Class 1E emergency buses. All motor operated valves (MOVs) associated with the AFWS are powered from the 480V AC emergency buses and fail as-is.

Steam generator level instrumentation, AFWS pump breaker and valve controls are powered from their associated Class 1E emergency buses. Although the AFWS instruments and associated wiring are not Class 1E, they are powered from Class 1E emergency buses..

The steam for the turbine-driven pump is received from the main steam system via a series of valves as shown in Figure 1. Steam is introduced to the turbine via a normally closed motor operated steam admission valve, and steam flow is regulated by a turbine throttle valve in series with the admission valve.

X.5.1.4 Instrumentation and Control

X.5.1.4.1 Controls

The AFWS can be controlled from either of two control stations, one at the main control room;the other at the remote shutdown panel.

X.5.1.4.2 Information Available to the Operator

The following indications are available, except as indicated, at both control stations:

1. SG level
2. Pump turbine RPM (control room only)
3. Pump motor current (control room only)
4. MOV valve positions
5. Pump motor breaker position
6. CST level
7. PWST level (control room only)
8. Auxiliary feed flow
9. Pump discharge pressure

The following alarms annunciate at both control stations:

1. CST low level
2. SG low level

X.5.1.4.3 Initiating Signals for Automatic Operation

Not applicable since AFW is manually initiated.

X.5.1.5 Testing

The systems are tested monthly in accordance with plant Technical Specification requirements. In addition to the periodic testing, the systems are retested in the recirculation mode in accordance with the surveillance tests subsequent to performing maintenance.

The systems are tested using the recirculating lines, at which time discharge pressures and pump motor currents are monitored. In addition, valve positions are verified monthly.

The licensee uses the system routinely during startup and shutdown thus verifying valve positions.

X.5.1.6 Technical Specifications

The Limiting Condition For Operation (LCO) for the system is 48 hours upon a failure of one of the AFWS trains (e.g., a pump motor failure). If the affected AFWS train is not restored within 48 hours, the unit must be brought to a hot shutdown in the next 12 hours.

A review of the Technical Specifications indicated that these specifications cover LCOs and periodic surveillance testing consistent with current Standard Technical Specifications.

X.5.2 Reliability Evaluation Results

X.5.2.1 Dominant Failure Modes

Failure modes of the AFWS were assessed for three loss of main feedwater initiating events. The dominant failure modes for each transient type are discussed below.

- Loss of Main Feedwater (LOFW) with Offsite Power Available

The dominant failure mode of the AFWS for this transient is failure of the operator to manually actuate the system. Upon the loss of main feedwater, the licensee estimates that the operator has 15 to 45 minutes, depending on the initiating transient, to actuate the AFWS before the steam generators would boil dry. Because of this time restriction, failure to perform the required actuation prior to boiling the SG dry has been assessed to be the dominant failure mode for this transient.

- LOFW With Only Onsite AC Power Available

This transient is very similar to the transient discussed above, except that the offsite AC power system is not available. Additional failure modes related to the onsite AC power system were considered; however, these did not have a significant impact on the dominant failure mode. As such, the dominant failure mode discussed above (i.e., failure of the operator to actuate the AFWS) is also dominant for this transient.

- LOFW with Only DC Power Available

For this event no AC power (onsite or offsite) is available; therefore, the AFWS is reduced to the steam-driven pump train. Failures which can fail this train include hardware failures of the pump or valves, maintenance outages, and human errors.

The dominant failure mode for this event is failure of the operator to manually open two normally closed valves (the steam admission valve and the AFW discharge valve) in the turbine-driven train within the aforementioned 15 to 45 minutes after the demand. The valves are AC motor-operated and are normally powered from offsite power or from the diesel-generators on loss of offsite AC power. Since neither of these power sources is available during this event, local manual opening of the valves is required.

X.5.2.2 Principal Dependencies

The most significant dependency found in this evaluation is the dependence on operator action to actuate the AFWS on demand.

The second significant dependency found is the dependence on AC power to actuate certain portions of the steam-driven pump train of the AFWS. This dependency is the dominant contributor to AFWS unavailability upon the total loss of AC power.

Location dependencies, such as component proximity to high energy lines, were considered but do not appear to be significant.

X.5.3 Recommendations

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system availability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant-specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.5.3.1 Short-Term

1. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

2. Recommendation GS-5 - The plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

3. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:
   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

4. Recommendation GS-8 - The licensee should install a system to automatically initiate AFW system flow. For the short-term, this system need not be safety-grade; however, it should meet the criteria listed below, which are similar to Item 2.1.7a of NUREG-0578. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-1.
   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

   - Testability of the initiating signals and circuits should be a feature of the design.
   - The initiating signals and circuits should be powered from the emergency buses.
   - Manual capability to intiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
   - The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
   - The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to intiate the AFW system from the control room.

X.5.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator should be provided in the control room.

   "The auxiliary feedwater flow instrument channels should be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

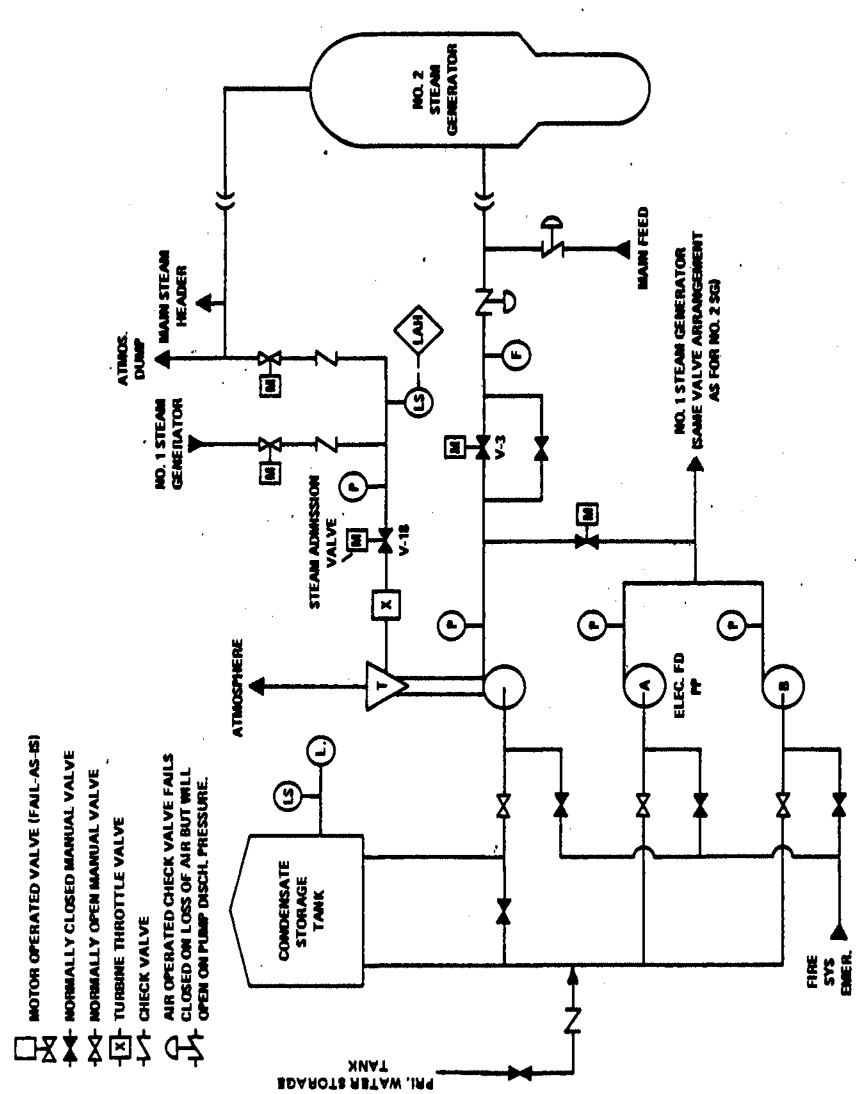
X.5.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-1 - Licensees with plants having a manual starting AFW system should install a system to automatically initiate the AFW system flow. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation.

2. <u>Recommendation</u> - GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being independently of any alternating current power source for at least two hours. Conversion of direct power to alternating current is acceptable.

AUXILIARY FEEDWATER SYSTEM



Auxiliary Feedwater System
Millstone 2
Figure 1

ENCLOSURE 2

BASIS FOR AUXILIARY FEEDWATER SYSTEM FLOW REQUIREMENTS

As a result of recent staff reviews of operating plant auxiliary feedwater systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
    1) Loss of Main Feed (LMFW)
    2) LMFW w/loss of offsite AC power
    3) LMFW w/loss of offsite and onsite AC power
    4) Plant cooldown
    5) Turbine trip with and without bypass
    6) Main steam isolation valve closure
    7) Main feed line break
    8) Main steam line break
    9) Small break LOCA
    10) Other transient or accident conditions not listed above.

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:
    - Maximum RCS pressure (PORV or safety valve actuation)
    - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperatur
    - RCS cooling rate limit to avoid excessive coolant shrinkage
    - Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cooldown the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant conditions considered in 1.a. above including:
   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.
   b. Time delay from initiating event to reactor trip.
   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).
   d. Minimum steam generator water level when initiating event occurs.
   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.
   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.
   g. Minimum number of steam generators that must receive AFW flow, e.g., 1 of 2, 2 of 4?
   h. RC flow condition - continued operation of RC pumps or natural circulation.
   i. Maximum AFW inlet temperature.
   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accomodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.
   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.
   l. Operating condition of steam generator normal blowdown following initiating event.
   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.
   n. Time at hot standby and time to cooldown RCS to RHR (or SCS) system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 30, 1979

Docket No. 50-255

Mr. David Bixel
Nuclear Licensing Administrator
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Dear Mr. Bixel:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT PALISADES NUCLEAR PLANT

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Combustion Engineering-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Combustion Engineering report CEN-114-P (Amendment 1-P) entitled, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

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ENCLOSURE 1

X.6 (CE) PALISADES

AUXILIARY FEEDWATER SYSTEM (AFWS)

X.6.1 System Description

X.6.1.1 Configuration - Overall Design

A simplified flow diagram of Palisades AFWS is presented in Figure 1. The AFWS includes a motor-driven pump and a turbine-driven pump, each capable of supplying 100% flow requirements for decay heat removal. Each pump has a capacity of 415 gpm at 2730 feet. The pumps discharge to a common AFWS header which branches and connects to the main feed headers to each steam generator (SG). Only one SG is needed to cool the plant down to the temperature where the Shutdown Cooling system (SCS) can be used to bring the plant to safe shutdown. The licensee estimates that the steam generator would boil dry in 15 minutes without AFW flow following the worst case condition of loss of main feedwater with reactor trip.

The two pumps are located in the same room and could be rendered inoperable as a result of a pipe break causing the flooding of the room.

The primary source of water for the AFWS is a 125,000 gallon condensate storage tank. In addition, 75,000 gallons of water from the primary system make-up storage tank and 275 gpm from the make-up demineralizer system can be supplied to the condensate storage tank via pneumatic-operated valves which are opened from the control room. The technical specifications

require that a total of 100,000 gallons of water inventory be available for the AFWS. The licensee estimates that this inventory is approximately an 8 hour supply. The condensate storage tank is also connected to the main condenser hotwell through two make-up valves connected in parallel.

An alternate, long-term source of water to the AFWS, if needed, is from Lake Michigan and is directed to the AFWS pump suctions via the three fire protection system pumps. Two of the fire pumps are driven by dedicated diesel engines and the other pump is driven by an electric motor which is powered from one of the two station emergency diesel generators.

X.6.1.2 Components - Design Classification

The condensate storage tank is the only source of AFWS water which has a seismic Category 1 classification. The steam turbine-driven auxiliary feedwater pump and associated steam inlet valves and piping is designed to withstand a 0.05 earthquake as stated in Appendix A of the FSAR. The motor-driven auxiliary feedwater pump and associated piping and valves are classified seismic Category 1.

X.6.1.3 Power Sources

The motor-driven auxiliary feedwater pump is supplied from one of the two AC emergency buses. The turbine-driven pump can receive motive-power steam from either steam generator.

The pneumatic-operated valves in the discharge header of both pumps receive control power from separate AC vital instrument buses. These buses are normally supplied from an AC emergency bus and backed up by the corresponding DC emergency bus via an inverter. The pneumatic-operated valves in each steam line from the steam generator to the turbine-driven auxiliary feedwater pump are controlled from independent DC emergency buses.

X.6.1.4 Instrumentation and Controls

X.6.1.4.1 Controls

The AFWS is manually initiated and feedwater flow to the steam generator(s) is manually controlled from the control room. Steam generator level indication (narrow range only) is available to the operator in the control room. The narrow range level channels are designed in accordance with protection system requirements.

X.6.1.4.2 Information Available to Operator

System information available to the operator in the control room to assess the performance of the auxiliary feedwater system is as follows:

- Status indicating lights for the motor driven auxiliary feedwater pump.
- Position indication of auxiliary feedwater flow path control valves.

- Primary and secondary source water level indications.
- Auxiliary feedwater flow indication.
- Auxiliary feedwater pressure indication.
- Steam pressure at inlet of turbine driven auxiliary feedwater pump.
- Steam generator level.

X.6.1.4.3 Initiating Signals for Automatic Operation

The auxiliary feedwater system is initiated manually from the control room.

In the event of a steam or feedwater (main or auxiliary) line break, isolation of the auxiliary feedwater flow paths to the affected SG is accomplished manually.

Main steam line break isolation is accomplished automatically by the MSIV whereas feedwater line break isolation is accomplished manually.

A turbine trip will result in a reactor trip if reactor power is initially above 15 percent of rated power. A reactor trip will always result in a turbine trip.

X.6.1.5 Testing

Subsequent to the completion of this review, the license has been amended to incorporate new Technical Specification requirements as follows:

APPLICABILITY

Applies to periodic testing requirements of the turbine-driven and motor-driven auxiliary feedwater pumps.

OBJECTIVE

To verify the operability of the auxiliary feedwater system and its ability to respond properly when required.

SPECIFICATIONS

a. The operability of the motor- and steam-driven auxiliary feed pumps shall be confirmed as required by Specification 4.3c.[1/]
b. The operability of the auxiliary feedwater pumps' discharge valves CV-0736A and CV-0737A shall be confirmed at least every three (3) months.

[1/] Specification 4.3c reads as follows: Inservice testing of ASME Class 1, 2 and 3 pumps, as determined by 10 CFR 50, Section 50.55a and Regulatory Guide 1.26 shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code with applicable addenda as required by 10 CFR 50, Section 50.55a(g), except where specific relief has been granted by the NRC.

X.6.1.6 Technical Specifications

The limiting conditions of operation are in accordance with the Technical Specifications as follows:

Steam and Feedwater Systems

Applicability

Applies to the operating status of the steam and feedwater systems.

Objective

To define certain conditions of the steam and feedwater system necessary to assure adequate decay heat removal.

Specifications

The primary coolant shall not be heated above 325°F unless the following conditions are met:

a. Both auxiliary feedwater pumps operable or one auxiliary feedwater pump and one fire pump operable.

b. A minimum of 100,000 gallons of water in the condensate storage and primary coolant system makeup tanks combined and a backup source of additional water from Lake Michigan by the operability of one of the fire protection pumps.

c. All valves, interlocks and piping associated with the above components required to function during accident conditions, are operable.

d. The main steam stop valves are operable and capable of closing in five seconds or less under no-flow conditions.

The licensee has committed to implement the following plant operating procedures, until an approved Technical Specification revision in this regard is established:

With the primary coolant system temperature greater than 325°, both auxiliary feedwater pumps and one fire pump will be operable except as follows:

a. One auxiliary feedwater pump may be inoperable for a period of 72 hours.

b. The firewater makeup to the auxiliary feedwater pump suction may be inoperable for a period of 72 hours.

If an inoperable auxiliary feedwater pump is not restored to service in 72 hours, the plant will be placed in hot shutdown within the next 12 hours.

The licensee has proposed Technical Specification changes in this regard which are under staff evaluation.

X.6.2 Reliability Evaluation Results

The Palisades AFWS consists of two full capacity subsystems, either of which, when delivering its pump capacity, can provide for adequate decay heat removal. The system is manually actuated from the plant control room. The failure modes expected to dominate the overall demand unavailbility of the AFWS were assessed given three transient events for which operation of the AFWS would be required. The dominant failure modes for these three transient events are summarized below:

X.6.2.1 Loss of Main Feedwater (LOFW)

The dominant failure mode assessed for the AFWS design for this transient was failure of the plant operator to start at least one of the AFW system trains. This potential failure mode was estimated to contribute roughly 90% to the AFWS unavailability. The next level of dominant failure modes identified was principally composed of double faults. These double faults included: (1) failures in the turbine and electric pump trains due to hardware faults or allowed maintenance outages, and (2) inadvertent closure of manual valve (A) from the condensate storage tank to the pumps sucton and not reopening this valve or, as backup, the operator failing to activate the fire water supply to the AFWS before pump damage occurs.

X.6.2.2 LOFW and Loss of Offsite AC Power (Only Onsite AC Power Available)

The dominant faults identified for this transient were essentially the same as described above.

X.6.2.3 LOFW with Only DC Power Available

The dominant failure modes identified for this given event were (1) operator failing to actuate system from the control room and (2) demand failures in the turbine train due to single hardware faults and to the allowed outage time for this train of AFWS. For this LOFW event, it was noted that air operated valves (E) and (F) in the steam supply line for the turbine driven pump could eventually fail close after being actuated open. These valves fail closed on loss of air supply and with time, the air supply in the air accumulator could decay to a point where the valves would close. The operator can, however, manually open these valves (locally) and reestablish operation of the turbine driven pump. If the plant operator fails to do this, the AFWS will experience delayed failure.

X.6.2.4 Principal Dependencies/Interaction Identified

The principal dependencies are described above. One other potential interaction identified is due to the fact that both AFWS pumps are located in the same room. Thus, the pumps appear vulnerable to flooding. No high energy lines were said to exist within this room; however, the room has flooded in the past to a depth of about six inches due to lack of flow through the floor drain. The room is current being inspected for flooding every shift.

X.6.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant-specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.6.3.1 Short-Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - Those in which the primary water supply is not initially available. The procedures for this case should include any operator actions

required to protect the AFW system pumps against self-damage before water flow is initiated; and,

- In case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

    Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

    The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

4. Recommendation GS-8 - The licensee should install a system to automatically initiate AFW system flow. For the short-term, this system need not be safety-grade; however, it should meet the criteria listed

below, which are similar to Item 2.1.7a of NUREG-0578. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-1.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiating signals and circuits should be a feature of the design.
- The initiating signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

5. Recommendation The licensee should verify that the air accumulator will hold the air operated valves in the turbine driven pump steam supply line open for at least two hours following loss of all AC power.

6. Recommendation - The motor driven pump and the pneumatic-operated valve(G) through which AFW flow to steam generator A is controlled receive motive and control power from Division I emergency buses. Pneumatic valve (F), which supplies steam from steam generator A to the turbine driven AFW pump, receives control power from a Division II emergency bus. Similarly AFW flow control valve (H) and steam supply valve (E) associated with steam generator B receive power from Division II and I emergency buses respectively. Upon loss of air or power, the AFW flow control valves (G) and (H) fail open and the turbine driven pump steam admission valves (E) and (F) fail closed. It is recognized that the AFW flow control valves are designed to fail open upon loss of air or power so that AFW flow to the steam generators should be assured. However, it cannot be concluded that all failures will result in opening these valves. Degradation of Division I buses could potentially result in loss of the entire AFW system. The licensee should analyze the consequences of Division I voltage degradation as well as other failures (e.g., restricted air flow) to assure that there is no Division I failure mode that can result in loss of the entire AFW system. Until this analysis is

completed or the AFW system is modified to preclude such an occurrence, emergency procedures should be prepared to retain AFW system capability. (See long term recommendation 4.b).

7. Recommendation - Each steam generator has two pneumatic-operated atmospheric steam dump valves connected in parallel. These four valves have the same controller which presumably receives power from only one source, and therefore is vulnerable to a single failure event. Concern was expressed to the licensee as to whether the steam supply to the turbine AFW pump is adversely affected by the potential simultaneous opening of all atmospheric dump valves due to a single failure at the controller or its power source. The licensee has indicated that the pressure drop across the valves is sufficiently large to assure adequate steam supply to the turbine driven pump from the steam generators. We require that the licensee provide analyses to confirm this assertion. (See long term recommendation 4a.)

X.6.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders

Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indiction of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic test on one AFW system train, and there is only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.6.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-1 - Licensees with plants having a manual starting AFW system, should install a system to automatically initiate the AFW

system flow. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation.

2. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water suplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s) should install redundant valve(s) from the alternate water supply upon low pump suction pressure. The licensee should propose Technical Specifiations to incorporate appropriate periodic inspections to verify the valve positions.

3. Recommendation - The licensee should evaluate the following concerns:

   a. The discharge lines of both AFW pumps combine into a single header through which all AFW water must flow. A pipe break in this single flow path could result in the loss of the entire AFW system function.

   b. The Palisades AFW system design does not meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position 10-1; namely, that the AFW system should maintain the capability to supply the required AFW flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines concurrent with a single active failure.

   The licensee should evaluate the postulated pipe breaks stated above and (1) determine any AFW system design changes or procedures necessary to detect and isolate the break and direct the required feedwater flow to the steam generator(s) before they boil dry or (2) describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.

4. The licensee should evalute the following concerns:

   a. Each steam generator has two pneumatic-operated atmospheric steam dump valves connected in parallel. These four valves have the same controller which presumably receives power from only one source. The consequences of single failures would be reduced by supplying power to the dump valves of each steam generator from separate power divisions. (See short-term recommendation 7).

b. This concern is a follow-up to that in short-term recommendation 6, (i.e., loss of the AFWS due to a degraded power system division). Valves (G) and (F) are both in AFWS train A but receive power from different DC divisions as do valves (E) and (H) which are in AFWS train B. Thus, the effect of degradation of one power division would be reduced by having valves (G) and (F) powered from the same division; similarly for valves (E) and (H).

c. Wide range steam generator level instrumentation is not provided in the control room. Evaluate the need for such instrumentation to facilitate proper operator action considering transients and accident conditions.

Based on the results of the above evaluations, the licensee should (1) determine any AFW system design changes necessary to mitigate the concern or (2) describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.

6.3.4 <u>Considerations Based on the Systematic Evaluation Program</u>

The following items are under review by the Systematic Evaluation Program (SEP) and supplement the above long-term recommendations.

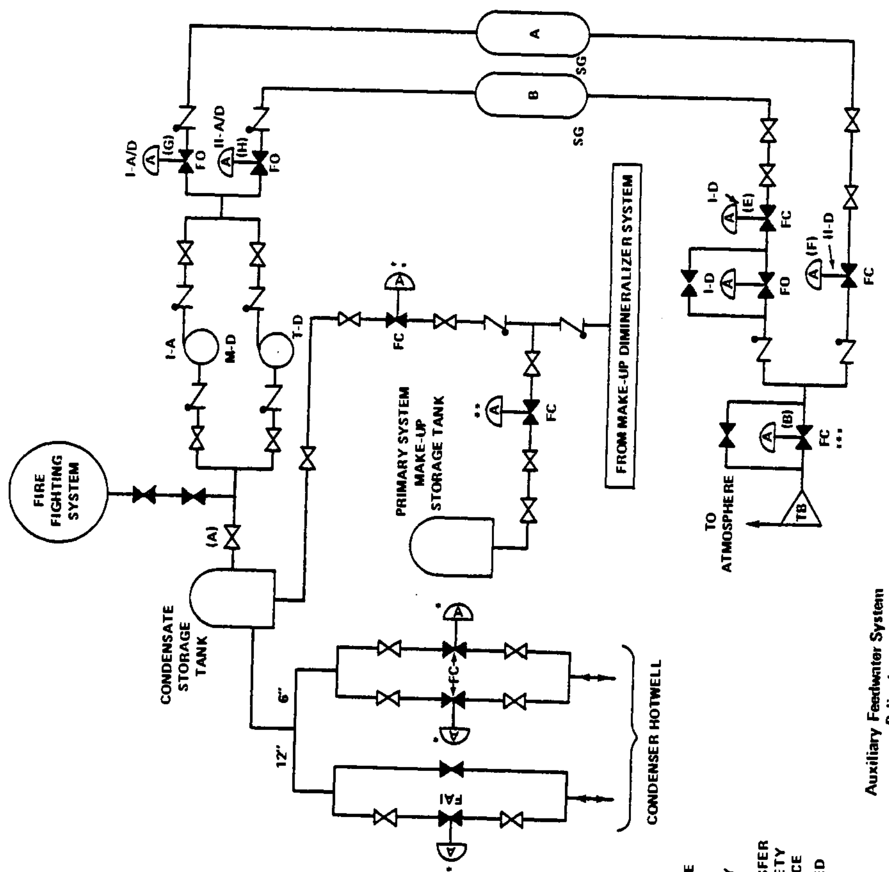
1. The Palisades Plant including the AFWS will be reevaluated during the SEP with regard to internally and externally generated missiles, pipe whip and jet impingement, quality and seismic design requirements, earthquakes, tornadoes, floods, and the failure of nonessential systems.

2. The staff will reassess the need for a water level alarm system in the AFWS pump room.

3. The Palisades AFWS is not automatically initiated and the design does not have capability to automatically terminate feedwater flow to a depressurized steam generator and provide flow to the intact steam generator. This is accomplished by the control room operator. The effect of this provision will be assessed in the main steam line break evaluation for Palisades.

4. A lack of system redundancy exists because the turbine-driven AFWS pump is not seismic Class 1. The staff will consider the need for upgrading the seismic classification of the pump in the SEP integrated assessment of Palisades.

5. The staff will assess the need for increasing the technical specification inventory limit for the seismic Class I AFWS water supply.



Auxiliary Feedwater System
Palisades
Figure 1

LEGEND:
- M-D MOTOR DRIVEN
- T-D TURBINE DRIVEN
- NORMALLY OPEN
- NORMALLY CLOSED
- AIR OPERATED
- SG STEAM GENERATOR
- I,II,III POWER DIVISIONS
- A ALTERNATING CURRENT
- D DIRECT CURRENT
- TB TURBINE
- FO FAIL OPEN
- FC FAIL CLOSE
- FAI FAIL AS IS

NOTES:
- • VALVES POWERED FROM SAME D-C EMERGENCY BUS
- •• VALVES POWERED NORMALLY FROM SAME NON SAFETY A-C BUS. MANUAL BACKUP TRANSFER CAPABILITY OF THE NON SAFETY BUS TO AN EMERGENCY SOURCE
- ••• NO CONTROL POWER REQUIRED

ENCLOSURE 2

BASIS FOR AUXILIARY FEEDWATER SYSTEM FLOW REQUIREMENTS

As a result of recent staff reviews of operating plant auxiliary feedwater systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of offsite and onsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above.

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:
   - Maximum RCS pressure (PORV or safety valve actuation)
   - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperat...
   - RCS cooling rate limit to avoid excessive coolant shrinkage
   - Minimum steam generator level to assure sufficient steam generator heat tr... fer surface to remove decay heat and/or cooldown the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant conditions considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

   g. Minimum number of steam generators that must receive AFW flow, e.g., 1 of 2, 2 of 4?

   h. RC flow condition - continued operation of RC pumps or natural circulation.

   i. Maximum AFW inlet temperature.

   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accomodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

   l. Operating condition of steam generator normal blowdown following initiating event.

   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

   n. Time at hot standby and time to cooldown RCS to RHR (or SCS) system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 17, 1979

Docket No. 50-335

Mr. Robert E. Uhrig, Vice-President
Advanced Systems and Technology
Florida Power and Light Company
P. O. Box 529100
Miami, Florida 33152

Dear Mr. Uhrig:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT ST. LUCIE UNIT 1

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Combustion Engineering-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Combustion Engineering report CEN-114-P (Amendment 1-P)

entitled, "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut
Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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ENCLOSURE 1

X.7 ST. LUCIE UNIT 1

AUXILIARY FEEDWATER SYSTEM

X.7.1 System Description

X.7.1.1 Configuration, Overall Design

A simplified flow diagram of the St. Lucie auxiliary feedwater system (AFWS) is shown in Figure 1. The AFWS consists of one full capacity turbine-driven pump (500 gpm @ 1200 psi) and two half capacity (250 gpm @ 1200 psi) motor-driven pumps. One turbine pump or both motor driven pumps are required to adequately remove decay heat. The turbine-driven pump supplies feedwater to two steam generators (SG) by means of two separate lines each with its own motor operated control valve. Each motor-driven pump normally supplies feedwater to one steam generator. A cross connection with two remote manual normally closed isolation valves is provided to enable the routing of feed flow of the two motor driven pumps to either steam generator. The AFWS is manually started from the control room. The AFW system can supply water to the SG(s), assuming a single active component failure with loss of offsite or onsite power. The licensee states that the AFWS is capable of cooling the plant down to the condition where the shutdown cooling system can be used to continue the safe plant shutdown process.

The primary water supply of the AFWS is maintained in a 250,000 gal.

seismic Category 1 condensate storage tank (CST) connected to the pumps' suction by redundant lines with locked open manual valves. A minimum of 168,000 gal. is reserved strictly for the AFWS by administrative control. The reserved water inventory is sufficient to maintain the plant at hot standby condition for 8 hours following a reactor trip, and subsequently cool the plant down to the shutdown cooling system cut-in temperature.

Low water level in the CST will alarm and annunciate in the main control room. The AFW pump suctions are connected only to the CST. Additional water may be supplied either from the SG Blowdown Monitor Storage Tank or the city water tanks via the CST as shown in Figure 1. Supplying water from these alternate sources requires considerable operator action and is estimated to take 3 hours to accomplish.

X.7.1.2 Components - Design, Classification

All components of the AFWS, including the primary water supply, are designed to seismic Category 1 requirements.

X.7.1.3 Power Sources

The steam turbine driven pump uses steam from the main steam lines taken upstream of each main steam isolation valve (MSIV) and exhausts to the atmosphere. The steam is supplied via an AC powered motor operated valve (MOV) from each steam generator. These valves are normally closed and fail as-is. Downstream of these valves there is a single

steam supply header with a DC powered MOV which is normally closed and fails as-is. The two motor driven pumps are powered from the Division A and B emergency diesel generators respectively in case of a loss of normal AC power.

X.7.1.4 Instrumentation and Controls

X.7.1.4.1 Controls

The control of auxiliary feedwater flow and steam generator water level is accomplished from the control room by remote manually operated control valves. A local control station is provided to facilitate plant shutdown if the control room is not accessible. All manually operated valves in the AFWS are locked open. The motor operated valves will fail in the "as-is" position.

X.7.1.4.2 Information Available to Operator

The important information available to the operator includes AFW discharge header flow, AFW discharge header pressure, CST level, steam generator level, steam pressure to steam driven AFW pump, and control valve position indication.

Additional information available is listed in the following instrument list:

Auxiliary Feedwater Parameters Available on RTGB 102. Vertical Section

1. Aux Feedwater Flow, Header 'A' FI-09-2A
2. Aux Feedwater Flow, Header 'B' FI-09-2B

3. Aux Feedwater Flow, Header 'C' FI-09-2C
4. Aux Feedwater Press. Header 'A' PI-09-8A
5. Aux Feedwater Press. Header 'B' PI-09-8B
6. Aux Feedwater Press. Header 'C' PI-09-8C
7. Steam Press. to Aux Feedpump 'C' PI-08-5
8. Condensate Storage Tank Level LIS-12-11
9. Aux Feed Pump '1A' Amperes
10. Aux Feed Pump '1B' Amperes

Auxiliary Feedwater Parameters on RTGB 102-Horizontal Section

1. AFW Pump 1A disch steam generator (SG) 1A MV-09-9. Switch and valve position lights.
2. AFW pump 1B disch to SG 1B MV-09-10 - switch and valve position lights.
3. AFW pump 1C disch to SG 1A MV-09-11 - switch and valve position lights.
4. AFW pump 1C Disch to SG 1B MV-09-12 - switch and valve position lights.
5. AFW pump 1A disch to SG 1B MV-09-13 - switch and valve position lights. (crossconnect valve)
6. AFW 1B disch to SG 1A MV-09-14 - switch and valve position lights. (crossconnect valve)
7. Start and stop switches for 1A, 1B and 1C Aux Feed Pumps and indicator lights.
8. 1C Aux Feed Pump speed controller and speed indicator.

9. 1C Aux Feed Pump steam inlet from 1A main steam line, MV-08-14 indicator lights.
10. 1C Aux Feed Pump steam inlet from 1B main steam line MV-08-13 indicator lights.
11. 1C Aux Feed Pump steam inlet MV-08-3 indicator lights.

X.7.1.4.3 Initiating Signals for Automatic Operation

The St. Lucie AFWS is a manually started system. In the event of a loss of main feedwater pumps or offsite power, followed by reactor trip the licensee estimates that the operator has approximately 13 minutes in which to start the AFW pump and open the AFW flow control valves to the steam generators to prevent the steam generators from boiling dry.

X.7.1.5 Testing

Each month the motor operated feed water valves are cycled from closed to full open to closed, after which each pump is started and operated at least 15 minutes. Specified minimum discharge pressure is verified while the pumps are operating. No manual valve lineup changes are required for this testing. Condensate storage tank level is verified at or above minimum at least once per 12 hours.

X.7.1.6 Technical Specifications

1. The two motor driven AFW pumps and the steam turbine driven AFW pump are all required to be operable when the reactor coolant system is above 325°F, the maximum operating temperature of the shutdown cooling system.

2. If any one pump is inoperable, it must be returned to operable status within 72 hours or the plant must be placed in hot standby within 12 hours.

3. If two or more pumps are inoperable, the plant must be in hot standby within 1 hour and in cold shutdown within 30 hours unless at least one pump is returned to operation and the unit is back under 2 above using the time intervals of the initial discovery.

4. The CST is required to have minimum volume of 116,000 gallons when the RCS temperature is above 325°F. If the volume is below minimum it must be restored within 4 hours or the plant must be in hot standby within the next 6 hours and in cold shutdown within the following 30 hours.

X.7.2 <u>Reliability Evaluation Results</u>

X.7.2.1 <u>Dominant Failure Modes</u>

The St. Lucie Unit 1 AFWS consists of two subsystems, i.e, one subsystem of two one-half capacity motor-driven pumps and another subsystem of a single full capacity turbine-driven pump. Either subsystem, when delivering its pumping capacity to at least one steam generator can provide for adequate decay heat removal for the three loss of main feedwater events considered.

The following failure modes were found to dominate the demand unavailability of the St. Lucie AFWS:

- <u>Loss of feedwater (LOFW) with offsite AC available</u>
  Failure to manually actuate the St. Lucie Unit 1 AFWS, was assessed to be the dominant failure mode and this fault contribution to the overall AFWS unavailability is estimated to be approximately 80 percent.

- <u>LOFW with only onsite AC available</u>
  St. Lucie Unit 1 uses a swing tie bus ("AB") that furnishes AC power to valves in the steam turbine driven portion of the AFWS. This bus is interlocked to prevent tie to more than one emergency diesel generator (EDG) simultaneously. The "AB" bus is normally tied to the "A" EDG. Thus, the limiting EDG failure would be failure of the "A" EDG. This failure requires human action to transfer bus "AB" to the available "B" EDG. The impact of this human action on the overall AFWS was assessed and found not to significantly alter the above results. Thus, failure to manually actuate AFWS remains the common dominant failure mode identified.

- <u>LOFW with only DC available</u>
  For this event, the St. Lucie Unit 1 AFWS design requires a plant operator to proceed to the local valve stations for the steam turbine driven train of the AFWS and open four AC motor

operated valves (2 steam and 2 water) that are normally closed. The licensee assessment of accessibility and the opening times indicate that this operation could be successfully accomplished by two men in about 5 minutes and one man in about 10 minutes. Human failure to open these valves has been assessed as the dominant fault contributor (~60%) for this event.

X.7.2.2. Principal Dependencies

The principal dependencies identified were those associated with human actions required to actuate the St. Lucie Unit 1 AFWS for the above three events.

X.7.3 Recommendations for this Plant

The short-term recommendations identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.7.3.1 Short-Term Recommendations

1. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

2. Recommendation GS-5 - The plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be

dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power sources is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

3. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

   The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW

   system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

4. Recommendation GS-8 - The licensee should install a system to automatically initiate AFW system flow. For the short term, this system need not be safety-grade; however, it should meet the criteria listed below, which are similar to Item 2.1.7a of NUREG-0578. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-1.

   The design should provide for the automatic initiation of the auxiliary feedwater system flow.

   The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

   Testability of the initiating signals and circuits should be a feature of the design.

   The initiating signals and circuits should be powered from the emergency buses.

   Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses. The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.7.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lesson Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

> "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.
>
> The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.7.3.3 Long-Term Recommendations

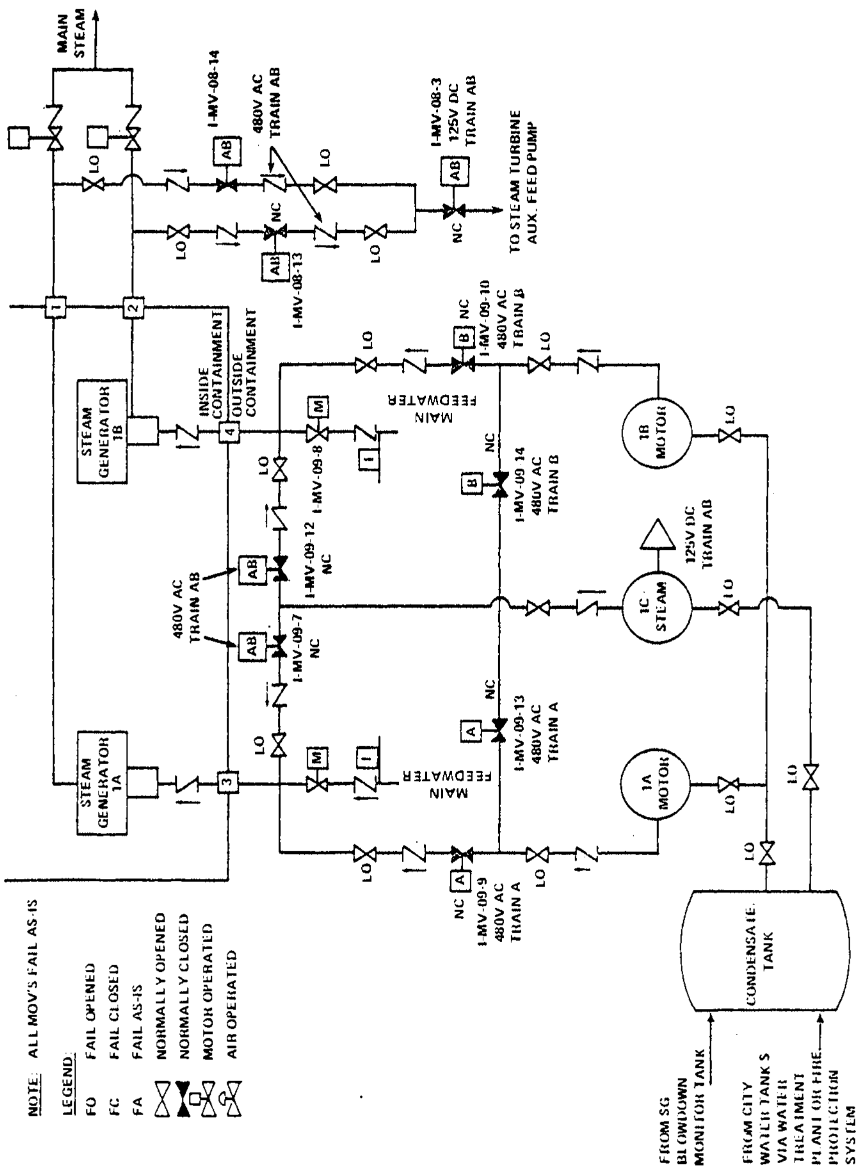
Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-1 - Licensees with plants having a manual starting AFW system, should install a system to automatically initiate the AFW system flow.. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation..

2. Recommendation - GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least 2 hours. Conversion of direct current power to alternating current is acceptable.

3. Recommendation - The present method of supplying water from the alternate water sources to the CST for the AFWS requires considerable operator action and is estimated to take approximately three hours to accomplish. The licensee should modify the design to provide means to supply water to the AFWS from the alternate sources within one-half hour or less.

4. Recommendation - The St. Lucie plant needs one full capacity train of AFW flow (2 motor-driven or 1 turbine-driven AFW pump) for safe plant shutdown. This AFWS design does not meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position 10-1; namely that the AFWS should maintain the capability to supply the required AFW flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines plus a single active failure. The licensee should (1) complete an evaluation assuming such an event and determine any AFW system modifications or procedures necessary to maintain the required AFW flow to the steam generator(s), or (2) describe how the plant can be brought to a safe shutdown condition by use of other available systems following such a postulated event.



Auxiliary Feedwater System
St. Lucie, Unit 1
Figure 1

ENCLOSURE 2

Basis for Auxiliary Feedwater
System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

   g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

   h. RC flow condition - continued operation of RC pumps or natural circulation.

   i. Maximum AFW inlet temperature.

   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

   l. Operating condition of steam generator normal blowdown following initiating event.

   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

   n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCTOBER 1 1 1979

Docket No.: 50-334

Mr. C. N. Dunn, Vice-President
Operations Division
Duquesne Light Company
435 Sixth Avenue
Pittsburgh, Pennsylvania

Dear Mr. Dunn:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT BEAVER VALLEY UNIT 1

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell Eisenhut

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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ENCLOSURE 1

X (W) BEAVER VALLEY UNIT 1

AUXILIARY FEEDWATER SYSTEM

X.1.1 System Description

X.1.1.1 Configuration, Overall Design

A simplified flow diagram of the Beaver Valley Plant, Unit No. 1, Auxiliary Feedwater System (AFWS) is shown in Figure 1. The AFWS consists of one turbine driven pump (700 gpm @ 2696 ft head), and two motor driven pumps (350 gpm @ 2696 ft head). The pump discharge headers are connected to permit auxiliary feedwater delivery to any one or all three steam generators by any AFW pump. The licensee states that for normal and transient plant operation, including loss of main feedwater flow, only one pump is required to cool the plant down to the condition where the RHR system can be put into operation to continue safe plant shutdown. However, in the event of an unisolable main steam or main feed line break, either one turbine-driven AFW pump or both motor-driven pumps are required to prevent dryout of the steam generators.

The primary water supply of the AFWS is maintained in a 140,000 gallon seismic Category I, primary plant demineralized water storage tank (DWST). The tank is reserved strictly for the AFWS pump usage. The reserved water inventory is sufficient to maintain the plant at hot standby condition for 8 hours following a reactor trip. Low water level in the DWST will alarm and annunciate in the main control room. The secondary water supply is the seismic Category I river water system with an additional backup source from the fire protection system.

X.1.1.2 Components - Design, Classification

All pumps, valves, piping, instrumentation and controls associated with the auxiliary feadwater system are designed to seismic Category I requirements.

The primary water source (Demineralized Water Storage Tank) and the secondary water source (Rivar Water System) are also designed to seismic Category I requirements. The additional backup water source from the fire protection system is not designed to seismic Category I requirements.

X.1.1.3 Power Sources

The turbine driven pump is supplied with steam from each steam generator outlet header upstream of the main steam isolation valve (MSIV) and exhausts to the atmosphere. The motor driven pumps receive power from the 4160 V AC vital buses. In the event of a loss of offsite power, the pumps are powered by the Division A and B emergency diesel generators, respectively.

X.1.1.4 Instrumentation and Controls

X.1.1.4.1 Controls

The control of auxiliary feedwater flow and steam generator water level is accomplished from the main control room by manually operated control valves. These valves can also be manually operated from the local shutdown control panel if the control room is not accessible. All manually operated valves in the main flow path of the AFWS are either "lock-opened" or "lock-closed" in their normal position. The motor operated valves fail in their "as is" position.

X.1.1.4.2 Information Available to Operator

The important information available to the operation includes AFW discharge header pressure, AFW flow to each steam generator, DWST water level, steam generator water level, steam pressure to turbine driven AFW pump and control valve position indicators. Additional information available is in the following instrument list:

SPECIFIC INSTRUMENTATION AND CONTROL

Flow

Auxiliary feed flow to 1A (B, C) Steam Generator

FI-FW-100A (B, C)

Readout location: Vertical Board - Section C

FI-FW-100A1 (B, C)

Readout location: Emergency Shutdown Panel

AUX STEAM GEN FEED PUMP AUTO START-STOP
PRI PLNT DEMIN WTR STRGE TNK LVL H-L CH 1
PRI PLNT DEMIN WTR STRGE TNK LVL H-L CH 2
1/3 STM GEN 1A HI-HI LEVEL
1/3 STM GEN 1A LOW LOW WATER LEVEL
STM GEN 1A LOW WTR LEVEL CH 1
STM GEN 1A LOW WTR LEVEL CH 2
1/3 STM GEN 1B HI-HI LEVEL
1/3 STEAM GEN 1B LOW LOW WATER LEVEL
STM GEN 1B LOW WTR LEVEL CH 1
STM GEN 1B LOW WTR LEVEL CH 2
1/3 STM GEN 1C HI-HI LEVEL
1/3 STM GEN 1C LOW LOW WTR LEVEL
STM GEN 1C LOW WATER LEVEL CH 1
STM GEN 1C LOW WATER LEVEL CH 2
AUX STM GEN FEED PUMP 3A START-STOP
AUX STM GEN FEED PUMP 3B START-STOP
AUX STM GEN FD PP 3A MTR INBD BRG TEMP
AUX STM GEN FD PP 3B MTR INBD BRG TEMP
AUX STM GEN FD PP 3A MTR OUTBD BRG TEMP
AUX STM GEN FD PP 3B MTR OUTBD BRG TEMP

X.1.1.4.3 Initiating Signals for Automatic Operation

The AFWS is automatically initiated. It can also be started manually from the main control room. In addition, the pumps can be manually started from the local shutdown control panel. The automatic initiating signals are as follows.

1) Turbine Driven Pump
   a) 1/3 Steam Generator Lo-Lo Level (1 out of 3 channel logic)
   b) Under Voltage

2) Motor Driven Pumps
   a) 2/3 Steam Generator Lo-Lo Level
   b) Both Main Feed Pumps Trip
   c) Safety Injection Signal
   d) Turbine Driven AFW pump low discharge pressure consistent with a start signal on turbine driven pump.
   e) Loss of offsite power

X.1.1.5 Testing

The systems are tested periodically in accordance with technical specification requirements. The frequency of periodic testing is 31 days. In addition, the particular system is tested in accordance with the technical specification after performing system maintenance. The systems are tested using the recirculating lines, with various plant parameters noted (suction and discharge pressures, etc). The

instrumentation system is checked periodically, in accordance with the technical specifications, on a per shift, monthly or refueling time frame basis.

X.1:1.6 Technical Specifications

A review of the technical specifications indicated that these specifications cover limiting conditions of operation (LCO) and periodic surveillance testing consistent with current standard Technical Specifications.

X.1.2 Reliability Evaluation

X.1.2.1 Dominant Failure Modes

The following failure modes were found to dominate the demand unavailability of the Beaver Valley Unit 1 AFWS.

Loss of Feedwater (LOFW) with Offsite AC Available

The dominant failure mode (≥90%) for this transient event was assessed to be those possible coupled human errors in testing, i.e., leaving two or more of the manual block valves closed in the discharge side of the pumps while performing the type of pump flow testing required by the Technical Specifications.

The licensee has recognized this possible common mode error and is planning to chain lock all manual valves into their correct alignment state. Further, the licensee will, in the future, stagger his pump test program such that no more than one of the three pumps will be tested in any one shift. Considering implementation of these procedures, the overall availability of the Beaver Valley AFWS design should be improved by roughly a factor of three.

LOFW with only Onsite AC Available

Assessment of the AFWS, given this transient event, indicated that there would be no significant change in the predicted unavailability of the Beaver Valley AFWS. Human error concerning mispositioned block valves in the AFWS discharge remained the dominant failure mode.

LOFW with only DC Available

In this transient event, the Beaver Valley Unit 1 AFWS would be expected to automatically actuate and the human could serve as backup to open any of the valving that failed to electrically respond.

The dominant contributors to AFWS unavailability in this event were:

- allowed test and maintenance outage
- hardware faults (principally the failure of steam turbine pump)

X.1.2.2 Principal Dependencies Identified

The principal dependency identified was the human error (common mode) vunerability associated with manual closure of the AFWS discharge block valves and failure to reopen them.

X.1.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS and plant specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL and plant specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981 or as soon thereafter as is practicable.

X.1.3.1 Short Term

1. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

   The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

- The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

4. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements as indicated in Recommendation GL-5.

The design should provide for the automatic initiation of the auxiliary feedwater system flow.

The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

Testability of the initiation signals and circuits shall be be a feature of the design.

- The initiation signals and circuits should be powered from the emergency buses.

- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

5. Recommendation - The normally closed manually operated suction valves from the river water system to the AFWS should be periodically tested and the position verified. The licensee should propose appropriate Technical Specifications to incorporate these provisions.

6. Recommendation - The strengthened administrative procedures described in Section X.1.2.1 above should be implemented; namely, the locking of manual valves in the correct position and staggered testing of the AFW system pumps. The licensee has advised us that it plans to implement such strengthened procedures before Beaver Valley Unit 1 (currently shut down for reasons unrelated to this AFW system review) returns to power.

7. Recommendation - As shown in Figure 1, the locked block valves in each AFW pump discharge line are aligned so that the combined flow from one motor-driven pump plus one turbine-driven pump is supplied to the steam generators via one AFW header while flow from the remaining motor-driven pump is supplied to the steam generators via the redundant AFW header. As indicated in Section 1.1.1, the licensee states that, in the event of an unisolable main steam or main feed line break, the flow from both motor-driven pumps or from the turbine-driven pump is required to prevent dryout of the steam generators. The licensee should review the present alignment of the AFW pump discharge block valves and modify as necessary to provide the AFW required for normal, transient, and accident conditions.

X.1.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

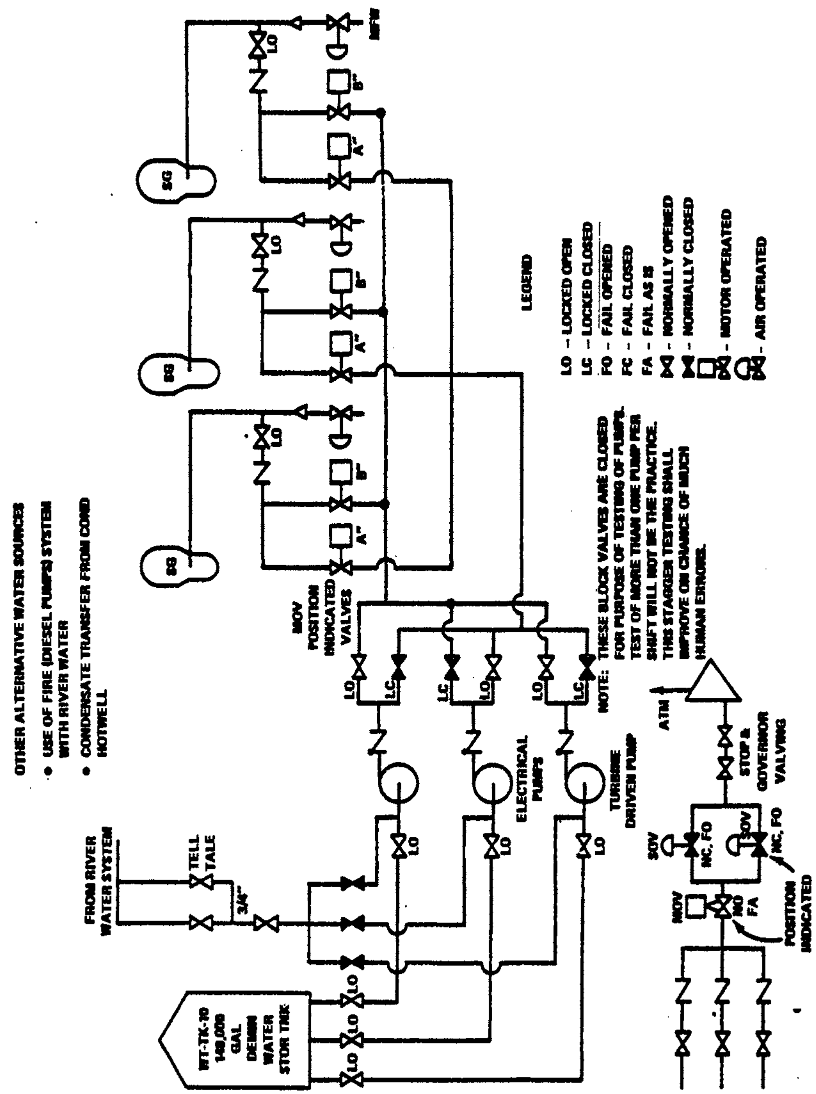
4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.1.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

2. Recommendation - As indicated in Section X.1.1.1, the plant requires flow from two motor-driven pumps or one turbine-driven pump for accident conditions. This design does not meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position 10-1; namely, that the AFWS should maintain the capability to supply the required AFW flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines concurrent with a single active failure. The licensee should complete an evaluation assuming such an event and (1) determine any AFW system modifications or procedures necessary to maintain the required AFW flow to the steam generator(s) or (2) describe how the plant can be brought to a safe shutdown condition by use of other available systems following such a postulated event.



Auxiliary Feedwater System
Beaver Valley, Unit-1

Figure 1.

Enclosure 2

Basis for Auxiliary Feedwater
System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

   - Maximum RCS pressure (PORV or safety valve actuation)
   - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
   - RCS cooling rate limit to avoid excessive coolant shrinkage
   - Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:
   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.
   b. Time delay from initiating event to reactor trip.
   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).
   d. Minimum steam generator water level when initiating event occurs.
   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 30, 1979

Docket Nos.: 50-315
50-316

Mr. John Dolan, Vice-President
Indiana and Michigan Electric Company
Indiana and Michigan Power Company
P. O. Box 18
Bowling Green Station
New York, New York 10004

Dear Mr. Dolan:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the Donald C. Cook Nuclear Plant, Units 1 and 2 (Cook 1 & 2). These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Based on our review of the auxiliary feedwater systems at Cook 1 & 2, we have identified requirements which are applicable to the current auxiliary feedwater system design. Enclosure 1 contains these requirements. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to Cook 1 & 2. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

Your letter of August 9, 1979 to Mr. Harold R. Denton, Director of Nuclear Reactor Regulation, submitted Amendment No. 84 to the Cook 1 & 2 Final Safety Analysis Report. This amendment, which contains a number of proposed modifications to the auxiliary feedwater systems, is currently under staff review. The requirements contained in Enclosure 1 are based on our review of the as-built auxiliary feedwater system design at Cook 1 & 2. We recognize that you have proposed modifications to the as-built design. However, to expedite matters, we request that you evaluate the Cook 1 & 2 design and procedures against the applicable requirements specified in Enclosure 1 to determine the degree to which your facility currently conforms to these requirements. You should also indicate how this degree of conformance will be affected by the proposed design modifications contained in Amendment No. 84.

The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the Cook 1 & 2 facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
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Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: See attached lists

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ENCLOSURE 1

X.2 (W) DONALD C. COOK UNITS 1 and 2

AUXILIARY FEEDWATER SYSTEM

X.2.1 System Description

X.2.1.1 Configuration and Overall Design

The auxiliary feedwater system (AFWS) is designed to supply water to the steam generators for reactor coolant system sensible and decay heat removal when the main feedwater system is not available. The AFWS for Cook 1 & 2 is utilized in the event of either a malfunction such as loss of offsite power, or an accident, and during certain periods of normal startup and shutdown. The AFWS is automatically actuated under certain transient and accident conditions.

The AFWS is shown in simplified form on Figure 1 attached. The AFWS, consists of a steam turbine driven pump for each unit which supplies AFW flow to 4 steam generators of its associated unit and two cross-connected motor driven pumps each of which supplies flow to 4 steam generators, two in each unit.

The motor driven and turbine driven auxiliary feed pumps of each unit normally take suction from the condensate storage tank associated with that each unit. A cross-tie line connects the condensate storage tanks and auxiliary feed pump suctions of the two units. An air-operated valve which can be controlled from the control room is provided in the cross-tie line. This valve is normally closed and will fail in the closed position. This valve can also be manually operated at a local valve station.

Each condensate storage tank has a capacity of 500,000 gallons of which 175,000 gallons are reserved by Technical Specification for AFW system use. The licensee estimates that this reserve capacity is sufficient for approximately 12 hours of operation and is adequate to bring the unit to RHR operation capability. Each condensate storage tank is located outdoors and is non-seismic Category 1 design. However, the tanks have been analyzed to show that they can withstand the operating basis earthquake (OBE). If neither condensate storage tank is available, the AFW pumps can take suction from the Essential Service Water System (ESWS) through a normally closed motor operated valve and a normally closed manual isolation valve at each auxiliary feed pump suction. The ESWS piping and valves are designed in accordance with B31.1. However, the entire ESWS system is analyzed to withstand the safe shutdown earthquake (SSE).

All manual valves located between the condensate storage tank and the AFW pumps suctions are locked in the open position.

A manual duplex strainer is installed in each auxiliary feed pump suction to prevent pump damage from debris and/or scale in the water. Also automatic backwash duplex trainers are installed in the ESWS pump discharge lines.

The motor driven AFW pumps supply two steam generators in each unit (i.e., the Unit 1 motor driven pump supplies steam generators No. 2 and No. 3 of Unit 1 and steam generators No. 1 and No. 4 of Unit 2 and the Unit 2 motor driven pump supplies steam generators No. 2 and No. 3 of Unit 2 and steam generators No. 1 and No. 4 of Unit 1.) Each of the motor driven pump supply lines to the steam generators has a normally closed motor-operated valve for flow control and isolation. On loss of power these valves fail AS-IS.

The motor driven AFW pumps are sized to prevent actuation of the pressurizer safety or relief valves in the event of loss of all main feedwater supply in conjunction with loss of power to the reactor coolant pump buses. Each motor driven AFW pump has a capacity of 450 gpm with a TDH of 2714 feet. These pumps are powered from separate emergency electrical buses. The capacity of the two pumps (900 gpm) is sufficient to maintain the level in 4 steam generators above the lower limit of the wide range level indicator.

The turbine driven AFW pumps meet the same criteria as the motor driven AFW pumps except their capacity is 900 gpm with a TDH of 2714 feet.

Steam to each of the turbine driven AFW pumps is supplied from its associated Units' No. 2 and No. 3 steam generators taken upstream of the main steam isolation valves. Each of the turbine driven AFW supply lines to the steam generators has a normally open motor-operated valve for flow control and isolation.

The discharge pipe header and individual supply lines to each steam generator for the motor driven and turbine driven auxiliary feedwater pumps are designed to seismic Category 1 requirements, AEPSC quality level 4 which is equivalent to ASME Class II.

Each AFW pump is provided with an emergency leakoff line and a test line. The emergency leakoff line ensures that a minimum flow through the pump is maintained to prevent pump overheating and possible damage.

Upon automatic startup of the motor driven auxiliary feed pumps, the steam generator blowdown valves and pump test line close. The motor driven valves in the pump discharge lines to the unaffected unit remain closed while the valves to the affected unit open automatically.

A high flow rate through the motor driven or turbine driven pump causes the associated pump's motor-operated isolation valves to the steam generators to automatically close to an intermediate position. The valves may then be operated as necessary from the control room.

X.2.1.2 Component Design Classification

1. The condensate storage tanks are non-seismic but have been analyzed to withstand the OBE.

2. The suction piping and valves from the condensate storage tanks to the AFW pumps are designed to B31.1 with quality control to

the requirement of B31.7. The suction piping is analyzed to withstand the SSE.

3. The ESWS (alternate AFW supply) piping, valves and components are designed to B31.1 requirements. However, the ESWS is analyzed to withstand the SSE.

4. The turbine driven pumps and motor driven pumps are designed to seismic Category I requirements.

5. The turbine driven pump discharge header and steam generator supply lines and motor driven pump discharge header and supply lines associated with each Unit are designed to seismic Category 1, AEPSC quality level 4 which is equivalent to ASME Class II.

6. Each motor driven and turbine driven AFW pump is located in a separate seismic Category I enclosure and protected from tornado missiles.

7. Motors, cables and other electrical components required for the AFW system operation are Class 1E.

X.2.1.3 Power Sources

Each Unit has two class 1E power system trains A and B. Each power train contains a 250 V DC Station battery, 4 KV diesel generator and power distribution system.

Each unit's turbine driven AFW pump and associated valves are powered from power train B of its own unit.

Each motor driven AFW pump and associated support system is powered from the A power train of its associated unit.

The motor driven AFW pump discharge valves are powered from the A power train of the unit served. i.e., Unit 1 motor driven AFW pump supply valves to Unit 1 steam qenerators are powered from train A Unit 1; Unit 1 motor driven AFW pump supply valves to Unit 2 steam generators are powered from train A Unit 2; Unit 2 motor driven AFW pump supply valves to Unit 2 steam generators are powered from train A Unit 2; Unit 2 motor driven AFW pump supply valves to Unit 1 steam generators are powered from train A Unit 2.

The AFW system as presently installed meets redundancy requirements, however, it is dependent on both AC and DC power for automatic operation.

Intended modifications (currently in progress) to the turbine driven AFW pumps will remove the turbine driven AFW pumps AC power dependence for automatic system initiation.

X.2.1.4 Instrumentation and Controls

The instrumentation and control power is supplied form the 120 V AC vital bus system. There are four vital buses, each supplied by an inverter receiving power either from the 600 V AC Class 1E auxiliary

buses or the 250 V DC power system. The motor driven pump breaker controls are powered from the Class 1E 250 V DC power system. The Class 1E station batteries are maintained at full charge by battery chargers supplied from the Class 1E auxiliary buses.

X.2.1.3.1 Controls

Controls for the AFW pumps and their associated valves are located in the control room of the unit with which the pump is associated, and are duplicated at the hot shutdown panel and other unit control panels.

a) FMO-211, -221, 231, and -241 (Unit 1) or (Unit 2) are the Steam generator supply valves from the turbine driven auxiliary feed pump (TDAFP). These 4-inch motor operated (Globe type) valves are normally open, but each may be closed by the control room operator in the event of a feedwater or steam line break at the steam generator with which it is associated. They also may be throttled to regulate steam generator level. In the event of a steam line break and rapid depressurization of a steam generator, or upon detection of a high flow at the TDAFP, these valves are automatically driven to an intermediate position to prevent pump runout. On loss of power, the above valves fail AS-IS.

b) FMO-212, -222, -232 and -242 (Unit 1) or (Unit 2) are the steam generator supply valves from the motor-driven auxiliary feed pumps (MDAFP). These 4-inch motor operated (Globe type) valves are normally closed and are opened and/or throttled as described in (a) above. These valves open automatically, as a result of any of the signals which require MDAFP start for that unit. The steam generator supply valves in the other unit will get a signal to close. On loss of power, these valves fail AS-IS.

c) WMO-753, and -754 are the essential service water (EWS) supply valves to the turbine driven and motor-driven auxiliary feed pumps. These 4-inch motor operated (Butterfly type) valves are normally closed, and except for testing under closely controlled conditions, are opened by the control room operator only if water is unavailable from the condensate storage tanks. In addition to the above normally closed MOVs, the ESW supply to the AFW pumps contains normally closed, manually operated, butterfly valves ESW-109, -115, -145 and -240. These valves must be locally opened in the event ESW is required. Operation of these valves can be achieved in less than 10 minutes. On loss of power, the motor operated valves fail AS-IS.

d) MCM-221 and -231 are the steam supply isolation valves to auxiliary feed pump turbines. These 4-inch motor operated gate valves are normally open, allowing steam pressure to be available up to the trip and throttle (T&T) valve at each turbine. The motor operated steam isolation valves MCM-221 and -231 can be opened or closed from the control room and on loss of power they fail AS-IS. The T&T valve opens automatically when the turbine driven AFW pump receives a start signal; however, it is AC-powered and fails AS-IS.

e) FRV-257 and -258 are the emergency leakoff valves for motor-driven and turbine driven auxiliary feed pumps. The valves are 1-inch, air operated diaphragm, globe type, normally open and spring actuated to fail open on loss of air pressure. The valves are automatically modulated (open or closed) when feedwater flow rate to the steam generators or through the test line is below or above the required minimum pump leakoff flow rate setpoint.

f) CRV-51 is the condensate storage tank cross-tie valve. This 8-inch, air operated diaphragm, globe type, valve is normally closed and is spring actuated to fail closed on loss of air pressure. The valve connects Unit 1 and Unit 2 Condensate storage tanks. This valve is opened by energizing a solenoid valve in the air supply line. Controls for remote operation of this valve are located in the control room. Power to energize the solenoid valve is supplied from the normal station AC power supply.

Opening this valve and manually realigning others in the AFW systems pump suctions permits the AFW systems of both Units to draw condensate from one tank in the event the other tank is not available.

X.2.1.4.2 Information Available to the Operator

The following is information available to the operator on the Main Control Board or on the Hot Shutdown Panel:

1. Flow, gpm to each steam generator
2. Steam generator levels
3. Breaker position (motor driven pump)
4. Motor current and voltage (motor driven pump)
5. Motor operated valve status lights from limit switches
6. Steam pressure to auxiliary feed turbine (as steam generator pressure)
7. Pump discharge pressure
8. Condensate storage tank level
9. Turbine driven auxiliary feed pump speed control
10. Operational alarms and annunciations shown on the attached Table A and B

X.2.1.4.3 Initiation Signals for Automatic Operation

The turbine driven and motor driven auxiliary feedwater pump start signals are listed below.

A. Turbine Driven

The following signals in one unit will start that unit's turbine driven auxiliary feed pump:

(1) Low-Low level in any 2 of 4 steam generators (possible loss feedwater or steam line break)

(2) Reactor coolant pumps bus undervoltage (anticipation of loss of offsite power)

(3) Manually

B. Motor Driven

The following signals in either unit will start both motor driven auxiliary feedwater pumps:

(1) Low-Low level in any steam generator (possible loss of feedwater or main steam line break)

(2) Trip of main feed pumps in either unit

(3) Any safety injection signal derived from Reactor Protection System and/or containment pressure - High at 1.2 psi.

(4) Loss of offsite power* (Pump is sequenced ON when emergency diesel generator is energizing safeguards bus)

(5) Manually

*Note

There is a delay of <60 seconds in starting the motor driven pump. The reason for this delay is to limit the loads during emergency diesel generator loading.

X.2.1.5 Testing

The AFW system and components are tested in accordance with Technical Specification requirements. The frequency of periodic testing of the pumps is 31 days. In addition, the particular system is tested in accordance with the Technical Specification after performing system maintenance. The systems are also tested using the pump recirculation lines test lines with various plant parameters noted (as called out by ASME Section XI). The instrumentation systems are checked in accordance with the technical specifications, on a per shift, monthly or refueling time frame basis.

FRV-255 and -256 are test valves for the motor and turbine-driven auxiliary feed pumps. These normally closed, 3-inch, air-operated, globe type valves are capable of passing approximately the design flow rates for each pump and are used to performance test the pumps on a periodic basis. The valves are diaphragm, spring close type and on loss of air pressure they fail closed. Operating air to each test valve is controlled by a solenoid valve installed in the air supply line. Should test valves FRV-255 and -256 be left in the open position, an automatic start of the auxiliary feed pumps will automatically close the valves.

X.2.1.6 Technical Specifications

A review of the technical specifications indicated that these specifications cover limitng conditions of operation (LCO) and periodic surveillance testing consistent with current standard technical specifications.

X.2.2 Reliability Evaluation

X.2.2.1 Dominant Failure Modes

The D. C. Cook auxiliary feedwater system was analyzed to determine the dominant failure modes under three transient conditions:

(a) LOFW with offsite power available

(b) LOFW with onsite power available

(c) LOFW with only DC power available

Results of the analysis are summarized below.

LOFW with Offsite Power Available

No significant failure modes were identified in the analysis. No significant single or double failures were noted. The most dominant failure modes appear to be triple failures involving maintenance in one of the pumps trains and independent failures in the other.

LOFW with Onsite Power Available

The system was analyzed to determine if the dominant failure modes would be significantly different given loss of offsite power. As in the previous case, the dominant failure modes would appear to involve three independent failures. The most dominant mode would involve maintenance of one pump train and hardware failures in another, and failure of the diesel powering the third.

LOFW with Only DC Power Available

Assuming loss of all AC power, the turbine-driven train must be manually actuated by locally opening the turbine-driven AFW pump trip and throttle valve. Assuming that this action is performed, the system should operate successfully, unless failures were to occur in this train or maintenance was being performed at the time of loss of power. The dominant failure modes for this case appear to be:

(a) operator fails to manually open the steam admission valve;
(b) turbine train unavailable due to maintenance.

It should be noted that the licensee is installing the capability to open the steam admission valves by use of a DC source.

X.2.2.2 Dependencies

No locational or environmental dependencies were identified which could cause common-mode failures of the system. The four pumps for the station are located in separate rooms equipped with adequate drains and protected against pipe whip and missiles. In addition, no common dependencies on AC or DC power were identified.

X.2.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system realiability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate

potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant-specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.2.3.1 Short Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-5 - Modifications currently are being implemented to make the turbine driven trains independent of any alternating current power source. The following recommendation should be met in the interim. The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operate would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations

should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:
   - Procedure should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

     The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation GS-7 - The licensee should verify that the automatic start AFW signals and associated circuitry are safety grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.2.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating

plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and a low level alarm in the control room for the AFW system primary water supply to allow the operator anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety related equipment in the room.

3. Recommendation - The licensee should implement the following requirements which are identical to Item 2.1.7.b of NUREG-0578:

Safety-grade indication of AFW flow to each steam generator should be provided in the control room.

The auxiliary feedwater flow instrument channels should be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

4. Recommendation - Licensees with plants require local manual realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.2.3.3 Long Term

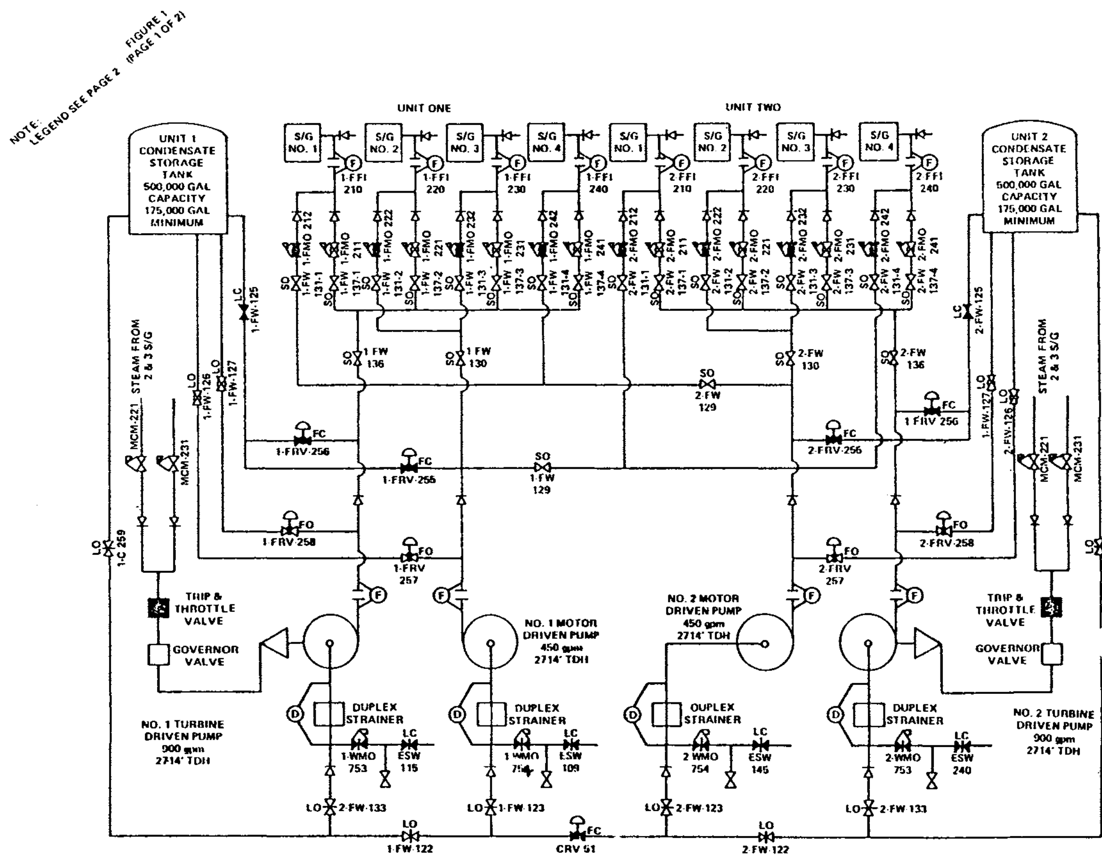
Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-3 - The licensee is currently performing modifications to make the turbine driven train independent of A-C power sources. The following recommendation should be met when these modifications are complete. At least one AFW system pump and its associated flow path and essential instrumentation

should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

2. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

3. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.



Auxiliary Feedwater System
D.C. Cook
Figure 1 (Sheet 1 of 2)

LEGEND:

- = MOTOR OPERATED GLOBE VALVE – OPEN
- = MOTOR OPERATED GLOBE VALVE – CLOSED
- = MOTOR OPERATED BUTTERFLY VALVE – OPEN
- = MOTOR OPERATED BUTTERFLY VALVE – CLOSED
- = MOTOR OPERATED GATE VALVE – OPEN
- = MOTOR OPERATED GATE VALVE – CLOSED
- = MANUAL GATE VALVE – OPEN
- = MANUAL GATE VALVE – CLOSED
- = MANUAL GLOBE VALVE – OPEN
- = MANUAL GLOBE VALVE – CLOSED
- = MANUAL BUTTERFLY VALVE – OPEN
- = MANUAL BUTTERFLY VALVE – CLOSED
- = CHECK VALVE

FO = FAILS OPEN
FC = FAILS CLOSED
LO = LOCKED OPEN (LOCK AND KEY)
LC = LOCKED CLOSED (LOCK AND KEY)
SO = SEALED OPEN (DETENT)

Auxiliary Feedwater System
D.C. Cook
Figure 1 (Sheet 2 of 2)

ENCLOSURE 2

Basis for Auxiliary Feedwater System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

   - Maximum RCS pressure (PORV or safety valve actuation)
   - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
   - RCS cooling rate limit to avoid excessive coolant shrinkage
   - Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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OCTOBER 1 2 1979

Docket No.: 50-348

Mr. Alan R. Barton
Senior Vice-President
Alabama Power Company
P. O. Box 2641
Birmingham, Alabama 35291

Dear Mr. Barton:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut /for

Darrell G. Eisenhut, Acting Director
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ENCLOSURE 1

X.3 (W) FARLEY 1

AUXILIARY FEEDWATER SYSTEM

X.3.1 System Description

X.3.1.1 Configuration, Overall Design

The auxiliary feedwater (AFW) system as shown in the attached simplified diagram consists of three pumps (2 motor driven, 1 turbine driven) each of which is normally lined up to feed all three steam generators. The motor driven pump discharges are cross connected through manually operated, locked-open valves upstream of the motor operated isolation valves to each steam generator. The turbine driven pump supplies each steam generator down-stream of the motor operated steam generator isolation valves and the auxiliary feedwater control valves. Check valves are provided downstream of the feedwater control valves that will prevent reverse flow through the control valves.

The primary water supply source for the AFW system is a 500,000 gallon capacity condensate storage tank (CST). 150,000 gallons are reserved for decay heat removal in the event of an accident. The licensee states that this reserve capacity is sufficient to maintain the plant at hot standby for two hours and cooldown to conditions that the RHR system can be operated. The CST is normally lined up to supply water to the AFW pumps, through redundant lines (one to the motor driven pumps and one to the turbine driven pump) through locked-open, manually operated isolation valves.

A backup long term source of water supply is provided by redundant service water trains. Two normally closed motor operated valves in series isolate the service water trains from the auxiliary feedwater system. These valves can be operated from the control room (key locked) to initiate service water flow to the AFW system. One SWS train is normally lined up to supply a motor driven pump and a turbine driven pump; the redundant SWS train normally supplies the other motor driven pump. With manual valve operation outside the control room, each SWS train can supply all three AFW pump suctions.

X.3.1.2 Components

All components of the auxiliary feedwater system are designed to Quality Group C, seismic Category I requirements including motor, pumps, piping, valves, and valve operators. The auxiliary feedwater control valves, which are the only normally closed valves in the system flow path, are air operated and DC power controlled. The air system is non-safety grade and the control valves will fail open on loss of air or DC control power.

X.3.1.3 Power Sources

The motor driven pumps are powered from independent Class 1E emergency buses supplied by the diesel generators. All valves in the motor driven trains are A-C motor operated or manual valves with the exception of the auxiliary feedwater control valves which are air operated and controlled by DC power. One air compressor can be powered by the diesel-generators to supply air, but it is not safety-grade. The control valves will fail open for maximum AFW flow on loss of air or DC power.

The turbine driven train can be operated and controlled from the control room, independently of AC power. The steam inlet valves are air operated, fail closed, with a one-hour air accumulator available for valve operation upon loss of air. The accumulator will open the valve and keep it open upon loss of instrument air supply. DC control power is used to actuate valve operation.

The backup service water system supply series isolation valves are powered from the same Class 1E bus that powers the service water system train and motor driven AFW pump train. Therefore a single failure of one bus will not disable both backup service water supplies.

There are two motor operated series isolation valves in the flow paths to each steam generator from the motor driven pumps that are powered from separate Class 1E buses such that a single failure of a bus will not prevent isolation of an affected steam generator following a main steam or feedwater line break.

The circuit breakers for the motor driven pumps require DC control power to operate and energize the AFW motors. These breakers can be manually closed locally without DC power.

X.3.1.4 Instrumentation and Controls

X.3.1.4.1 Controls

Steam generator level is controlled manually from the control room. Flow to the steam generators from the motor driven pumps is controlled by

ɔdulating three flow control valves, one to each steam generator. These valves open for full flow to the steam generators following AFW system initiation. This flow cannot be varied if the system was automatically started by a safety injection signal, until the injection signal has been reset which is 60 seconds after the receipt of the signal.

The flow to each steam generator from the turbine driven pump will normally be controlled from the control room by varying turbine speed. The flow control valves to each steam generator from the turbine driven pump will normally be kept full open.

Each pump and all motor operated and air operated valves can be operated from the control room and are powered from essential Class 1E buses.

X.3.1.4.2 Information Available to Operator

The control room operator has the following indications and alarms available in the control room.

1. Motor driven Aux Feedwater Pumps
   a. Ammeter
   b. Breaker Status
   c. Monitor Light (Pump Running)
   d. Fault Trip Alarm (overcurrent)
   e. Pump in Local Control Alarm
   f. Breaker Fails to Close Alarm (Loss of Offsite Power & SIAS) indication and alarm

2. St .m Admission Valves Turbine Driven AFW Pump
   a. Valve Status indication
   b. Monitor Light
   c. Valve in Local Control Alarm
   d. Turbine Driven Pump - Fault Alarm - (Overspeed Trip, Steam Valves Closed with Demand Signal)

3. General
   a. Valve Position Indication for all motor and air operated valves
   b. Turbine Speed
   c. Turbine Steam Pressure
   d. Flow to each Steam Generator
   e. Pump Discharge Pressure
   f. Pump Suction Pressure
   g. Condensate Storage Tank Level
   h. Steam Generator Level
   i. Low Suction Pressure Alarm to each pump
   j. Loss of Ventilation Cooler for each motor driven pump room
   l. Hi/Lo Suction flow alarm

X.3.1.4.3 Initiating Signals for Automatic Operation

Motor Driven Pumps

1. Lo-Lo S/G level 2 out of 3 detectors to any one steam generator

2. Both Main F ad Pumps Trip (senses stop valve to turbine driven main feed pumps)

3. Loss of offsite power or two out of three undervoltage condition on respective ESF buses.

4. Safety Injection Signal

Turbine Driven Pump

1. Lo-Lo Steam Generator Level 2 out of 3 detectors to any 2 Steam Generators

2. Undervoltage to any two of three reactor coolant pump buses.

X.3.1.5 Testing

1. The motor driven pumps and the turbine driven pump are tested for operability by recirculation back to the condensate storage tank monthly. Each valve in the auxiliary feedwater system flow path or bypass flow path that is not locked, sealed or otherwise secured in position is verified to be in its correct position at least once per month. Motor operated stop check valves in AFW discharge to each steam generator are verified to be open with the breaker to the valve operators locked open at least once a month.

2. At least once per 18 months during shutdown:
   a. Verify that the motor driven pumps will start upon receipt of the following signals:

(1) Loss of main feedwater pumps

(2) safety injection signal

(3) steam generator water level low-low from one steam generator

(4) loss of offsite power

b. Verify that the steam turbine driven pump starts automatically upon receipt of the following:

(1) Blackout Signal (undervoltage to RCP buses)

(2) Steam generator low-low water from two steam generators

Valve operability tests are performed quarterly on motor and air operated valves. Stroke tests for these valves are performed quarterly.

X.3.1.6 Technical Specifications

With any one auxiliary feedwater pump inoperable, restore three auxiliary feedwater pumps (2 motor, 1 steam) to operable status within 72 hours or be in Hot Shutdown within next 12 hours. This is in accordance with limiting conditions for operation of Standard Technical Specifications.

X.3.2 Reliability Evaluation

X.3.2.1 Dominant Failure Modes

The dominant failure modes are expressed for three transient situations. Success criterion is the operation of at least one of the three pump trains.

LOFW with Offsite Power Available

The unavailability of the AFWS during this type of transient is dominated by several combinations of three failure elements. These include test and maintenance outages and hardware failures in various combinations and a combination of failures in source lines from the condensate storage tank along with failure of the service water system backup.

Test and maintenance outages of turbine driven pump train and motor driven pump trains are based on monthly pump tests as well as 72 hour allowable maintenance periods for each train. The hardware failures of for the motor driven pump trains include pump failure, in-line valve failures and control signal failure to the pumps.

The hardware failures for the turbine driven pump train include pump failure, in-line valve failures and valve failures in the steam supply lines, including control of steam inlet valves. While the determination of dominant failure contributors is based on systems of this type in general, specific failure data for Farley in its early life shows a series of failure on demand due to trip throttle valve action at the steam inlet to the turbine driven pump. Failures of the supply line valves in the closed position are considered independent human errors.

LOFW with loss of Offsite Power but with Onsite AC Availabl

The conditional unavailability of the AFWS during this type of transient is dominated by the same failure contributors as in the LOFW with Offsite Power Available transient with the addition that failure of one of the two motor driven pump trains can come from potential one train failure of onsite power .

LOFW with loss of all AC, DC Available

Only the steam turbine driven pump train can be operable in this type of situation and failure contributors include test and maintenance and hardware single failure elements. Also included is a failure to manually reset steam inlet valves which are operated by AC derived compressed air.

X.3.2.2 Interdependencies

The principal noted dependency is the AC derived compressed air which operates the steam turbine steam inlet valves. Loss of AC and compressed air supply result in eventual bleed-off and fail-closed of the steam inlet valves.

X.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic,

denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.3.3.1 Short-Term

1. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

2. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube

oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

3. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

4. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.3.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level

alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

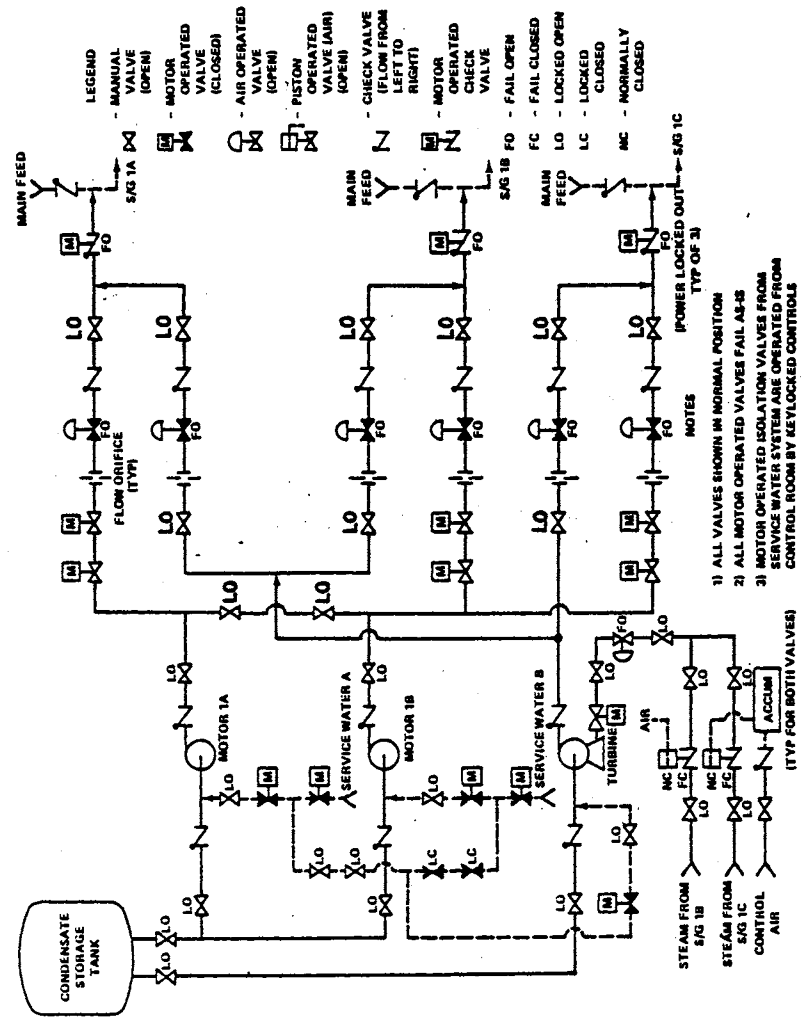
4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.3.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

2. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.



Auxiliary Feedwater System
Farley 1
Figure 1

ENCLOSURE 2

Basis for Auxiliary Feedwater System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
    1) Loss of Main Feed (LMFW)
    2) LMFW w/loss of offsite AC power
    3) LMFW w/loss of onsite and offsite AC power
    4) Plant cooldown
    5) Turbine trip with and without bypass
    6) Main steam isolation valve closure
    7) Main feed line break
    8) Main steam line break
    9) Small break LOCA
    10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:
   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.
   b. Time delay from initiating event to reactor trip.
   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).
   d. Minimum steam generator water level when initiating event occurs.
   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 22, 1979

Docket No.: 50-244

Mr. Leon D. White, Jr., Vice-President
Electric & Steam Production
Rochester Gas and Electric Corporation
89 East Avenue
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Dear Mr. White:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT ROBERT E. GINNA NUCLEAR POWER PLANT, UNIT 1

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
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X.4 GINNA

AUXILIARY FEEDWATER SYSTEM

X.4.1 System Description

X.4.1.1 Configuration -Overall Design

A simplified flow diagram of the Ginna auxiliary feedwater system (AFWS) is presented in figure 1. The AFWS consists of a main (M) AFWS and a standby (SB) AFWS. The (SB) AFWS was installed subsequent to the (M) AFWS and has recently been placed in service. The (M) AFWS consists of 3 pumps (2 motor-driven pumps, each 200 gpm, and 1 turbine-driven 400 gpm). Normally, each motor-driven pump supplies one steam generator (SG) but, with operator action either motor-driven pump can provide feedwater to both steam generators (SG). The turbine-driven pump normally provides feedwater to both SGs. Only the flow from one motor-driven AFW pump to one SG is needed to cool the plant down to the temperature where the RHR system can be used to bring the plant to safe shutdown. The steam generator would boil dry in approximately 30 minutes without any feedwater flow and a reactor trip.

All three of the (M) AFWS are located in the same room and could be rendered inoperable as a result of a high energy line break. The (SB) AFWS was added to provide independent AFWS capability following such an event.

The (SB) AFWS is in a separate plant area from the (M) AFWS. The (SB) AFWS consists of 2 motor-driven pumps. Each motor pump has a capacity of 200 gpm. The pumps are in the same room but separated by a partial wall. Thus the (SB) AFWS functions independent of the (M) AFWS.

The primary source of water for the (M) AFWS are two 30,000 gallon condensate storage tanks (CST). The tanks are non-seismic Category I and are cross-connected through locked-open manual operated valves. The (M) AFWS pumps can draw from either tank. The two condensate tanks are connected to the condenser hotwell and can be connected to a 100,000 gallon non-seismic Category I condensate storage tank. The pump that would transfer water from either the condenser hotwell or the 100,000 gallon tank to the 30,000 gallon tanks is powered from a non-safety grade supply. There is an emergency procedure for connecting to these water sources. Connection to either of these water sources requires operator action, which takes approximately 15 minutes. The (M) AFWS also has a secondary seismic Category I water source; namely, the service water system (SWS). The primary water source for the (SB) AFWS is the SWS. The SWS draws water from Lake Ontario. It is estimated to take approximately 5 minutes to connect the (M) AFWS to the SWS. There is an emergency procedure for connecting the (M) AFWS to the SWS.

X.4.1.2 Components - Design Classification

The (M) AFWS, the (SB) AFWS, and the SWS have a Class I seismic qualification. The primary source (two 30,000 gallon condensate storage tanks) and associated supply lines to the (M) AFWS pumps suction are non-Class I seismic.

X.4.1.3 Power Sources

The main and standby auxiliary feedwater systems are powered from the emergency buses. The two motor-driven pumps, associated valves and lube oil cooling system for the turbine driven pump in the main auxiliary feedwater system receive motive power from two redundant and independent AC emergency buses. The steam admission and water discharge valves and lube oil cooling systems associated with the steam turbine-driven pump in the main auxiliary feedwater system receive power from the electrical divisions indicated in Figure 1. The two motor-driven pumps and valves in the standby auxiliary feedwater system are supplied from redundant and independent AC emergency buses. The (SB) AFWS is interlocked with the (M) AFWS so that both are not simultaneously loaded onto the vital AC buses to prevent overloading the vital buses on loss of offsite power.

X.4.1.4 Instrumentation and Controls

X.4.1.4.1 Controls

Upon loss of the main feedwater system, the (M) AFWS is automatically initiated to supply water to the steam generators. Thereafter, the level in the steam generator is manually controlled from the control room by adjusting valve positions. The (SB) AFWS is manually initiated and manually controlled from the control room by adjusting valve positions.

X.4.1.4.2 Information Available to Operator

System information available to the operator in the control room to assess the performance of the auxiliary feedwater system is as follows:

.Indicating red (open) and green (close) lights associated with each electrical and pneumatic operated valve.
.Steam generator level
.Steam genarator pressure
.Auxiliary feedwater flow indication in each of the two water paths to the steam generators as related to the (M) AFWS.
.Auxiliary feedwater flow indication in each of the two water paths to the steam generator as related to the (SB) AFWS.

The (M) AFW pumps are not automatically tripped as a result of low pump suction pressure conditions. This was a potential concern because the non-seismic condensate storage tank supply lines could be severed by a seismic event causing the loss of suction to the (M) AFWS pumps. There is also no alarm or indication in the control room to alert the operator of low suction pressure conditions at AFWS pumps. However the operator does have CST level and pump discharge pressure and flow indication. Further, however, in the event of

seismic damage to the (M) AFWS primary water source, the (SB) AFWS would be available since its water souce (SWS) is seismic Category I.

X.4.1.4.3 Initiating Signals for Automatic operation

The steam turbine-driven and motor-driven pumps and corresponding valves in the (M) AFWS are automatically initiated by the following signals:

·Motor-Driven Auxiliary Feedwater Pump A
··2/3 lo-lo level in either SG
··Both main feedwater pumps trip
··Safety injection initiation
Motor-Driven Auxiliary Feedwater Pump B
··2/3 lo-lo level in either SG
··Both main feedwater pumps trip
..Safety injection initiation
·Steam Admission Valve to the Turbine-Driven Auxiliary Feedwater Pump
··2/3 lo-lo level in both steam generators
··Loss of voltage on both 4 KV buses (non-safety buses)
.Motor and Turbine Driven Pumps Discharge Valves
···pump start

The (SB) AFWS is manually initiated.

Both the main and standby auxiliary feedwater systems flow paths to the steam generators are not isolated automatically as a result of a steam or feedwater (main or auxiliary) line break. The isolation is accomplished manually.

X.4.1.5 Testing and Technical Specifications

Subsequent to this review, the licensee proposed a Technical Specification revision which provides limiting conditions of operation and periodic testing for both the (M) and (SB) AFWS. These proposed revisions have been revewed by the staff (Systematic Evaluation Program) and found acceptable. The Technical Specification revisions were approved in Amendment 29 to the Ginna operating license (DPR-18) dated August 24, 1979.

X.4.2 Reliability Evaluation

X.4.2.1 Dominant Failure Modes

LOFW with offsite power available

Failure of operator to throttle pumps and failure of operator to switch to service water supply and failure of operator to actuate the (SB) AFWS.

The condensate storage tanks have 15,000 gallons dedicated to the (M) AFWS. When the system starts, all 3 pumps have the possibility of starting. Their total capacity is 800 gpm. However, only 200 gpm

flow to one SG is necessary. To achieve the 200 gpm flow rate, the operator must either throttle or shut off some pumps. If this action is not taken, the CSTs could empty in 20 minutes. The short time interval may not allow the operator time enough to valve in the backup water source from the hotwells and 100,000 gallon tank. A procedure is available; however, it requires operator action outside the control room. The next alternative is to open a service water system valve which is outside the control room. A procedure exists and the licensee estimates 5 minutes to take this action. The final alternative is to valve in from behind the control panel the (SB) AFWS for which procedures exist. If the operator throttles the pumps correctly initially, there should be adequate time and supply to prevent a problem. The licensee estimates the steam generator boil dry time to be approximately 30 minutes which should allow sufficient time to valve in the service water.

LOFW with onsite power available

Same as for LOFW with offsite power available. For this event the condenser hotwell and 100,000 gallon backup condensate storage tank are not available since the transfer pumps are powered from non-vital AC. bus.

LOFW with only DC available

Failure of the turbine pump train.

This is the short term failure. For this condition, the turbine could eventually fail since the AC powered service water pumps are not operating. Thus, there is no water flow to cool the turbine pump lube oil. The CST (assuming 15,000 gal level) could go dry in 40 minutes and also cause failure. The backup sources from the service water and (SB) AFWS are all AC dependent and would not be available. See Recommendations.

X.4.2.2 Principal Dependencies

1. All (M) AFWS pumps are in the same room with high energy piping over-head. However, a postulated high energy line break in this room is mitigated by the installation of the (SB) AFWS in a separate plant area.

2. The DC controlled turbine lube oil pump forces oil through a heat exchanger which depends on the AC powered service water system to cool the oil. In a total loss of AC, the turbine could fail. See Recommendations.

X.4.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as

soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.4.3.1 Short-Term

1. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

   The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

2. Recommendation - The plant has AC dependent service water cooling of the lube oil for the turbine driven pump. The

turbine driven feedwater pump has an AC lube oil pump and a DC lube oil pump. These pumps direct the oil through a heat exchanger which depends on the AC powered service water system pumps to cool the oil. In the event of a total loss of AC power, lube oil cooling capability for the turbine-driven pump will be lost due to the loss of AC power to the service water pumps. The turbine-driven pump could cease to function due to the loss of lube oil cooling. The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. Subsequent to this review, the licensee conducted a test to demonstrate that the turbine-driven pump could operate for two hours without lube oil cooling water flow. The test was run for one hour and 45 minutes with the final one hour and 15 minutes of the test with the pump at rated speed,but at 50% of required plant flow. Preliminary test results indicate the pump and turbine bearing temperatures remained within allowable limits. The staff is evaluating these test results to determine if the test data will support a conclusion that the required AFW flow can be provided independent of any AC power source. Until this evaluation is complete, interim emergency procedures should be established which provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all

alternating current power to monitor pump/turbine bearng and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control fo the AFW system is needed. (See Recommendation GL-3 for the longer term resolution of this concern).

3. Recommendation GS-6 The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independetly verify that the valves are properly aligned.

- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

4. Recommendation GS-7 - The licensee should verify that the automatic start (M) AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the (M) AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.

   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

   - Testability of the initiation signals and circuits shall be a feature of the design.

   - The initiation signals and circuits should be powered from the emergency buses.

   - Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

   - The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

   - The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.4.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW systems designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits wth respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

> "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.
>
> The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.4.3.3 Long-Term

Long Term recommendations for improving the system are as follows:

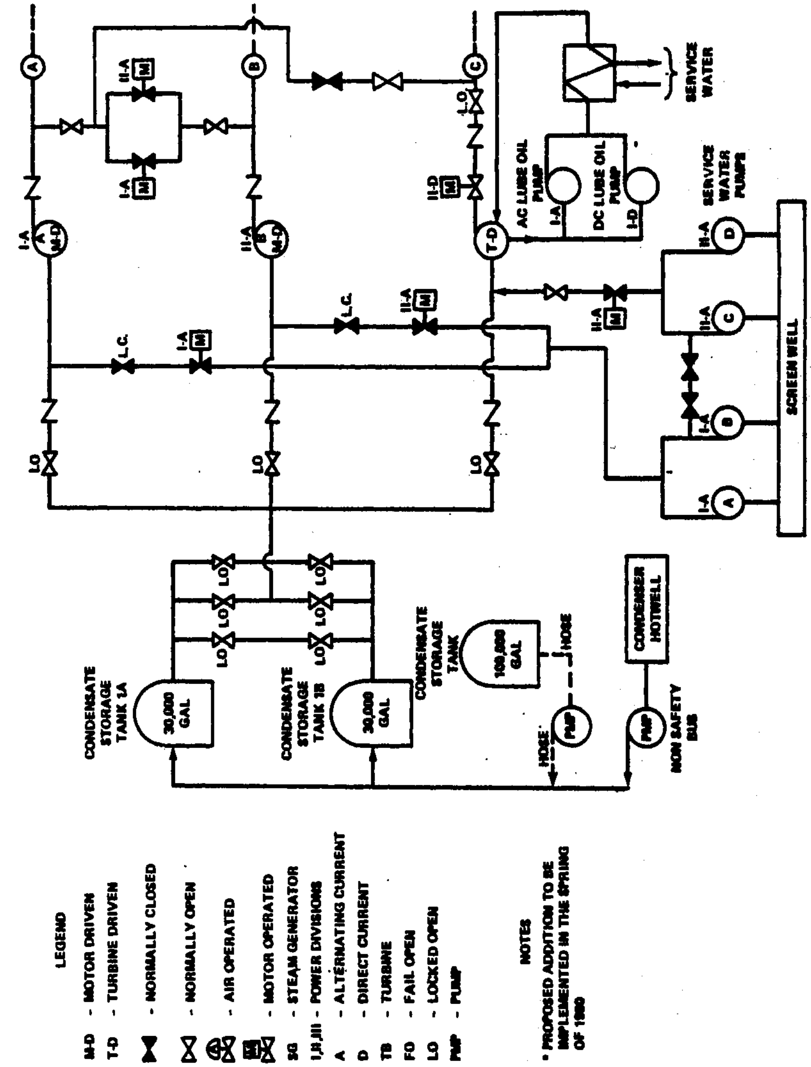
1. Recommendation - GL-3. At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least 2 hours. Conversion of direct current power to alternating current is acceptable.

2. Recommendation - The licensee should evaluate the water source capabilities (AC powered service water pumps, condensate transfer pumps and the limited inventory of condensate storage tank water gravity feed to the turbine pump suction to assure that there is a water source sufficient to supply the required AFW flow for 2 hours independent of any AC power source.

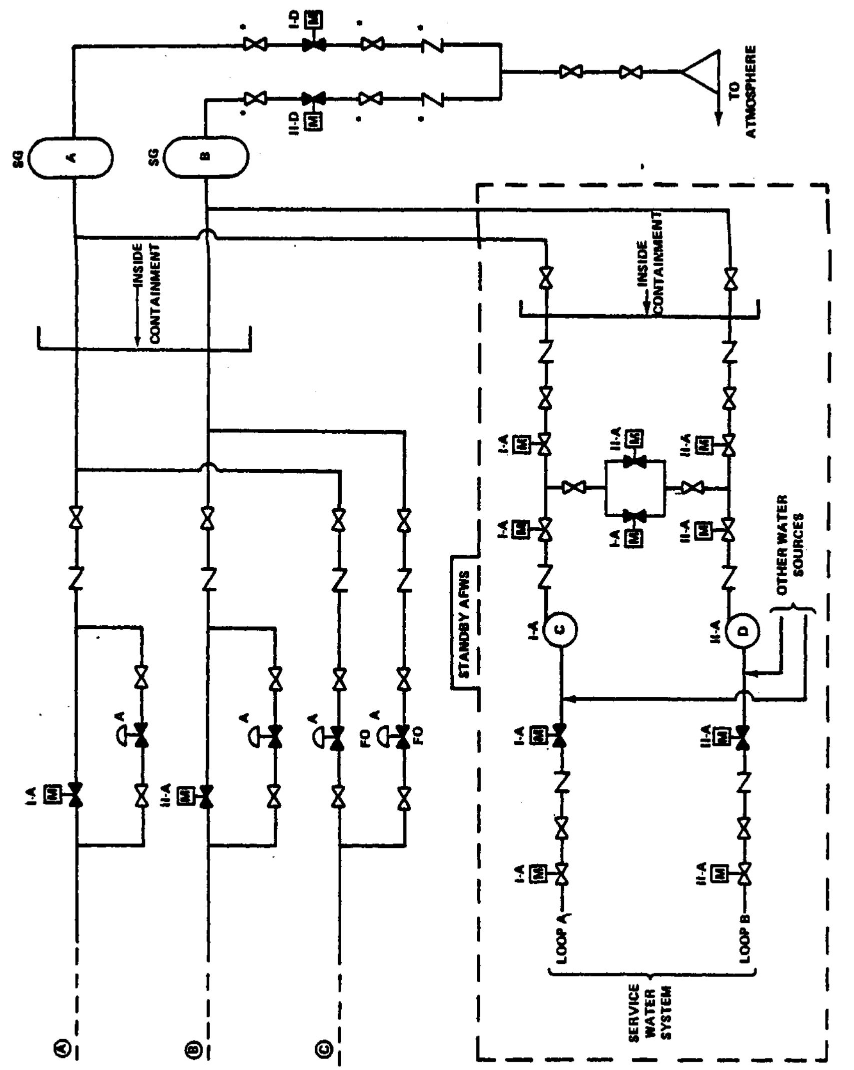
3. Recommendation - GL-5. The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

4. There is no provision for either the main or standby AFWS 's to automatically terminate flow to a depressurized steam generator and automatically provide flow to the intact steam generator. This is accomplished by the control room operator. The lack of this automatic capability will be further evaluated as part of the Systematic Evaluation Program.

5. The main and standby AFWSs will be reevaluated for internal and external missiles, seismic design requirements, and flood and tornado protection as part of the Systematic Evaluation Program.



Auxiliary Feedwater System
Ginna
Figure 1 (Sheet 1)



Auxiliary Feedwater System
Ginna
Figure 1 (Sheet 2)

ENCLOSURE 2

Basis for Auxiliary Feedwater
System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

   g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

   h. RC flow condition - continued operation of RC pumps or natural circulation.

   i. Maximum AFW inlet temperature.

   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

   l. Operating condition of steam generator normal blowdown following initiating event.

   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

   n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 21, 1979

Docket No.: 50-261

Mr. J. A. Jones
Senior Vice-President
Carolina Power and Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

Dear Mr. Jones:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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ENCLOSURE 1

X.6 (W) H. B. ROBINSON

AUXILIARY FEEDWATER SYSTEM

X 6.1 System Description

X 6.1.1 Configuration Overall Design

A simplified drawing of the H. B. Robinson auxiliary feedwater system (AFWS) is shown in Figure 1. Basically the system consists of two motor driven pumps located in the auxiliary building, each with a capacity of 300 gpm at 1300 psi. and a turbine driven pump located in the seismic portion of the turbine building with a capacity of 600 gpm at 1300 psi. The turbine-driven pump is not tornado missile protected. All three pumps take their primary suction from the seismic Category I condensate storage tank. The system is automatically started by signals identified in Section 6.1.4.3. The two motor driven pumps take suction from a common header and feed all three steam generators through lines which are cross-connected in the pump room as shown in Figure 1. There is a normally closed motor operated valve in each line to the steam generators. The AFWS discharge lines from the motor driven pumps connect to the main feed lines inside containment.

The turbine driven pump takes its source of water from the CST common header and feeds into the main feed system through three normally closed motor-operated valves. The auxiliary feedwater lines

from the turbine driven pump train connect to the feedwater regulating valve bypass lines for each individual steam generator outside containment.

The system was evaluated for high energy line breaks in the main steam, main feed lines and the AFWS itself. For the main feed and steam line breaks, at least one train of the AFWS will be able to feed at least one steam generator which is sufficient to safely shut down the plant. Remote manual action would be required to isolate the affected steam generator and AFWS line feeding that generator. For the high energy line break in the Auxiliary Feedwater System, the worst break is in the motor driven pump trains' cross connection line. In this case, since the pumps and associated motor operated valves are in the same room, these trains could be shorted out if the cross connection line is not isolated in time. The steam driven train is, however, still available to shut down the plant, provided the pump does not fail. If the pump fails, auxiliary feedwater flow would be lost, but main feedwater could still be used to supply water to the steam generators.

Sources of Water

There are three sources of water for the auxiliary feedwater system. The primary source is from a 200,000 gallon seismic Category I condensate storage tank (CST), of which 35,000 gallons are dedicated to the auxiliary feedwater system. This will last a minimum of two hours. The CST is not protected against tornado missiles. All valves from the tank to the AFWS are normally open, and are local manually operated valves.

The secondary source of water as well as long term cooling is the seismic Category I service water system and the ultimate heat sink. The piping for this system is buried or in the auxiliary building so it is protected against tornado missiles, however, the pumphouse which contains the service water pumps is not protected against tornado missiles. The valves connecting this system to the AFWS are locked closed manual valves. Thus, it would take time to open these valves. There is, however, sufficient time to open these valves before the condensate storage tank is depleted or the steam generators boil dry.

The back-up source of water is the non-seismic deep well system which has a capacity of 600 gpm. The valves that connect this system with the AFWS are manually locked closed valves.

X.6.1.2 Components - Design and Classification

Component	Environmental Qualification	Design Classification	Seismic Category
Motor Driven Pump	Ambient	ASME VIII	I
Turbine Driven Pump	"	ASME VIII	I
Piping	"	B31.1	I
Valves/Actuators	"	B31.1	I
Control & Actuation System	"	-	I
Indication	"	-	N.S.
Condensate Storage	"	-	I
Service Water System	"	-	I
Deep Well System	"	-	N.S.
Main Steam Lines to Turbine Driven Pump (connects upstream MSIV)	"	-	I
Main Feed Lines from Main Feed Block Valves to Steam Generators	"	-	I

N.S - Non Seismic Category I

X.6.1.3 Power Sources

Each motor driven pump is supplied power from its respective emergency bus which receives power from normal station transformers or separate diesel generators (DG). The three motor operated discharge valves (MOV) in the motor driven pump trains are powered from the emergency buses. The valve for steam generator A is powered from emergency bus E1 or E2. The valve for steam generator (SG) B is powered from bus E2 and the valve for SG C is from bus E1. The MOV's are normally closed and fail-as-is. The instrumentation for these trains is taken off the station batteries.

The turbine driven pump is supplied steam from all three steam generators. The steam is taken off upstream of the MSIV and passes through a motor operated valve, a check valve and goes into a common header which feeds the turbine. The motor operated valves take their power from the emergency buses. The valves from SG B&C are connected to bus E1, and the valve from SG A is connected to bus E2. The motor operated valves in the turbine pump discharge lines to the steam generator also take their power from the emergency buses with valves for SG A&C from bus E2 and SG B from bus E1. The above MOV's are normally closed and fail-as-is.

The system does not meet NRC's current power source diversity position with respect to the turbine driven pump train valves although manual action can be taken at the valves. (See recommendation GS-5)

In addition, cooling to the lube oil coolers to the turbine driven pump is from the service water system, which takes its power from the emergency busses. Upon station blackout (loss of all AC), cooling is lost to the turbine which could result in a possible shaft seizure or wiped bearings in the turbine within a short time (approximately 10 to 20 minutes), thus resulting in the loss of all AFW flow. However, the lube oil cooling water piping and valves are arranged so that

the lube oil cooler can be cooled by AFW pump flow; but the valve alignment must be changed (See Recommendation GS-5).

X.6.1.4 Instrumentation and Controls

X.6.1.4.1 Controls

The following AFWS manual controls are available in the control room:

1. Motor Driven Pump Start-Stop
2. Steam Inlet Line to Turbine - Motor Operated Valves Open-Close
3. AFW Discharge Line Motor Operated Valves - Open-Close

All other valves as well as the above can be controlled at the local stations. Steam generator level is controlled manually at the motor operated discharge valves locally or in the control room by starting and stopping the pumps.

X.6.1.4.2 Information available to Operator

The following information is available to the operator in the control room:

1. Motor Driven Pumps Start-Stop
2. Motor Operated Valves (All) Opened-Closed
3. Motor and Turbine Driven Pumps Discharge Pressure
4. Steam Generator Level
5. Steam Generator and Steam Header Pressure

This information is also available at the local control stations.

The following alarms are located in the control room:

1. Condensate Storage Tank Low Level
2. Equipment on Local Control
3. Steam Generator(s) Low Low Level Alarms
4. Steam Generator High Level Alarm
5. Low AFW Pump Discharge Pressure Alarm and Trip
6. Loss of Lube Oil

X.6.1.4.3 Initiation Signals for Automatic Operation

The following signals initiate automatic operation of the AFWS:

1. Low-Low Level on Steam Generator
   a. 2 out of 3 on one steam generator initiates the motor driven pump trains.

   b. 2 out of 3 on two steam generators initiates the turbine pump train.

2. Loss of Both Main Feed Water Pumps starts Motor Driven Pump Trains.

3. Loss of Offsite Power starts motor driven pump trains.

4. Safety Injection Signal starts motor driven pump trains.

5. Loss of Voltage (<70%) on buses 1 and 4 starts the steam driven pump train. Steam inlet valves (MOV) are operated from buses E1 and E2 and therefore will open.

X.6.1.5 Testing

The system is tested on a monthly bases with some exceptions. The pumps are run in recirculating mode monthly and the motor operated discharge valves are stroked monthly with the pumps off. All other valves in the system including the AFW pump turbine steam inlet valves are tested quarterly; however one steam inlet valve is operated monthly in conjunction with the monthly tests of the turbine driven pump. All valves are checked at the end of the tests for correct positioning both in the control room and locally. The piping in the system is hydrostatically tested every 10 years. The system is tested as a whole during refueling cycle as part of the ECCS actuation test. The only pieces of equipment not tested on a periodic bases are the locked closed valves to the service water and deep well systems. The periodic test requirements are as follows:

AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the turbine-driven and motor-driven auxiliary feedwater pumps.

Objective

To verify the operability of the auxiliary feedwater system and its ability to respond properly when required.

Specification

4.8.1 Each motor driven auxiliary feedwater pump will be started at intervals not to exceed one month, run for 15 minutes, and determined that it is operable.

4.8.2 The steam turbine driven auxiliary feedwater pump by using motor operated steam admission valves will be started at intervals not to exceed one month, run for 15 minutes, and determined that it is operable when the reactor coolant system is above the cold shutdown condition. When periods of reactor cold shutdown extend this interval beyond one month, the test shall be performed immediately following reactor heatup.

4.8.3 The auxiliary feedwater pump discharge valves will be tested by operator action at intervals not greater than one month.

4.8.4 These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

X.6.1.5 Technical Specifications

The following technical specifications apply to H. B. Robinson. The salient features are that one pump could be out of service for an indefinite period of time with no limiting condition for operation on the plant, that the instrumentation for the system could be out for all trains without limiting condition for operation.

SECONDARY STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of turbine cycle.

Objective

To define conditions of the turbine cycle steam-relieving capacity. Auxiliary Feedwater System and Service Water System operation is necessary to ensure the capability to remove decay heat from the core.

Specification

3.4.1 The reactor coolant shall not be heated above 350°F unless the following conditions are met:

a. A minimum turbine cycle steam relieving capability of twelve (12) main steam safety valves operable.

b. Two of the three auxiliary feedwater pumps must be operable.

c. A minimum of 35,000 gallons of water in the condensate storage tank and an unlimited water supply from the lake via either leg of the plant Service Water System.

d. Essential features including system piping and valves directly associated with the above components are operable.

e. The main steam stop valves are operable and capable of closing in five seconds or less.

3.4.2 The specific activity of the secondary coolant system shall be ≤ 0.10 μCi/gram DOSE EQUIVALENT I-131 under all modes of operation from cold shutdown through power operation. When the specific activity of the secondary coolant system is >0.10 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours.

The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1-2.

3.4.3 If, during power operations, any of the specifications in 3.4.1 above cannot be met within 24 hours, the operator shall initiate procedures to put the plant in the hot shutdown condition. If any of these specifications cannot be met within 48 hours, the operator shall cool the reactor below 350°F using normal procedures.

X 6.2 Reliability Evaluation

X 6.2.1 Dominant Failure Modes

The system was analyzed for three cases:

(a) loss of feedwater with offsite power available;

(b) loss of feedwater with onsite AC power available;

(c) loss of feedwater with only DC power available.

The dominant failure modes for each case are summarized below.

X 6.2.1.1 <u>LOFW with Offsite Power available</u>

The dominant failure modes are as follows:

(1) loss of condensate storage tank supply due to failure of valves in the supply line plus failure to manually actuate backup service water supply by locally opening closed valves.

(2) one train out indefinitely for maintenance plus hardware and maintenance outages in other two trains.

X.6.2.1.2 <u>LOFW with Onsite AC power available</u>

The system was analyzed assuming loss of offsite power, considering the possible loss of one of the diesel generators. The dominant failure modes for this case are similar to those discussed in the previous case.

X.6.2.1.3 <u>LOFW with Only DC Power Available</u>

The system will fail in the long-term due to reliance of turbine lube oil cooling on AC power without operator action to realign cooling water valves (See Section 6.1.3). In the short-term ($\leq$45 minutes), unavailability is dominated by maintenance and hardware failures of the turbine driven pump train and failure to manually open the steam and water MOVs which do not open without AC power.

- 13 -

(b) loss of feedwater with onsite AC power available;

(c) loss of feedwater with only DC power available.

The dominant failure modes for each case are summarized below.

X 6.2.1.1 LOFW with Offsite Power available

The dominant failure modes are as follows:

(1) loss of condensate storage tank supply due to failure of valves in the supply line plus failure to manually actuate backup service water supply by locally opening closed valves.

(2) one train out indefinitely for maintenance plus hardware and maintenance outages in other two trains.

X.6.2.1.2 LOFW with Onsite AC power available

The system was analyzed assuming loss of offsite power, considering the possible loss of one of the diesel generators. The dominant failure modes for this case are similar to those discussed in the previous case.

X.6.2.1.3 LOFW with Only DC Power Available

The system will fail in the long-term due to reliance of turbine lube oil cooling on AC power without operator action to realign cooling water valves (See Section 6.1.3). In the short-term ($\leq$45 minutes), unavailability is dominated by maintenance and hardware failures of the turbine driven pump train and failure to manually open the steam and water MOVs which do not open without AC power.

X.6.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumention can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

3. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

- The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current

power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed., (See Recommendation GL-3 for the longer-term resolution of this concern.)

5. Recommendation GS-6 - The licensee shoul confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system shoul be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circui should be upgraded to meet safety-grade requirements as indicated in

Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to intiate the AFW system from the control room.

7. The licensee should propose modifications to the Technical Specifications to provide for periodic testing of the normally locked closed service water and deep well manual valves.

8. The licensee should propose modifications to the Technical Specifications to provide for monthly testing of all steam admission valves to the turbine pump.

X.6.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for 1 hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technnical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for

operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.6.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-2 - Licensees with plants in which all (primary and alterna'.e) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

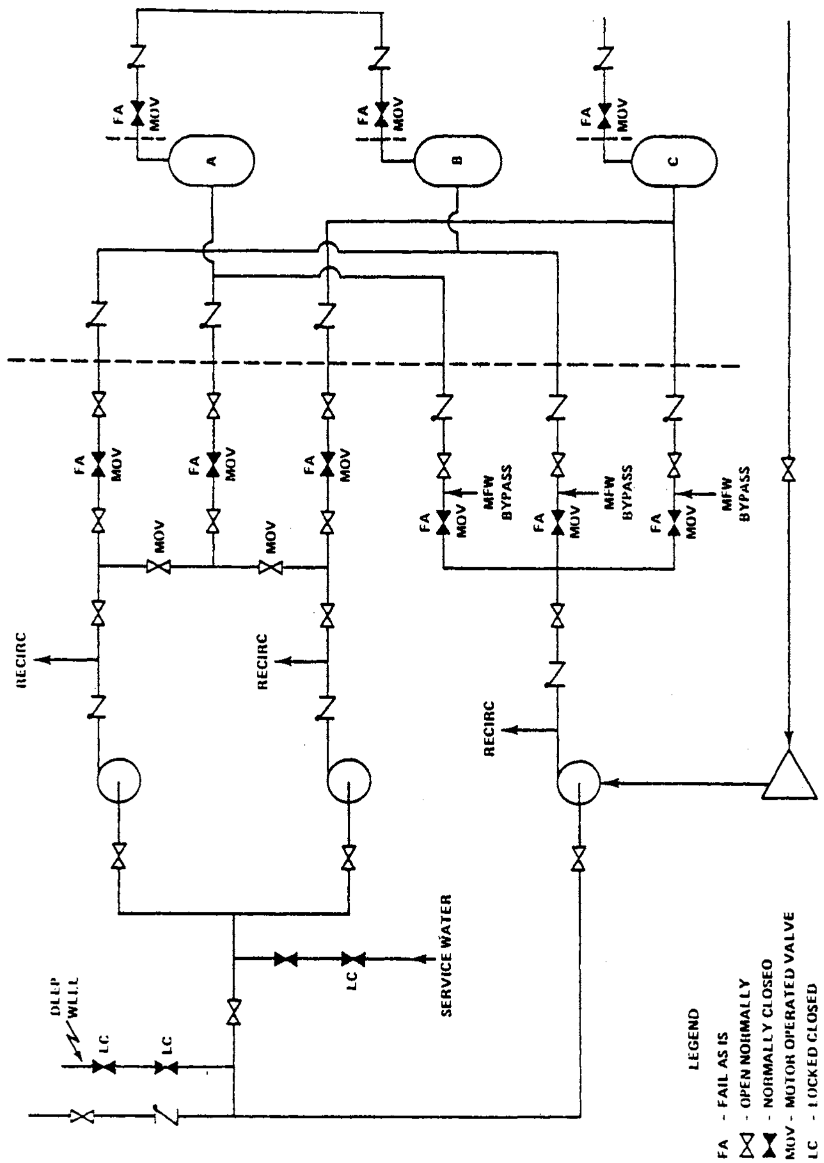
   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

3. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety grade requirements.

4. None of the AFW water sources are protected against tornado missiles. The licensee should complete an evaluation considering a postulated tornado plus a single active failure to determine any AFW system modifications or procedures necessary to assure a sufficient AFW water supply or assure that the plant can be brought to a safe shutdown condition in such an event.



Auxiliary Feedwater System
H.B. Robinson

Basis for Auxiliary Feedwater
System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

   g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

   h. RC flow condition - continued operation of RC pumps or natural circulation.

   i. Maximum AFW inlet temperature.

   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

   l. Operating condition of steam generator normal blowdown following initiating event.

   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

   n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 11, 1979

Docket No.: 50-213

Mr. D. C. Switzer, President
Connecticut Yankee Atomic Power Company
P. O. Box 270
Hartford, Connecticut 06101

Dear Mr. Switzer:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT HADDAM NECK NUCLEAR POWER PLANT

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
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ENCLOSURE 1

X.5 (W) HADDAM NECK

AUXILIARY FEEDWATER SYSTEM

X.5.1 System Description

X.5.1.1 Configuration Overall Design

Figure 1 is a simplified flow diagram of the Haddam Neck auxiliary feedwater system (AFWS). The AFWS consists of two steam turbine driven pumps* which take water through a common underground header from the demineralized water storage tank and inject it into four steam generators via main feedwater piping. The pumps discharge to a common header which supplies water via either of two possible parallel flow paths. One path feeds to the bypass line around the main feed regulating valves in the turbine building. By using the bypass line, the main feedwater bypass control valve can be used to regulate flow to the steam generators individually. The other flow path supplies water from the discharge header through a motor operated valve to the main feedwater piping downstream of the feedwater check valve inside containment.

Steam to the turbine driven pumps is taken from all four steam generators upstream of the main steam isolation valves from a common header.

*The licensee indicated that it plans addition of a motor driven AFWS pump.

The header is normally split in such a manner that two steam generators supply one turbine pump while the other two steam generators supply the remaining turbine pump.

The system has no automatic initiation capability and relies on manual initiation from the control room for all conditions. However, on loss of control air, for whatever reason, the turbine driven pumps would start due to the fail-open feature of steam inlet valves and deliver AFW through the main feedwater bypass control valves which also fail open on loss of air.

No electrical power is necessary to operate these valves because the controls at the panel mechanically initiate or remove control air. (Control air passes through panel via copper tubing.)

The primary source of water is from the demineralized water storage tank (Minimum capacity 50,000 gallons by Technical Specifications) which is always lined up to the pump suction header via locked open manually operated valves. The secondary source of water is the primary water storage tank (Minimum volume of 80,000 gallons by Technical Specifications) which must be transferred to the demineralized water storage tank before use. As a backup to these sources, the recycle water storage tank (100,000 gallons) is normally always available and also must be transferred to the demineralized water storage tank before use. Long term sources of makeup water include the water treatment system using a well pump, the well pump without use of the water treatment system and a diesel driven fire protection system pump. All water sources must eventually come via the demineralized water storage tank.

X.5.1.2 Components Design Classification

The seismic design and safety classification of components for the Haddam Neck plant are being reviewed as part of the Systematic Evaluation Program. The safety classification and seismic design requirements for the plant as compared to today's requirements are too detailed and complex to provide a meaningful explanation in this report. Refer to the details available as part of SEP for this information. The overall design of the auxiliary feedwater system, including the demineralized water storage tank and primary water storage tank,are considered to be seismic Category I based on the Licensee's standards. The adequacy of these seismic criteria are also being evaluated as part of SEP.

X.5.1.3 Power Sources

No electric power sources are directly used for valve operation or turbine pump startup to use the main feedwater bypass control valve flow path. To use the alternate flow path directly to the feedwater inlet piping at the steam generators, a single motor operated valve, powered from a vital bus is used.

Compressed air is used to operate the steam inlet valves and the main feedwater bypass line control valves. These valves are opened or closed at the control panel by controls that are essentially control valves that control the air pressure from the compressed air header to the valve operators. All valves fail in the open position upon loss of air pressure. The compressed air system includes three air compressors and three air receivers for control air. All of the compressors can be powered by the diesel generators.

The AFW pumps have a self-contained lube oil pumping system (shaft driven) but require service water to cool the lube oil. The service water is supplied on a continuous basis to the lube oil coolers (one service water train to each pump). However, the pumps will start and operate for an unspecified time without cooling water. Subsequent to this review, the licensee indicated it is presently in the process of modifying the AFW system to eliminate the need for service water for the AFW turbine driven pump lube oil coolers. The modification will provide a self-contained bearing oil cooling system for each AFW pump. Water will be drawn from the pump first stage discharge and will circulate through all necessary pump and turbine pump and turbine bearings and will return to the AFW pump suction.

X.5.1.4 Instrumentations and Controls (In Control Room)

X.5.1.4.1 Controls

Steam generator level is controlled manually from the control room by varying turbine speed or throttling the feedwater bypass control valve or a combination of both. When the alternate path to the steam generators is used through the motor operated valve directly to the feedwater piping inside containment, level is controlled by turbine speed control.

Controls for the valves to initiate the auxiliary feedwater system through either of the two flow paths are located in the control room. The controls for the normal flow path through the feedwater bypass line are independent of electrical power.

X.5.1.4.2 Information Available to the Operator

I. Alarms

1. Demineralized water storage tank low level
2. Control air system low pressure alarm
3. Discharge header high temperature alarm (indicates backflow from main feedwater system to discharge header via leaky check valve)
4. Hi/Lo steam generator level alarms

II. Indication

1. Electrical position indication for motor operated isolation valve in an alternate flow path
2. Output pressure of controllers to bypass flow control valves and turbine inlet valves (indirect indication of valve position and turbine speed)
3. Steam pressure at inlet to turbines
4. Discharge pressure from pumps
5. Steam generator level
6. Demineralized water storage tank level and temperature

X.5.1.4.3 Initiating Signals for Automatic Control

Not applicable - manual AFWS initiation

X.5.1.5 Testing

The auxiliary feedwater pumps, steam inlet valves, and controls are tested monthly by isolating pump discharge and starting pump from the control room and checking discharge pressure. This same test is performed following return of system to operation after maintenance. A flow test of the auxiliary feedwater pumps is performed annually.

Valve position is verified monthly and the active valves are cycled quarterly. All valves,active and manual,are cycled annually and the stroke times of the active valves verified.

The controls for all valves are used during valve testing for control operability check.

Technical Specifications

The reactor shall not be critical (except for determination of "just critical" rod position and low power tests at or below 10 percent of full power) unless the following conditions are met:

1. One steam driven auxiliary feedwater pump available
2. A minimum of 50,000 gallons in the demineralizer water storage tank and an additional 80,000 gallons in the primary water storage tank.
3. System piping and valves directly associated with the above components operable.

Licensee is planning to convert to standard Technical Specifications and communications with NRC have been started in this regard. In a letter dated June 1, 1979 in response to Bulletin 79-06A, the licensee submitted a license amendment request proposing more comprehensive technical specifications to further assure the availability of the AFW system. The proposed changes include a requirement that both AFW pumps be operable when the reactor is critical and a provision that limits the time that one AFW pump train can be inoperable. The proposed change is currently under staff review.

X.5.2 Reliability Analysis

X.5.2.1 Dominant Failure Modes

LOFW with Offsite Power Available

The principal dominant failure modes include two single failures associated with human failure. One is a human failure to restore to open, following a maintenance action, the suction line valve from the demineralized storage water tank. The second is the human failure to initiate the AFWS upon evidence of need. The latter contributor is reduced to some extent due to recent NRC Bulletin 79-06A for operator personnel specifically dedicated for AFWS initiation.

Other dominant failure modes include failure to reopen valves in both of two systems, and long term allowable maintenance in one pump system combined with hardware or human failure associated with the other pump system.

LOFW with Loss of Offsite Power

Same as above.

LOFW with Loss of Offsite and Onsite AC

The dominant failure is loss of both pumps due to lack of lube oil cooling from loss of all AC.

X.5.2.2 Interdependencies

The principal interdependencies noted are the common valve in the storage tank line and the AC dependence for cooling of the steam driven pumps.

X.5.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.5.3.1 Short Term

1. Recommendation GS-1 - The licensee should propose* modifications to the Technical Specifications to limit the time period that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

*As discussed in Section 5.15 the licensee has proposed Technical Specification modifications for AFW system which are currently under review by the staff.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be incorporated into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

3. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable.* Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed.

*As noted in Section 5.1.3, the licensee is proceeding with AFW system modifications to provide cooling of the turbine driven AFW pump lube oil which is independent of alternating current power.

5. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:
   - Procedure should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendetion GS-8 - The licensee should install a system to automatically initiate AFW system flow. This system need not be safety-grade; however, in the short-term, it should meet the criteria listed below, which are similar to Item 2.1.7a of NUREG-0578. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.
   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

- Testability of the initiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- Any alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

7. Recommendation -

a. According to Haddam Neck surveillance procedure No. 5.1-13, the monthly operational check of the auxiliary feedwater pumps is currently performed by closing a manual valve in the common discharge header of both pumps, isolating the normal flow path of the auxiliary feedwater system. A parallel flow path is available by manual operation from the control room through motor operated valve MOV-35. The

monthly pump test should be performed by isolating the pumps individually such that one pump is always available for normal AFW system operation. When the system is converted to automatic operation, then the existing procedure will have to be changed to individual pump isolation tests to allow automatic initiation.

b. According to Haddam Neck surveillance procedure No. 5.1-14, the annual flow capacity test of the AFW pumps is currently performed either at power or in hot standby. During the test temporary piping is connected to a valved flange in the common discharge header to divert flow away from the normal flow paths and direct it to the yard sewers via the temporary piping. This diverts flow from both AFW pumps while the isolation valve in the flange connection is open. This test should not be conducted when the plant is at power since both AFW pumps' availability is affected.

X.5.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety related equipment in the room.

3. Recommendation - The licensee should implement the following requirements which are specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require a local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.5.3.3 Long Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

Licensees with plants in which the primary AFW system water supply passes through valves in a single flowpath, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation- GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

3. Recommendation - There is a common crossconnect line with no isolation valves between the two parallel flow paths on the S/G's. A break in this section cannot be isolated in the present design and the total system would be unavailable. It is recommended that some modifications be made (such as isolation valves) to provide isolation when necessary and assure a means of supplying AFW flow following isolation of such a break. The licensee has begun design plans to add a motor driven pump to the system. The licensee should introduce the flow from this third pump in such a manner that a break in this crossconnect

line will not result in the loss of all pumps. Also the licensee should 1) install the third pump with appropriate valves in the pump discharge line connections to meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position ASB 10-1; namely, to maintain the capability to supply the required AFW flow to the steam generators with a postulated pipe break anywhere in the AFW pump discharge lines plus a single active failure, or 2) describe how the plant can be brought to a safe shutdown condition by use of other available systems following such a postulated event.

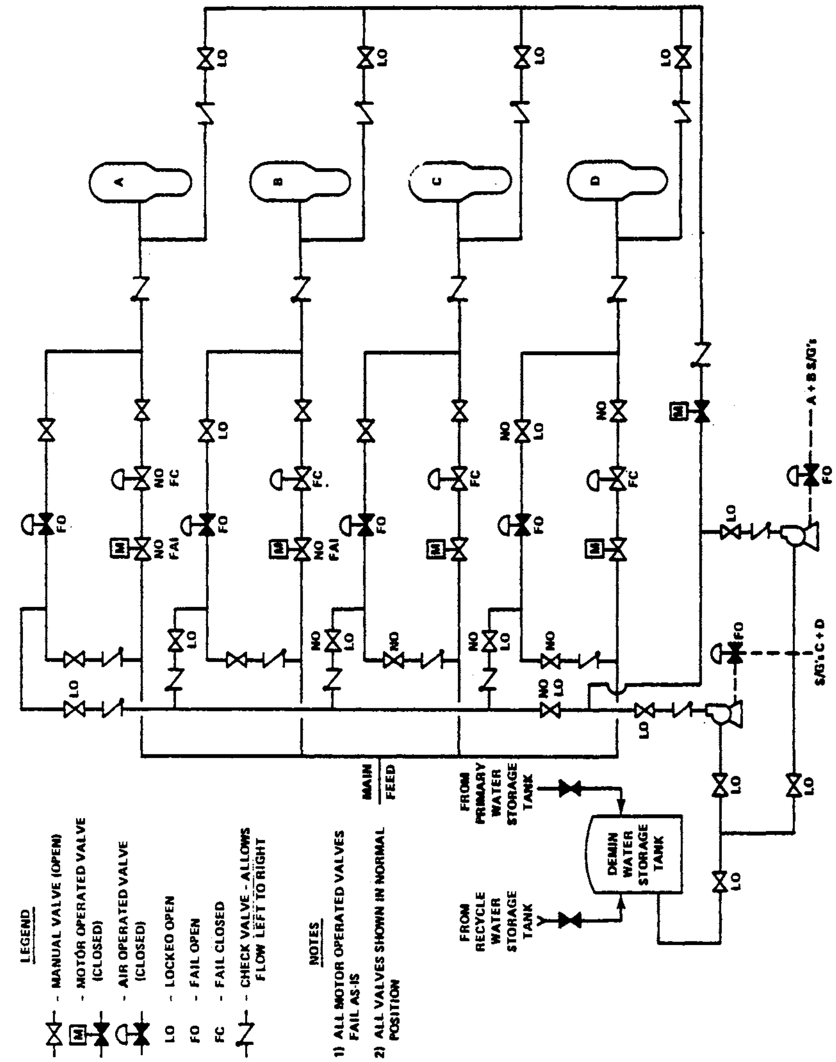
4. The AFW system itself is not designed to withstand a passive failure at all points within the system. A pipe break in a normally pressurized portion of the AFW system can be isolated by operation of manual valves outside the control room. An alternate flow path to all four S/G's would be available following such isolation. The motor driven main feedwater pumps may also be available in this event since no transient should result to cause a loss of non-vital power. For the same reasons, the main feed pumps may also be available following a break in any portion of the AFW system that is not normally pressurized even though the AFW system could be disabled. Further review, including the main feedwater system and time available for operator action, should be conducted to determine if this design has protection equivalent to today's requirements (pipe break

and single active failure). This review is being conducted as a part of Systematic Evaluation Program (SEP).

5. The Systematic Evaluation Program (SEP) will re-evaluate the plant with regard to

   a. internally and externally generated missiles, pipe whip and jet impingement quality and seismic design requirements earthquakes, tornadoes, floods and failure of nonessential systems

   b. the possible need for automatic termination of feedwater flow to a depressurized steam generator and providing flow to the intact steam generator(s). This is accomplished by the control room operator.



Auxiliary Feedwater System
Haddam Neck
Figure 1

ENCLOSURE 2

Basis for Auxiliary Feedwater System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
    1) Loss of Main Feed (LMFW)
    2) LMFW w/loss of offsite AC power
    3) LMFW w/loss of onsite and offsite AC power
    4) Plant cooldown
    5) Turbine trip with and without bypass
    6) Main steam isolation valve closure
    7) Main feed line break
    8) Main steam line break
    9) Small break LOCA
    10) Other transient or accident conditions not listed above that require AFW for mitigation

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

    - Maximum RCS pressure (PORV or safety valve actuation)
    - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
    - RCS cooling rate limit to avoid excessive coolant shrinkage
    - Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 7, 1979

Docket No. 50-247

Mr. William J. Cahill, Jr.
Vice-President
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Dear Mr. Cahill:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT INDIAN POINT NO. 2 NUCLEAR POWER PLANT

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subjec facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
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ENCLOSURE 1

X.7 (W) INDIAN POINT 2 & 3

AUXILIARY FEEDWATER SYSTEM (AFWS)

X.7.1 System Description

X.7.1.1 Configuration - Overall Design

A simplified drawing of the AFW systems for Units 2 and 3 are shown in Figures 1 and 2 respectively. The system is basically the same for both units, although there are differences in the actuation system and diesel generators. The system consists of one turbine-driven pump (capacity 800 gpm at 1350 psia) and two motor-driven pumps, each with a capacity of 400 gpm at 1350 psia. A flow of 200 gpm to each of two out of four steam generators (SG) is required for safe shutdown. Indian Point Unit 2 and 3 steam generators would boil dry in approximately 35 minutes and 24 minutes respectively without any feedwater flow, assuming a reactor trip.

The AFW water supply consists of a primary source, a secondary source, and a long-range source. The primary source is one seismic Category I condensate storage tank with a total capacity of 600,000 gallons. Of this total volume, 360,000 gal. is dedicated for AFWS use. When the water level in the condensate storage tank reaches the 360,000 gal. low value, a valve automatically closes isolating the condensate storage tank outlet from all other systems. The secondary water source is a 1.5 million gal. city water storage tank which is shared between Units 2 & 3. This backup water supply can be manually initiated from the control room. The long range water source is the city water supply. Each motor-driven AFW pump supplies water to two steam generators. The turbine-driven pump is headered to supply all four steam generators. Motive steam to the turbine-driven pump is from two steam generators; the piping configuration is such that either one or both of these steam generators can provide steam to the turbine-driven pump. The AFW system is automatically actuated, but the operator has to control flow rate to the steam generators remote-manually.

X.7.1.2 Components - Design Classification

The components of the AFWS for IP-2 and IP-3 are classified seismic Class I. The motor-driven pumps and AFW system instrumentation and controls are supplied from Class 1E power sources, except for IP-2 flow control valves (see Section 7.1.4.1 below).

X.7.1.3 Power Sources

In IP-2 and IP-3, the motor-driven pumps receive power from independent AC emergency buses. Pneumatic-operated alves in the steam inlet line to the turbine-driven pump and the AFW flow paths receive power as noted in Figures 1 and 2 and as discussed below.

X.7.1.4 Instrumentation and Controls

X.7.1.4.1 Controls

The AFWS is automatically initiated. Flow control to the four steam generators is through eight air-operated valves located on the discharge side of the pumps, which are normally 35% open. After actuation

of the FW pumps, level in the SGs is maintained manually from the control room by positioning the flow control valves. Each valve can be positioned from the control room via electric/air converters. Each motor-driven auxiliary FW pump has discharge flow paths to two steam generators, each provided with a valve position controller. The turbine-dirven pump has discharge flow paths, each provided with a valve position controller. Air to these valves is from a common header which is supplied by independent air compressors powered from separate emergency diesel generators. The air supply to the valves is backed up by an emergency high pressure nitrogen (bottle) system.

In IP-3, the valve position controllers associated with one motor-driven pump (31) and the turbine-driven pump receive power from independent safety grade instrument buses with backup battery inverters. The controllers associated with the remaining motor-driven pump currently receive power from a safety grade instrument bus. However, the licensee has indicated that a design modification is in progress to supply this bus with a battery inverter system.

In IP-2, all the valve position controllers receive power from the same non-safety grade bus, but fail open on loss of power. (See short term recommendation #7.)

In addition to remote control from the control room, all of the AFW pumps and regulating valves can be operated locally in the auxiliary feedwater building. All regulating valves are equipped with manual operators and equalizing valves for the control air to take the pnuematic operators out of service. With the local steam generator level indication noted in X.7.1.4.2. the level in each of the steam generators can be maintained and controlled from the auxiliary feedwater building without any assistance from the control room.

X.7.1.4.2 Information Available to Operator

System information available to the operator in the control room to assess the performance of the auxiliary feedwater system is as follows:

- Pump on-off-auto trip lights
- Aux feedwater flow path control valves position indication
- Primary source water level indication and alarm
- IP-2 and 3 secondary source high and low water level alarm (alarms located only in IP-2 control room.)
- Aux FW flow indication to each steam senerator
- Steam generator levels

X.7.1.4.3 Initiating Signals for Automatic Operation

IP-3

The auxiliary feedwater pumps are automatically started on receipt of any of the following signals:

Steam Driven Feedwater Pump

1) 2/3 low-low water level in any 2/4 SGs
2) Loss of offsite power concurrent with a main turbine-generator trip

Motor-Driven Feedwater Pumps

1) 2/3 Low-Low Water Level in any one steam generator
2) Loss of either main feed pump
3) Safety injection trip signal
4) Loss of offsite power concurrent with a main turbine-generator trip

IP-2

Steam Driven Feedwater Pump

1) 2/3 Low-Low Water Level in any 2/4 SGs
2) Loss of offsite power concurrent with a main turbine-generator trip.

Motor-Driven Feedwater Pumps

1) 2/3 Low-Low Water Level in any one steam generator
2) Loss of either main feed pump
3) Safety injection trip signal
4) Loss of offsite power concurrent with a main turbine-generator trip

Main steam or main feedwater line break isolation is accomplished automatically in IP-2 and IP-3.

The design of the AFWS does not have the capability to automatically terminate feedwater flow to a depressurized steam generator and provide flow to the intact steam generator. This is accomplished by the operator.

X.7.1.5 Testing

The AFWS is tested periodically in accordance with the following Technical Specification requirements:

Indian Point 2 Specification - Testing Requirements

1.a Each motor-driven auxiliary feedwater pump will be started at intervals not greater than every month with full flow established to the steam generators once every refueling.

b The steam turbine driven auxiliary feedwater pump will be started at intervals not greater than six months with full flow established to the steam generators once every refueling.

c The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.

2. These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

Indian Point 3 - Specification - Testing Requirements

1.a Each auxiliary feedwater pump will be started manually from the control room at monthly intervals with full flow established to the steam generators once every refueling.

b The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.

c Backup supply valves from the city water system will be tested once every refueling.

2. Acceptance levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least fifteen minutes.

X.10.1.6 Technical Specification

The limiting conditions of operation for Indian Point 2 and 3 AFWS are contained in the following Technical Specifications:

Indian Point 2 - Specification

A. The reactor shall not be heated above 350°F unless the following condition are met:

(1) A minimum ASME code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Two of the three auxiliary feedwater pumps must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tanks and a backup supply from the city water supply.

(4) System piping and valves directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) The total iodine activity of I-131 and I-133 on the secondary side of the steam generator shall be less than or equal to 0.15 μCi/cc.

B. If during power operations any of the conditions of 3.4.A above can not be met within 48 hours the operator shall start to shutdown and cool the reactor below 350°F using normal operating procedures.

Indian Point 3 - Specification

A. The reactor shall not be heated above 350°F unless the following condition are met:

(1) A minimum ASME code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Two of the three auxiliary feedwater pumps must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tank.

(4) System piping and valves directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) Two steam generators capable of performing their heat transfer function.

(7) City water system piping and valves directly associated with providing backup supply to the auxiliary feedwater pumps are operable.

B. If during power operations any of the conditions of 3.4.A above can not be met within 48 hours the operator shal start to shutdown and cool the reactor below 350°F using normal operating procedures.

X.7.2 Reliability Evaluation

X.7.2.1 Dominant Failure Modes

Loss of MFW with offsite power available

The dominant failure mode for this transient is the failure to provide sufficient water to the suction of the AFWS pumps. There are two manual valves in the supply line from the condensate storage tank; the inadvertent closure of either of these valves cuts off this water supply. In the event of an AFWS demand, operator action would be required to either open the closed valve (locally) or to manually open the valves in the supply line from the alternate water sources (city water) before pump damage occurs. Thus the dominant failure mode is the human error of inadvertently closing a valve in the CST supply line, coupled with the failure of the operator to manually reopen the closed valve or open the valves from the backup water supply.

A second important failure mode was also noted in this evaluation. The Indian Point Technical Specifications and LCOs require only that two of the three AFWS be operable, thus allowing the possibility that one train could be out of service indefinitely. This, in effect, reduces a three train system to a two train system, and thus reduces the predicted AFWS reliability to some degree. Revision of the Technical Specifications/LCOs to the present requirements (in the standard Technical Specifications) would make this failure mode much less significant.

Loss of MFW with only onsite AC power available

Because the dominant failure modes discussed above are not dependent on the source of AC power (onsite or offsite), these modes are also dominant for this transient event.

Loss of MFW with only DC power available

In this transient, loss of both offsite and onsite AC power is postulated to occur, so that the AFWS is reduced to only the steam-driven pump train. Thus failures in this train alone would be sufficient to fail AFWS, for this transient. The dominant failure mode for this case is that the train is out of service for maintenance, for the reason that current Technical Specifications and LCOs specify no time limit that the train could be out of service. Thus the revision of the Technical Specifications and LCOs mentioned for the above cases also would be of significant benefit for this case.

X.7.2.2 Principal Dependencies

The principal dependency found in this analysis is, as discussed above, the manual valves located in the feedwater supply line common to all AFWS pumps and the possible unlimited outage of one pump.

X.7.3 Recommendations for this Plant*

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system realiability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant-specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

* Recommendations apply to IP-2 and 3 unless otherwise stated.

X.7.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

3. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

   The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

4. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures shouls include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

5. Recommendation GS-6 - The licensee should confirm flow path available ability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals ans associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation sginals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.
   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
   - Testability of the initiation signals and circuits shall be a feature of the design.

- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and curcuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

Indian Point 2

7. Recommendation - The pneumatic-operated valves in the steam supply line to the turbine-driven AFW pump, and all of the pneumatic-operated AFW flow control valves derive their power from the same non-safety grade bus. Although these valves are designed to fail open upon the loss of air or power, thereby assuring auxiliary feedwater flow to the steam generators upon such losses, it cannot be concluded that all failures will result in opening the valves. The consequences of voltage degradation should be analyzed as well as other failures (e.g., restricted air flow)

to assure that such events would not incapacitate the auxiliary feedwater system the licensee should establish suitable emergency procedures to assure AFWS function for such events.

X.7.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accompolished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria

should include demonstrating that the pumps emain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operatin should propose Technical Specification to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.7.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s) should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

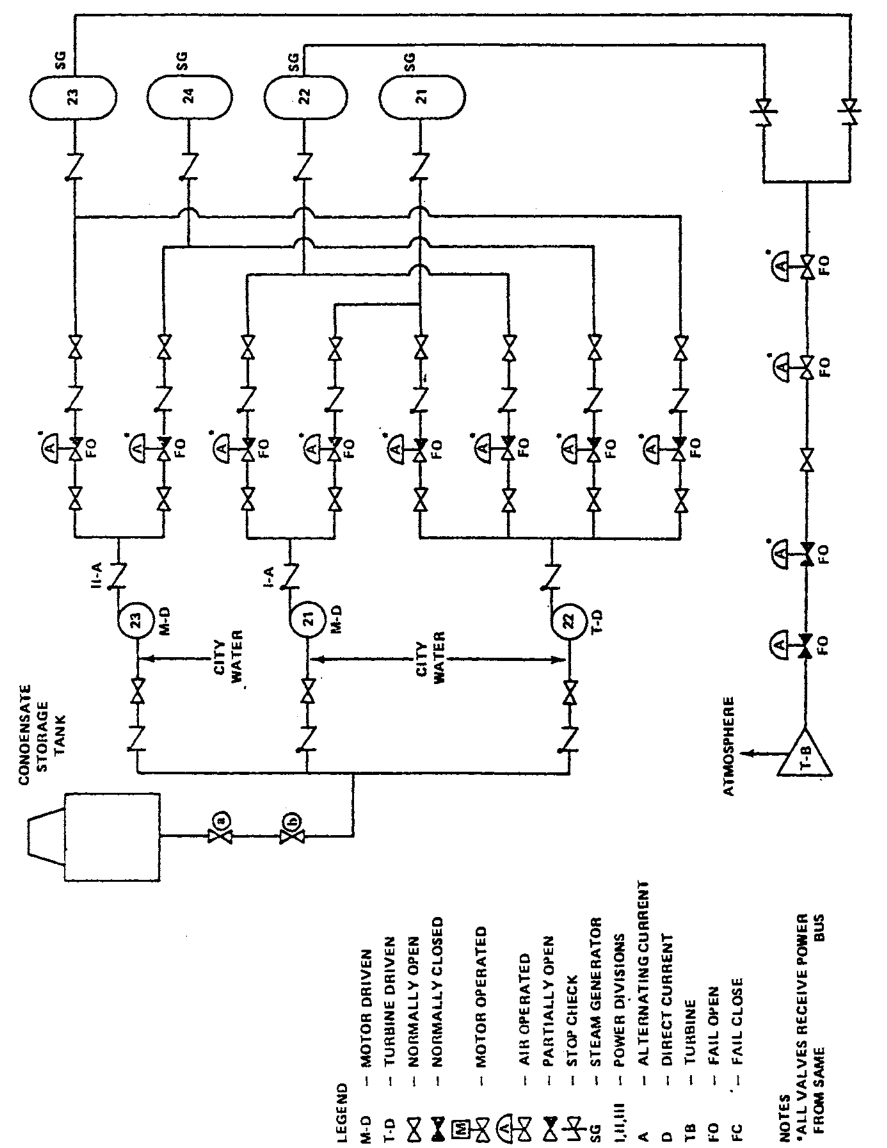
3. Recommendation - The two motor-driven pumps and the turbine driven pump are located in the same room. The licensee should evaluate the capability of the design to withstand a) environmental conditions (steam, flooding, pipe whip and jet impingement) resulting from a pipe break, b) internally generated missiles.

The licensee should evaluate the postulated pipe breaks stated above and (1) determine any AFW system design changes or procedures necessary to detect and isolate the break and direct the required feedwater flow to the steam generator(s) before they boild dry or (2) describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.

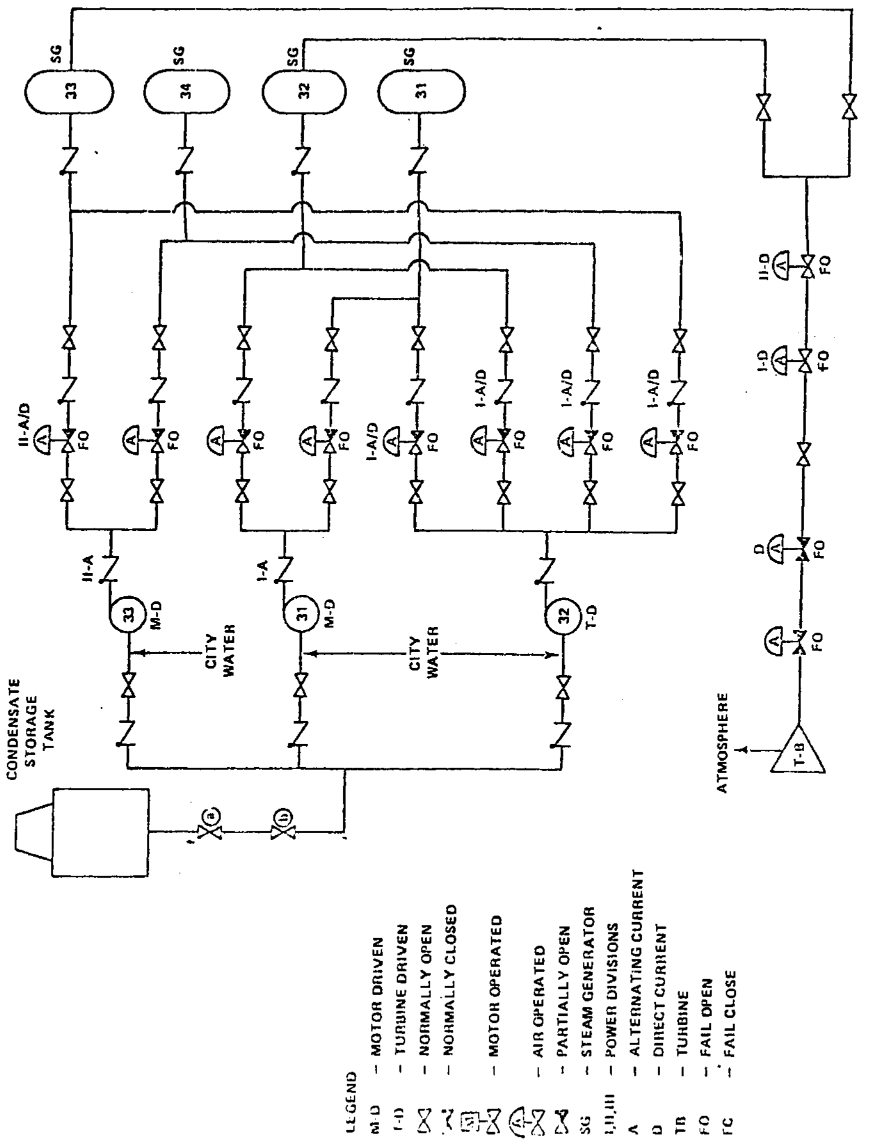
Indian Point 2

4. Recommendation - This is the same concern as that addressed in short term recommendation number 7.

The licensee should complete the modification described in Section 7.1.4.1 above that will supply power to these controllers from separate safety grade buses.



Auxiliary Feedwater System
Indian Point-2
Figure 1



Auxiliary Feedwater System
Indian Point-3
Figure 2

ENCLOSURE 2

Basis for Auxiliary Feedwater
System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

   g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

   h. RC flow condition - continued operation of RC pumps or natural circulation.

   i. Maximum AFW inlet temperature.

   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

   l. Operating condition of steam generator normal blowdown following initiating event.

   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

   n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 7, 1979

Docket No.: 50-286

Mr. George T. Berry
General Manager and Chief Engineer
Power Authority of the State of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT INDIAN POINT NO. 3 NUCLEAR POWER PLANT

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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ENCLOSURE 1

X.7 (W) INDIAN POINT 2 & 3

AUXILIARY FEEDWATER SYSTEM (AFWS)

X.7.1 System Description

X.7.1.1 Configuration - Overall Design

A simplified drawing of the AFW systems for Units 2 and 3 are shown in Figures 1 and 2 respectively. The system is basically the same for both units, although there are differences in the actuation system and diesel generators. The system consists of one turbine-driven pump (capacity 800 gpm at 1350 psia) and two motor-driven pumps, each with a capacity of 400 gpm at 1350 psia. A flow of 200 gpm to each of two out of four steam generators (SG) is required for safe shutdown. Indian Point Unit 2 and 3 steam generators would boil dry in approximately 35 minutes and 24 minutes respectively without any feedwater flow, assuming a reactor trip.

The AFW water supply consists of a primary source, a secondary source, and a long-range source. The primary source is one seismic Category I condensate storage tank with a total capacity of 600,000 gallons. Of this total volume, 360,000 gal. is dedicated for AFWS use. When the water level in the condensate storage tank reaches the 360,000 gal. low value, a valve automatically closes isolating the condensate storage tank outlet from all other systems. The secondary water source is a 1.5 million gal. city water storage tank which is shared between Units 2 & 3. This backup water supply can be

manually initiated from the control room. The long range water source is the city water supply. Each motor-driven AFW pump supplies water to two steam generators. The turbine-driven pump is headered to supply all four steam generators. Motive steam to the turbine-driven pump is from two steam generators; the piping configuration is such that either one or both of these steam generators can provide steam to the turbine-driven pump. The AFW system is automatically actuated, but the operator has to control flow rate to the steam generators remote-manually.

X.7.1.2 Components - Design Classification

The components of the AFWS for IP-2 and IP-3 are classified seismic Class I. The motor-driven pumps and AFW system instrumentation and controls are supplied from Class 1E power sources, except for IP-2 flow control valves (see Section 7.1.4.1 below).

X.7.1.3 Power Sources

In IP-2 and IP-3, the motor-driven pumps receive power from independent AC emergency buses. Pneumatic-operated valves in the steam inlet line to the turbine-driven pump and the AFW flow paths receive power as noted in Figures 1 and 2 and as discussed below.

X.7.1.4 Instrumentation and Controls

X.7.1.4.1 Controls

The AFWS is automatically initiated. Flow control to the four steam generators is through eight air-operated valves located on the discharge side of the pumps, which are normally 35% open. After actuation of the FW pumps, level in the SGs is maintained manually from the control room by positioning the flow control valves. Each valve can be positioned from the control room via electric/air converters. Each motor-driven auxiliary FW pump has discharge flow paths to two steam generators, each provided with a valve position controller. The turbine-dirven pump has discharge flow paths, each provided with a valve position controller. Air to these valves is from a common header which is supplied by independent air compressors powered from separate emergency diesel generators. The air supply to the valves is backed up by an emergency high pressure nitrogen (bottle) system.

In IP-3, the valve position controllers associated with one motor-driven pump (31) and the turbine-driven pump receive power from independent safety grade instrument buses with backup battery inverters. The controllers associated with the remaining motor-driven pump currently receive power from a safety grade instrument bus. However, the licensee has indicated that a design modification is in progress to supply this bus with a battery inverter system.

In IP-2, all the valve position controllers receive power from the same non-safety grade bus, but fail open on loss of power. (See short term recommendation #7.)

In addition to remote control from the control room, all of the AFW pumps and regulating valves can be operated locally in the auxiliary feedwater building. All regulating valves are equipped with manual operators and equalizing valves for the control air to take the

pnuematic operators out of service. With the local steam generator level indication noted in X.7.1.4.2, the level in each of the steam generators can be maintained and controlled from the auxiliary feedwater building without any assistance from the control room.

X.7.1.4.2 Information Available to Operator

System information available to the operator in the control room to assess the performance of the auxiliary feedwater system is as follows:

- Pump on-off-auto trip lights
- Aux feedwater flow path control valves position indication
- Primary source water level indication and alarm
- IP-2 and 3 secondary source high and low water level alarm (alarms located only in IP-2 control room.)
- Aux FW flow indication to each steam senerator
- Steam generator levels

X.7.1.4.3 Initiating Signals for Automatic Operation

IP-3

The auxiliary feedwater pumps are automatically started on receipt of any of the following signals:

Steam Driven Feedwater Pump

1) 2/3 low-low water level in any 2/4 SGs

2) Loss of offsite power concurrent with a main turbine-generator trip

Motor-Driven Feedwater Pumps

1) 2/3 Low-Low Water Level in any one steam generator
2) Loss of either main feed pump
3) Safety injection trip signal
4) Loss of offsite power concurrent with a main turbine-generator trip

IP-2

Steam Driven Feedwater Pump

1) 2/3 Low-Low Water Level in any 2/4 SGs
2) Loss of offsite power concurrent with a main turbine-generator trip.

Motor-Driven Feedwater Pumps

1) 2/3 Low-Low Water Level in any one steam generator
2) Loss of either main feed pump
3) Safety injection trip signal
4) Loss of offsite power concurrent with a main turbine-generator trip

Main steam or main feedwater line break isolation is accomplished automatically in IP-2 and IP-3.

The design of the AFWS does not have the capability to automatically terminate feedwater flow to a depressurized steam generator and provide flow to the intact steam generator. This is accomplished by the operator.

X.7.1.5 Testing

The AFWS is tested periodically in accordance with the following Technical Specification requirements:

Indian Point 2 Specification - Testing Requirements

1.a Each motor-driven auxiliary feedwater pump will be started at intervals not greater than every month with full flow established to the steam generators once every refueling.

b The steam turbine driven auxiliary feedwater pump will be started at intervals not greater than six months with full flow established to the steam generators once every refueling.

c The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.

2. These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

Indian Point 3 - Specification - Testing Requirements

1.a Each auxiliary feedwater pump will be started manually from the control room at monthly intervals with full flow established to the steam generators once every refueling.

b The auxiliary feedwater pumps discharge valves will be tested by operator action at intervals not greater than six months.

c Backup supply valves from the city water system will be tested once every refueling.

2. Acceptance levels of performance shall be that the pumps start, reach their required developed head on recirculation flow, and operate for at least fifteen minutes.

X.10.1.6 Technical Specification

The limiting conditions of operation for Indian Point 2 and 3 AFWS are contained in the following Technical Specifications:

Indian Point 2 - Specification

A. The reactor shall not be heated above 350°F unless the following condition are met:

(1) A minimum ASME code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Two of the three auxiliary feedwater pumps must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tanks and a backup supply from the city water supply.

(4) System piping and valves directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) The total iodine activity of I-131 and I-133 on the secondary side of the steam generator shall be less than or equal to 0.15 μCi/cc.

B. If during power operations any of the conditions of 3.4.A above can not be met within 48 hours the operator shall start to shutdown and cool the reactor below 350°F using normal operating procedures.

Indian Point 3 - Specification

A. The reactor shall not be heated above 350°F unless the following condition are met:

(1) A minimum ASME code approved steam-relieving capability of twenty (20) main steam valves shall be operable (except for testing).

(2) Two of the three auxiliary feedwater pumps must be operable.

(3) A minimum of 360,000 gallons of water in the condensate storage tank.

(4) System piping and valves directly associated with the above components operable.

(5) The main steam stop valves are operable and capable of closing in five seconds or less.

(6) Two steam generators capable of performing their heat transfer function.

(7) City water system piping and valves directly associated with providing backup supply to the auxiliary feedwater pumps are operable.

B. If during power operations any of the conditions of 3.4.A above can not be met within 48 hours the operator shall start to shutdown and cool the reactor below 350°F using normal operating procedures.

X.7.2 Reliability Evaluation

X.7.2.1 Dominant Failure Modes

Loss of MFW with offsite power available

The dominant failure mode for this transient is the failure to provide sufficient water to the suction of the AFWS pumps. There are two manual valves in the supply line from the condensate storage tank; the inadvertent closure of either of these valves cuts off this water supply. In the event of an AFWS demand, operator action would be required to either open the closed valve (locally) or to manually open the valves in the supply line from the alternate water sources (city water) before pump damage occurs. Thus the dominant failure mode is the human error of inadvertently closing a valve in the CST supply line, coupled with the failure of the operator to manually reopen the closed valve or open the valves from the backup water supply.

A second important failure mode was also noted in this evaluation. The Indian Point Technical Specifications and LCOs require only that two of the three AFWS be operable, thus allowing the possibility that one train could be out of service indefinitely. This, in effect, reduces a three train system to a two train system, and thus reduces the predicted AFWS reliability to some degree. Revision of the Technical Specifications/LCOs to the present requirements (in the standard Technical Specifications) would make this failure mode much less significant.

Loss of MFW with only onsite AC power available

Because the dominant failure modes discussed above are not dependent on the source of AC power (onsite or offsite), these modes are also dominant for this transient event.

Loss of MFW with only DC power available

In this transient, loss of both offsite and onsite AC power is postulated to occur, so that the AFWS is reduced to only the steam-driven pump train. Thus failures in this train alone would be sufficient to fail AFWS, for this transient. The dominant failure mode for this case is that the train is out of service for maintenance, for the reason that current Technical Specifications and LCOs specify no time limit that the train could be out of service. Thus the revision of the Technical Specifications and LCOs mentioned for the above cases also would be of significant benefit for this case.

X.7.2.2 Principal Dependencies

The principal dependency found in this analysis is, as discussed above, the manual valves located in the feedwater supply line common to all AFWS pumps and the possible unlimited outage of one pump.

X.7.3 Recommendations for this Plant*

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system realiability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant-specific) identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

* Recommendations apply to IP-2 and 3 unless otherwise stated.

X.7.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumentation can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

3. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

   The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main

feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

4. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures shouls include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

5. Recommendation GS-6 - The licensee should confirm flow path available ability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals ans associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation sginals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.
   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
   - Testability of the initiation signals and circuits shall be a feature of the design.

   - The initiation signals and circuits should be powered from the emergency buses.
   - Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
   - The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
   - The automatic initiation signals and curcuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

Indian Point 2

7. Recommendation - The pneumatic-operated valves in the steam supply line to the turbine-driven AFW pump, and all of the pneumatic-operated AFW flow control valves derive their power from the same non-safety grade bus. Although these valves are designed to fail open upon the loss of air or power, thereby assuring auxiliary feedwater flow to the steam generators upon such losses, it cannot be concluded that all failures will result in opening the valves. The consequences of voltage degradation should be analyzed as well as other failures (e.g., restricted air flow)

to assure tha' such events would not incapacitate the auxiliary feedwater system the licensee should establish suitable emeroencv procedures to assure AFWS function for such events.

X.7.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accompolished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria

should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operatin should propose Technical Specification to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.7.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s) should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

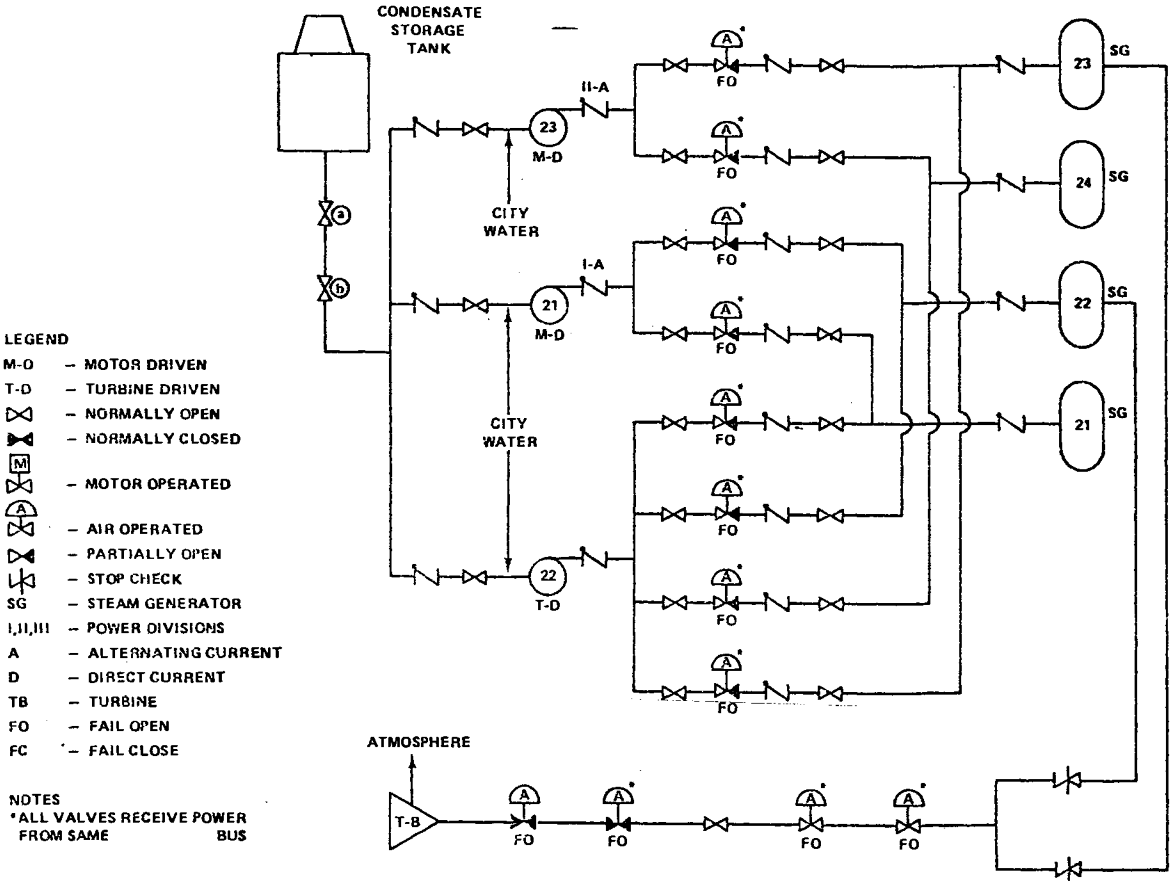
3. Recommendation - The two motor-driven pumps and the turbine driven pump are located in the same room. The licensee should evaluate the capability of the design to withstand a) environmental conditions (steam, flooding, pipe whip and jet impingement) resulting from a pipe break, b) internally generated missiles.

The licensee should evaluate the postulated pipe breaks stated above and (1) determine any AFW system design changes or procedures necessary to detect and isolate the break and direct the required feedwater flow to the steam generator(s) before they boild dry or (2) describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.

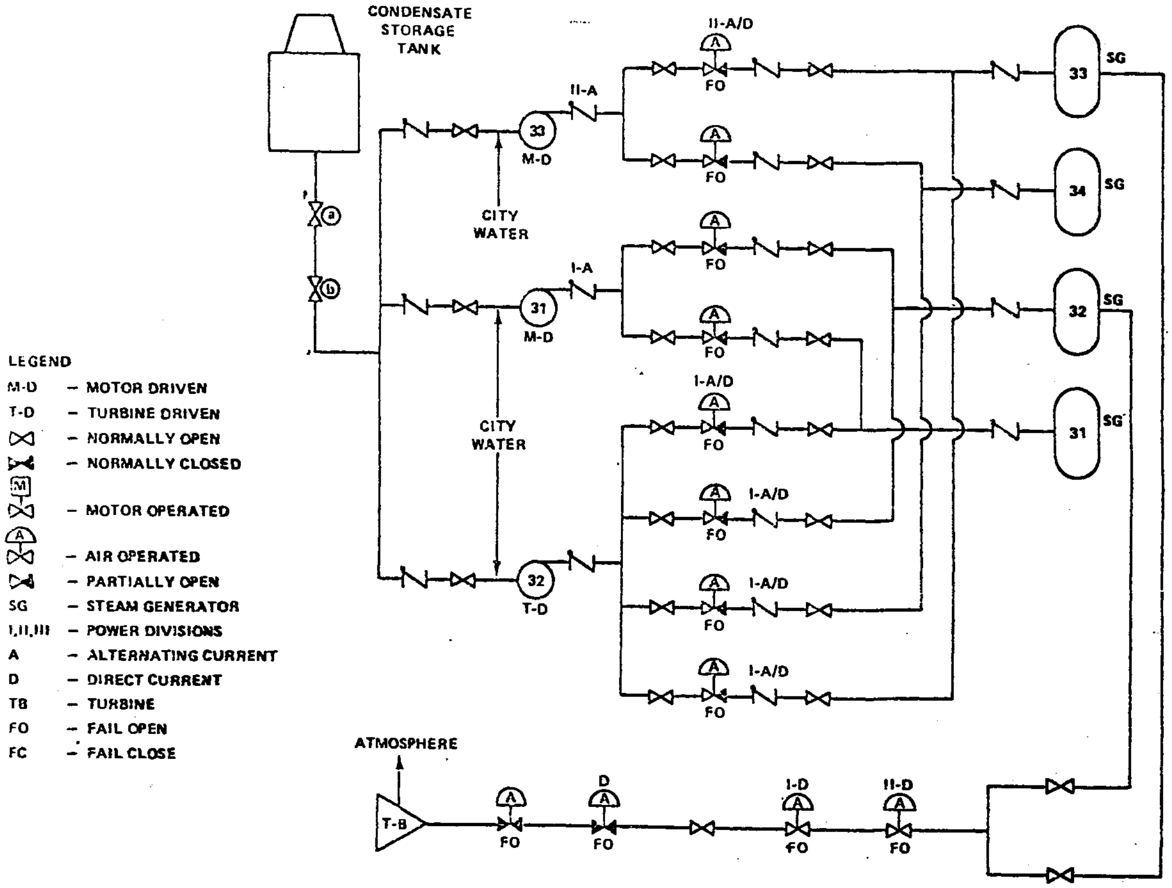
Indian Point 2

4. Recommendation - This is the same concern as that addressed in short term recommendation number 7.

   The licensee should complete the modification described in Section 7.1.4.1 above that will supply power to these controllers from separate safety grade buses.



Auxiliary Feedwater System
Indian Point-2
Figure 1



Auxiliary Feedwater System
Indian Point-3
Figure 2

ENCLOSURE 2

Basis for Auxiliary Feedwater System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

    1) Loss of Main Feed (LMFW)
    2) LMFW w/loss of offsite AC power
    3) LMFW w/loss of onsite and offsite AC power
    4) Plant cooldown
    5) Turbine trip with and without bypass
    6) Main steam isolation valve closure
    7) Main feed line break
    8) Main steam line break
    9) Small break LOCA
    10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 21, 1979

Docket No.: 50-305

Mr. Eugene R. Mathews, Vice-President
Power Supply and Engineering
Wisconsin Public Service Corporation
P. O. Box 1200
Green Bay, Wisconsin 54305

Dear Mr. Mathews:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT KEWAUNEE PLANT

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Acting Director
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ENCLOSURE 1

X.8 (W) KEWAUNEE

AUXILIARY FEEDWATER SYSTEM

X.8.1 System Description

X.8.1.1 Configuration, Overall Design

Figure 1 is a simplified flow diagram of the Kewaunee auxiliary feedwater system (AFWS). The Kewaunee AFWS design includes three auxiliary feedwater pumps, two motor driven and one turbine driven, which supply feedwater to two steam generators. All three pumps are normally lined up to take suction from the non-safety grade condensate storage tanks through a common header. A redundant seismic Category I source of water is available to the pump suction from the service water system. Each train of the service water system will supply one motor driven pump, and both service water trains can supply the turbine driven pump. A failure in the common pump suction header will not affect this supply since the service water is connected directly to the individual pump suctions. All operations to connect the service water system are done from the control room via safety grade equipment.

The motor driven pump discharge lines are cross connected downstream of the AFW control valves. There are two normally open motor operated valves (DC powered) in the crossconnect line with the turbine driven pump discharge connected between the two valves. Manual operation from the control room is available to separate the headers or direct the turbine driven pump discharge to a specific steam generator. This is accomplished by closing one or both of the cross connect isolation valves.

Each motor driven pump (240 gpm) has its own air operated flow control valve located at the pump discharge. These valves are normally open and fail open on loss of air. Each of the three pumps has an individual normally open manual isolation valve at the pump discharge. There is no flow control valve for the turbine driven pump which operates at full flow capacity continuously on demand (240 gpm).

There are no other valves in the flowpath between the pumps and the steam generators other than a check valve at each pump discharge and a check valve in each of the two discharge lines at the main feedwater system connection.

There are two condensate storage tanks (capacity 75,000 gallons each). Technical specifications require at least 75,000 gallons total be available when the reactor is above 350°F or the plant must be cooled down below 350° within 48 hours. One condensate tank is normally lined up to the AFW system suction, with the other tank is used for normal secondary system demands.

Each motor pump uses a startup lube oil pump, powered from the same electrical train as that of the pump. The turbine driven start-up

lube oil pump is d-c powered such that the turbine driven train is independent of A-C power. Electrical interlocks are provided such that a pump will not start unless its respective lube oil startup pump develops sufficient lube oil pressure. Once up to speed, lube oil can be supplied by shaft driven lube oil pumps independent of the startup system. While running, a loss of lube oil pressure will result in a pump trip.

X.8.1.2 Components, Design Classification

The condensate storage tank is classified non-safety grade and non-seismic Category I. The AFW piping, valves and pumps are classified safety Class I and seismic Category I. All safety systems at Kewaunee, including the reactor coolant system, are classified safety Class 1 and have the same quality assurance requirements. Steam generator level indication, valve position indication and all control equipment are safety grade. The rest of the system instrumentation is non-safety grade.

X.8.1.3 Power Sources

The valves which initiate steam flow to the turbine driven pump are operated by a safety grade d-c power supply. These are the normally open motor operated isolation valves from the steam lines and a normally closed motor operated valve at the turbine inlet. Each motor driven pump receives power from separate emergency A-C buses capable of being supplied by the diesel generators.

The normally open flow control valves from each motor driven pump are air operated and fail open on loss of air. Control power is from the same emergency bus as its respective pump. The cross connect isolation valves are D-C powered motor operated valves supplied by safety related D-C buses.

All instrumentation and controls associated with the auxiliary feedwater systems are powered from onsite electrical systems.

X.8.1.4 Instrumentation and Controls

X.8.1.4.1 Controls

The Kewaunee AFW design has a minimum number of control features because of the small number of valves associated with the system. These control are:

1) Motor Driven Pump Start/Stop Switches
2) Turbine driven Pump Steam inlet valve open/close for start and stop operations
3) Modulation Control of AFW flow control valves for motor driven pumps from full open to full closed
4) Open/Close control of discharge header cross connect isolation valves (each isolation valves will also isolate turbine pump discharge from individual steam generators).
5) Open/Close control of turbine steam isolation valves which isolate turbine from main steam lines.
6) Open/Close control of service water system isolation valves to AFW pump suctions.

All controls are located in the control room. Steam generator level will normally be controlled by modulating the flow control valves on the discharge of the motor driven pumps. The turbine driven pump will deliver full flow when operating and will be secured manually from the control room if flow to generators is more than necessary.

X.8.1.4.2 Information Available to Operator

The following alarms and indications are available to the operator in the control room:

I. Indication

1) System Actuation Light - Actuates when motor is energized with sufficient pump discharge pressure. (Both signals necessary)
2) Discharge Pressure for each pump
3) Flow rate to each steam generator
4) Condensate storage tanks level indication
5) Steam Generator Level
6) Valve position indication for steam valves to turbine, feedwater control valves, crossconnect isolation valves and service water system supply to AFW isolation valves.

II. Alarms (Turbine Pump)

1) Steam inlet valve open to turbine coincident with low lube oil pressure to turbine driven pump
2) Steam inlet valve open coincident with low discharge pressure at turbine driven pump

3) Steam isolation valve (either of two) from main steam system to turbine header not open

4) Turbine throttle valve - not open

5) Discharge valve not open - either of two crossconnect isolation valves

6) Valve Control Power - In pullout position (cannot actuate valve at steam inlet)

III. Alarms (Motor Driven pumps)

1) Breaker closed coincident with low lube oil pressure

2) Breaker closed Low Discharge Pressure

3) Breaker in pull out - Pump cannot be started

4) Breaker open coincident with auxiliary feedwater control valve closed.

X.8.1.4.3 Initiating Signals for Automatic Operation

I. Motor Driven Pumps

1) Safety Injection Signal

2) Loss of Bus Voltage (Loss of Offsite Power - LOOP)

3) Low-Low Level in either steam generator-two out of three detectors

4) Tripping of Both Main feedwater pumps. (Motor Driven Main Feed Pumps - Signal taken from contact on circuit breaker)

II. Turbine Driven Pump

1) Low-Low level in both steam generators-2 out of 3 detectors

2) Loss of Voltage on both 4-KV busses (Reactor Coolant Pump and Main Feedwater Pump supply bus)

All pumps can be manually started from the control room and will automatically supply full flow to steam generators when started.

The main feedwater pump trip automatic start of the motor driven auxiliary feedwater pumps is bypassed during startup by breaker "pull out" switch in control room.

X.8.1.5 Testing

Pump operability is tested once per month, by closing the auxiliary feedwater control valves and manual isolation valve at pump discharge and verifying discharge pressure while recirculating to the condensate storage tank. This same test also verifies valve operability. The service water system isolation valves to the auxiliary feedwater pumps are cycled quarterly to verify operability.

X.8.1.6 Technical Specifications

Limiting conditions for Operation with regard to the auxiliary feedwater system are:

1) The reactor shall not be above 350°F unless the following conditions are met.

   a) Two of three AFW pumps are operable

   b) System piping and valves for 2 pump trains are available

   c) Minimum of 75,000 gallons of water is available in the condensate storage tanks and the service water system is capable of delivering an unlimited supply from Lake Michigan.

2) If, when the reactor is above 350°F, any of the above conditions are not met within 48 hours, the reactor shall be shutdown and cooled to below 350° using normal operating procedures.

X.8.2 Reliability Evaluation

X.8.2.1 Dominant Failure Modes

The dominant failure modes are expressed for three transient situations. Success criterion is the operation of at least one of the three pump trains.

LOFW with Offsite Power Available

The unreliability of the AFWS during this type of transient is dominated by two types of failure combinations. The first involves initiation of the AFW pumps with inadvertent closure, and delayed discovery of the manual valves in the CST supply line combined with human failure to switch to the service water source.

The second failure combination type is based on maintenance outages combined with hardware failure. The Kewaunee technical specification permits unlimited outage of one of the three subsystems and permits outage or test of a second system for up to 48 hours prior to a required shutdown.

LOFW With Loss of Offsite Power but With Onsite AC Power Available

The conditional unreliability of the AFWS during this type of transient is dominated by the same failure mode as in the previous section with a more significant contribution from the triple hardware failure. In this situation failure of one of the two electrical loops can come from partial (one train) failure of onsite power.

LOFW with Loss of All AC, DC Available

The conditional unreliability of the AFWS during this type of transient is dominated by the test and maintenance contribution by the turbine driven pump train. Since only the turbine driven pump train is useable under these conditions, unlimited possible outage of that train makes a high probability for AFWS outage possible.

X.8.2.2 Interdependencies

None noted.

X.8.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.8.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumention can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

3. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:
   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.
   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
   - Testability of the initiation signals and circuits shall be be a feature of the design.
   - The initiation signals and circuits should be powered from the emergency buses.

   - Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
   - The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
   - The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to intiate the AFW system from the control room.

X.8.3.2 Additional Short Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditons (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technnical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.8.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

3. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

ENCLOSURE 2

Basis for Auxiliary Feedwater System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
      1) Loss of Main Feed (LMFW)
      2) LMFW w/loss of offsite AC power
      3) LMFW w/loss of onsite and offsite AC power
      4) Plant cooldown
      5) Turbine trip with and without bypass
      6) Main steam isolation valve closure
      7) Main feed line break
      8) Main steam line break
      9) Small break LOCA
      10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

   - Maximum RCS pressure (PORV or safety valve actuation)
   - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
   - RCS cooling rate limit to avoid excessive coolant shrinkage
   - Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

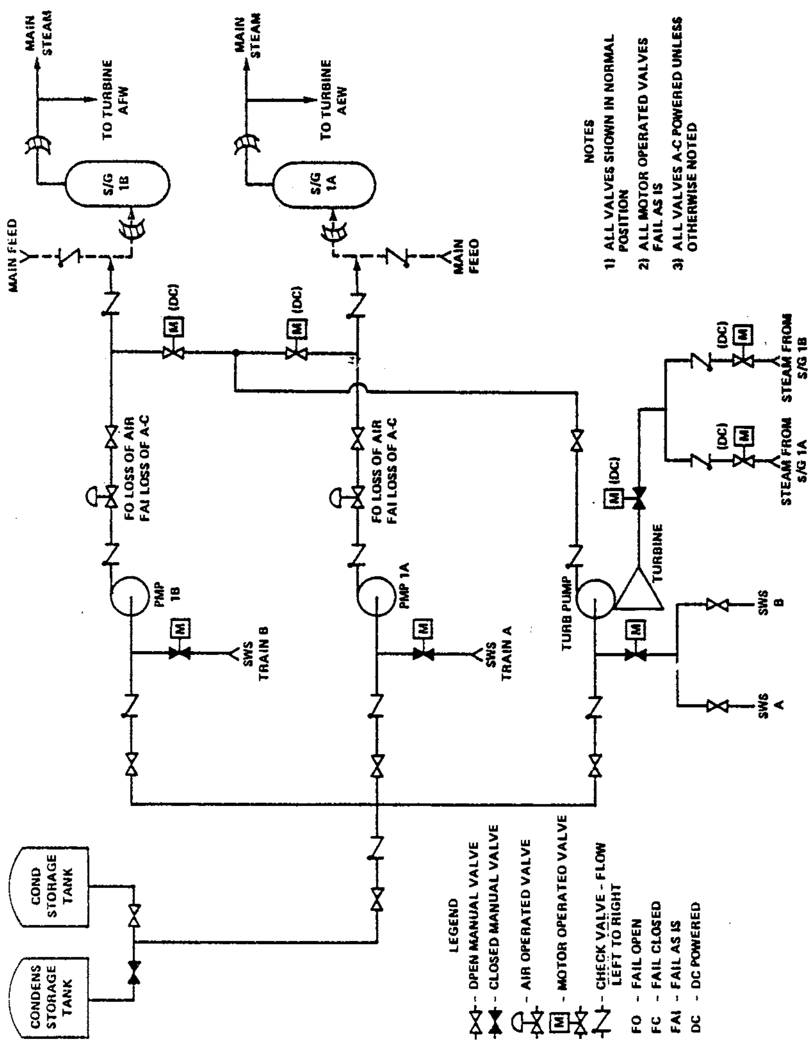
k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.



Auxiliary Feedwater System
Kewaunee
Figure 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 28, 1979

Docket No.: 50-338

Mr. W. L. Proffitt
Senior Vice-President - Power
Virginia Electric and Power Company
P. O. Box 26666
Richmond, Virginia 23261

Dear Mr. Proffitt:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT NORTH ANNA POWER STATION, UNIT 1

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

D. Eisenhut, Acting Director
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Office of Nuclear Reactor Regulation

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ENCLOSURE 1

X.9 (W) NORTH ANNA UNIT 1

AUXILIARY FEEDWATER SYSTEM

X.9.1 System Description

X.9.1.1 Configuration and Overall Design

The auxiliary feedwater system (AFWS) is designed to supply water to the steam generators (SG) for reactor coolant system sensible and decay heat removal when the normal feedwater system is not available. The AFWS can be utilized during certain periods of normal startup and shutdowns, in the event of malfunctions such as loss of main feedwater flow, loss of offsite power and also in the event of an accident. The AFW system is automatically initiated upon receipt of the following signals: low steam generator level, safety injection, loss of offsite power, and main feed pump trip.

The AFWS is shown in simplified form in Figure 1 attached. The system consists of two motor driven pumps (3A, 3B), and one steam driven turbine pump (2). Each motor driven pump has a design flow of 350 gpm, the turbine driven pump has a design flow of 700 gpm but is orfice limited to 350 gpm when pumping in the normal lineup to SG-A. Taps from each main steam line at a point upstream of the main steam isolation valves provide the source of steam to the turbine. The motor driven pumps are connected to separate emergency power buses

(Class 1E). Normally the pumps take suction from the 110,000 gallon emergency condensate storage tank. This provides 8 hours of operation for decay heat removal. The emergency condensate storage tank is designed to seismic Category I requirements and is protected from tornado missiles. An additional supply of 300,000 gallons is available from a non-seismic condensate make-up storage tank. In addition to the 300,000 gallon supply, an unlimited supply of water is available from the seismic Category I service water and fire protection systems which are supplied from Lake Anna and the spray cooling pond, which is seismic Category I designed.

Referring to Figure 1, each pump is lined up normally to a specific steam generator; pump 3B to SG B, 3A to SG C and 2 to SG A. The pumps can be aligned to other steam generators in the event of line breaks, pump failures, etc., by positioning the manual control valves to suit. AFW flow to steam generator C is normally remote manually controlled by an air operated valve. AFW flow to steam generators A and B is similarly manually controlled by an AC motor operated valve in each supply line. In the event of loss of offsite and onsite AC power, realignment of manual valves is necessary to supply AFW flow to all steam generators from the turbine-driven pump. The instrument air supply system, for the air operated AFW flow control valves, includes a 16.7 cu. ft. accumulator tank charged to 100 psig. This capacity is sufficient to operate the air operated valve(s) from 30 minutes to 8 hours depending on frequency of valve adjustment.

X.9.1.2 Component Design Classification

The turbine pump train and motor pump trains (110,000 gallon tank, pumps, valves, motors, piping, service water and fire protection systems) are seismic Category 1 and tornado missile protected, designed to Quality Group C. (Class 1E for electrical equipment). The 300,000 gallon condensate make up tank is non-seismic.

X.9.1.3 Power Sources

The motor driven pumps and motor operated valves are supplied from the Class 1E A-C emergency buses which may be powered by the diesel generators, 3A from Emergency Bus 1H, 3B from emergency bus 1J. The steam admission valves for the turbine pump are air-operated using DC solenoids and are energized from the emergency battery buses.

Instrumentation and Controls

The instrumentation and Control power supplies are from the 120 VAC vital bus system. There are 4 vital buses, each supplied by an inverter from the 125 VDC power system. The motor driven pump breaker controls are powered from the 125 VDC power system provided from the Class 1E emergency DC buses.

X.9.1.4 Controls

Steam generator level is controlled remote manually from either the Main Control Room (MCR) or the Auxiliary Shutdown Panel (ASP) with safety grade instrumentation provided (level and flow indications).

The valves in the water flow lines to the steam generators consist of three motor operated valves from one header and three air operated valves from the other header. Any pump can supply either header by operating manual valves in the pumps discharge. (For normal alignment refer to Section X.9.1.1.) The air operated valves are normally open and fail open on loss of power, the MOVs are normally open and fail as-is on loss of power. The MOVs or air operated valves are positioned by the operator to maintain proper level in the steam generators.

The steam admission valves to the turbine are air operated, normally closed and fail open. These valves can be controlled from the MCR or ASP.

X.9.1.5 Information Available to the Operator

The following indications are available at both operating stations except as noted.

1. Position indication of the MOVs and air operated valves
2. Flow, gpm to each steam generator (Main Control Board (MCB)) only
3. Pump Current and Voltage (MCB only)
4. Steam Pressure to Aux feed pump turbine (MCB Only)
5. Steam Generator Levels
6. Pump Discharge and suction pressures (MCB only)
7. Breaker (motor driven pumps) position
8. Condensate storage tank level emergency

X.9.1.6 Initiating Signals for Automatic Operations

The following signals start the pump motors and open the steam control valves to the turbine:

1. Steam generator water level, low-low in any steam generator (2 out of 3 signals)
2. Safety Injection signal* (Delay of 35-60 seconds on Motor driven pumps)
3. Loss of offsite power
4. Main Feed pump trip (loss of all Main feed pumps)
5. Manual

X.9.1.7 Testing

The systems are tested periodically in accordance with tech spec requirements. The frequency of periodic testing is 31 days. In addition, the particular system is tested in accordance with the technical specifications after performing system maintenance. The systems are tested using the recirculating lines, with various plant parameters noted. (Suction and discharge pressure, etc.) The instrumentation systems are checked periodically, in accordance with the technical specifications, on a per shift, monthly or refueling time frame basis.

*There is no delay of the S.I. signal to the turbine driven pump control. The reason for the delay in the motor driven pump control circuit is to limit the loads during emergency diesel generator loading.

X.9.1.8 Technical Specifications

A review of the technical specifications indicated that these specifications cover limiting conditions of operation (LCO) and periodic surveillance testing consistent with standard Technical specifications.

X.9.2 Reliability Evaluation

X.9.2.1 Dominant Failure Modes

Successful delivery of feedwater is considered to be the flow of at least 350 gpm to one (or more) of the three steam generators for the transients considered here.

Failure modes of the AFWS were assessed for three types of initiating transients. The dominant failure modes for each transient type are discussed below.

Loss of MFW with Offsite Power Available

The reliability analysis of the North Anna AFWS based on this initiating transient did not identify any single failures or double failures which would fail the entire AFWS. This assessment indicates the dominant AFWS failure mode to be a combination of the failure of both actuation trains to actuate their respective components, coupled with failure of the operator to detect the non-actuation of the system and to manually actuate it.

Loss of MFW with Only Onsite AC Power Available

This transient is somewhat different in character than in the case above, in that reliance for AC power is now on the station diesel generators rather than offsite power. In essence, this adds failure modes such as diesel-generator failure to start to the overall list of failure modes of trains of the AFWS.

The dominant failure modes for the AFWS discussed for the above case are not dependent on the actual source of AC power (i.e., offsite vs. onsite). Thus the probability of the failure mode discussed above should not change. Further, the addition of the diesel-generator failure mode to other train failure modes is not sufficiently important to make the probability of such a combination of modes significant. For these reasons,the AFWS failure probability for this case is still dominated by the coincident loss of both actuation trains, coupled with the failure of the operator to subsequently manually actuate the AFWS.

Loss of MFW with Only DC Power Available

In this case, no AC power (offsite or onsite) is available; the AFWS is thus reduced to the one steam-driven train for feedwater delivery. A number of single failures within this train can fail the AFWS (e.g., hardware failure in the pump and valves, control system failures, etc.). The dominant failure mode for this train is that the train is out of service for maintenance when the transient occurs.

X.9.2.2 Principal Dependencies

The potential for location dependencies was noted during this reliability evaluation, in that some portions of the AFWS were located in common rooms. However, because no location dependencies were found which could potentially affect all trains of the AFWS, these dependencies do not appear to be ä significant concern.

XIII.9.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.9.3.1 Short Term

1. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
    - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
    - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

2. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:
    - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
    - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

3. Recommendation GS-7 - The licensee should verify that the automatic start AFW signals and associated circuitry are safety

grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.9.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.
2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

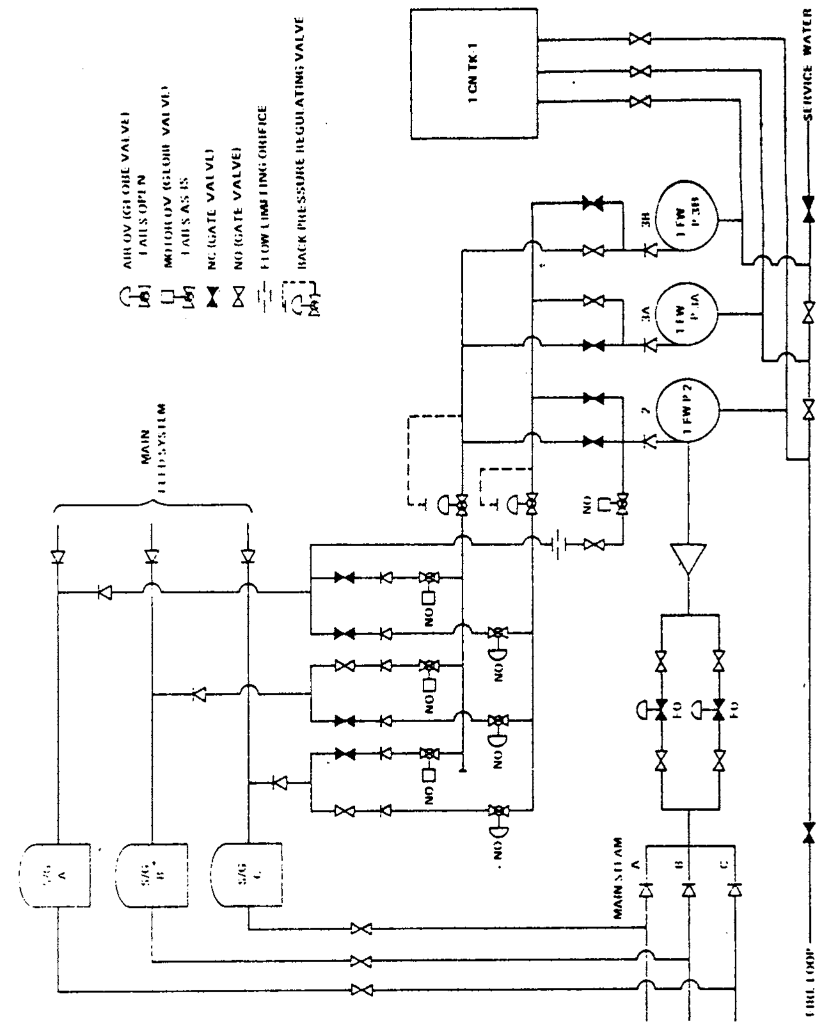
   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.9.3.3 Long Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.



Auxiliary Feedwater System
North Anna
Figure 1

Basis for Auxiliary Feedwater System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

   - Maximum RCS pressure (PORV or safety valve actuation)
   - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
   - RCS cooling rate limit to avoid excessive coolant shrinkage
   - Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 21, 1979

Docket Nos.: 50-266
50-301

Mr. Sol Burstein
Executive Vice-President
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

Dear Mr. Burstein:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc: w/enclosure
See next page

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ENCLOSURE 1

X.11 (W) POINT BEACH 1 AND 2

AUXILIARY FEEDWATER SYSTEM (AFWS)

X.11.1 System Description

X.11.1.1 Configuration, Overall Design

A simplified flow diagram of Point Beach 1 and 2 AFWS is shown in Figure 1. The automatically initiated auxiliary feedwater (AFW) system for each Point Beach Unit is partially shared between units 1 and 2 to supply AFW to both steam generators of each unit. Each AFW system uses a turbine drive pump and a motor driven pump. The turbine driven pump of one unit feeds both steam generators of that unit only. The motor driven pump of each unit feeds one steam generator in each unit and therefore is shared between units. The turbine driven pumps supply AFW to the main feedwater piping inside containment through a motor operated valve for each steam generator of their respective units. The motor operated valves (MOV 1A and 2A for Unit 1, MOV 1B and 2B for Unit 2 on Figure 1) are normally opened to a throttled position to supply design flow to each steam generator. On loss of power these valves fail as-is.

Each of the two motor driven pumps supplies AFW to one steam generator of each unit through individual motor operated isolation valves which are normally open and fail as-is on loss of power. (MOV 3A and 3B from one pump and MOV 4A and 4B from the other pump). A pressure control valve (PCV-1 for Unit 1, PCV-2 for Unit 2) at the discharge of each pump controls flow to two steam generators (one generator per unit) by maintaining a constant pressure at the pump discharge. The set point of this controlled pressure determines flow to the steam generators and can be varied by the control room operator. The PCV's are air operated and fail open upon loss of air.

All four AFW pumps normally take suction from two non-seismic Category I condensate storage tanks (45,000 gallons capacity each) through manually operated locked open isolation valves. The condensate storage tanks are normally lined up in parallel to the common suction header of the AFW pumps.

The minimum total capacity of the condensate storage tanks (by Technical Specifications) is 10,000 gallons per operating unit. The total capacity (20,000 gallons) will allow at least 25 minutes of supply with both turbine drive AFW pumps running (400 gpm per turbine-driven pump) or 50 minutes supply with both motor-driven pumps running (200 gpm per motor driven pump). The service water system serves as the seismic Category I source of water to the AFWS and is capable of unlimited supply. The service water system (SWS) connects directly to the suction of each AFW pump down-stream of the suction check valves and is therefore unaffected by malfunctions in the condensate tank supply portion of the AFW system. SWS supply is initiated in the control room by opening a motor operated valve in the SWS to each AFW pump suction. The system is arranged such that a failure of either of the two diesel generators on site will not prevent water from being supplied to the AFW system for either unit.

Since all valves in the flow path to the steam generators are normally open and fail as-is (with exception of PCV-1 and 2 which fail open) a loss of A-C or D-C power does not require valve manipulation. The motor operated steam valves at the inlet to the turbines (MS-1A and 2A for Unit 1, MS-1B and 2B for Unit 2) are D-C motor operated valves and will automatically open in the event of a loss of all A-C power.

In the event of an unisolable main steam or feedwater line break coincident with a worst case single active failure, operator action within the control room will isolate AFW flow to the affected steam generator and assure flow to the unaffected steam generator. The licensee estimates >30 minutes to boil dry.

A break anywhere in the auxiliary feedwater system discharge piping would not prevent automatic AFW flow to at least one steam generator on demand. A single active failure coincident with a break could disable automatic AFW to both steam generators, depending on break location. In either case, breaks could be isolated by operator action within the control room.

11.1.2 <u>Component Design Classification</u>

All pumps, valves, piping, instrumentation and controls associated with the auxiliary feedwater system (except Condensate Storage Tanks) are designed safety Class I which includes seismic Category I requirements.

The condensate storage tank and associated instrumentation are not designed to safety grade requirements. The piping from the tank to the auxiliary feedwater system is classified as safety class I which includes seismic Category 1 requirements.

X.11.1.3 Power Sources

Power sources for all instrumentation and controls are taken from the emergency buses which are supplied by the safety related diesel generators or safety related station batteries. Steam generator water level control and the automatic initiation system are designed as a safety related system, including seismic Category I.

Each motor driven pump and associated instrumentation and controls are powered by a separate diesel-generator, such that a failure of one diesel generator will only disable one motor driven train.

The turbine driven pump for each unit receives steam from both steam generators of its respective unit through parallel d-c motor operated isolation valves. The parallel valves are powered from separate D-C buses such that a loss of one d-c system will not prevent operation of either turbine driven pump.

X.11.1.4 Instrumentation and Controls

X.11.1.4.1 Controls

All controls for the active components of the auxiliary feedwater system can be operated from the control room. Normally steam generator

level is controlled in the control room by adjusting the pressure set point of the pressure control valves at the discharge of each motor driven pump. If it is necessary to control turbine pump flow for level control, the motor operated valves in the discharge lines from the turbine driven pump each steam generator can be throttled from the control room.

Each control actuator in the control room is located in a basic system layout (MIMIC Bus) to help identify the control switch function in addition to the identifying name plate.

X.11.1.4.2 Information Available to the Operator

I. Alarms

a) Hi/Lo Steam Generator Level

b) Low Level - Condensate Storage Tank

c) Service Water System Header Pressure Low

II: Indication

a) Steam Generator Level

b) Condensate Storage Tank Level

c) AFW pump discharge pressure

d) Service Water Header pressure

e) Valve Position Indication - All Active Valves

f) Pump Running Lights - Motor Drive

g) Pump Breaker Trouble Light - (Did Not Close on Demand)

h) Pressure Set Point - Pressure Control Valve

All valve position indicators are located with their respective controls on the "MIMIC Board" such that the valves are readily identified.

X.11.1.4.3 AFW Initiating Signals

I. Turbine Pumps

a) Lo-Lo Level in both S/G's of its respective unit - automatic

b) Loss of both 4 KV busses (Supply reactor coolant Pumps) - automatic

c) Manual - From Control Room

II. Motor Driven Pumps

a) Lo-Lo Level in any one S/G of either unit - automatic

b) Trip of both Main Feed Pump - either unit - automatic

c) Safety Injection Signal - either unit - automatic

d) Manual - from Control Room

X.11.1.5 Testing

1) Valve position is verified monthly

2) Service Water System supply valves are cycled monthly

3) Operational tests of AFW pumps are performed monthly by verifying pump suction and discharge pressure (Tests are staggered)

4) Flow verification tests from condensate tanks to S/G's are performed at each refueling or whenever in cold shutdown (Not more frequently than quarterly)

5) Automatic initiation of the AFW system is verified during each refueling.

6) Control and initiating circuits are tested with each pump and valve test

7) Following maintenance on the system, an operational test is performed to bring the system back in service.

X.11.1.6 Technical Specifications

A. When the reactor coolant is heated above 350°F the reactor shall not be taken critical unless the following conditions are met:

1a. Two Unit Operation - Three of the four auxiliary feedwater pumps are operable.

1b. Single Unit Operation - Either the turbine driven pump associated with that unit together with one of the two motor driven pumps or both motor driven pumps must be operable.

2. A minimum of 10,000 gallons of water per operating unit in the condensate storage tanks and an unlimited water supply from the lake via either leg of the plant service water system.

3. System piping and valves required to function during accident conditions directly associated with the above components must be operable.

B. During power operation, the requirements are modified to allow the following components to be inoperable for a specified time.

If the system is not restored to meet the above requirements within the time period specified the appropriate reactor(s) shall be placed in the hot shutdown condition. If they are not satisfied within an additional 48 hours, the appropriate reactor(s) shall be cooled down to less than 350°F.

1. Two Unit Operation - One of the three operable auxiliary feedwater pumps may be out-of-service provided a pump is restored to operable status within 24 hours.
2. Single Unit Operation - One of the two operable auxiliary feedwater pumps may be out-of-service provided a pump is restored to operable status within 24 hours.

X.11.2 Reliability Evaluation

X.11.2.1 Dominant Failure Modes

The dominant failure modes are expressed for three transient situations and two operational configurations, single unit operation and double unit operation.

Limiting conditions for single unit operation are a single motor-driven pump and associated turbine driven pump operable or both motor driven pumps operable. Any one can be out of service for 24 hours.

Limiting conditions for double unit operation are three of four auxiliary feedwater pumps operable. Any one can be out of service for 24 hours.

LOFW with Only DC Power Available

Single Unit Operation

The dominant failure contributor is loss of both motor-driven pumps and subsequent failure of the turbine driven pump due to loss of service water (AC) cooling to steam turbine pump bearing oil.

Double Unit Operation

Same failure as single unit operation.

X.11.2.2 Interdependencies

The principal noted dependency is the design for AC cooling of the turbine driven pumps.

X.11.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.11.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumention can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

4. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

- The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

- The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

5. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:
   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system shou be modified in the short-term to meet the functional requirements listed

.11.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator actions, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions

(temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

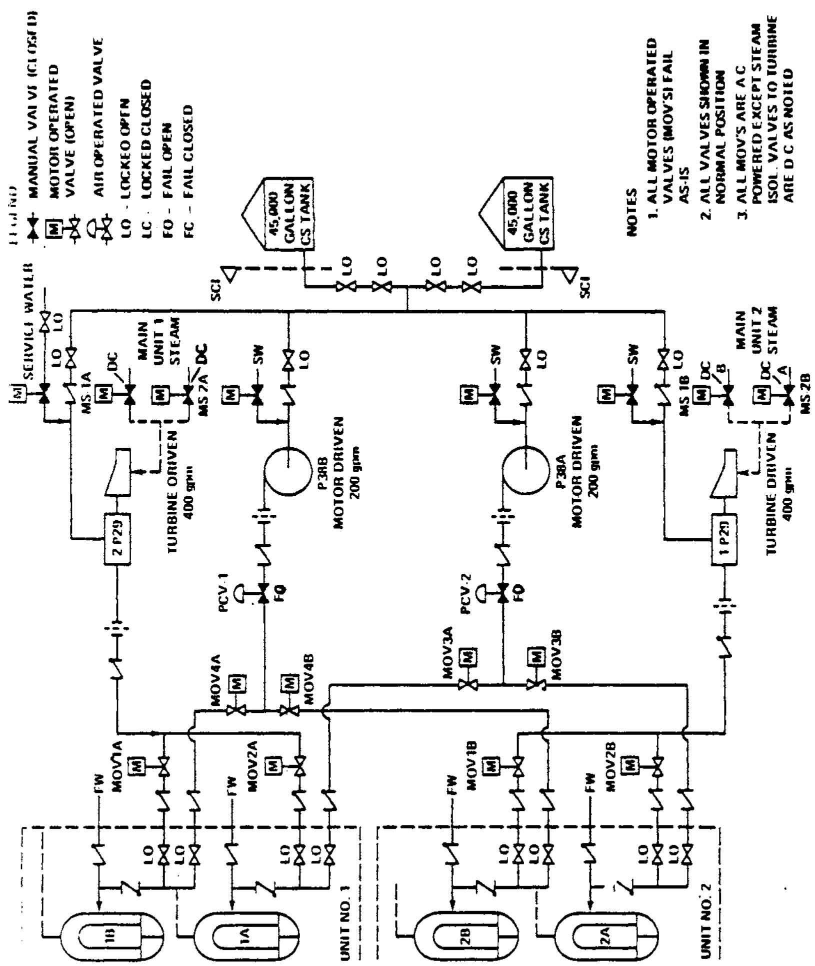
X.11.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

2. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

3. Recommendation-GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.



Auxiliary Feedwater System
Pt. Beach 1 & 2
Figure 1

ENCLOSURE 2

Basis for Auxiliary Feedwater
System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCTOBER 16 1979

Docket Nos.: 50-282
50-306

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Mayer:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Robert L. Tedesco / for

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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ENCLOSURE 1

X.10 (W) PRAIRIE ISLAND 1 & 2

AUXILIARY FEEDWATER SYSTEM

X.10.1 System Description

X.10.1.1 Configuration and Overall Design

The auxiliary feedwater system (AFWS) is designed to supply water to the steam generators for reactor coolant system decay heat removal when the normal feedwater system is not available. It is also used for plant startups and shutdown (below the point where main feed system flow capacity is not required).

The AFWS is shown in simplified form on Figure 1. The system consists of two steam turbine-driven pumps 11 and 22, (220 gpm rated flow with 20 gpm recirculation flow each), one for each unit, capable of delivering feedwater to either or both steam generators of the same unit. There is no interconnection between the discharge line of the two turbine-driven pumps of either unit. In addition, there are two motor-driven pumps 12 and 21 (220 gpm rated flow with 20 gpm recirculation flow each), one for each unit, capable of delivering feedwater to either or both steam generators of the same unit. Referring to Figure 1 pumps 11 and 12 are normally lined up to feed the steam generators of Unit #1, pumps 21 and 22 are normally lined up to feed the steam generators of Unit #2. The two motor-driven pump discharge headers are interconnected by two normallv closed valves. By opening these valves the Unit #1 pump #12 can supply water to Unit #2 steam generators or the Unit #2 pump #21 can supply water to Unit #1 steam generators.

Normal feedwater supply to the auxiliary feedwater pumps consists of 150,000 gallon condensate storage tanks, one for Unit 1, two for Unit 2 with a common (isolable) header. A backup water supply to the pumps is provided by the cooling water system. The cooling water system consists of five pumps, 2000 gpm capacity each - three motor driven, two diesel driven. Normally two-motor driven pumps are operating. Actuation of the third motor driven pump is automatic on low cooling water header pressure. If low discharge pressure persists (~75 psi) and/or AC power is lost, the diesel driven pumps are automatically started. In addition, 260 gpm of water can be supplied via the non-seismic demineralized water treatment system to the condensate storage tank(s).

The AFW system is automatically actuated. The licensee states that the steam generators would lose their ability to transfer heat in approximately 40 minutes. The valves in the AFWS lines to the steam generators are motor operated and are normally open. The steam admission valve in the steam supply line to the steam turbine is motor operated and is normally closed. Two steam supply valves, one from each steam generator, are motor operated and are normally open.

X.10.1.2 Component Design Classification

The turbine pump trains and motor pump trains (pumps, valves, motors, piping) are seismic Category I, tornado missile protected and designed to Quality Group I. Electrical equipment is designed as Class IE.

The sources of water and associated piping are classified as follows:

1. Condensate storage tank (Unit 1) } Type 3 (Non-seismic)
   Condensate storage tanks (Unit 2)
2. Cooling Water System - Seismic Category I, Tornado Missile Protected
3. Suction Piping
   from condensate storage tanks - Class 2B (non-seismic)
   from cooling water system (seismic - Category 1)
4. Demineralized water treatment system (non-seismic)

X.10.1.3 Power Sources

The motor driven pumps are supplied from the Class IE emergency buses, (Bus #16 - Train B for #12 pump, Bus 26 - Train A for #21 pump) Motor operated valve (MOV) power is from the emergency buses on a train basis. The emergency buses are capable of being powered from the diesel generators. Steam for the turbine driven pumps is supplied from each steam generator of the respective reactor unit

The instrumentation and control power supplies are from the 120 VAC vital bus system. There are four vital buses/unit, each supplied by an inverter connected to the 480 VAC emergency bus and the 125 VDC power system. The motor driven pump breaker controls are powered from the 125 VDC control batteries which are charged by battery chargers connected to the 480 VAC emergency buses.

X.10.1.4 Instrumentation and Control

Controls

Any of the MOV's can be controlled from either the Main Control Room or the Hot Shutdown Panel (local station).

Steam generator level is controlled by positioning the MOV's in the flow discharge lines. Level and flow indication is provided for operator information.

The valves are motor operated and fail as-is on loss of power.

X.10.1.4.1 AFW System Information Available to the Operator (At both remote and local stations except as noted)

1. MOV Position
2. S/G Level and pressure indication (alarm-control room only)
3. S.I. Ready Panel-abnormal valve position and AFW pump operability status-(control room only)
4. Discharge Pressure
5. Discharge Flow
6. CST Level Indication (low level alarm-control room only)

X.10.1.4.2 Initiating Signals for Automatic Operation

The following signals start the pump motors and open the steam admission control valve to the turbine of the affected unit:

1. Low-low water level in either steam generator
2. Trip of both main feedwater pumps
3. Safety injection
4. Undervoltage of both 4.16 kv normal buses (turbine driven pump only)
5. Manual

X.10.1.6 Testing

Auxiliary feed system surveillance tests are required on a monthly and refueling interval in accordance with tech spec requirements. The monthly tests involve (a) stroking MOV's and observing stem travel and (b) pump curve point check. The test is performed by shutting the appropriate pump discharge valves and recirculating back to the CST. After the test, the valves are positioned to normal lineup and all valve positions are verified.

X.10.1.7 Technical Specifications

The present limiting condition of operation (LCO) permits one unit operation with one motor-driven pump operable and either one turbine or one motor driven pump operable and if failure occurs and is not fixed in 48 hours - go to cold shutdown. Two unit operation is permitted with all four AFW pumps operable. If a failure occurs and repair is not completed within 7 days so that the four pump requirement is met, one unit must be taken to cold shutdown.

X.10.2 Reliability Evaluation Results

X.10.2.1 Dominant Failure Modes

Failure modes of the AFWS were assessed for three types of initiating transients. The dominant failure modes for each transient type are discussed below.

It should be noted that the failure modes discussed below as dominant presume LCOs of 48 hours on all AFWS trains. Currently, the LCO allows both the turbine-driven trains to be unavailable indefinitely when only one of the two units is operating. See Recommendation GS-1.

Loss of MFW with Offsite Power Available

A dominant failure mode of the AFWS for this transient is assessed to be the blockage of flow to the two steam generators due to inadvertent closure of two manual valves in the pump discharge lines inside containment. These valves could possibly be closed prior to the AFWS demand because of, for example, a personnel error in failing to reopen them after maintenance on the AFWS. Because these valves do not have remote position indication and are located inside the containment, there could be a considerable delay in gaining access to and reopening of the valves after an AFWS demand. However, the licensee states that any inadvertent closure of these valves would be detected prior to reactor startup or at least before the reactor exceeds 2% power since (a) plant startup procedures require a valve alignment check to verify the AFWS flow path and (b) the AFWS is used during normal plant startup to maintain steam generator water level

before initiating operation of the main feedwater system after reaching 2% reactor power; thus inoperability of the AFWS would have been detected before proceeding further. See Recommendations GS-2 and GS-6.

Loss of MFW with Only Onsite AC Power Available

This transient is very similar to the transient discussed above. Additional failure modes related to the onsite AC power system were considered; however, these did not have a significant impact. As such, a dominant failure mode for the case described above (closure of two manual valves in the AFWS discharge lines inside containment) is also considered to be dominant for this transient.

Loss of MFW with Only DC Power Available

In this transient no AC power (onsite or offsite) is available, so that the AFWS is reduced to one steam-driven pump train. The dominant failure mode of this train in such a transient is assessed to be failure of the operator to open the normally closed steam-admission valve before the steam generator water level decreases to the point where it loses its ability to transfer heat. This valve is motor-operated and is normally powered from either offsite AC power or from the diesel-generators. Since neither of these power sources is available in this transient, local, manual opening of the valve would be required.

X.10.2.2 Principal Dependencies

The principal dependency found in this evaluation is the common-cause failure of all trains due to closure of the manual valves in the two AFWS discharge lines.

The second significant dependency found is the dependence on AC power to run the turbine-driven pump train of the AFWS.

Because of physical separation of the AFWS pumps, location dependencies do not appear to be significant in this plant.

The AFWS pumps require cooling from the plant cooling water system. However, since this system can be run from offsite or onsite AC power supplies, and also has separate diesel-driven pumps, this potential common cause failure does not appear to be of significance.

X.10.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant specific) identified in this section represent actions to improve AFW systems reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations (both generic, denoted by GL, and plant specific) identified in this section involve system design evaluations and/or modifications to improve AFW system

reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.10.3.1 Short Term

1. Recommendation GS-1 - The Technical Specification LCO for one unit operation allows the turbine-driven pump train of that unit to be unavailable indefinitely. Consequently, the plant could not provide AFW flow in the event of loss of offsite and onsite AC power. The licensee should propose modifications to the Technical Specifications to limit the time that a turbine-driven pump train can be inoperable during single unit operation. The licensee should update the Technical Specification LCO for both one and two unit operation to conform with current standard Technical Specifications; namely 72 hours and 12 hours for the outage time limit and action time.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow, including the manual valves V12 and V25 inside containment. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

3. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearing may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the

emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

5. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW signals and associated circuitry are safety grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet

the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.10.3.2 <u>Additional Short-Term Recommendations</u>

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletin and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at <u>W</u>- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. <u>Recommendation</u> - The licensee should provide redundant level indications and a low level alarm in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.
2. <u>Recommendation</u> - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   ". Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train, and there is only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.10.3.3 Long Term

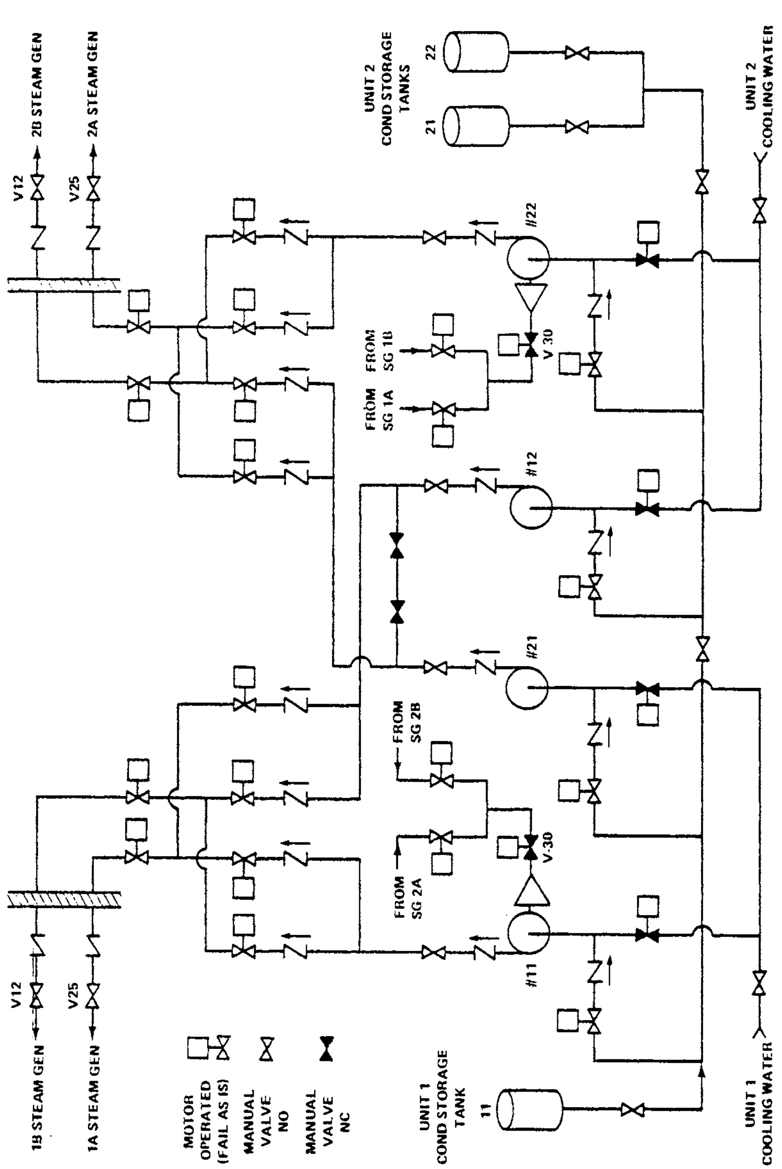
Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being

operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

2. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pump is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available for the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trip on low suction pressure or upgrading the normal source of water to meet seismic Category I tornado protection requirements.

3. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.



Auxiliary Feedwater System
Prairie Island
Figure 1

ENCLOSURE 2

Basis for Auxiliary Feedwater
System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

   g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

   h. RC flow condition - continued operation of RC pumps or natural circulation.

   i. Maximum AFW inlet temperature.

   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

   l. Operating condition of steam generator normal blowdown following initiating event.

   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

   n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 21, 1979

Docket No.: 50-272

Mr. F. P. Librizzi, General Manager
Electric Production
Public Service Electric and Gas Company
80 Park Place, Room 7221
Newark, New Jersey 07101

Dear Mr. Librizzi:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT SALEM UNIT 1

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Acting Director
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Enclosures:
As stated

cc: w/enclosures
See next page

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ENCLOSURE 1

X.12 (W) SALEM 1

AUXILIARY FEEDWATER SYSTEM

X.12.1 System Description

X.12.1.1 Configuration, Overall Design

A simplified flow diagram of the Salem 1 Auxiliary Feedwater System (AFWS) is shown in Figure 1. The AFWS consists of one steam turbine driven pump (880 gpm @ 1550 psi) and two motor driven pumps (440 gpm @ 1300 psi). The turbine driven pump is connected such that it can supply feedwater to all four steam generators. Each of the motor driven pumps is connected to supply feedwater to two different pairs of the four steam generators (SG). The licensee states that each of the three pumps is capable of cooling the plant down to the condition where the RHR system can be used to continue the safe plant shutdown process.

The primary water supply for the AFWS is maintained in the 220,000 gallon seismic Category I Auxiliary Feedwater Storage Tank (AFWST). The water inventory is sufficient for 8 hours decay heat removal and to maintain the water inventory of the steam generators at or above the minimum allowable water level. Low water level or low water temperature in the AFWST will alarm and annunciate in the main control room. The secondary water supply is taken from the non-seismic Demineralized Water Storage Tanks (500,000 gallons each). In addition, two other backup water sources are available from the fire protection and domestic water storage tank and the seismic Category I service water system. A spool piece is required for connection between the AFWS and either of these two other backup water systems. The licensee estimates that approximately 1/2 hour will be needed for manual connection of the spool piece. (see recommendation GS-4)

X.12.1.2 Components - Design, Classification

The AFWS including the primary water supply is classified as an engineered safety-related system and designed according to seismic Category 1 and Quality Group C requirements.

X.12.1.3 Power Sources

The steam turbine driven pump uses steam from two of the four main steam lines taken from a point upstream of the main steam isolation valves (MSIV) and exhausts directly to the atmosphere. Separate isolation valves are provided for the steam supply connections to the AFWS turbine pump. The motor driven pumps receive power from the 4 KV vital buses. The steam supply valve to the turbine driven pump is DC operated (Channel C). The turbine driven pump discharge valves are also DC operated (Channel C). The motor driven pump discharge valves are AC operated (either A or B vital bus).

X.12.1.4 Instrumentation and Controls

X.12.1.4.1 Controls

The instrumentation and controls within the AFWS have been designed as seismic Category I and IEEE 279 components. The SG level is remote manually controlled in the main control room. When the level in any SG is $\leq$ 10% on the narrow range instrumentation, the feedwater flow to the SG is limited to $\leq$ 1.2 in/min. All power operated valves can be manually controlled from the control room; however, on loss of AC or DC power, these valves can be operated locally.

X.12.1.4.2 Information Available to Operator

The information available to the operator includes pump operability (suction pressure, discharge pressure and discharge flow), AFWST discharge pressure, AFWST level and temperature, steam generator steam flow, steam generator water level and control valve position indication. The system instrument designation on Figure 1 and the associated function are listed below.

Instruments

PD-1043 Auxiliary Feed Storage Tank Discharge Pressure.
PA-1676 Auxiliary Feed Storage Tank Discharge Pressure.
PL-1675 Auxiliary Feed Storage Tank Discharge Pressure.
PA-1039 No. 11 Auxiliary Feedpump suction pressure.
PL-3448 No. 11 Auxiliary Feedpump suction pressure.
PT-1677 No. 11 Auxiliary Feedpump suction pressure.

PA-1040 No. 12 Auxiliary Feedpump suction pressure.
PL-3447 No. 12 Auxiliary Feedpump suction pressure.
PT-1683 No. 12 Auxiliary Feedpump suction pressure.
PA-1041 No. 13 Auxiliary Feedpump suction pressure.
PL-3446 No. 13 Auxiliary Feedpump suction pressure.
PT-1685 No. 13 Auxiliary Feedpump suction pressure.
PA-3450 No. 11 Auxiliary Feedpump discharge pressure.
PA-1081 No. 11 Auxiliary Feedpump discharge pressure.
PL-1678 No. 11 Auxliary Feedpump discharge pressure.
PA-3449 No. 12 Auxiliary Feedpump discharge pressure.
PA-1082 No. 12 Auxiliary Feedpump discharge pressure.
PL-1684 No. 12 Auxiliary Feedpump discharge pressure.
PA-1083 No. 13 Auxiliary Feedpump discharge pressure.
PL-1686 No. 13 Auxiliary Feedpump discharge pressure.
FA-1037 No. 11 Auxiliary Feedpump discharge flow.
FA-1038 No. 12 Auxiliary Feedpump discharge flow.
FA-1087 No. 11 Steam Generator steam flow.
FA-1091 No. 12 Steam Generator steam flow.
FA-1095 No. 13 Steam Generator steam flow.
FA-1097 No. 14 Steam Generator steam flow.
TD-3608 Auxiliary Feedwater Storage Tank temperature.
LD-2955 Auxiliary Feedwater Storage Tank level.
LD-3601 Auxiliary Feedwater Storage Tank level.
LL-3443 Auxiliary Feedwater Storage Tank level.
LA-1688 Auxiliary Feedwater Storage Tank level.

X.12.1.4.3 Initiating Signals for Automatic Operation

The motor driven pumps will start automatically on any of the following conditions: loss of offsite power, loss of main feedwater flow, safeguards sequence signal, or low-low level signal from any one of the four steam generators. When either of these pumps is started, a start indication is shown on the status panel in the control room, as well as the remote control station and local indication at its local AFW control panel. The turbine driven pump is started by any one of the following conditions: loss of offsite power, low-low level in two of the four steam generators or undervoltage signal in the RCP group buses.

X.12.1.5 Testing and Technical Specifications

The licensee indicated that after each AFW train maintenance outage, the train would be flow tested to the SG to ensure system flow path alignment. Technical Specifications, Auxiliary Feedwater Systems, Surveillance Requirements 4.7.1.2 provide for certain testing at least once per 31 days in accordance with Station Procedure SP(0) 4.7.1.2(a). Testing the 13 steam driven feed pump does not prevent the two motor driven feed pumps or their associated flow paths from performing their intended function. Technical Specifications Section 4.0.5, in-service testing of pumps, requires periodic testing of each motor driven feed pump in accordance with Station Procedure SP(0) 4.0.5-P. Testing of either motor driven (11 or 12) feed pump does not prevent the remaining motor driven feed pump or the steam driven feed pump from performing its individual safety function.

Surveillance Procedure SPO(0) 4.0.5-P Precautions 3.0 states "Do Not Test More Than One Pump At A Time," and Technical Specifications Action Statement 3.7.12 does not permit more than 1 pump to be out of service for more than 72 hours, otherwise be in Hot Shutdown within the next 12 hours.

X.12.2 Reliability Evaluation Results

X.12.2.1 Dominant Failure Modes Identified

Normally, any one of three subsystems supplying their pump capacity to at least 2 of the 4 steam generators can provide for adequate decay heat removal (given those three transient events considered).

Presently, however, the flow from the AFWS is throttled back to reduce the potential for occurrence of water hammer due to rapid condensation of steam in the steam generator associated with feedwater addition when the SG water level has dropped below the feed ring. Because of this initially throttled operation of the AFWS ($\leq$ 1.2 in/min), the operator must take steps shortly after AFWS actuation to increase pump flow and the rate of fill of the SGs until the desired water level is reestablished in the SGs.

This throttle back procedure 1) serves to reduce the installed AFWS capacity; 2) it reduces the initial flow capacity that can be claimed to exist for the AFWS designs; 3) it interposes the operator initially into the operation of the AFWS; 4) it creates a risk of delayed refill of the SGs thereby increasing the chance of operation (and sticking open) of the PORVs on the pressurizer; and 5) it affects the overall availability that might otherwise be estimated to exist for the Salem Unit 1 AFWS design. For those reasons, we conclude that the need to maintain such AFWS throttle-back procedures should be reassessed by the licensee.

The following failure modes were found to dominate the demand unavailability of the Salem Unit 1 AFWS.

X.12.2.1.1 Loss of Feedwater (LOFW) with Offsite AC Available

The single manual valve (IAFI) in the suction line from the primary water storage tank was assessed to be the dominant fault contribution ($\lesssim$ 70%) for the Salem 1 AFWS. Presently this manual value is not locked open and if inadvertently closed would result in delay or possibly failure of the AFWS on demand. The pumps could be damaged unless prompt operator action was taken to shut the pumps off and take subsequent actions to reestablish a water supply to the AFWS.

The independent failure of both the steam turbine pump subsystem and either of the motor driven pumps subsystems comprised $\lesssim$ 30% of the

next level of dominant fault contributors. These fault contributions would diminish somewhat (such as a factor of two) if the throttle-back procedures were removed.

LOFW with Onsite AC Available

The dominant failure modes for this transient are essentially the same as those described above.

LOFW with Only DC Power Available

For this transient event, the electric AFWS pumps would be unavailable but feedwater to the SG would be automatically provided by the single steam turbine-driven pump. (No AC dependencies were identified for the steam driven portion of the AFWS.) The dominant fault contributions would be those associated with failure of the turbine-driven pumps or the unavailability of this subsystem due to maintenance outage.

X.12.2.2 Principal Dependencies Identified

The principal dependencies identified were those associated with the single manual valve (IAFI) in the AFWS suction line to the principal water supply tank and the throttle-back practice currently being used. Both of these dependencies give rise to potential human errors that disable the availability of the AFWS.

X.12.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.12.3.1 Short Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

2. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

3. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

- Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

The design snould provide for the automatic initiation of the auxiliary feedwater system flow.

- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.12.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.12.3.3 Long-Term

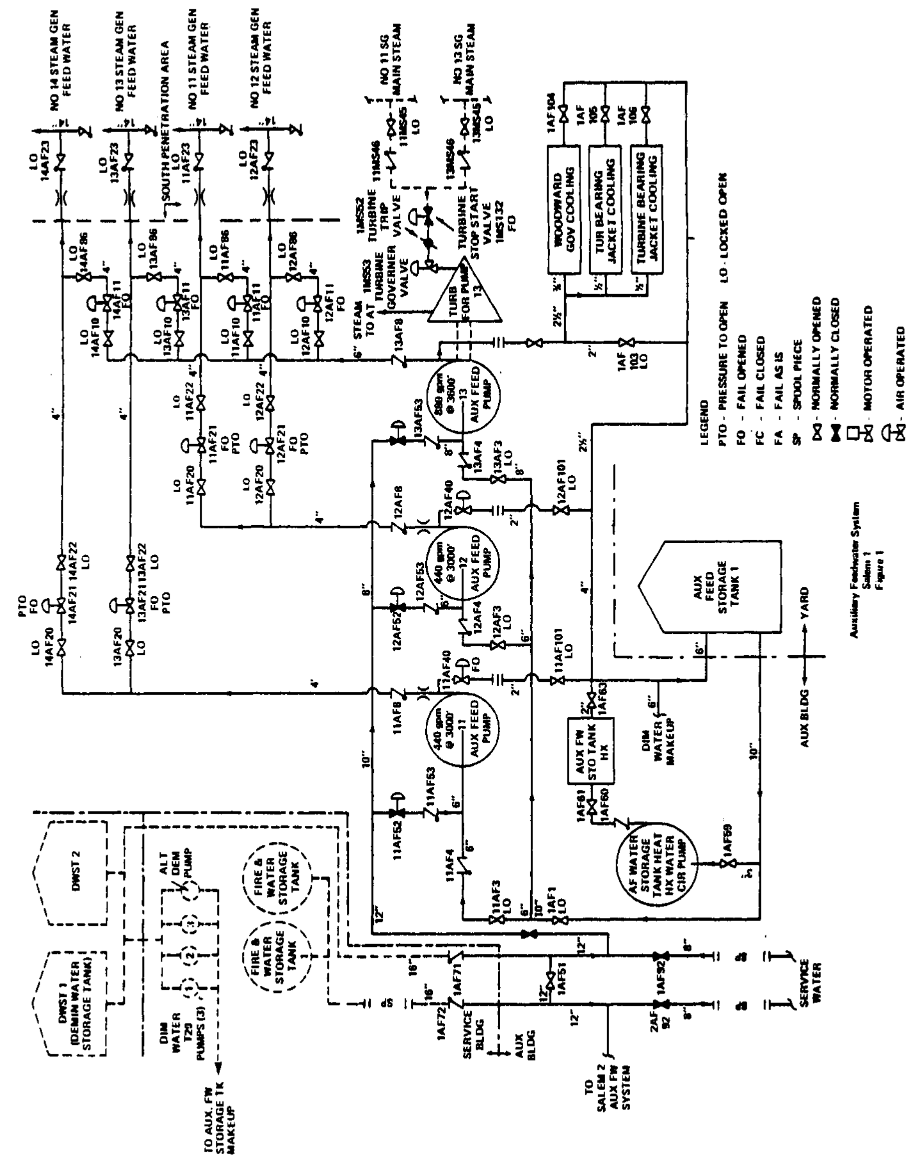
Long-term recommendations for improving the system are as follows:

1. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the prir:ary AFW system water supply passes through valves in a single flo.. path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.



Auxiliary Feedwater System
Salem 1
Figure 1

ENCLOSURE 2

Basis for Auxiliary Feedwater
System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

    1) Loss of Main Feed (LMFW)
    2) LMFW w/loss of offsite AC power
    3) LMFW w/loss of onsite and offsite AC power
    4) Plant cooldown
    5) Turbine trip with and without bypass
    6) Main steam isolation valve closure
    7) Main feed line break
    8) Main steam line break
    9) Small break LOCA
    10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

Enclosure 2

Basis for Auxiliary Feedwater System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
    1) Loss of Main Feed (LMFW)
    2) LMFW w/loss of offsite AC power
    3) LMFW w/loss of onsite and offsite AC power
    4) Plant cooldown
    5) Turbine trip with and without bypass
    6) Main steam isolation valve closure
    7) Main feed line break
    8) Main steam line break
    9) Small break LOCA
    10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 15, 1979

Docket No.: 50-206

Mr. James H. Drake, Vice-President
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Dear Mr. Drake:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

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ENCLOSURE 1

X.13 (W) SAN ONOFRE 1

AUXILIARY FEEDWATER SYSTEM

X.13.1 System Description

X.13.1.1 Configuration, Overall Design

A simplified diagram of the San Onofre Unit 1 Auxiliary Feedwater System (AFWS) is shown in Figure 1. Basically, the Auxiliary Feedwater System (AFWS) is a manually operated system which consists of two auxiliary feedwater pumps (AFP), one motor-driven pump whose capacity is 235 gpm at 1035 psi and one steam driven pump whose capacity is 300 gpm at 1110 psi. Both pumps have common suction and discharge piping and valves.

Flow from the AFW pump can be directed to the three steam generators via two paths. The normal path is from the pumps to the main feed header through connections upstream and downstream of high pressure feed heater. The second path is the emergency feedwater line which is a four-inch line which can be supplied by either AFP. This line branches into three three-inch lines which join the main feed lines for each of the three steam generators between the main feedwater regulating valves (FRVs) and the main feed line containment penetrations. Normally closed isolation valves in the three-inch lines must be manually opened locally to supply feedwater through the emergency lines. Control of AFW flow through the normal path is by means of air-operated auxiliary feedwater regulating valves(AFRVs) which bypass the main FRVs. Another bypass line exists around each of the FRVs. This line has a two-inch manual valve which may be opened to allow feedwater to bypass a failed-closed FRV. The FRVs and AFRVs are air-operated and controlled from the control room. On loss of air, the FRVs fail open while the AFRVs fail closed. Portions of the main feedwater system are also used for safety injection; the main feed pumps are electric motor-driven and are safety-related and are powered from the emergency buses.

Isolation of failed portions of the AFW flow paths can be accomplished by manual valves.

If the motor-driven AFP fails due to electrical or mechanical problems, the turbine-driven AFP is available to provide the necessary steam generator makeup during a shutdown. The flow from the turbine-driven AFP (300 gpm) is sufficient to control and raise steam generator level about four minutes after a scram. The motor-driven auxiliary feedwater pump flow (235 gpm) is sufficient to control and raise steam generator level approximately seven to eight minutes after a reactor scram.

Both AFW pumps receive water via a four-inch line from the condenser make up and reject line, which is connected to the condensate storage tank (CST), with the CST being the primary source of water.

Sources of Water

There are three sources of water for AFW System. The primary source is the Condensate Storage Tank (CST). This tank holds 240,000 gallons of

which 15,000 gallons is dedicated to the AFW System. This will last for approximately three hours. All valves to the AFWS are in the normally open position and are manually operated. This tank is not tornado missile protected.

The secondary source of water is the Primary Plant Make-up Tank (PPMT). This tank holds 150,000 gallons of which a maximum of 105,000 gallons is reserved for the AFWS. The Technical Specifications require a total of 105,000 gallons be available either from this tank or the service water reservoir. The PPMT is not tornado missile protected. The licensee estimates that the 105,000 gallons will last approximately 39 hours. The licensee estimates that conservatively 30 minutes may be required to line up the system, since one manual valve must be opened and a primary plant make-up pump (one is normally operating at all times), is used to put water into the CST.

The back-up source and long term cooling is from the Service Water Reservoir through the service water and fire protection systems. This reservoir has a capacity of 3 million gallons of which 105,000 gallons is dedicated as stated above. Portions of these systems, at least the pumps and some of the piping, are not tornado missile protected and would take about 30 minutes to line-up, since manual valves must be opened and a fire hose must be connected to the CST. Complete loss of water sources to the auxiliary feedwater system such as by extensive tornado damage to the CST would disable the AFWS; whereas, tornado damage to the service piping from the service water reservoir affects the availability of the long term supply of water for the AFW system.

X.13.1.2 Components - Design and Classification

Component	Environmental Qualification	Design Classification	Seismic Category
Motor-Driven Pump	Ambient*	Safety Related	B
Steam-Driven Pump	Ambient*	Safety Related	B
AFWS Piping	—	Safety Related	B
Main Feed Piping After main feed pumps	—	Safety Related	
AFWS Valves	Ambient	Safety Related	B
MFWS Valves - After MFW Pumps	Ambient	Safety Related	
Condensate Storage Tank	Ambient	Safety Related	A
Primary Plant Make-Up Tank	Ambient	Non-Safety Related	B
Primary Plant Make-Up Tank Piping System	Ambient	Non-Safety Related	B
Service Water Reservior	Ambient	Safety Related	A
Service Water System at Pumps	Ambient	Safety Related	
Main Steam Piping	—	Safety Related	A

* 40-104°F 100% Humidity

Seismic Catetory

A = Designed for SSE

B = Designed for OBE

C = Non seismic

The system has been reviewed based on documents which are now available to the Staff for postulated breaks in high energy lines including the Main Steam, Main Feed and Auxiliary Feedwater Systems. As a result of the review, we conclude that for a break in the AFW System discharge piping with or without a single active failure, water can be supplied to the steam generator via the main feed pumps and the main feed system assuming these pumps are available and that there is no safety injection signal. A break in the main feed or main steamline outside containment may result in environmental conditions for which components in the main feed and AFWS have not been demonstrated to be operable. A break in the steam line to the turbine-driven AFW pump at the pump may also result in local environmental conditions for which main feed and AFWS components have not been demonstrated to be operable. In this latter case, one train of the main feedwater system would not be affected and would, therefore, remain available to provide feedwater to the steam generators provided there is no safety injection signal. Based on the above, postulated breaks in the main steam and main feed lines may result in local environmental conditions which may disable conventional means to feed the steam generators and result in steam generators boiling dry.

In conjunction with high energy pipe breaks, the licensee states that in accordance with the criteria established by the NRC and previously approved by the NRC for San Onofre Unit 1, the licensee's analysis of pipe breaks outside containment did not postulate breaks in the annulus between the containment and the turbine building. However, in order to protect against the effects of cracks along pipes in this area, the main steam and main feedwater lines were enclosed in metal sleeves. The licensee stated that they consider that steam released from pipe cracks would, for the most part, condense on the sleeves and drip out the ends and any steam which did go out the ends would tend to rise to the open atmosphere. In view of these considerations. The licensee does not consider credible that the environment at the manual auxiliary feedwater valves located approximately 14 feet below the high energy lines would be such as to prevent an operator from opening the valves.

In the feedwater mezzanine area, in order to preclude breaks in the main steam and feedwater lines, an augmented ISI Program has been established. However, breaks were postulated in smaller piping. In addition, cracks were postulated in all piping. To protect cable trays located below the high energy line from jet impingement from the breaks or cracks, the floor grating was replaced with a plate barrier. Although the cable trays penetrate the turbine building wall about 5 feet above the manual auxiliary feedwater valves, the licensee believes that steam is inhibited from passing through these pentrations by the plate barrier. Although steam could pass through penetraions at the elevation of the main steam and feedwater lines, the licensee considers that this steam would tend to rise to the open atmosphere. In the area of the manual auxiliary feedwater valves (about 14 feet below these lines) the licensee believes that the environment would not be so adverse as to prevent an operator from opening the valves. See Section 13.3.3, Long Term Recommendations, and Section 13.3.4, Systematic Evaluation Program Considerations for recommendations relating to high energy pipe breaks.

X.13.1.3 Power Sources

The steam supply for the turbine-driven AFW pump is provided from the main steam header from a connection upstream of the main steam stop valves. The turbine-driven AFW pump is started by local manual startup of the turbine. An air operated valve supplies steam to the turbine and takes power from D.C. Bus 1. On loss of air pressure this valve would fail closed. However, it can be opened manually to control the turbine locally.

The motor-driven pump can be started from the control room, the auxiliary control panel, or with the local operation of its breaker in the 480V switchgear room. The motor receives power from 480V switchgear bus #3. This bus can receive electrical power from both offsite and onsite sources. During a loss of offsite power, emergency diesel generator #1 supplies power to switchgear bus #3 via 4160V bus 1C after the electrical system has been realigned.

The main feedwater regulator motor-operated block valves take their power from the A-C buses and fail in the as-is position on loss of power. These valves can be manually operated locally.

Upon loss of all A-C power, the turbine pump will provide water to the steam generators via manually operated valves. The pump bearings will be cooled by gravity feed from the service water reservior.

X.13.1.4 Instrumentation and Controls

X.13.1.4.1 Controls

All controls for the system are local, manual controls except for the motor driven pump on-off control, the main feedwater regulating valves control and the auxiliary feedwater regulating valve control. These controls are located locally as well as at the remote shutdown panel and the control room. The motor-operated block valves are controlled only locally or in the control room.

X.13.1.4.2 Information Available to the Operator

The following information is available to the operator in the control room.

1. CST, PPMT, and SWR water level alarms
2. CST and PPMT tank level indication
3. Steam Generator Level
4. Steam Generator Low Level Alarms
5. Flow at feed flow control valves
6. Main steam pressure
7. Main Feed Line Pressure
8. Main Feed Flow Control Valve Position Indication
9. Electric AFW pump operation and ammeter

The following information is available at the remote shutdown panel.

1. Steam generator level indication.
2. Electric AFW pump operation

All other information needed by the operator can be found at the local stations.

X.13.1.5 Initiating Signals for Automatic Operation

Since the system is a manually initiated system this section is not applicable, but manual initation is started on loss of main feed pumps and low steam generator level. Subsequent to the staff review of the San Onofre 1 AFW system, the licensee completely revised his emergency operating instructions related to abnormal steam generator water level (including loss of main feed pumps) and steam generator high energy pipe break. These revised procedures identify plant symptoms and provide specific immediate and subsequent action requirements for the control room operator and the dedicated operator stationed at the redundant AFW system manually operated control valves to initiate AFW system operation.

X.13.1.6 Testing

Both Auxiliary Feedwater Pumps are required to be tested bi-weekly, but the licensee states that they are presently being tested weekly in the recirculating mode of operation. The turbine-driven pump is tested every six months in an overspeed condition.

The feed control valves are used continuously for plant operation. All other normally closed valves are not tested except when in use.

The two diesel generators are tested monthly on a staggered bases.

X.13.1.7 Technical Specifications

The Technical Specifications for the plant that are applicable to the Auxiliary Feedwater System are as follows:

TURBINE CYCLE

Operating Status

Applicability: Applies to the operating status of the turbine cycle.

Objective: To define conditions of the turbine cycle necessary to ensure the capability to remove decay heat from the core.

Specification: The reactor shall not be pressurized above 500 psig unless the following conditions are met:

(1) A minimum turbine cycle steam-relieving capability of 5,706,000 lb/hr (except for resting of the main steam safety valves).

(2) Both auxiliary feedwater pumps operable, or the steam-driven auxiliary feedwater pump is continuous operation when the residual decay heat levels are greater than the natural heat losses from the reactor coolant system.

(3) A minimum of 15,000 gallons of water in the condensate storage tank, and an additional 105,000 gallons in the service-water reservoir and/or the primary plant makeup tank.

(4) System piping and valves directly associated with the above components operable.

After criticality is achieved, one auxiliary feedwater pump may be removed from service for maintenance for a period not to exceed 24 consecutive hours.

X.13.2 Reliability Evaluation

X.13.2.1 Dominant Failure Modes

The San Onofre auxiliary feedwater system was analyzed to determine the dominant failure modes under three transient conditions:

a. LOFW with offsite power available

b. LOFW with onsite power available

c. LOFW with only DC power available.

Results of the Anaysis are summarized below.

LOFW with Offsite Power Available

Unavailability of the auxiliary feedwater system is dominated by the following:

a. Operator failure to actuate system upon demand;

b. Failure of the single manual valve in the supply line from the condensate storage tank.

The operator must recognize conditions requiring auxiliary feedwater, start the pumps (electric pump from the control room or turbine pump locally) and locally open the normally closed manual discharge valves. Despite having a dedicated man at the local station, his actions are dependent upon instruction from the control room operator. The availability of the system is, thus, dependent upon the knowledge and actions of the control room operator.

Despite several sources of water, all water is drawn from the condensate storage tank through a single manual valve. Should this valve fail closed, the system will be unavailable.

<u>LOFW with Onsite Power Available</u>

The unavailability of the system is dominated by the same factors as the case discussed above. Postulating loss of one of the two diesel generators does not effect the dominant failure modes.

<u>LOFW with Only DC Power Available</u>

Despite loss of all AC power, the turbine-driven pump train could continue to supply the necessary auxiliary feedwater. Sufficient cooling should be supplied by gravity feed to keep the pump bearings cool.

Short term system unavailability ($\leq$ 30 minutes) is dominated by the potential for maintenance being performed on the turbine driven pump and by the possibility of the discharge block valve inadvertantly being left closed following maintenance on the pump.

Long term unavailability ($\geq$ 30 minutes) is dependent upon assuring that the steam admission valve remains open. This air operated valve will fail closed upon subsequent loss of air which is dependent upon AC power. Manual action is required to open this valve or to provide an air supply after about 30 minutes when local air reservoirs could be depleted.

X.13.2.2 <u>Dependencies</u>

One potential dependency was identified in the analysis. Both pumps of the auxiliary feedwater system are in a common location making them susceptible to any locally adverse conditions such as high energy breaks or fires.

X.13.3 <u>Recommendations for this Plant</u>

The short-term recommendations identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.13.3.1 Short-Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendations GL-2 for the longer term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered up by the procedures:

   The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   The case in which the primary water supply is being depleted. The procedures for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes should be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

   The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system vlaves in their normal alignment.

5. Recommendation GS-8 - The licensee should install a system to automatically initiate AFW system flow. This system need not be safety-grade; however in the short-term, it should meet the criteria listed below, which are similar to Item 2.1.7a of NUREG-0578. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-1.

   The design should provide for the automatic initiation of the auxiliary feedwater system flow.

- The automatic initiation signals and circuits should be designed so that a single failure will not results in the loss of auxiliary feedwater system function.
- Testability of the initiating signals and circuits should be a feature of the design.
- The initiating signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
- The automatic initiation signals and circuits should be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

6. Recommendation - The licensee should propose modifications to Technical Specifications so that manual valves that are normally closed will be tested periodically.

7. Recommendation - The licensee should install valve operators that can be controlled from the control room on all the normally closed manual discharge valves. This will reduce the time delay inherent in

the present manual set-up as discussed in Section 13.2.1. The AFW system could then be operated from the control room until the system has been fully automated. (See Recommendation 5 above).

8. Recommendation - To reduce dependence on a single flow path from the water sources and increase the quantity of water reserved and readily available for the AFW system, the licensee should connect temporary piping or a fire hose from the Service Water Reservoir/ fire protection system directly to the AFWS pump suction header.

X.13.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or trasnfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shutdown and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578: "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in the Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on the AFW system

train, and there is only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.13.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-1 - Licensees with plants having a manual starting AFW system, should install a system to automatically initiate the AFW system flow. This system and associated automatic initiation signals should be designed and installed to meet safety-grade requirements. Manual AFW system start and control capability should be retained with manual start serving as backup to automatic AFW system initiation.

2. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the primary AFW system water supply passes throughvavles in a single flow path, but the alternate AFW

system water supplies connected to the AFW system pump suction piping downstream of the above valve(s) should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure. The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

3. Recommendation - GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

4. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category 1 and tornado protection requirements.

5. Recommendation - The licensee should evaluate the following concerns:

   a) A break in the main feed or main steamline outside containment or a break in the steamline to the turbine driven AFW pump may result in environmental conditions for which the main feed and AFW system components are not qualified.

   b) The San Onofre Unit 1 AFW system design does not meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position 10-1; namely, that the AFW system should maintain the capability to supply the required AFW flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines concurrent with a single active failure.

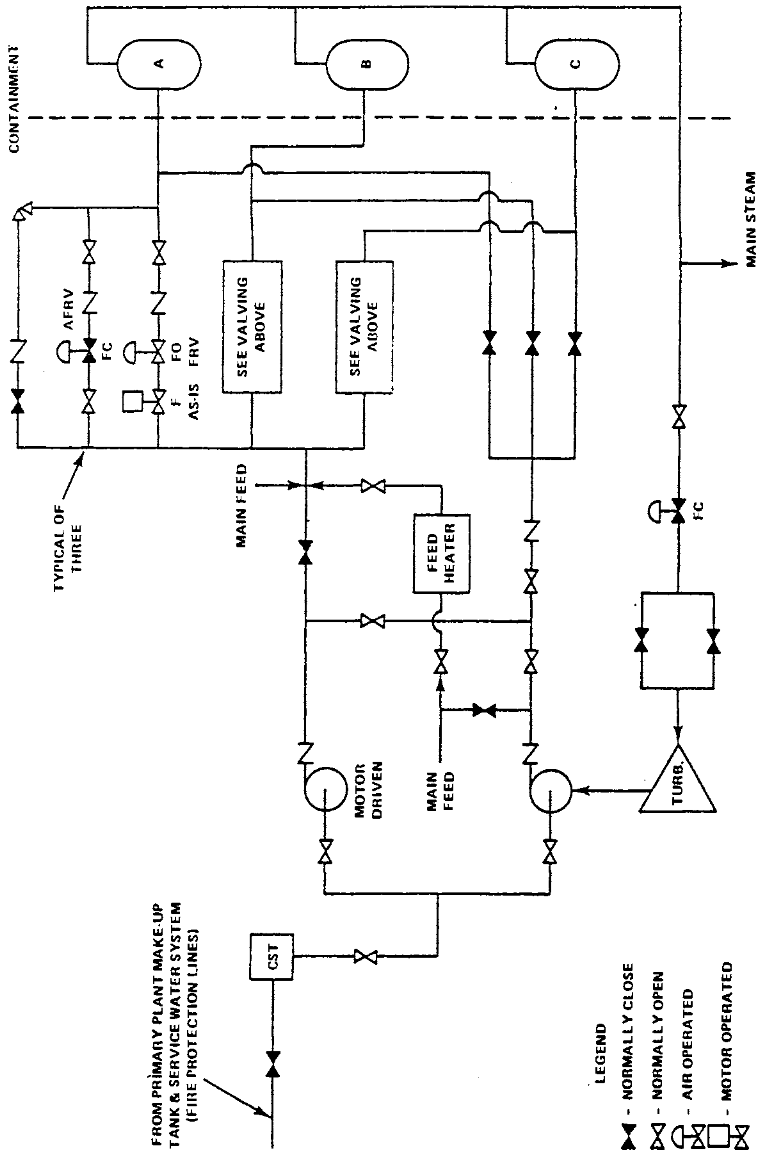
   The licensee should evaluate the postulated pipe breaks stated above and (1) determine any AFW and main feedwater system design changes including environmental qualification, or procedures necessary to detect and isolate the break and direct the required feedwater flow to the steam generator(s) before they boil dry or (2) describe how the plant can be brought to a safe shutdown condition by use of other systems which would be available following such postulated events.

13.3.4 Systematic Evaluation Program Considerations

The following items are still under review by the Systematic Evaluation Program (SEP) and supplement the above long term recommendations:

1. The San onofre Unit 1 plant, including the AFW System, will be reevaluated during the SEP with regard to internally and externally generated missiles, pipe whip and jet impingement including main steam and main feed line breaks inside and outside containment, quality and seismic design requirements, and the effects of earthquakes, tornadoes and floods.

2. The San Onofre Unit 1 AFW System is not automatically initiated and the design does not have capability to automatically terminate AFW flow to a depressurized steam generator and provide flow to the intact steam generator in the event of a main steam or main feed line break. The effect of this design will be assessed in the design basis event evaluations for San Onofre Unit 1.



Auxiliary Feedwater System
San Onofre 1
Figure 1

ENCLOSURE 2

Basis for Auxiliary Feedwater
System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

   g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

   h. RC flow condition - continued operation of RC pumps or natural circulation.

   i. Maximum AFW inlet temperature.

   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

   l. Operating condition of steam generator normal blowdown following initiating event.

   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

   n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 25, 1979

Docket Nos.: 50-280
50-281

Mr. W. L. Proffitt
Senior Vice-President - Power
Virginia Electric and Power Company
P. O. Box 26666
Richmond, Virginia 23261

Dear Mr. Proffitt:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT SURRY POWER STATION, UNITS 1 AND 2

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Brian Grimes
for Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

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ENCLOSURE 1

SURRY 1 & 2

AUXILIARY FEEDWATER SYSTEM

X.14.1 System Description

X.14.1.1 Configuration and Overall Design

The auxiliary feedwater system (AFWS) is designed to supply water to the steam generators for reactor coolant system sensible and decay heat removal when the normal feedwater system is not available. The AFWS can be utilized during other periods, such as during startup and shutdown, in the event of malfunctions such as loss of offsite power and also in the event of an accident.

The AFWS is designed to seismic Category I requirements and is tornado missile proof.

A simplified flow diagram is shown on Figure 1. The system consists of two motor driven pumps (3A and 3B) and one steam driven pump (2). Each motor driven pump has a net capacity of 350 gpm; the turbine driven pump has a net capacity of 700 gpm. Taps from each main steam line at a point upstream of the main steam isolation valves provide the source of steam to the turbine through two parallel valves, one motor and one air operated. The motor driven pumps are connected to separate Class IE 480VAC emergency buses.

Normally, the pumps take suction from the emergency condensate storage tank, with a capacity of 110,000 gallons. This provides approximately five hours of operation at system design flow of one motor-driven pump. This tank is designed to seismic Category I requirements and is protected from tornado missiles. Additional supplies for the AFW system are as follows:

1. 300,000 gallons from a non-seismic condensate storage tank.
2. 110,000 gallons underground storage tank - seismic Category I and missile protected.
3. 2-300,000 gallon Fire Main Supply Tanks (non-seismic Category I), Fire Main Supply Piping (seismic Category I).
4. A cross-connect to the other unit's water supply, consisting of the same supplies listed in 1, 2, and 3 above.

Each of the three pumps discharge into two headers, aligned by manual valves. There are three lines from each header, and each line contains a motor operated valve located inside containment. The lines join downstream of the MOVs and form a common discharge line supplying each steam generator via the associated main feed line. In the event of failure of one header, the supplies from the pumps may be isolated from the failed header by manual operated valves to assure steam generator water flow from the other header. The motor operated valves (MOV) in the system flow path are normally open, and fail as-is. The air operated valve in the turbine steam supply system is normally closed, and fails open; the parallel MOV is normally closed and fails as-is. The AFWS discharge lines of both units are cross connected but are isolated by normally closed MOV valves. Operator action will permit the AFWS of one unit to supply water to the steam generators of the opposite unit.

X.14.1.2 System Design Classification

The turbine pump train and motor pump trains (pumps, valves, motors, piping) are seismic Category I and tornado missile proof (Class 1E for electrical equipment).

X.14.1.3 Power Sources

The motor driven pumps and valves are supplied from the Class 1E A-C emergency buses; 3A from Emergency bus 1H, 3B from emergency bus 1J. The air operated turbine pump steam admission valve is D.C. solenoid operated and fails open. The parallel MOV is powered by Class 1E A-C power and fails as is.

X.14.1.4 Instrumentation and Control

The instrumentation and control power supplies are from the 120 VAC vital bus system. There are four vital buses, two supplied from inverters connected to the emergency DC power supplies and two regulated power supplies connected to the AC emergency buses.

X.14.1.4.1 Controls

Steam generator level is controlled manually from either the main control room or the auxiliary shutdown panel by operating the appropriate MOV in the AFW line. The valves are motor operated, are normally open and fail as-is on loss of power. Class 1E instrumentation is provided (level and flow indications).

Information Available to the Operator

Except as noted, the following indications are available at both operating stations:

1. MOV position indication
2. Air operated valves position indication, turbine control system
3. Auxiliary feedwater flow (Control Room only)
4. SG level-wide range
5. Auxiliary feed pump amperage (Control Room only)
6. Breaker (motor driven pump) position
7. Condensate (110,000 gal) tank level (Control Room only)

Initiating signals for Automatic Operation

1. The following signals start the motor driven pump motors:
   a. Low-Low level from any steam generator
   b. Undervoltage on transfer buses D & F
   c. Safety injection
   d. Trip of both main feed pumps
   e. Manual

2. The following signals open the steam control valves starting the steam turbine:
   a. Undervoltage on the Station Service bus (2 out of 3 logic)
   b. Low-Low steam generator level-2 out of 3 steam generators
   c. Manual

X.14.1.5 Testing

The systems are tested every 30 days in accordance with technical specification requirements. In addition to the periodic tests, operational tests are performed in accordance with surveillance tests following maintenance on a particular system or component. The instrumentation systems are tested periodically, per shift, every 30 days and every 18 months in accordance with technical specification requirements.

X.14.1.6 Tech Specs

The limiting condition of operation (LCO) permits plant operation if two of the three auxiliary feedwater pumps are operable. This could result in unrestricted plant operation if any of the three (including the steam driven pump train) remains inoperable.

X.14.2 Reliability Evaluation Results

X.14.2.1 Dominant Failure Modes

Successful delivery of feedwater is considered to be the flow of at least 350 gpm to one (or more) of the three steam generators, for the transients considered here.

Failure modes of the AFWS were assessed for three types of initiating transients. The dominant failure modes for each transient type are discussed below.

Loss of MFW with offsite power available

The reliability analysis of the Surry AFWS based on this initiating transient did not identify any single failures or double failures which would fail the entire AFWS. Consideration was given to combinations of three failures such as a combination of one pump out for maintenance, coupled with hardware failures in the other two lines. However, the dominant failure mode was assessed to be a common cause failure resulting from the failure to reopen all the manual pump discharge valves following test, coupled with the failure to either reopen at least one valve within approximately one-half hour after a demand on the AFWS, or to draw feedwater from the other unit's AFWS through the unit intertie connections.

Loss of MFW with only onsite AC power available

The response of the Surry AFWS to this transient should not be significantly different from that for the case discussed above. As such, it is again concluded that the dominant failure mode is the human error of failing to reopen the manual pump discharge valves after test, coupled with the failure to reopen one valve or to realign the other unit's AFWS within 30 minutes after an AFWS demand.

Loss of MFW with only DC power available

In this transient neither onsite or offsite AC power is available; thus the AFWS is reduced to the one steam-driven pump train. Failure of this train can occur in a number of ways. The results of this examination indicate that the dominant mode of failure is that the steam-driven train is out of service due to maintenance. The current Surry Technical Specifications and LCOs permit the outage of one AFWS pump indefinitely, so that the possibility of the steam train being out of service in a station blackout incident could be high. Revision of the Surry Technical Specifications and LCOs to normally require the operability of all three trains except for limited maintenance outages (as in the standard Tech Specs) would improve the reliability of the AFWS substantially for this transient.

X.14.2.2 Potential Dependencies

The potential for a common-cause failure of the AFWS due to human error is, as discussed above, the most significant dependency found in this analysis. A second potential common-cause failure due to commonalities of equipment location was also noted; however, since the unit intertie system was installed specifically to alleviate this possible problem, this does not appear to be a significant concern.

X.14.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.14.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumention can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that

     the valves are properly aligned.

   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.

   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

   - Testability of the initiation signals and circuits shall be a feature of the design.

- The initiation signals and circuits should be powered from the emergency buses.

- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.

- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

6. Recommendation - Procedures should be established to lock open and periodically verify open position of all manual AFWS valves inside containment.

7. Recommendation-The licensee should require staggering of the periodic pump train tests (e.g., one train at North Anna is tested every 10 days rather than all three trains tested at once on a monthly basis). This reduces the potential for inadvertently leaving closed the discharge valves of all trains after test.

8. Recommendation - Emergency procedures should be available to the operators for operating the AFWS of one unit such that it is supplying water to the steam generator(s) of the opposite unit in the event that such an operating mode should be necessary.

X.14.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and

run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room

be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X-14.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

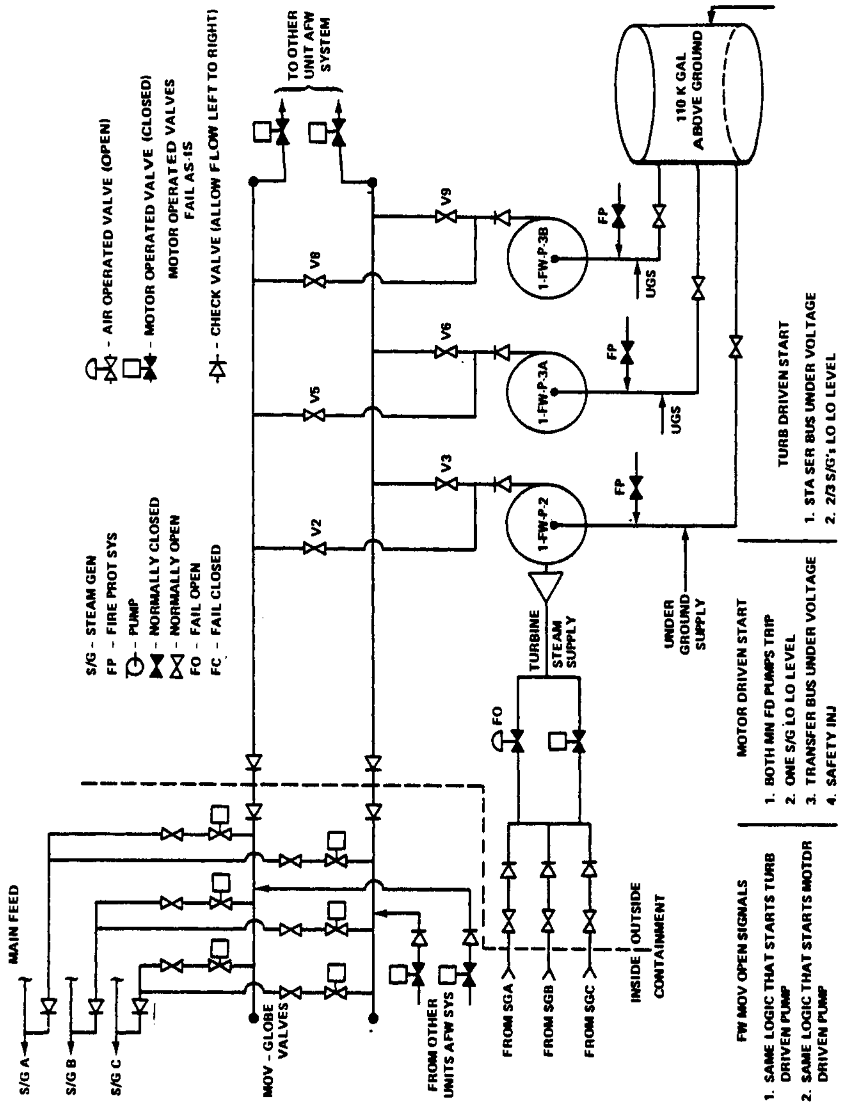
1. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

2. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

3. Recommendation - The AFWS flow control valves for both the motor and turbine pump trains are AC powered, normally open, fail as-is motor operated valves which are located inside containment. Also, manual, normally open valves are located inside containment. The AFWS design should be reevaluated, including the possibility of relocating the valves outside containment, assuming an accident inside containment which necessitates AFWS operation and which creates a containment environment (humidity, radiation) that precludes access to the valves. The reevaluation should consider the following:

a. A possible common mode failure (environmentally induced) causing spurious closure or failure of the MOV's in a throttled position.

b. An AFWS line break downstream of the MOV's and failure of the MOV's to operate.



Auxiliary Feedwater System
Surry Nuclear Station
Figure 1

ENCLOSURE 2

Basis for Auxiliary Feedwater System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

   - Maximum RCS pressure (PORV or safety valve actuation)
   - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
   - RCS cooling rate limit to avoid excessive coolant shrinkage
   - Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket No.: 50-344

OCT 3 1979

Mr. Charles Goodwin
Assistant Vice President
Portland General Electric Company
121 S. W. Salmon Street
Portland, Oregon 97204

Dear Mr. Goodwin:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT TROJAN NUCLEAR PLANT

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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ENCLOSURE 1

X.15 (W) TROJAN

AUXILIARY FEEDWATER (AFW) SYSTEM

X.15.1 System Description

X.15.1.1 Configuration, Overall Design

The Auxiliary Feedwater (AFW) system for Trojan uses two full capacity pumps (1 turbine driven, 1 diesel driven - 880 gpm per pump) to feed four steam generators. A simplified flow diagram of the Trojan AFW is shown in Figure 1. The licensee is currently planning installation of a third motor driven pump (non-safety grade) for startup and shutdown. Each of the two installed pumps takes suction from a non-safety grade condensate storage tank through manually locked open valves via a common suction header. The seismic Category I classification of the AFW system stops at the check valve in each pump's suction line. A long term seismic Category I supply of water from the service water system (SWS) connects to the piping between the check valves and the pumps via normally closed motor operated isolation valves. These valves can be opened or closed from the control room.

Each pump feeds all four steam generators through a discharge line which branches into four lines to supply the four steam generators. Each pump discharge line is provided with a check valve and an isolation valve to permit maintenance of the pump and the check valve. The discharge line then branches into four lines to supply the four steam generators. Each of these four lines is provided with a motor-operated control valve, a check valve downstream, and manually-operated isolation valves upstream and downstream of the control valve.

Each of these four auxiliary feedwater lines joins with a corresponding line from the second pump into a single line in which a flow indicator is provided for remote and local indication. Also, flow restrictors are located upstream of each motor-operated control valve in each of the two supply lines to each steam generator. In the event of a pipe break downstream of the MOV, a high-flow signal from a flow element at one of these restrictors will automatically close the motor-operated valve associated with the orifice. The single auxiliary feed line then joins with the steam generator main feedwater line in the Seismic Category I section between the feedwater line isolation check valve and the containment.

The system is designed to automatically start both AFW pumps upon receipt of initiating signals. All valves in the system flowpath are normally open and fail as-is. The steam turbine driven pump (880 gpm to the S/G's) is auto/manual started by opening motor operated isolation valves from the steam lines of all four steam generators and by opening the turbine trip throttle valve in the common header downstream of the four inlet valves. Service water to cool the lube oil of the turbine driven pump is automatically initiated by opening of a

MOV from the service water system whenever the turbine pump gets a signal to start. The licensee is presently revising this design such that the lube oil will be cooled by the discharge of the turbine driven pump.

The diesel driven pump has its own starting battery which automatically starts the diesel. The initiating signals also start a reduction gear lube oil priming pump, and open an MOV to supply service water for jacket cooling and lube oil cooling. The diesel has a day tank with a 500 gallon capacity good for 10 hours of diesel operation. Automatic transfer of oil to the day tank from the emergency diesel fuel oil transfer system is controlled by day tank level.

Both the diesel and turbine driven pumps use governors that control the speed to automatically maintain a set pressure differential between the pump discharge and the steam generators. This pressure differential can be selected by the control room operator to help control steam generator level.

X. 5.1.2 <u>Components, Design Classification</u>

The condensate storage tank and the piping from the condensate storage tank to the check valve in each pump's suction line are non-safety grade (non-seismic). The recirculation lines from the pump discharge to the condensate storage tank are also non-safety grade. The pumps, piping, valves, and valve actuators for the rest of the AFW system are seismic Category I.

The controls, instrumentation and power supplies for the operation of the auxiliary feedwater system are seismic Category I, Class IE. However, the actual indicators in the control room are not designed to meet seismic Category I requirements.

The ventilation supply fans, diesel fuel oil and lube oil system, service water cooling supply and water source supply are designed to seismic Category I requirements.

X.15.1.3 Power Sources

The turbine driven pump (train A) is associated with the train A electrical buses,including the train A ESF channels for automatic operation.

The diesel driven pump is supplied by train B in the same manner as the turbine driven pump.

Neither train is independent of AC power. The steam inlet valves to the turbine driven pump are operated from the train A vital 480 volt AC bus. Manual operation of these valves is required to open these MOV's upon loss of the train A vital bus,since they fail as-is and are normally closed. These valves are normally closed to protect

against the effects of a steam line break in the supply line downstream of these valves.

The diesel driven pump may start and operate without AC power but due to lack of cooling water to the jacket and lube oil, and due to lack of ventilation, operation of the diesel could not be sustained. The licensee estimates that this diesel will trip on over-temperature in 5-10 minutes.

The vital DC buses are used to supply control power to the speed governors for both pumps and for operation of the turbine trip throttle valve to the turbine driven pump.

X.15.1.4 Instrumentation and Controls

X.15.1.4.1 Controls

All controls for normal operation for the AFW system are Class 1E and operated from the control room. These include steam inlet valves, the trip throttle, the steam control valve in the steam line to the turbine, the AFW flow control valves (2 to each S/G), and pump start and stop.

Steam Generator level is controlled from the control room by controlling pump speed and opening/closing of the AFW flow control valves to the steam generator. Steam generator level transmitters and instrumentation circuits are safety grade but the level indicators on the control panel are not.

X.15.1.4.2 Information Available to the Operator

I. Alarms

1. Hi/Lo Steam Generator Level
2. Lo Condensate Storage Tank Level
3. Low AFW Pump Suction Press-each pump
4. Local-Control Override for each pump
5. Hi Conductivity AFW

II. Indication

1. Steam Generator Level
2. Condensate Storage Tank Level
3. Valve Position Indication for all MOV's, including SWS
4. Steam Pressure at Turbine
5. Suction Press each pump
6. Discharge Press each pump
7. Auto/Man Light for Pump Control
8. AFW Flow to each Generator (not for each pump)
9. Differential Pressure - Pump discharge and steam generator

X.15.1.4.3 Initiating Signals for Auto Operation

Both Pumps
1. Safety Injection Signal
2. Lo-Lo Level in any steam generator (2 out of 3 detectors)

3. Loss of Both main Feed Pumps
4. Loss of Offsite Power - Sensed on Vital Bus

X.15.1.5 Testing

Pumps and motor operated valves are tested monthly. All the MOV's, including the service water system supply isolation valves, are cycled during their monthly test. The pumps are tested by closing the flow control valve for the pump, starting the pump, and checking pressure and recirculation flow. All testing is done from the control room.

Every 18 months, a flow verification test from the condensate storage tank to the steam generators is performed. Also every 18 months, automatic start of the AFW pumps from the auto-start logic is tested. Every 18 months, a routine instrumentation and controls calibration check is performed.

X.15.1.6 Technical Specifications

1. At least two independent steam generator auxiliary feedwater pumps and associated flow paths shall be operable with:
   a. One feedwater pump capable of being powered by an operable diesel with $\geq$450 gallons of fuel in its day tank, and
   b. One feedwater pump capable of being powered from an operable steam supply.

2. With one AFW pump inoperable, restore the inoperable pump to operable status within 72 hours or be in hot shutdown within the next 12 hours.

X.15.2 Reliability Evaluation

X.15.2.1 Dominant Failure Modes

LOFW with Offsite Power Available

The dominant failure contributors include a combination of human errors associated with the two water sources (manual valve left closed and failure to take corrective actions). Other dominant contributors are combinations of hardware failures associated with each pump train, test and maintenance outages, and a human error resulting in manual valves left closed in the pump discharge lines, undetected by control room indication or by pump test indications.

LOFW with Loss of Offsite Power with Onsite AC Power Available

The dominant failure contributors are the same as for the non-LOP transient with the addition of a single emergency AC train failure in combination with other failures in the other pump train.

LOFW with Loss of All AC, DC Available

Under present design, assuming completion of the modification to provide bearing cooling water from the AFW turbine pump line, the dominant failure contributors are single failures. They include the human failure to open a condensate storage tank manual valve, the hardware, test and maintenance and human error contributors associated

with the turbine train-human failure to open the AC steam inlet MOV's by hand and AC power dependence for cooling the diesel driven pump.

X.15.2.2 Interdependencies

The principal noted dependency is the design for AC cooling of the diesel driven pump and for operation of the turbine steam inlet valves.

X.15.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.15.3.1 Short Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position.

These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions.

Since the water for cooling of the lube oil for the turbine-driven pump bearing may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures and, if necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer term resolution of this concern.)

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:
   - Proceduresshould be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation GS-7 - The licensee should verify that the automatic start AFW signals and associated circuitry are safety grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety grade requirements as indicated in Recommendation GL-5.

- The design should provide for the automatic initiation of the auxiliary feedwater system flow.
- The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
- Testability of the initiation signals and circuits shall be a feature of the design.
- The initiation signals and circuits should be powered from the emergency buses.
- Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
- The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

- The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room.

X.15.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.
2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test

acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

?.10.3.3 Long Term

Long-term recommendations for improving the system are as follows:

1. Recommendation - GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s) should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation - GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

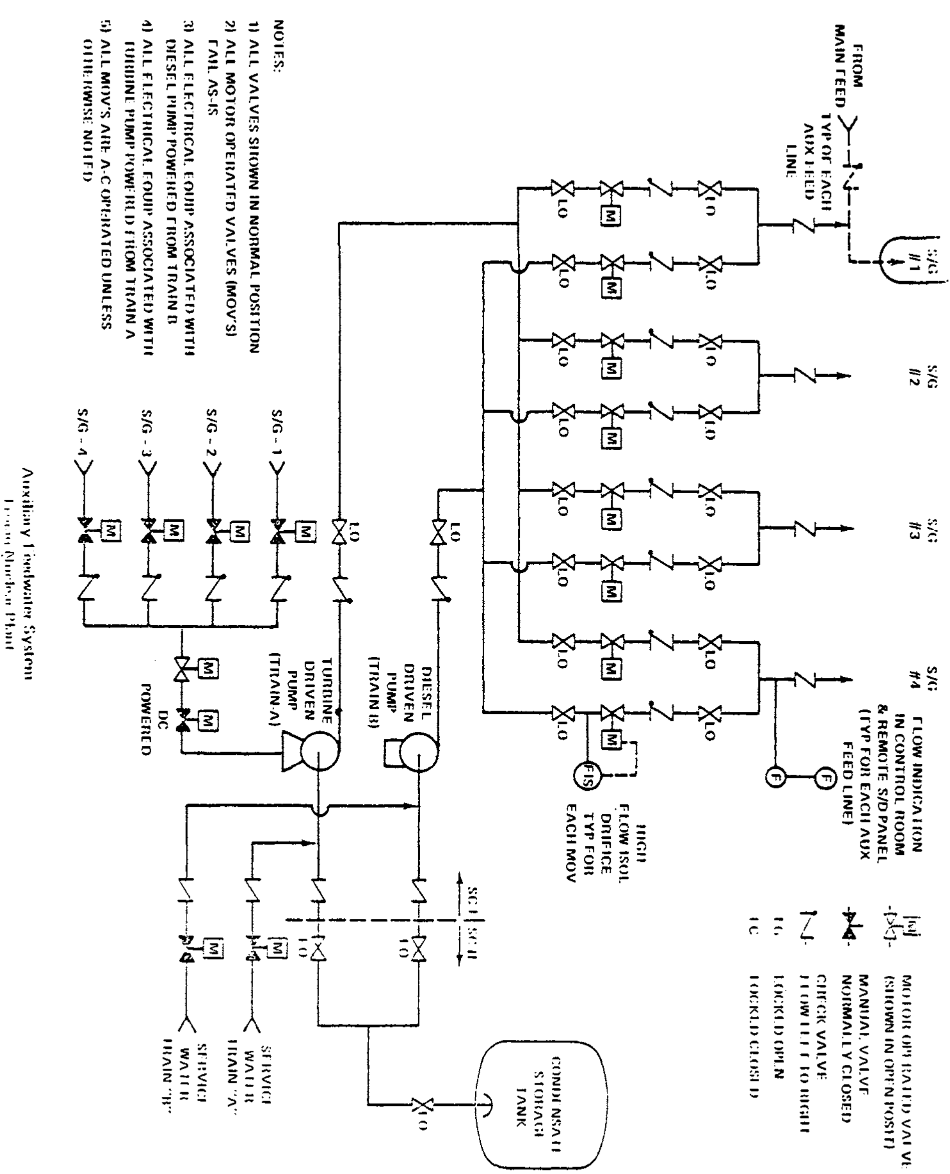
3. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of

their AFW systems to determine if automatic protection of the pump is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available for the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

4. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

5. Recommendation - A motor driven pump is currently being installed or is planned to be installed by the licensee. Present plans are for a non-safety grade motor driven pump system. Based on past experience of the problems associated with the speed control (overspeed trips) of both the diesel and turbine driven pumps and other Licensee Event Reports on the Trojan AFW system, the licensee should further review the proposed installation to determine if the motor driven pump should be safety grade and automatically actuated by the AFW automatic start logic.

6. Recommendation - A pipe break in certain locations of the turbine driven auxiliary feedwater pump discharge piping may affect both AFW trains, since portions of this piping pass through the diesel driven pump room. The motor driven pump to be installed should be located such that a break in the AFW system (not associated with the motor driven pump train) could not affect the motor drive pump. Also the licensee should 1) install the motor pump with appropriate valves in the pump discharge line connections to meet the high energy line break criteria in SRP 10.4.9 and Branch Technical Position 10-1; namely, the AFWS should maintain the capability to supply the required AFW flow to the steam generator(s) assuming a pipe break anywhere in the AFW pump discharge lines plus a single active failure or 2) describe how the plant can be brought to a safe shutdown condition by use of other available systems following such a postulated event.



Auxiliary Feedwater System

ENCLOSURE 2

Basis for Auxiliary Feedwater System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
    1) Loss of Main Feed (LMFW)
    2) LMFW w/loss of offsite AC power
    3) LMFW w/loss of onsite and offsite AC power
    4) Plant cooldown
    5) Turbine trip with and without bypass
    6) Main steam isolation valve closure
    7) Main feed line break
    8) Main steam line break
    9) Small break LOCA
    10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:
   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.
   b. Time delay from initiating event to reactor trip.
   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).
   d. Minimum steam generator water level when initiating event occurs.
   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.
   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.
   g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?
   h. RC flow condition - continued operation of RC pumps or natural circulation.
   i. Maximum AFW inlet temperature.
   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.
   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.
   l. Operating condition of steam generator normal blowdown following initiating event.
   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.
   n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCTOBER 1 8 1979

Docket Nos.: 50-250
50-251

Dr. Robert E. Uhrig, Vice-President
Advanced Systems and Technology
Florida Power and Light Company
P. O. Box 529100
Miami, Florida 33152

Dear Dr. Uhrig:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT TURKEY POINT PLANT, UNITS 3 AND 4

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Robert L. Tedesco for
Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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ENCLOSURE 1

X.16 (W) TURKEY POINT UNITS 3 & 4

AUXILIARY FEEDWATER SYSTEMS

X.16.1 System Description

X.16.1.1 Configuration-Overall Design

The auxiliary feedwater system (AFWS) for the Turkey Point plant (Units 3 & 4), as shown in Figure 1, consists of three steam turbine driven pumps, i.e., one pump normally aligned to each unit and the third pump is a shared standby for either unit. Each pump normally delivers 600 gpm (@ 2775 ft. head) feedwater to the three steam generators (SG) in each unit. Also, the control room operator can manually direct flow from any pump to all three steam generators of either unit. Under a design basis accident, only one pump would be required in order to cool the plant down to a condition where the RHR system can be put into operation to continue the safe plant shutdown process.

Primary water supply for the AFWS comes from the seismic Category I condensate storage tanks (CST) of both units. Each CST has a capacity of 250,000 gallons with a minimum reserved storage capacity of 185,000 gallons of demineralized water. With this quantity of water, the licensee indicated that the unit can be kept at hot standby condition for 15 hours and then cooled to 350°F, at which point the RHR system can be put in service, or the unit can be kept at hot standby condition for about 23 additional hours. All the manually operated valves associated with CST's are locked open. A secondary water supply comes from the non-seismic Category 1 water treatment system. An additional feedwater supply can be provided from the main feedwater system of the adjacent Units 1 & 2 (non-nuclear power plant).

X.16.1.2 Components - Design, Classification

The AFWS is designed according to seismic Category I requirements. The AFWS is classified as an engineered safety related system and its associated instrumentation and controls are designed accordingly.

X.16.1.3 Power Sources

The turbine driven pumps are supplied with steam from the main steam line of either or both units upstream of the MSIV. The operator normally selects the steam supply from the Unit which has lost its normal feedwater supply. The turbines have an atmosphere exhaust. Steam can also be supplied from the Unit having normal feedwater supply and from an auxiliary steam system connection to Units 1 & 2. The turbine driven pump steam supply line has a normally closed AC motor operated valve in series with a normally closed DC solenoid air operated valve. The pump discharge control valves are DC solenoid operated air valves.

X.16.1.4 Instrumentation and Control

X.16.1.4.1 Controls

The steam generator water level is manually controlled by the control room operator using either one of the DC solenoid operated air valves.

Local manual operation of these valves can be performed on loss of compressed air. The AFW pump feedwater discharge rate is always greater than the turbine steam consumption when the steam pressure is higher than 120 psig. When the steam pressure is reduced to 120 psig, the RHR system is started and the AFW pumps are shut down.

X.16.1.4.2 Information Available to Operator

Low water level in the condensate storage tank will alarm and annunciate in the main control room. In addition, AFW flow indication, SG water level, and control valve position indication are provided in the control room.

X.16.1.4.3 Initiating Signals for Automatic Operation

All three AFW pumps will automatically start by any of the following signals from either Unit:

(a) safety injection

(b) low-low water level in any of the three steam generators

(c) loss of voltage on both 4160V buses

(d) loss of both main feedwater pumps.

Any one of these signals will also automatically open the normally closed motor operated and air operated valves in series which isolate the main steam line from the steam supply header of each AFW pump turbine. Air to operate the AFW control valves to the steam generators is supplied when the steam supply valves commence opening. The AFWS can also be started manually in the control room or from the local station.

X.16.1.5 Testing

The Turkey Point Units 3 and 4 Technical Specifications require the following testing of the auxiliary feedwater system.

1) Monthly test of each auxiliary feedwater pump to run for 15 minutes and verify a flow rate of 600 gpm to the steam generators.
2) Tests of auxiliary feedwater discharge valves during the monthly pump tests.
3) Tests of steam supply and turbine pressure valves during monthly pump tests.

These tests are designed to verify the operability of the auxiliary feedwater system and its ability to respond properly when required.[1]

X.16.1.6 Technical Specifications

The Turkey Point Units 3 and 4 Technical Specifications provide for the following limiting conditions for operation with respect to the Auxiliary Feedwater System:

1) Two out of three AFWS pumps must be operable for single nuclear unit operation.
2) Three out of three AFWS pumps must be operable for dual nuclear unit operation.

[1]The licensee advised that the type of periodic (monthly) testing performed for the AFWS includes full flow path discharge to the SG's, i.e., a single actuation of AFWS and delivery to SG's while power is being produced.

3) The condensate storage tank must contain a minimum of 185,000 gallons of water.

4) System piping, interlocks and valves must be operable.

If any of the above conditions cannot be met within 48 hours, the reactor must be shut down and the reactor coolant temperature must be reduced to less than 350°F.

X.16.2 Reliability Evaluation

X.16.2.1 Dominant Failure Modes

The AFWS simplified flow diagram for Turkey Point Unit 3 is illustrated in Figure 1. This AFWS design reflects a redundant, highly shared, system between Units 3 and 4. Operation of any one of the three steam turbine driven pumps would be expected to result in successful decay heat removal from either Units 3 or 4. Accordingly, the success criterion selected for this reliability evaluation was: Failure of AFWS is insufficient AFWS flow from one AFWS pump to 2 of 3 steam generators in one unit.

The following failure modes were found to dominate the demand unavailability of the Turkey Point AFWS.

LOFW with Offsite AC Available

The Turkey Point AFWS was found to be highly redundant in that there was no obvious single faults (active components, manual valves or human errors) identified that dominate the availability of the AFWS.

The periodic testing practice followed involves full flow path testing to the steam generators. This type of testing is of quality in that it yields an advantage on detectability of valves that might be mispositioned through human errors. Also, the AFWS manual valves are locked open and this practice further reduces the chance of inadvertent closure through human error.

Several unlikely common mode vulnerabilities were identified that might serve to limit the availability of the highly redundant Turkey Point AFWS; their ultimate impact should be further considered in a longer time assessment. These were:

a) The possible common sharing of the lube oil cooling by the service (city) water system which is DC powered.

b) The potential for common disabling of Unit #3 and/or #4 AFWS by a single failure of the connecting piping between the headers in the AFWS pump discharge and steam supply paths.

LOFW with Only Onsite AC Available

The impact of shared emergency diesel generators (EDG) and their contribution to the unavailability of the Turkey Point AFWS were estimated to be very small. The steam admission valves to the turbine pumps are AC operated, but either of the two EDG's operating would suffice to operate at least one or more of the three AC valves in each header in Unit 3 and 4. Further, the human can serve as backup to open these valves if for some reason, the AC or DC valves in either Unit 3 or 4 steam admission header failed to operate electrically.

The dominant faults appear to remain similar to those discussed for the preceeding LOFW transient event.

LOFW with Only DC Available

As noted above the steam admission valves are AC operated in Turkey Point Units #3 and #4. The dominant fault contribution for this event was assessed to be failure of the human to open at least one of the steam admission valves by local manual action. The licensee estimated that such actions could be accomplished within about 10 minutes.

X.16.2.2 Principle Dependencies Identified

One dependency identified was the AC dependency for the steam admission valves that, for the event including complete loss of AC, would require local manual action to initiate the AFWS.

Several additional dependencies were identified that should be considered further, but on a longer term consideration as to their ultimate impact on the AFWS. These were (a) the potential for common lubrication cooling faults in the service (city) water system and (b) the potential for common disabling of the AFWS due to breaks in the single line in the AFWS discharge headers and in the steam supply headers to all turbine driven AFW pump turbines.

X.16.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.16.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time period that one AFW system pump and its associated flow train and essential instrumentation can be inoperable.

   The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are

locked and in the open position. These inspections should be proposed for incorporated into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

3. Recommmendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operators when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

4. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation of flow control is

required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling the lube oil for the turbine-driven pump may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations for manual initiation and control of the AFW system should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer term resolution of this concern).

5. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:
   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

- The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

6. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.
   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
   - Testability of the intiation signals and circuits shall be a feature of the design.
   - The initiation signals and circuits should be powered from the emergency buses.
   - Manual capability to initiate the auxiliary feedwater system from the control room should be retained and

should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
   - The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultanous and/or sequential) of the loads to the emergency buses.
   - The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to intiate the AFW system from the control room.

X.16.3.2 Additional Short Term Recommendations

The following additional short term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins & Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs in W- and C-E-designed operating plants. They have not been examined for specific applicability to ths facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator actions, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain with design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperatures, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technnical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for

operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.16.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

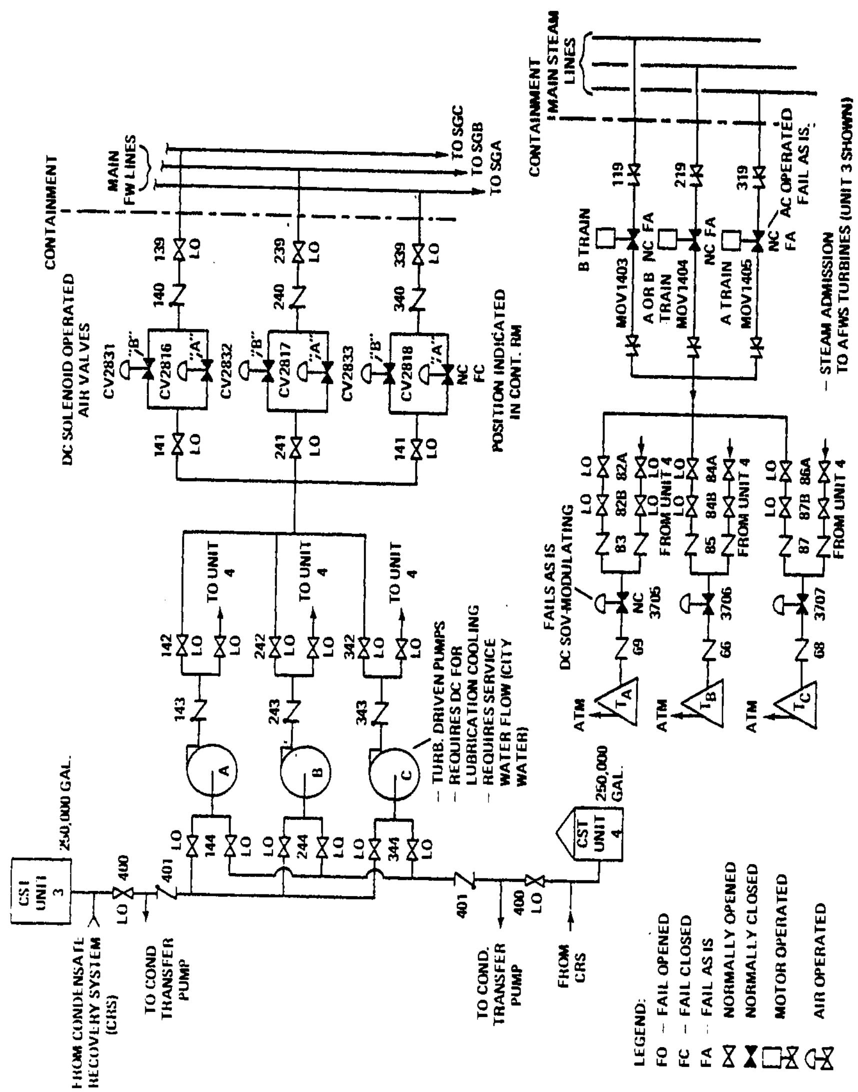
1. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

2. Recommendation - GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

3. Recommendation - The AFW pump discharge lines and turbine driven AFW pump steam supply lines for each unit combine into single lines through which all water and steam respectively from either unit must flow. A pipe break in either of these single flow paths would cause loss of the capability to provide AFW flow to all the steam generators of one unit. The licensee should evaluate the consequences of a

postulated pipe break in these sections of the AFW discharge or steam supply, assuming a concurrent single active failure and 1) determine any AFW system modifications or procedures necessary to detect and isolate the break, and direct the required AFW flow to the steam generators before they boil dry or 2) describe how the plant can be brought to a safe shutdown condition by use of other available systems following such a postulated pipe break.

4. Recommendation - The lube oil cooling of the three turbine driven AFW pumps is provided from a common source; namely the service (city) water system. The licensee should evaluate this cooling water system to determine if there are potential common mode (electrical or mechanical) failures that could disable the lube oil cooling for all three turbine driven pumps. The licensee should provide the results of the evaluation and 1) indicate any system modifications or procedures necessary to prevent a common mode failure of the lube oil cooling system or 2) provide information that demonstrates that the turbine driven AFW pumps can operate for at least two hours without lube oil cooling water and independent of AC power.



Auxiliary Feedwater System
Turkey Point (Units 3 and 4)
Figure 1

ENCLOSURE 2

Basis for Auxiliary Feedwater
System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

   1) Loss of Main Feed (LMFW)
   2) LMFW w/loss of offsite AC power
   3) LMFW w/loss of onsite and offsite AC power
   4) Plant cooldown
   5) Turbine trip with and without bypass
   6) Main steam isolation valve closure
   7) Main feed line break
   8) Main steam line break
   9) Small break LOCA
   10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 9, 1979

Docket No.: 50-029

Mr. Robert H. Groce, Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Dear Mr. Groce:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT YANKEE-ROWE NUCLEAR POWER STATION, UNIT 1

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report on Small

Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Acting Director
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As stated

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ENCLOSURE 1

YANKEE ROWE

AUXILIARY FEEDWATER SYSTEM

X.17.1 System Description

X.17.1.1 Configuration and Overall Design

The auxiliary feedwater system (AFWS) is designed to supply water to the steam generators for reactor coolant system decay heat removal when the normal feedwater system is not available. The AFWS is not normally used for other plant operations such as startup or shutdown. The system can also be used for performing hydrostatic tests during plant shutdown. A dedicated operator for initiating flow for this system is available with direct communication with the control room operator. The auxiliary feedwater (emergency boiler feed pump-EBFP) must be started locally and four normally closed manual valves in parallel in the EBF pump discharge lines must locally opened. After starting the pump, the flow can be controlled from the control room. However, the dedicated operator remains on station even if flow is being controlled from the control room.

The AFWS is shown in simplified form on Figure 1 attached. The system consists of a steam turbine driven positive displacment pump, with steam being supplied from each steam generator into a common header to the pump turbine. Discharge from the pump feeds into a common header which supplies each of four steam generators via the main feedwater piping. Each of the AFWS lines containsa normally closed manual isolation valve. The pump capacity is ≥80 gpm at 1200 psi and takes suction from a 30,000 gallon Demineralized Water Tank (DWT).

A secondary source of water is available from a 135,000 gallon Primary Water Storage Tank (PWST). Water from the 135,000 gallon tank is gravity fed to the 30,000 gallon demineralized water tank by opening one manual valve or directly fed to pump suction by opening a different manual valve. Level indication from the 30,000 gallon and 135,000 gallon tanks are provided in the control room, with high and low level alarms in the control room for the 30,000 gallon tank.

A backup method of supplying feedwater to the steam generators in the event of failure in the AFWS is the plant's primary coolant system charging pumps with total capacity of ~100 gpm (33 gpm/pump). Two of the pumps have variable speed motors. The system is connected permanently by a spool piece that connects to the main feedwater header. The operation of ten manual valves (two drains and eight isolaton) is required to initiate flow from this source. The water supply to the charging pumps is the 135,000 gallon Primary Water Storage Tank.

The High Pressure Safety Injection and Low Pressure Safety Injection pumps provide another backup method of supplying feedwater to the steam generators. Flow from this source is obtained by the operation of the same manual valves used when the charging system is the source,

plus the operation of one of two redundant motor operated valves (MOV). Flow is then directed to the steam generators through the same permanently connected spool piece used for the charging pump path as described above. The flow available from this source is 200 gpm per train (three trains available).

AFW flow is controlled by the normal feedwater control valves in the main feedwater (MFW) lines to the steam generators. The preferred system to be used upon demand is the steam driven turbine pump (AFW) system. The charging pumps or the S.I. pumps are backups to the AFW system. The minimum AFW flow required for decay heat removal is approximately 80 gpm.

The turbine driven pump steam admission valve is a manual valve, which is in the auxiliary steam header. The auxiliary steam header is isolated on receipt of a containment isolation signal by operation of an air operated trip valve. Capability is provided to override the containment isolation signal from the control room. The trip valve also closes on loss of air pressure. An alternate supply of nitrogen is provided (in tanks) in the event of loss of the normal air supply. A number of normally open isolation valves are also located in the header between the admission valve and the trip valve that feeds steam to various steam auxiliary systems.

X.17.1.2 Component Design Classification

The steam piping and primary piping (charging, SI systems) are non-seismic systems, Safety Class 2, classified in accordance with ANSI 18.2, which requires either safety Class 2 or 3 piping. Control and Instrumentation Systems are non-Class IE.

X.17.1.3 Power Sources

Power for the charging pumps and MOVs is supplied from separate nonsafety 480V AC buses, which are capable of being fed by the emergency 480V AC buses by remote manual operation of circuit breakers. The injection pumps are connected to the 480V emergency buses. Offsite power normally feeds the emergency buses. Diesel generators are automatically connected to the emergency 480V AC buses on loss of offsite power.

The plant electrical bus arrangement consists of three independent divisions of 2400V AC buses, one bus fed by one offsite line, a second fed from an independent offsite line, and the third fed from the unit generator. Capability exists to manually transfer from one supply to the other. The three 2400V AC divisions then feed three independent 480V AC through transformers.

The instrumentation and control power is 120V AC from an inverter connected to the 125V DC battery supply.

X.17.1.4 Instrumentation and Controls

X.17.1.4.1 Controls

The water level for each steam generator is controlled manually from the control room by the feedwater controllers that normally are used in the main feed system lines. Steam generator water level can also be controlled locally at the controllers. All MOVs can be remote manually operated from the control room. The charging pumps and SI pumps can be started from the control room.

X.17.1.4.2 Information Available to the Operator

The following indications are available in the control room:

1. Level Indication - 30,000 gallon demineralized water tank
2. Level Indication - 135,000 gallon PWST
3. Flow to steam generators when SI system used
4. Charging pump discharge pressure
5. Steam generator water level
6. Steam generator steam pressure

X.17.1.4.3 Initiating Signals for Automatic Operation

The AFWS initiation is manual. (See section 17.1.1 for manual operation)

X.17.1.5 Testing

Steam Turbine System

The steam turbine is tested every 15 days and operated for 15 minutes. The discharge pressure is monitored to verify rated output (950 psi).

In addition to the operational test, the valve lineup of the system is verified.

The SI system is tested weekly on a staggered basis. The flow is recirculated to the supply tank and pump current is monitored (vibration tests are performed monthly for both the AFWS and SI system). In addition, at the completion of the operational test, valve position of the system is verified.

X.17.1.6 Technical Specifications

The AFWS must be operable or the unit must be in hot standby in one hour and hot shutdown in next 12 hours.

X.17.1.7 Additional Information

The AFW system is manually actuated, however, approximately one hour of steam generator water inventory is available subsequent to loss of feedwater and reactor shutdown.

The offsite power is exceptionally reliable, having experienced only one outage in 19 years of operation.

No challenges to the AFW system have been made during the entire operational history.

X.17.2 Reliability Evaluation Results

X.17.2.1 Dominant Failure Modes

Failure modes of the AFWS were assessed for three types of initiating transients. The dominant failure modes for each transient type are discussed below.

Loss of MFW with Offsite Power Available

The dominant failure mode of the AFWS for this transient results from a set of human errors. The first human error, which causes the unavailability of the AFWS, is the inadvertent closure of one of six manual valves in the steam supply line to the AFWS pump turbine. Upon a demand for the AFWS, the operator has up to an hour to detect this fault and correct it (i.e., open the valve). An alternative for the operator is to manually open the valves from the charging pumps and supply water to the steam generators from these pumps. Thus, the dominant failure mode is the combination of a human error inadvertently closing one of the steam supply line valves and the error of failing to reopen the valve, or realigning the charging pumps, within about one hour after a demand on the AFWS.

Loss of MFW with Only Onsite AC Power Available

AC power dependencies were considered as potential faults for this analysis. It was concluded that the dependence on onsite power instead of offsite power does not significantly alter the results of the assessment. Thus the dominant failure mode of combinations of human errors before and after the transient event is considered to be dominant for this transient also.

Loss of MFW with Only DC Power Available

For this event, the probability of AFWS failure is reduced to the probability of failure of the steam driven pump train. The dominant failure mode within this train is failure to provide steam to the turbine, caused by the inadvertent closure of any 1 of 6 valves in the steam supply line, coupled with failure to reopen the closed valve(s) within approximately one hour after a demand on the AFWS.

X.17.2.2 Principal Dependencies

Within this plant, the principal dependency is the requirement for human actions, such as valve manipulations, to start the AFWS or the backup systems such as the charging pumps or the safety injection pumps. No other dependencies of significance were identified in this evaluation.

X.17.3 Recommendations for this Plant

The short-term recommendations (both generic, denoted by GS, and plant-specific) identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term (both generic, denoted by GL, and plant-specific) recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.17.3.1 Short-Term

1. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer term resolution of this concern.

2. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:

   The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,

   The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

3. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation of flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done, the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiaton and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

4. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.

The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

5. Recommendation - The AFW surveillance tests should require that the normally closed manually operated valves in the connection between the charging pumps/safety injection pumps and the AFW system be cycled each quarter.

X.17.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system design at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator actions, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Postion 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic test on one AFW system train, and there is only one remaining AFW train available for operation should propose Technical Specifications to provide that a dedicated individual who is in communiciation with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would realign the valves in the AFW system train from the test mode to its operational alignment.

X.17.3.3 Long-Term

Long-term recommendations for improving System are as follows:

1. Recommendation - At least one AFW system pump, its associated flow path and essential instrumentation should be capable of being initiated from the control room and being operated independently of any alternating current for at least two hours. Conversion of direct current to alternating current is acceptable.

2. Recommendation - Initiation of AFW flow (including flow from the backup systems-charging/SI) to the steam generators requires several local manual operator actions outside the control room. Even though there is a reasonable time period (up to one hour before the S/G's will boil dry) for operator action and a dedicated operator, the licensee should improve the reliability of initiating AFW flow by providing the capability to start the pumps and open the valves of the AFWS by operator action from the control room. Local manual operation capability should be retained as a backup to remote manual operation capability.

3. Recommendation - A pipe break in the Main Feedwater header upstream of the control valves could cause loss of all AFW flow to all steam generators since the AFW pump and the charging/SI pumps connect to this header. The licensee should evaluate the consequences of a pipe break in this section of the MFW header and 1) determine any system design changes or emergency procedures necessary to detect and isolate the break and direct the required AFW flow to the steam generators before they boil dry or 2) describe how the plant can be brought to a safe shutdown condition by use of other available systems following such a postulated event.

4. Recommendation - The air operated trip valve in the auxiliary steam header which supplies steam to the turbine driven AFW pump closes upon receipt of a containment isolation signal. The

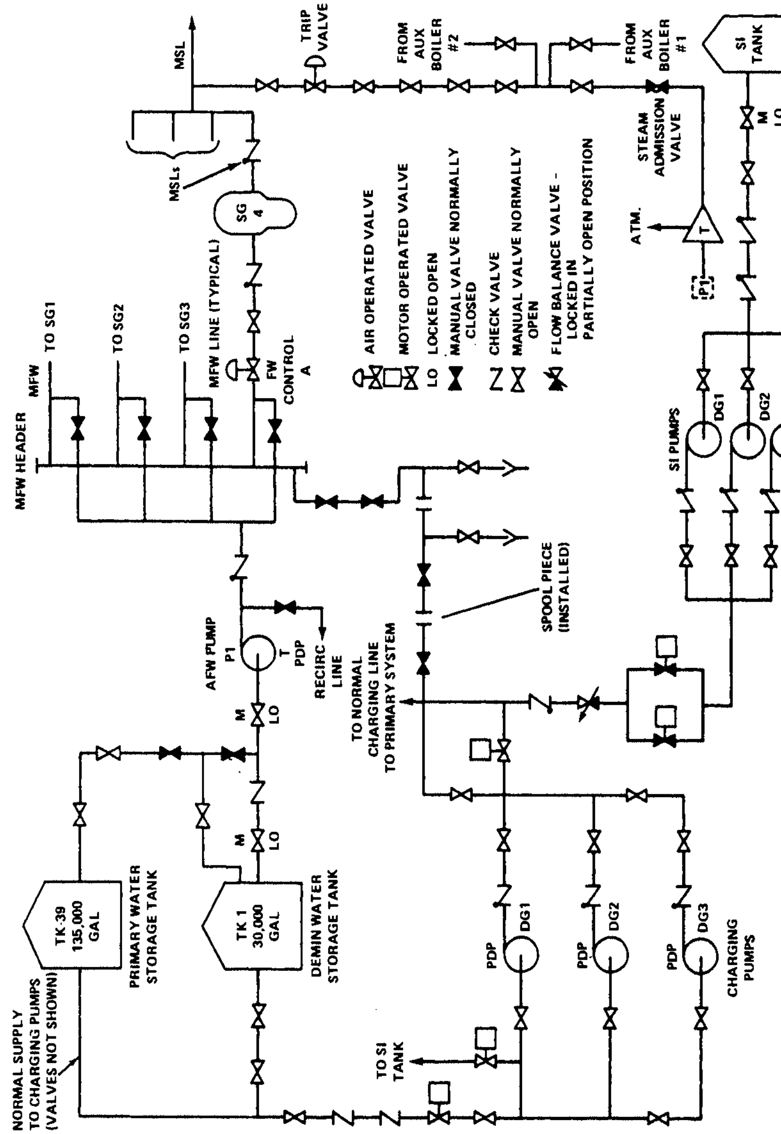
licensee should review the design basis for this circuit logic to determine whether all events that can generate a containment isolation signal should in fact, shutdown the AFWS. As a result of this review, describe any design changes of procedure changes that will be proposed to assure AFW system and containment isolation capability.

5. Recommendation - The licensee should evaluate the need for the charging pumps and associated instruments and control to be normally supplied by the emergency electrical buses since the charging pumps are backups to the one AFW pump.

6. Recommendation - The plant is within the scope of the Systematic Evaluation Program (SEP). The following additional long term concerns have been identified by SEP, and are applicable.

   a. The Yankee Rowe Nuclear Plant including the AFWS will be reevaluated during the SEP with regard to internally and externally generated missiles, pipe whip and jet impingement, quality and seismic design requirements, and earthquakes, tornadoes, and floods.

   b. The Yankee Rowe AFWS is not automatically initiated and the design does not have capability to automatically terminate feedwater flow to a depressurized steam generator and provide flow to the intact steam generator. This is accomplished by manual valve operation,either from the control or locally. The effect of this will be assessed in the main steam line break evaluation for the plant.



Auxiliary Feedwater System
Yankee Rowe
Figure 1

ENCLOSURE 2

Basis for Auxiliary Feedwater
System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
    1) Loss of Main Feed (LMFW)
    2) LMFW w/loss of offsite AC power
    3) LMFW w/loss of onsite and offsite AC power
    4) Plant cooldown
    5) Turbine trip with and without bypass
    6) Main steam isolation valve closure
    7) Main feed line break
    8) Main steam line break
    9) Small break LOCA
    10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

    - Maximum RCS pressure (PORV or safety valve actuation)
    - Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
    - RCS cooling rate limit to avoid excessive coolant shrinkage
    - Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:
   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.
   b. Time delay from initiating event to reactor trip.
   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).
   d. Minimum steam generator water level when initiating event occurs.
   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

h. RC flow condition - continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 18, 1979

Docket Nos.: 50-245
50-304

Mr. Cordell Reed, Assistant Vice-President
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. Reed:

SUBJECT: NRC REQUIREMENTS FOR AUXILIARY FEEDWATER SYSTEMS AT ZION STATION, UNITS 1 AND 2

The purpose of this letter is to advise you of our requirements for the auxiliary feedwater systems at the subject facility. These requirements were identified during the course of the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Enclosure 1 to this letter identifies each of the requirements applicable to the subject facility. These requirements are of two types, (1) generic requirements applicable to most Westinghouse-designed operating plants, and (2) plant-specific requirements applicable only to the subject facility. Enclosure 2 contains a generic request for additional information regarding auxiliary feedwater system flow requirements.

The designs and procedures of the subject facility should be evaluated against the applicable requirements specified in Enclosure 1 to determine the degree to which the facility currently conforms to these requirements. The results of this evaluation and an associated schedule and commitment for implementation of required changes or actions should be provided for NRC staff review within thirty days of receipt of this letter. Also, this schedule should indicate your date for submittal of information such as design changes, procedure changes or Technical Specification changes to be provided for staff review. You may also provide your response to the items in Enclosure 2 at that time.

In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletins and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident as described in the Westinghouse report WCAP-9600, "Report

on Small Break Accidents for Westinghouse NSSS System." Our specific concerns include systems reliability (other than the auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training.

We plan to identify, in separate correspondence, the requirements resulting from the additional items from the Bulletins and Orders Task Force review.

Sincerely,

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
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ENCLOSURE 1

ZION UNITS 1 AND 2
AUXILIARY FEEDWATER SYSTEM

X.18.1 System Description

X.18.1.1 Configuration, Overall Design

A simplified drawing of the Zion Auxiliary Feedwater (AFW) System is shown in Figure X.18-1. The design configuration is the same for both units. The system consists of two motor-driven pumps, each with a capacity of 450 gpm at 3099 feet head and one steam turbine driven pump with a capacity of 900 gpm at 3099 feet head. The motor-driven pumps feed a common header, which in turn feeds all four steam generators through motor-operated throttle valves that limit the flow to 105 gpm per steam generator. The turbine-driven pump feeds a common header, which feeds all steam generators through motor-operated throttle valves, which limit the flow to 105 gpm per steam generator. The normal auxiliary feedwater supply is the condensate storage tank, which supplies a common header that feeds all three pumps. There are cross-connections between all pump discharge trains, as shown in Figure X.18-1 with the cross-connection between the motor-driven pump and turbine-driven pump trains normally closed; all other valves are in a normally open-fail open or fail-as is position, as shown in the figure.

The AFW system was evaluated for high energy line breaks, with and without a single active failure, in the main steam system, main feedwater system and AFW system. In all cases considered, auxiliary feedwater can be put into the steam generator within 20 minutes of the break or indication of a pipe break, through one of the unaffected trains. There was only one place in the pipe chase where a break in a auxiliary feedwater line might damage the other auxiliary feedwater line. However, since the lines are the same size, this would not cause a break in the other line. Consequently, in case of a break in an auxiliary feedwater line, water can still be supplied to the steam generator(s). (Section 18.2.2 discusses an additional pipe break concern).

Source of Water

The auxiliary feedwater system has five sources of water. The primary source of water is the non-seismic condensate storage tank for the unit. This tank has a capacity of 500,000 gallons of which 170,000 gallons are administratively dedicated to the auxiliary feedwater system. These 170,000 gallons will provide a minimum of eight hours of water to the steam generator,with the steam being dumped to atmosphere. Valves in the supply line to the AFW system are normally open. The tank is not tornado missile protected, but the lines are buried, so some protection from tornado missiles is provided.

The secondary source of water is the condensate storage tank for the other unit. The cross connection has a normally closed manually operated valve, but there are other means of transferring water from tank to tank, if needed.

The backup source (and one of the long-term sources of water) is the seismic Category I service water system, which can supply both units. The source of service water is Lake Michigan. A common header from the service water system serves all pumps. In order to put the system into operation as a backup for the AFW system, motor-operated valves have to be opened. There is a manually-operated cross tie with the other unit's service water system. There is the possibility for minor flooding in the auxiliary building when the service water system is used for long-term cooling. There are some vents in the service water system which will

discharge about 20 gpm,if they are not closed when the service water system is used to supply the AFW system.

The other long-term source of water is the makeup demineralizer system, which is not designed to seismic Category I requirements. Although this system feeds directly to the condensate storage tanks at a maximum rate of 1200 gpm, it would take between 10-30 minutes to put the system in operation manually. The water source for this system is also Lake Michigan.

X.18.1.2 Components - Design, Classification

Table X.18-1

	Environmentally Qualified (line break)	Design Classification	Seismic Category
Steam Driven Pump and Turbine	Yes	1	I
Motor Driven Pumps and Motors	Yes	1	I
Piping - In Auxiliary Building	Yes	1	I
from Condensate Storage Tank to Auxiliary Building		3	N.S.*
Valves -	Yes	Same as piping	
Power Supplies	Yes	1	I
Instrumentation	No	3	N.S.*
Controls	Yes	1	I
Condensate Storage Tank	-	3	N.S.*
Service Water System	-	1	I
	Environmentally Qualified (line break)	**Design Classification**	**Seismic Category**
Make-up Demineralization System	-	3	N.S.*
Main Steam Supply for Turbine Driven Pump including Valves	-	1	I

*N.S. (non-seismic) - not designed to meet seismic Category I requirements.

X.18.1.3 Power Sources

The motor-driven pumps are powered from separate emergency diesel-generators; i.e., one pump to each diesel-generator. The eight motor-operated throttle valves take power from the diesels. Two out of four throttle valves for the motor-driven pump trains are run off one diesel-generator. The other two are on another diesel-generator. The four on the turbine-driven pump train are divided the same way. All motor-operated valves are in the open position and fail as-is.

The steam turbine-driven pump takes its steam from steam generators A and D through a common header, a normally open, motor operated, fail-as-is valve, and an air-operated, fail-open, control valve. The solenoid for the air-operated valve is powered from the direct current buses.

The alternating current power for the instrumentation is derived from the direct current buses, which take power from the onsite alternating current system, or from the station batteries through an inverter.

The power for the service water pumps comes from the alternating current power system. Since the motor-driven and turbine-driven pumps have lube oil coolers, the cooling from these pumps comes from the service water system. If the plant were to experience a station blackout (loss of offsite and onsite alternating current power), the licensee estimates that the turbine-driven pump would last a minimum of 15 minutes without cooling and the steam generators would boil dry approximately 30 minutes after the auxiliary feedwater pump stopped. We believe that this condition is unacceptable, and adequate lube oil cooling for the turbine-driven pump should be provided which is independent of alternating current. (See Recommendations).

X.18.1.4 Instrumentation and Controls

X.18.1.4.1 Controls

The AFW system is an entirely automatic system, except for the switching of the sources of water. This switching must be done either in the control room or locally, depending on the source. The manual controls located in the control room are the pump on-off switch, valves open-close switches, and the throttle valve control switch, which is used for steam generator level control. The controls at the remote shutdown panel and local AFW operating station are the same as those in the control room, except for the lube oil pump controls, which are only located at the remote shutdown panel which is also the local AFW operating station.

X.18.1.4.2 Information Available to Operator

The indication available to the operator is as follows:

1. Alarms (Control Room Only)
   a. Low suction pressure (common all pumps);
   b. Auxiliary feed pump not available (common all pumps);
   c. Condensate tank low and high level;
   d. Low lube oil pressure trip (common all pumps);

2. Indicators (Control Room and Remote Shutdown Panel, except as noted)
   a. AFW flow to each generator;
   b. Condensate storage tank level; (Control Room and Rad Waste Panel)
   c. AFW pump running;
   d. Valve indications (open-close);
   e. AFW pump turbine inlet steam pressure; (Control Room only)
   f. AFW pump discharge pressure;
   g. AFW pump motor current.

X.18.1.4.3 Initiating Signals for Automatic Operation

The AFW system flow is automatically initiated on any of the following signals:

1. Steam Generator low-low level (10% narrow range)
   a. Motor-driven pumps start on one steam generator low-low level;
   b. Turbine-driven pump starts on two steam generators low-low level.
2. Loss of offsite power;
3. Safety injection signal;

The AFW system can also be initiated by the operator from the control room and/or the Remote Shutdown Panel.

X.18.1.5 Testing

The AFW system is tested on a monthly basis, one train at a time. The procedure tests all components of the system. It is composed of closing the throttle valves for the train being tested, and running the system in the recirculating mode. The throttle valves are then opened to allow a flow of 105 gpm to the steam generators. All valves, including the service water system valves and the throttle valves, are stroked monthly. At the same time the monthly tests are being performed, a vibration test is performed on the pumps.

When a train is being brought back into service after a maintenance outage, the above tests are performed, except that the vibration test is deleted.

X.18.1.6 Technical Specifications

Table X.18-1 details the limiting conditions for operation and surveillance requirements for the Zion Station AFW system. Problem areas identified during our review are that (1) one pump can be out of service indefinitely with no action taken, and (2) the licensee does not consider the instrumentation as part of the system. Thus the instrumentation for one or more pumps could be out of service while the plant is allowed to operate. (See Recommendation).

X.18.2 Reliability Evaluation

X.18.2.1 Dominant Failure Modes

The Zion AFW system was analyzed for the following cases:

a. Loss of main feedwater with offsite power available;

b. Loss of main feedwater with onsite alternating current power available;

c. Loss of main feedwater with only direct current power available.

The results of the analysis are summarized below.

ZION TECHNICAL SPECIFICATIONS

LIMITING CONDITION FOR OPERATION

Auxiliary feedwater pump system

A. Two of the three auxiliary feedwater pump systems shall be operable whenever the reactor is going from cold shutdown to hot standby.

B. Two of the three auxiliary feedwater pump systems shall be operable whenever the reactor is in hot standby or operating except as specified in 3.7.2.C.

C. From and after the date that two of the three auxiliary feedwater pump systems are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days provided that during these 7 days the remaining auxiliary feedwater pump system shall be operable.

D If these conditions cannot be met the reactor shall be brought to the hot shutdown condition within four hours. After a maximum of 48 hours in the hot shutdown condition, if the system is not operable the reactor shall be brought to the cold shutdown condition within 24 hours.

SURVEILLANCE REQUIREMENT

Auxiliary feedwater pump system
(Table 4.7-2)

A. Surveillance and testing of the auxiliary feedwater pump systems shall be performed as follows:

   1. The auxiliary feedwater pumps shall be started manually from the control room each month. Performance will be acceptable if the pump starts upon actuation, operates for at least 10 minutes on recirculation flow, and the discharge pressure and flow are within ±10% of a point on the pump head curve.

   2. The service water power operated supply valves to the auxiliary feedwater pumps shall be stroked manually from the control room each month. Performance will be acceptable if valve motion is indicated upon actuation.

B. Not Applicable.

C. When it is determined that two of three auxiliary feedwater pump systems are inoperable, the one remaining system shall be started immediately and daily thereafter including the associated standby AC and DC power supplies (See Section 4.15.1.B.2 and 4.15.1.E.1).

D. Not Applicable.

Loss of Main Feedwater with Offsite Power Available

The unavailability of the AFW system for this case is dominated by the following failures:

a. maintenance being performed on the second manual valve in the supply line from the condensate storage tank + failure to manually initiate service water to supply the system upon demand, and,

b. turbine-driven pump train out for maintenance over an extended period of time + testing of one of the motor-driven pumps.

The first of these failures disables the supply water to the AFW system. The second disables AFW system discharge to the steam generators in the following manner. Maintenance on the turbine-driven pump train results in four lines to the steam generator not receiving flow; testing for the motor-driven pump train requires closing of the motor-operated valves in the other four lines to the steam generators, thus closing off all discharge flow to the steam generators for the period of the test.

Loss of Main Feedwater with Onsite Alternating Current Power Available

The AFW system was analyzed assuming loss of offsite power, considering possible loss of one of the diesel generators. Failure of a diesel-generator has no significant effect on system reliability. The dominant failure modes appear to be similar to those discussed in the previous case.

Loss of Main Feedwater with Only Direct Current Available

For this case, neither offsite nor onsite alternating current power are available. The system may not successfully operate for an extended time period without alternating current power due to the fact that the turbine-driven pump lube oil is dependent upon the alternating current

powered service water system for cooling. Without alternating current power, lube oil cooling is lost, which could result in failure of the only operable AFW pump.

In the short-term ($\leq$ 45 minutes), the turbine-driven pump train unavailability is dominated by maintenance, test, and single hardware failures.

X.18.2.2 Dependencies

In addition to the dependence of the turbine-driven pump train on alternating current power discussed above, one locational dependence was identified. The two motor-driven pumps and their associated motor-operated valves on the suction side are in a common location. There is only a short barrier (about six feet tall) between this cell and the cell containing the turbine-driven pump and its suction motor-operated valve. Thus, there are potential location-dependent interactions in this system. A high energy line (steam line to the turbine-driven pump) passes through this space. Although the pumps are qualified for a steam environment, it does not appear that the motor-operated valves are (See Recommendations).

X.18.3 Recommendations for this Plant

The short-term recommendations identified in this section represent actions to improve AFW system reliability that should be implemented by January 1, 1980, or as soon thereafter as is practicable. In general, they involve upgrading of Technical Specifications or establishing procedures to avoid or mitigate potential system or operator failures. The long-term recommendations identified in this section involve system design evaluations and/or modifications to improve AFW system reliability and represent actions that should be implemented by January 1, 1981, or as soon thereafter as is practicable.

X.18.3.1 Short-Term

1. Recommendation GS-1 - The licensee should propose modifications to the Technical Specifications to limit the time that one AFW system pump and its associated flow train and essential instrumention can be inoperable. The outage time limit and subsequent action time should be as required in current Standard Technical Specifications; i.e., 72 hours and 12 hours, respectively.

2. Recommendation GS-2 - The licensee should lock open single valves or multiple valves in series in the AFW system pump suction piping and lock open other single valves or multiple valves in series that could interrupt all AFW flow. Monthly inspections should be performed to verify that these valves are locked and in the open position. These inspections should be proposed for incorporation into the surveillance requirements of the plant Technical Specifications. See Recommendation GL-2 for the longer-term resolution of this concern.

3. Recommendation GS-3 - The licensee has stated that it throttles AFW system flow to avoid water hammer. The licensee should reexamine the practice of throttling AFW system flow to avoid water hammer.

   The licensee should verify that the AFW system will supply on demand sufficient initial flow to the necessary steam generators to assure adequate decay heat removal following loss of main feedwater flow and a reactor trip from 100% power. In cases where this reevaluation results in an increase in initial AFW system flow, the licensee should provide sufficient information to demonstrate that the required initial AFW system flow will not result in plant damage due to water hammer.

4. Recommendation GS-4 - Emergency procedures for transferring to alternate sources of AFW supply should be available to the plant operators. These procedures should include criteria to inform the operator when, and in what order, the transfer to alternate water sources should take place. The following cases should be covered by the procedures:
   - The case in which the primary water supply is not initially available. The procedures for this case should include any operator actions required to protect the AFW system pumps against self-damage before water flow is initiated; and,
   - The case in which the primary water supply is being depleted. The procedure for this case should provide for transfer to the alternate water sources prior to draining of the primary water supply.

5. Recommendation GS-5 - The as-built plant should be capable of providing the required AFW flow for at least two hours from one AFW pump train independent of any alternating current power source. If manual AFW system initiation or flow control is required following a complete loss of alternating current power, emergency procedures should be established for manually initiating and controlling the system under these conditions. Since the water for cooling of the lube oil for the turbine-driven pump bearings may be dependent on alternating current power, design or procedural changes shall be made to eliminate this dependency as soon as practicable. Until this is done,

the emergency procedures should provide for an individual to be stationed at the turbine-driven pump in the event of the loss of all alternating current power to monitor pump bearing and/or lube oil temperatures. If necessary, this operator would operate the turbine-driven pump in an on-off mode until alternating current power is restored. Adequate lighting powered by direct current power sources and communications at local stations should also be provided if manual initiation and control of the AFW system is needed. (See Recommendation GL-3 for the longer-term resolution of this concern.)

6. Recommendation GS-6 - The licensee should confirm flow path availability of an AFW system flow train that has been out of service to perform periodic testing or maintenance as follows:

   - Procedures should be implemented to require an operator to determine that the AFW system valves are properly aligned and a second operator to independently verify that the valves are properly aligned.
   - The licensee should propose Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary AFW system water source to the steam generators. The flow test should be conducted with AFW system valves in their normal alignment.

7. Recommendation GS-7 - The licensee should verify that the automatic start AFW system signals and associated circuitry are safety-grade. If this cannot be verified, the AFW system automatic initiation system should be modified in the short-term to meet the functional requirements listed below. For the longer term, the automatic initiation signals and circuits should be upgraded to meet safety-grade requirements as indicated in Recommendation GL-5.

   - The design should provide for the automatic initiation of the auxiliary feedwater system flow.
   - The automatic initiation signals and circuits should be designed so that a single failure will not result in the loss of auxiliary feedwater system function.
   - Testability of the initiation signals and circuits shall be be a feature of the design.
   - The initiation signals and circuits should be powered from the emergency buses.
   - Manual capability to initiate the auxiliary feedwater system from the control room should be retained and should be implemented so that a single failure in the manual circuits will not result in the loss of system function.
   - The alternating current motor-driven pumps and valves in the auxiliary feedwater system should be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.
   - The automatic initiation signals and circuits shall be designed so that their failure will not result in the loss of manual capability to intiate the AFW system from the control room.

8. Recommendation GS-8 - The licensee should propose modifications to the Technical Specifications to include a Limiting Condition of Operation when the condensate storage tank level falls below the 170,000 gallon level, considering both one- and two-unit operation.

X.18.3.2 Additional Short-Term Recommendations

The following additional short-term recommendations resulted from the staff's Lessons Learned Task Force review and the Bulletins and Orders Task Force review of AFW systems at Babcock & Wilcox-designed operating plants subsequent to our review of the AFW system designs at W- and C-E-designed operating plants. They have not been examined for specific applicability to this facility.

1. Recommendation - The licensee should provide redundant level indications and low level alarms in the control room for the AFW system primary water supply to allow the operator to anticipate the need to make up water or transfer to an alternate water supply and prevent a low pump suction pressure condition from occurring. The low level alarm setpoint should allow at least 20 minutes for operator action, assuming that the largest capacity AFW pump is operating.

2. Recommendation - The licensee should perform a 72-hour endurance test on all AFW system pumps, if such a test or continuous period of operation has not been accomplished to date. Following the 72-hour pump run, the pumps should be shut down and cooled down and then restarted and run for one hour. Test acceptance criteria should include demonstrating that the pumps remain within design limits with respect to bearing/bearing oil temperatures and vibration and that pump room ambient conditions (temperature, humidity) do not exceed environmental qualification limits for safety-related equipment in the room.

3. Recommendation - The licensee should implement the following requirements as specified by Item 2.1.7.b on page A-32 of NUREG-0578:

   "Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

   The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements for the auxiliary feedwater system set forth in Auxiliary Systems Branch Technnical Position 10-1 of the Standard Review Plan, Section 10.4.9."

4. Recommendation - Licensees with plants which require local manual realignment of valves to conduct periodic tests on one AFW system train and which have only one remaining AFW train available for operation, should propose Technical Specifications to provide that a dedicated individual who is in communication with the control room be stationed at the manual valves. Upon instruction from the control room, this operator would re-align the valves in the AFW system train from the test mode to its operational alignment.

X.18.3.3 Long-Term

Long-term recommendations for improving the system are as follows:

1. Recommendation GL-2 - Licensees with plants in which all (primary and alternate) water supplies to the AFW systems pass through valves in a single flow path should install redundant parallel flow paths (piping and valves).

   Licensees with plants in which the primary AFW system water supply passes through valves in a single flow path, but the alternate AFW system water supplies connect to the AFW system pump suction piping downstream of the above valve(s), should install redundant valves parallel to the above valve(s) or provide automatic opening of the valve(s) from the alternate water supply upon low pump suction pressure.

   The licensee should propose Technical Specifications to incorporate appropriate periodic inspections to verify the valve positions.

2. Recommendation GL-3 - At least one AFW system pump and its associated flow path and essential instrumentation should automatically initiate AFW system flow and be capable of being operated independently of any alternating current power source for at least two hours. Conversion of direct current power to alternating current is acceptable.

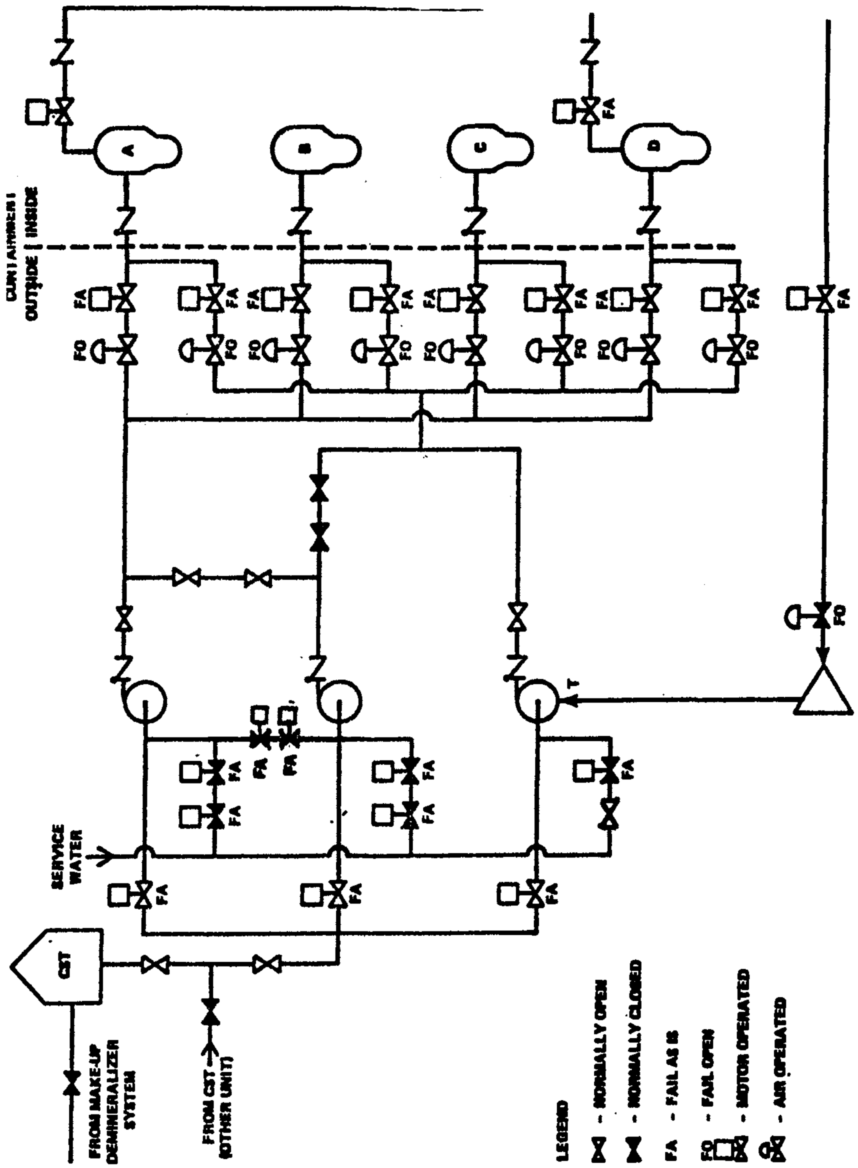
3. Recommendation - GL-4 - Licensees having plants with unprotected normal AFW system water supplies should evaluate the design of their AFW systems to determine if automatic protection of the pumps is necessary following a seismic event or a tornado. The time available before pump damage, the alarms and indications available to the control room operator, and the time necessary for assessing the problem and taking action should be considered in determining whether operator action can be relied on to prevent pump damage. Consideration should be given to providing pump protection by means such as automatic switchover of the pump suctions to the alternate safety-grade source of water, automatic pump trips on low suction pressure or upgrading the normal source of water to meet seismic Category I and tornado protection requirements.

4. Recommendation- GL-5 - The licensee should upgrade the AFW system automatic initiation signals and circuits to meet safety-grade requirements.

5. Recommendation - The licensee should evaluate the consequences of a postulated break in the steam line to the turbine-driven AFW pump to determine the need to qualify the AFW system valves, valve actuators, and instrumentation for the environmental conditions resulting from such a high energy line break in order to maintain operability of the motor-driven AFW pumps and their associated flow trains.

6. Recommendation - There are no valves in either of the common headers supplied from the motor-driven AFW pumps or the turbine-driven pump to all four steam generators. A pipe break in either header could cause loss of all AFW flow to all steam generators from either

the motor-driven or turbine-driven AFW pumps. The licensee should evaluate a postulated pipe break in either header and indicate the AFW system design changes or emergency procedures necessary to detect and isolate the break and direct the required AFW flow to the steam generators before they boil dry.



Auxiliary Feedwater System
Zion
Figure X.1R-1

ENCLOSURE 2

Basis for Auxiliary Feedwater System Flow Requirements

As a result of recent staff reviews of operating plant Auxiliary Feedwater Systems (AFWS), the staff concludes that the design bases and criteria provided by licensees for establishing AFWS requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

We require that you provide the following AFWS flow design basis information as applicable to the design basis transients and accident conditions for your plant.

1. a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:
    1) Loss of Main Feed (LMFW)
    2) LMFW w/loss of offsite AC power
    3) LMFW w/loss of onsite and offsite AC power
    4) Plant cooldown
    5) Turbine trip with and without bypass
    6) Main steam isolation valve closure
    7) Main feed line break
    8) Main steam line break
    9) Small break LOCA
    10) Other transient or accident conditions not listed above

   b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

   g. Minimum number of steam generators that must receive AFW flow; e.g. 1 out of 2?, 2 out of 4?

   h. RC flow condition - continued operation of RC pumps or natural circulation.

   i. Maximum AFW inlet temperature.

   j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

   k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

   l. Operating condition of steam generator normal blowdown following initiating event.

   m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

   n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

- Maximum RCS pressure (PORV or safety valve actuation)
- Fuel temperature or damage limits (DNB, PCT, maximum fuel central temperature)
- RCS cooling rate limit to avoid excessive coolant shrinkage
- Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

2. Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a. above including:

   a. Maximum reactor power (including instrument error allowance) at the time of the initiating transient or accident.

   b. Time delay from initiating event to reactor trip.

   c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and introduction of AFWS flow into steam generator(s).

   d. Minimum steam generator water level when initiating event occurs.

   e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences - identify reactor decay heat rate used.

   f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

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3. Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

F

APPENDIX F

LETTER TO LICENSEES OF ALL OPERATING REACTORS,

DATED OCTOBER 30, 1979

CONCERNING SHORT-TERM LESSONS LEARNED REQUIREMENTS

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

October 30, 1979

(TO ALL OPERATING NUCLEAR POWER PLANTS)

Gentlemen:

SUBJECT: DISCUSSION OF LESSONS LEARNED SHORT TERM REQUIREMENTS

On September 13, 1979, a letter was issued to each power reactor licensee which defined a set of "short term" requirements resulting from the NRC staff investigations of the TMI accident. Since the letter was issued, the staff has attempted to further define these requirements. During the week of September 24, 1979, seminars were held in four regions of the country to encourage industry feedback and dialogue on each short term requirement. As a result of these discussions, four topical meetings were held in Bethesda to discuss certain issues in further detail.

Enclosure 1 provides additional clarification of the NRC staff requirements. It should be noted that the intent of these requirements have not changed throughout this process and are restated in Enclosure 1.

Enclosure 2 is a chart of the NUREG-0578 items and their corresponding implementation schedules. The chart indicates which of the items require prior NRC review and approval and those for which post implementation NRC review is acceptable.

For those items requiring prior NRC approval, your design details should be submitted in a timely manner so that this approval and your implementation of the item can be completed by the required date. For those items which do not require prior NRC approval, you must document your method of implementation by the required completion date. These schedules assume that your methods are in complete agreement with the staff's requirements as previously documented in NUREG-0578, our September 13, 1979 letter, and clarified herein. Where your methods are not in complete agreement with the staff's requirements, a detailed description of your proposed methods along with justification for the differences, is required. Please provide this description and justification as soon as possible but no later than 15 days following receipt of this letter.

The schedule for completing each of the short term TMI followup requirements is firm. Some licensees, in responding to our September 13, 1979 letter have indicated an inability to meet the established implementation schedule. If your response was in this category you are requested to reconsider your implementation schedule with the purpose of improving your implementation dates to meet those required by the staff. Within fifteen days from receipt of this letter, you are requested to submit your revised schedule for implementation. If you are unable to commit to meeting any of the January 1, 1980 requirements, you must provide, for each item, a report on the degree of compliance expected on January 1, 1980, and a detailed justification for the delay.

Sincerely,

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Discussion of TMI Lessons Learned Short Term Requirements
2. Implementation Schedule

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(continued)

Enclosure 1

DISCUSSION OF

TMI LESSIONS LEARNED

SHORT TERM REQUIREMENTS

EMERGENCY POWER SUPPLY (2.1.1)

Pressurizer Heaters

POSITION

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

Pressurizer Heater Power Supply

1. The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

2. Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

3. The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

4. Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

CLARIFICATION

1. In order not to compromise independence between the sources of emergency power and still provide redundant capability to provide emergency power to the pressurizer heaters, each redundant heater or group of heaters should have access to only one Class 1E division power supply.

2. The number of heaters required to have access to each emergency power source is that number required to maintain natural circulation in the hot standby condition.

3. The power sources need not necessarily have the capacity to provide power to the heaters concurrent with the loads required for LOCA.

4. Any change-over of the heaters from normal offsite power to emergency onsite power is to be accomplished manually in the control room.

5. In establishing procedures to manually reload the pressurizer heaters onto the emergency power sources, careful consideration must be given to:

   a. Which ESF loads may be appropriately shed for a given situtation.
   b. Reset of the Safety Injection Actuation Signal to permit the operation of the heaters.
   c. Instrumentation and criteria for operator use to prevent overloading a diesel generator.

6. The Class IE interfaces for main power and control power are to be protected by safety-grade circuit breakers. (See also Reg. Guide 1.75)

7. Being non-Class IE loads, the pressurizer heaters must be automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal. (See item 5.b. above)

Emergency Power Supply (2.1.1)

Pressurizer Level and Relief Block Valves

POSITION

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15 17 and 20 of Aprendix A to 10 CFR Part 50 for the event of loss of offsite power, the following positions shall be implemented:

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

1. Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being suppied from either the offsite power source or the emergency power source when offsite power is not available.

CLARIFICATION

1. While the prevalent consideration from TMI Lessons Learned is being able to close the PORV/block valves, the design should retain, to the extent practical, the capability to open these valves.

2. The motive and control power for the block valve should be supplied from an emergency power bus different from that which supplies the PORV.

3. Any changeover of the PORV and block valve motive and control power from the normal offsite power to the emergency onsite power is to be accomplished manually in the control room.

4. For those designs where instrument air is needed for operation, the electrical power supply requirement should be capable of being manually connected to the emergency power sources.

PERFORMANCE TESTING FOR BWR AND PWR RELIEF AND SAFETY VALVES (2.1.2)

POSITION

Pressurized Water Reactor and Boiling Water Reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

CLARIFICATION

1. Expected operating conditions can be determined through the use of analysis of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70.

2. This testing is intended to demonstrate valve operability under various flow conditions, that is, the ability of the valve to open and shut under the various flow conditions should be demonstrated.

3. Not all valves on all plants are required to be tested. The valve testing may be conducted on a prototypical basis.

4. The effect of piping on valve operability should be included in the test conditions. Not every piping configuration is required to be tested, but the configurations that are tested should produce the appropriate feedback effects as seen by the relief or safety valve.

5. Test data should include data that would permit an evaluation of discharge piping and supports if those components are not tested directly.

6. A description of the test program and the schedule for testing should be submitted by January 1, 1980.

7. Testing shall be complete by July 1, 1981.

DIRECT INDICATION OF POWER-OPERATED RELIEF VALVE AND SAFETY VALVE POSITION FOR PWRs AND BWRs (2.1.3.a)

POSITION

Reactor System relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

CLARIFICATION

1. The basic requirement is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

2. The valve position should be indicated in the control room. An alarm should be provided in conjunction with this indication.

3. The valve position indication may be safety grade. If the position indication is not safety grade, a reliable single channel direct indication powered from a vital instrument bus may be provided if backup methods of determining valve position are available and are discussed in the emergency procedures as an aid to operator diagnosis and action.

4. The valve position indication should be seismically qualified consistent with the component or system to which it is attached. If the seismic qualification requirements cannot be met feasibly by January 1, 1980, a justification should be provided for less than seismic qualification and a schedule should be submitted for upgrade to the required seismic qualificiation.

5. The position indication should be qualified for its appropriate environment (any transient or accident which would cause the relief or safety valve to lift). If the environmental qualification program for this position indication will not be completed by January 1, 1980, a proposed schedule for completion of the environmental qualification program should be provided.

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

SUBCOOLING METER

POSITION

Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation " (see Section 2.1.9 of NUREG-0578)

In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

CLARIFICATION

1. The analysis and procedures addressed in paragraph one above will reviewed and should be submitted to the NRC "Bulletins and Orders Task Force" for review.

2. The purpose of the subcooling meter is to provide a continuous indication of margin to saturated conditions. This is an important diagnostic tool for the reactor operators.

3. Redundant safety grade temperature input from each hot leg (or use of multiple core exit in T/C's) are required.

4. Redundant safety grade system pressure measures should be provided.

5. Continuous display of the primary coolant saturation conditions should be provided.

6. Each PWR should have: (A.) Safety grade calculational devices and display (minimum of two meters) or (B.) a highly reliable single channel environmentally qualified, and testable system plus a backup procedure for use of steam tables. If the plant computer is to be used, its availability must be documented.

7. In the long term, the instrumentation qualifications must be required to be upgraded to meet the requirements of Regulatory Guide 1.97 (Instrumentation for Light Water Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident) which is under development.

8. In all cases appropriate steps (electrical, isolation, etc.) must be taken to assure that the addition of the subcooling meter does not adversely impact the reactor protection or engineered safety features systems.

9. The attachment provides a definition of information required on the subcooling meter.

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.) ________________

Display Type (Analog, Digital, CRT) ________________

Continuous or on Demand ________________

Single or Redundant Display ________________

Location of Display ________________

Alarms (include setpoints) ________________

Overall uncertainty (°F, PSI) ________________

Range of Display ________________

Qualifications (seismic, environmental, IEEE323) ________________

Calculator

Type (process computer, dedicated digital or analog calc.) ________________

If process computer is used specify availability. (% of time) ________________

Single or redundant calculators ________________

Selection Logic (highest T., lowest press) ________________

Qualifications (seismic, environmental, IEEE323) ________________

Calculational Technique (Steam Tables, Functional Fit, ranges) ________________

Input

Temperature (RTD's or T/C's) ________________

Temperature (number of sensors and locations) ________________

Range of temperature sensors ________________

Uncertainty* of temperature sensors (°F at 1)	________
Qualifications (seismic, environmental, IEEE323)	________
Pressure (specify instrument used)	________
Pressure (number of sensors and locations)	________
Range of Pressure sensors	________
Uncertainty* of pressure sensors (PSI at 1)	________
Qualifications (seismic, environmental, IEEE323)	________

<u>Backup Capability</u>

Availability of Temp & Press	________
Availability of Steam Tables etc.	________
Training of operators	________
Procedures	________

*Uncertainties must address conditions of forced flow and natural circulation

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

ADDITIONAL INSTRUMENTATION

POSITION

Licensees shall provide a decription of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

CLARIFICATION

1. Design of new instrumentation should provide an unambiguous indication of inadequate core cooling. This may require new measurements to or a synthesis of existing measurements which meet safety-grade criteria.

2. The evaluation is to include reactor water level indication.

3. A commitment to provide the necessary analysis and to study advantages of various instruments to monitor water level and core cooling is required in the response to the September 13, 1979 letter.

4. The indication of inadequate core cooling must be unambiguous, in that, it should have the following properties:

   a) it must indicate the existence of inadequate core cooling caused by various phenomena (i.e., high void fraction pumped flow as well as stagnant boil off).

   b) it must not erroneously indicate inadequate core cooling because of the presence of an unrelated phenomenon.

5. The indication must give advanced warning of the approach of inadequate core cooling.

6. The indication must cover the full range from normal operation to complete core uncovering. For example, if water level is chosen as the unambiguous indication, then the range of the instrument (or instruments) must cover the full range from normal water level to the bottom of the core .

CONTAINMENT ISOLATION (2.1.4)

POSITION

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to NRC.

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

CLARIFICATION

1. Provide diverse containment isolation signals that satisfy safety-grade requirements.

2. Identify essential and non-essential systems and provide results to NRC.

3. Non-essential systems should be automatically isolated by containment isolation signals.

4. Resetting of containment isolation signals shall not result in the automatic loss of containment isolation

DEDICATED H_2 CONTROL PENETRATIONS (2.1.5.a)

POSITION

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that the redundancy and single failure requirements of General Design Criterion 54 and 56 of Appendix A to 10 CFR 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

CLARIFICATION

1. This requirement is only applicable to those plants whose licensing basis includes requirements for external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere.

2. An acceptable alternative to the dedicated penetration is a combined design that is single-failure proof for containment isolation purposes and single-failure proof for operation of the recombiner or purge system.

3. The dedicated penetration or the combined single-failure proof alternative should be sized such that the flow requirements for the use of the recombiner or purge system are satisfied.

4. Components necessitated by this requirement should be safety grade.

5. A description of required design changes and a schedule for accomplishing these changes should be provided by January 1, 1980. Design changes should be completed by January 1, 1981.

CAPABILITY TO INSTALL HYDROGEN RECOMBINER AT EACH LIGHT WATER NUCLEAR POWER PLANT (2.1.5.c)

POSITION

The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering sheilding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

CLARIFICATION

1. This requirement applies only to those plants that included Hydrogen Recombiners as a design basis for licensing.

2. The shielding and associated personnel exposure limitations associated with recombiner use should be evaluated as part of licensee response to requirement 2.1.6.B, "Design review for Plant Shielding."

3. Each licensee should review and upgrade, as necessary, those criteria and procedures dealing with recombiner use. Action taken on this requirement should be submitted by January 1, 1980.

INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIALS FOR PWRs AND BWRs (2.1.6.a)

POSITION

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate Leak Reduction

    a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

    b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction

    Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

CLARIFICATION

Licensees shall, by January 1, 1980, provide a summary description of their program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident. Examplesof such systems are given on page A-26 of NUREG-0578. Other examples include the Reactor Core Isolation Cooling and Reactor Water Clea Cleanup (Letdown function) Systems for BWRs. Include a list of systems which are excluded from this program. Testing of gaseous systems should include helium leak detection or equivalent testing methods. Consider in your program to reduce leakage potential release paths due to design and operator deficiencies as discussed in our letter to you regarding North Anna and Related Incidents dated October 17, 1979.

DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS (2.1.6.b)

POSITION

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory,and 1% of the core solids, are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

CLARIFICATION

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. In order to assure that personnel can perform necessary post-accident operations in the vital areas, we are providing the following guidance to be used by licensees to evaluate the adequacy of radiation protection to the operators:

1. Source Term

   The minimum radioactive source term should be equivalent to the source terms recommended, in Regulatory Guides 1.3, 1.4, 1.7 and Standard Review Plant 15.6.5. with appropriate decay times based on plant design.

a. Liquid Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory and 1% of all others are assumed to be mixed in the reactor coolant and liquids injected by HPCI and LPCI.

b. Gas Containing Systems: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For gas containing lines connected to the primary system (e.g., BWR steam lines) the concentration of radioactivity shall be determined assuming the activity is contained in the gas space in the primary coolant system.

2. <u>Dose Rate Criteria</u>

The dose rate for personnel in a vital area should be such that the guidelines of GDC 19 should not be exceeded during the course of the accident. GDC 19 limits the dose to an operator to 5 Rem whole body or its equivalent to any part of the body. When determining the dose to an operator, care must be taken to determine the necessary occupancy time in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, we are providing the following dose rate criteria with alternatives to be documented on a case-by-case basis. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines provided occupancy is not required at the location of the hot spot. These doses are design objectives and are not to be used to limit access in the event of an accident.

a. Areas Requiring Continuous Occupancy: ≤15mr/hr. These areas will require full time occupancy during the course of the accident. The Control Room and onsite technical support center are areas where continuous occupancy will be required. The dose rate for these areas is based on the control room occupancy factors contained in SRP 6.4.

b. Areas Requiring Infrequent Access: GDC 19. These areas may require access on a regular basis, but not continuous occupancy. Shielding should be provided to allow access at a frequency and duration estimated by the licensee. The plant Radiochemical/Chemical Analysis Laboratory, radwaste panel, motor control center, instrumentation locations, and reactor coolant and containment gas sample stations are examples where occupancy may be needed often but not continuously.

AUTO INITIATION OF THE AUXILIARY

FEEDWATER SYSTEM (AFWS) (2.1.7.a)

POSITION

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term:

1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses.

5. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

6. The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

CLARIFICATION

Control Grade (Short-Term)

1. Provide automatic/manual initiation of AFWS.
2. Testability of the initiating signals and circuits is requied.
3. Initiating signals and circuits shall be powered from the emergency buses.
4. Necessary pumps and valves shall be included in the automatic sequence of the loads to the emergency buses. Verify that the addition of these loads does not comprimise the emergency diesel generating capacity.
5. Failure in the automatic circuits shall not result in the loss of manual capability to initiate the AFWS from the control room.
6. Other Considerations
    a. For those designs where instrument air is needed for operation, the electric power supply requirement should be capable of being manually connected to emergency power sources.

AUXILIARY FEEDWATER FLOW INDICATION TO STEAM GENERATORS (2.1.7.b)

POSITION

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

CLARIFICATION

A. Control Grade (Short-Term)

1. Auxiliary feedwater flow indication to each steam generator shall satisfy the single failure criterion.

2. Testability of the auxiliary feedwater flow indication channels shall be a feature of the design.

3. Auxiliary feedwater flow instrument channels shall be powered from the vital instrument buses.

B. Safety-Grade (Long-Term)

1. Auxiliary feedwater flow indication to each steam generator shall satisfy safety-grade requirements.

C. Other

1. For the Short-Term the flow indication channels should by themselves satisfy the single failure criterion for each steam generator. As

a fall-back position, one auxiliary feed water flow channel may be backed up by a steam generator level channel.

2. Each auxiliary feed water channel should provide an indication of feed flow with an accuracy on the order of $\pm$ 10%.

IMPROVED POST-ACCIDENT SAMPLING CAPABILITY (2.1.8.a)

POSITION

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift.

DISCUSSION

The primary purpose of implementing Improved Post-Accident Sampling Capability is to improve efforts to assess and control the course of an accident by:

1. Providing information related to the extent of core damage that has occurred or may be occurring during an accident;

2. Determining the types and quantities of fission products released to the containment in the liquid and gas phase and which may be released to the environment;

3. Providing information on coolant chemistry (e.g., dissolved gas, boron and pH) and containment hydrogen.

The above information requires a capability to perform the following analyses:

1. Radiological and chemical analyses of pressurized and unpressurized reactor coolant liquid samples;

2. Radiological and hydrogen analyses of containment atmosphere (air) samples.

CLARIFICATION

The licensee shall have the capability to promptly obtain (in less than 1 hour) pressurized and unpressurized reactor coolant samples and a containment atmosphere (air) sample.

The licensee shall establish a plan for an onsite radiological and chemical analysis facility with the capability to provide, within 1 hour of obtaining the sample, quantification of the following:

1. certain isotopes that are indicators of the degree of core damage (i.e., noble gases, iodines and cesiums and non-volatile isotopes),
2. hydrogen levels in the containment atmosphere in the range 0 to 10 volume percent,
3. dissolved gases (i.e., H_2 , O_2) and boron concentration of liquids.

or have in-line monitoring capabilities to perform the above analysis.

Plant procedures for the handling and analysis of samples, minor plant modifications for taking samples and a design review and procedural modifications (if necessary) shall be completed by January 1, 1980. Major plant modifications shall be completed by January 1, 1981.

During the review of the post accident sampling capability consideration should be given to the following items:

1. Provisions shall be made to permit containment atmosphere sampling under both positive and negative containment pressure.

2. The licensee shall consider provisions for purging samples lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for passive flow restrictions to limit reactor coolant loss or containment air leak from a rupture of the sample line.

3. If changes or modifications to the existing sampling system are required, the seismic design and quality group classification or sampling lines and components shall conform to the classification of the system to which each sampling line is connected. Components and and piping downstream of the second isolation valve can be designed to quality Group D and nonseismic Category I requirements.

The licensee's radiological sample analysis capability should include provisions to:

a. Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Lessons Learned Item 2.1.6.b. Where necessary, ability to dilute samples to provide capability for measurement and reduction of personnel exposure, should be provided. Sensitivity of onsite analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/gm to the upper levels indicated here.

b. Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

c. Maintain plant procedures which identify the analysis required, measurement techniques and provisions for reducing background levels.

The licensees chemical analysis capability shall consider the presence of the radiological source term indicated for the radiological analysis.

In performing the review of sampling and analysis capability, consideration shall be given to personnel occupational exposure. Procedural changes and/or plant modifications must assure that it shall be possible to obtain and analyze a sample while incurring a radiation dose to any individual that is as low as reasonably achievable and not in excess of GDC 19. In assuring that these limits are met, the following criteria will be used by the staff.

1. For shielding calculations, source terms shall be as given in Lessons Learned Item 2.1.6.b.

2. Access to the sample station and the radiological and chemical analysis facilities shall be through areas which are accessible in post accident situations and which are provided with sufficient shielding to assure that the radiation dose criteria are met.

3. Operations in the sample station, handling of highly radioactive samples from the sample station to the analysis facilities, and handling while working with the samples in the analysis facilities shall be such that the radiation dose criteria are met. This may involve sufficient shielding of personnel from the samples and/or the dilution of samples for analysis. If the existing facilities do not satisfy these criteria, then additional design features, e.g., additional shielding, remote handling etc. shall be provided. The radioactive sample lines in the sample station, the samples themselves in the analysis facilities, and other radioactive lines of the vicinity of the sampling station and analysis facilities shall be included in the evaluation.

4. High range portable survey instruments and personnel dosimeters should be provided to permit rapid assessment of high exposure rates and accumulated personnel exposure.

The licensee shall demonstrate their capability to obtain and analyze a sample containing the isotopes discussed above according to the criteria given in this section.

POSITION

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident", which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

1. Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

   a. Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

   b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (ALARA) concentrations to a maximum of 10^5 μCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

2. Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

3. In-containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

DISCUSSION

The January 1, 1980 requirement, were specifically added by the Commission and were not included in NUREG-0578. The purpose of the interim January 1, 1980 requirement is to assure that licensees have methods of quantifing radioactivity releases should the existing effluent instrumentation go offscale.

CLARIFICATION

1. Radiological Noble Gas Effluent Monitors

   A. January 1, 1980 Requirements

   Until final implementation in January 1, 1981, all operating reactors must provide, by January 1, 1980, an interim method for quantifying high level releases which meets the requirements of Table 2.1.8.b.1. This method is to serve only as a provisional fix with the more detailed, exact methods to follow. Methods are to be developed to quantify release rates of up to 10,000 Ci/sec for noble gases from all potential release points, (e.g., auxiliary building, radwaste building, fuel handling building, reactor building, waste gas decay tank releases, main condenser air ejector, BWR main condenser vacuum pump exhaust, PWR steam safety valves and atmosphere steam dump valves and BWR turbine buildings) and any other areas that communicate directly with systems which may contain primary coolant or containment gases, (e.g., letdown and emergency core cooling systems and external recombiners). Measurements/analysis capabilities of the effluents at the final release point (e.g., stack) should be such that measurements of individual sources which contribute to a common release point may not be necessary. For assessing radioiodine and particulate releases, special procedures must be developed for the removal and analysis of the radioiodine/particulate sampling media (i.e., charcoal canister/filter paper). Existing sampling locations are expected to be adequate; however, special procedures for retrieval and analysis of the sampling media under accident conditions (e.g., high air and surface contamination and direct radiation levels) are needed.

It is intended that the monitoring capabilities called for in the interim can be accomplished with existing instrumentation or readily available instrumentation. For noble gases, modifications to existing monitoring systems, such as the use of portable high range survey instruments, set in shielded collimators so that they "see" small sections of sampling lines is an acceptable method for meeting the intent of this requirement. Conversion of the measured dose rate (mR/hr) into concentration (μCi/cc) can be performed using standard volume source calculations. A method must be developed with sufficient accuracy to quantify the iodine releases in the presence of high background radiation from noble gases collected on charcoal filters. Seismically qualified equipment and equipment meeting IEEE-279 is not required.

The licensee shall provide the following information on his methods to quantify gaseous releases of radioactivity from the plant during an accident.

1. Noble Gas Effluents

   a. System/Method description including:

      i) Instrumentation to be used including range or sensitivity, energy dependence, and calibration frequency and technique,

      ii) Monitoring/sampling locations, including methods to assure representative measurements and background radiation correction,

      iii) A description of method to be employed to facilitate access to radiation readings. For January 1, 1980, Control room read-out is perferred: however, if impractical, in-situ readings by an individual with verbal communication with the Control Room is acceptable based on (iv) below.

iv) Capability to obtain radiation readings at least every 15 minutes during an accident.

v) Source of power to be used. If normal AC power is used, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous readout for 7 consecutive days.

b. Procedures for conducting all aspects of the measurement/ analysis including:

i) Procedures for minimizing occupational exposures

ii) Calculational methods for converting instrument readings to release rates based on exhaust air flow and taking into consideration radionuclide spectrum distribution as function of time after shutdown.

iii) Procedures for dissemination of information.

iv) Procedures for calibration.

B. January 1, 1981 Requirements

By January 1, 1981, the licensee shall provide high range noble gas effluent monitors for each release path. The noble gas effluent monitor should meet the requirements of Table 2.1.8.b.2. The licensee shall also provide the information given in Sections 1.A.1.a.i, 1.A.1.a.ii, 1.A.1.b.ii, 1.A.1.b.iii, and 1.A.1.b.iv above for the noble gas effluent monitors.

2. Radioiodine and Particulate Effluents

A. For January 1, 1980 the licensee should provide the following:

1. System/Method description including:

a) Instrumentation to be used for analysis of the sampling media with discussion on methods used to correct for potentially interfering background levels of radioactivity.

b) Monitoring/sampling location.

c) Method to be used for retrieval and handling of sampling media to minimize occupational exposure.

d) Method to be used for data analysis of individual radionuclides in the presence of high levels of radioactive noble gases.

e) If normal AC power is used for sample collection and analysis equipment, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous read-out for 7 consecutive days.

2. Procedures for conducting all aspects of the measurement analysis including:

a) Minimizing occupational exposure

b) Calculational methods for determining release rates

c) Procedures for dissemination of information

d) Calibration frequency and technique

B. For January 1, 1981, the licensee should have the capability to continuously sample and provide onsite analysis of the sampling media. The licensee should also provide the information required in 2.A above.

3. Containment Radiation Monitors

Provide by January 1, 1981, two radiation monitor systems in containment which are documented to meet the requirements of Table 2.1.8.b.2.

It is possible that future regulatory requirements for emergency planning interfaces may necessitate identification of different types of radionuclides in the containment air, e.g., noble gases (indication of core damage) and non-volatiles (indication of core melt). Consequently, consideration should be given to the possible installation or future conversion of these monitors to perform this function.

TABLE 2.1.8.b.1

INTERIM PROCEDURES FOR QUANTIFYING HIGH LEVEL ACCIDENTAL RADIOACTIVITY RELEASES

- Licensees are to implement procedures for estimating noble gas and radioiodine release rates if the existing effluent instrumentation goes off scale.

- Examples of major elements of a highly radioactive effluent release special procedures (noble gas).

  - Preselected location to measure radiation from the exhaust air, e.g., exhaust duct or sample line.

  - Provide shielding to minimize background interference.

  - Use of an installed monitor (preferable) or dedicated portable monitor (acceptable) to measure the radiation.

  - Predetermined calculational method to convert the radiation level to radioactive effluent release rate.

TABLE 2.1.8.b.2

HIGH RANGE EFFLUENT MONITOR

- NOBLE GASES ONLY
- RANGE: (Overlap with Normal Effluent Instrument Range)

- UNDILUTED CONTAINMENT EXHAUST	10^{+5}	μCi/CC
- DILUTED (>10: 1) CONTAINMENT EXHAUST	10^{+4}	μCi/CC
- MARK I BWR REACTOR BUILDING EXHAUST	10^{+4}	μCi/CC
- PWR SECONDARY CONTAINMENT EXHAUST	10^{+4}	μCi/CC
- BUILDINGS WITH SYSTEMS CONTAINING PRIMARY COOLANT OR GASES	10^{+3}	μCi/CC
- OTHER BUILDINGS (E.G., RADWASTE)	10^{+2}	μCi/CC

- NOT REDUNDANT - 1 PER NORMAL RELEASE POINT
- SEISMIC - NO
- POWER - VITAL INSTRUMENT BUS
- SPECIFICATIONS - PER R.G. 1.97 AND ANSI N320-1979
- DISPLAY*: CONTINUOUS AND RECORDING WITH READOUTS IN THE TECHNICAL SUPPORT CENTER (TSC) AND EMERGENCY OPERATIONS CENTER (EOC)
- QUALIFICATIONS - NO

*Although not a present requirement, it is likely that this information may have to be transmitted to the NRC. Consequently, consideration should be given to this possible future requirement when designing the display interfaces.

TABLE 2.1.8.b.3

HIGH RANGE CONTAINMENT RADIATION MONITOR

- RADIATION: TOTAL RADIATION (ALTERNATE: PHOTON ONLY)
- RANGE:
  - UP TO 10^8 RAD/HR (TOTAL RADIATION)
  - ALTERNATE: 10^7 R/HR (PHOTON RADIATION ONLY)
  - SENSITIVE DOWN TO 60 KEV PHOTONS*
- REDUNDANT: TWO PHYSICALLY SEPARATED UNITS
- SEISMIC: PER R. G. 1.97
- POWER: VITAL INSTRUMENT BUS
- SPECIFICATIONS: PER R.G. 1.97 REV. 2 AND ANSI N320-1978
- DISPLAY: CONTINUOUS AND RECORDING
- CALIBRATION: LABORATORY CALIBRATION ACCEPTABLE

*Monitors must not provide misleading information to the operators assuming delayed core damage when the 80 KEV photon Xe-133 is the major noble gas present.

IMPROVED IN-PLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS (2.1.8.c)

POSITION

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

CLARIFICATION

Use of Portable versus Stationary Monitoring Equipment

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments for the following reasons:

a. The physical size of the auxiliary/fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

b. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

c. Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

d. The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high dose rate areas.

Iodine Filters and Measurement Techniques

A. The following are short-term recommendations and shall be implemented by the licensee by January 1, 1980. The licensee shall have the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single channel analyzer (SCA). The SCA window should be calibrated to the 365 keV of ^{131}I. A representative air sample shall be taken and then counted for ^{131}I using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too must activity collected on the sampling cartridge.

B. By January 1, 1981:

The licensee shall have the capability to remove the sampling cartridge to a low background, low contamination area for further analysis. This area should be ventilated with clean air containing no airborne radionuclides which may contribute to inaccuracies in analyzing the sample. Here, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble bases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples and effluent charcoal samples under accident conditions.

TRANSIENT AND ACCIDENT ANALYSIS (2.1.9)

POSITION

See NUREG-0578, page A-44.

DISCUSSION

The scope of the requied transient and accident analysis is discussed in NUREG-0578. The schedule for these analyses is included in NUREG-0578 and is reproduced in the Implementation Schedule attachment to this letter. The Bulletins and Orders Task Force has been implementing these required analyses on that schedule. The analysis of the small break loss of coolant accident has been submitted by each of the owners groups. These analyses are presently under review by the B&O Task Force. The scope and schedule for the analysis of inadequate core cooling have been discussed and agreed upon in meetings between the owners groups and the B&O Task Force, and are documented in the minutes to those meetings.

The analysis of transients and accidents for the purpose of upgrading emergency procedures is due in early 1980 and the detailed scope and schedule of this analysis is the subject of continuing discussions between the owners groups and the B&O Task Force.

CONTAINMENT PRESSURE INDICATION

POSITION

A continuous indication of containment pressure should be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

CLARIFICATION

1. The containment pressure indication shall meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy, and testability.

2. The containment pressure monitor shall be installed by January 1, 1981.

CONTAINMENT WATER LEVEL INDICATION

POSITION

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

CLARIFICATION

1. The narrow range sump level instrument shall monitor the normal containment sump level vice the containment emergency sump level.

2. The wide range containment water level instruments shall meet the requirements of the proposed revision to Regulatory Guide 1.97 (Instrumentation for Light-Water Cooled Nuclear Power Plant to Assess Plant Conditions During and Following a Accident).

3. The narrow range containment water level instruments shall meet the requirements of Regulatory Guide 1.89 (Qualification of Class IE Equipment of Nuclear Power Plants).

4. The equivalent capacity of the wide range PWR level instrument has been changed from 500,000 gallons to 600,000 gallons to ensure consistency with the proposed revision to Regulatory Guide 1.97. It should be noted that this measurement capability is based on recent plant designs. For older plants with smaller water capacities, licensees may propose deviations from this requirement based on the available water supply capability at their plant.

5. The containment water level indication shall be installed by January 1, 1981.

CONTAINMENT HYDROGEN INDICATION

POSITION

A continuous indicaton of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

CLARIFICATION

1. The containment hydrogen indication shall meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy, and testability.

2. The containment hydrogen indication shall be installed by January 1, 1981.

REACTOR COOLANT SYSTEM VENTING

POSITION

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

1. A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

2. Analyses deomonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1), and Standard Review Plan Section 6.2.5.

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

CLARIFICATION

A. General

1. The two important safety functions enhanced by this venting capability are core cooling and containment integrity. For events within the present design basis for nuclear power plants, the capability to vent non-condensible gases will provide additional assurance of meeting the requirements of 10CFR50.46 (LOCA criteria) and 10CFR50.44 (containment criteria for hydrogen generation). For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of non-condensible gas without the loss of core cooling or containment integrity.

2. Procedures addressing the use of the RCS vents are required by January 1, 1981. The procedures should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be based on the following criteria: (1) assurance that the plant can meet the requirements of 10CFR50.46 and 10CFR50.44 for Design Basis Accidents; and (2) a substantial increase in the plants ability to maintain core cooling and containment integrity for events beyond the Design Basis.

B. BWR Design Considerations

1. Since the BWR owners group has suggested that the present BWR designs inherent capability of venting, this question relates to the capability of existing systems. The ability of these systems to vent the RCS of non-condensible gas must be demonstrated. In addition the ability of these systems to meet the same requirements as the PWR vent systems must be documented. Since there are important differences among BWR's, each licensee should address the specific design features of his plant.

2. In addition to reactor coolant system venting, each BWR licensee should address the ability to vent other systems such as the isolation condenser, which may be required to maintain adequate core cooling. If the production of a large amount of non-condensible gas would cause the loss of function of such a system, remote venting of that system is required. The qualifications of such a venting system should be the same as that required for PWR venting systems.

C. PWR Vent Design Considerations

1. The locations for PWR Vents are as follows:

   a. Each PWR licensee should provide the capability to vent the reactor vessel head.

   b. The reactor vessel head vent should be capable of venting non-condensible gas from the reactor vessel hot legs (to the elevation of the top of the outlet nozzle) and cold legs (through head jets and other leakage paths). Additional venting capability is required for those portions of each hot leg which can not be vented through the the reactor vessel head vent. The NRC recognizes that it is impractical to vent each of the many thousands of tubes in a U-tube steam generator. However, we believe that a procedure can be developed which assures that sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the reactor coolant system. Such a procedure is required by January 1981.

   c. Venting of the pressurizer is required to assure its availability for system pressure and volume control. These are important considerations especially during natural circulation.

2. The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly large range of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which we consider reasonable, is to specify a volume of non-condensible gas to be vented and a venting time i.e., a vent capable of venting a gas volume of 1/2 the RCS in one hour. Other criteria and engineering approaches should be considered if desired.

3. Where practical the RCS vents should be kept smaller than the size corresponding to the definition of a LOCA (10CFR50 Appendix A). This will minimize the challenges to the ECCS since the inadvertent opening of a vent smaller than the LOCA definition would not require ECCS actuation although it may result in leakage beyond Technical Specification Limits. On PWRs the use of new or existing valves which are larger than the LOCA definition will require the addition of a block valve which can be closed remotely to terminate the LOCA resulting from the inadvertent opening of the vent.

4. An indication of valve position should be provided in the control room.

5. Each vent should be remotely operable from the control room.

6. Each vent should be seismically qualified.

7. The requirements for a safety grade system is the same as the safety grade requirement on other Short Term Lessons Learned items, that is, it should have the same qualifications as were accepted for the reactor protection system when the plant was licensed. The exception to this requirement is that we do not require redundant valves at each venting location. Each vent must have its power supplied from an emergency bus. A degree of redundancy should be provided by powering different vents from different emergency buses.

8. For systems where a block valve is required, the block valve should have the same qualifications as the vent.

9. Since the RCS vent system will be part of the reactor coolant systems boundary, efforts should be made to minimize the probability of an inadvertent actuation of the system. Removing power from the vents is one step in the direction. Other steps are also encouraged.

10. Since the generation of large quantities of non-condensible gas could be associated with substantial core damage, venting to atmosphere is unacceptable because of the associated released radioactivity. Venting into containment is the only presently available alternative. Within containment those areas which provide good mixing with containment air are preferred. In addition, areas which provide for maximum cooling of the vented gas are preferred. Therefore the selection of a location for venting should take advantage of existing ventilation and heat removal systems.

11. The inadvertent opening of an RCS vent must be addressed. For vents smaller than the LOCA definition, leakage detection must be sufficient to identify the leakage. For vents larger than the LOCA definition, an analysis is required to demonstrate compliance with 10CFR50.46.

SHIFT SUPERVISOR RESPONSIBILITIES (2.2.1.a)

POSITION

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

   c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly secified.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

CLARIFICATION

The attachment provides clarification to the above position.

SHIFT SUPERVISOR RESPONSIBILITY (2.2.1.A)

NUREG-0578 POSITION (POSITION NO.)	CLARIFICATION
Highest Level of Corporate Management (1.)	V. P. For Operations
Periodically Reissue (1.)	Annual Reinforcement of Company Policy
Management Direction (1.)	Formal Documentation of Shift Personnel, All Plant Management, Copy to IE Region
Properly Defined (2.0)	Defined in Writing in a Plant Procedure
Until Properly Relieved (2.B)	Formal Transfer of Authority, Valid SRO License, Recorded in Plant Log
Temporarily Absent (2.C)	Any Absence
Control Room Defined (2.C)	Includes Shift Supervisor Office Adjacent to the Control Room
Designated (2.C)	In Administrative Procedures
Clearly Specified	Defined in Administrative Procedures
SRO Training	Specified in ANS 3.1 (Draft) Section 5.2.1.8
Administrative Duties (4.)	Not Affecting Plant Safety
Administrative Duties Reviewed (4.)	On Same Interval as Reinforcement: i.e., Annual by V. P. for Operations.

SHIFT TECHNICAL ADVISOR (Section 2.2.1.b)

POSITION

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The Shift Technical Advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the Shift Technical Advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

DISCUSSION

The NRC Lessons Learned Task Force has recommended the use of Shift Technical Adviors (STA) as a method of immediately improving the plant operating staff's capabilities for response to off-normal conditions and for evaluating operating experience.

In defining the characteristics of the STA, we have used the two essential functions to be provided by the STA. These are accident assessment and operating experience assessment.

1. Accident Assessment

   The STA serving the accident assessment function must be dedicated to concern for the safety of the plant. The STA's duties will be to diagnose off-normal events and advise the shift supervisor. The duties of the STA should not include the manipulatin of controls or supervision of operators. The STA must be available, in the control room, within 10 minutes of being summoned.

   The qualifications of the STA should include college level education in engineering and science subjects as well as training in reactor operations both normal and off-normal. Details regarding these qualifications are provided in paragraphs A.1, 2 and 3 of Enclosure 2 to our September 13, 1979 letter. In addition, the STA serving the accident assessment function must be cognizant of the evaluations performed as part of the operating experience assessment function.

2. Operating Experience Assessment

The persons serving the opeating experience assessment function must be dedicated to concern for the safety of the plant. Their function will be to evaluate plant operations from a safety point of view and should include such assignments as listed on pages A-50 and A-51 of NUREG-0578. Their qualifications are identical to those described previously under accident assessment and collectively this group should provide competence in all technical areas important to safety. It is desirable that this function be performed by onsite personnel.

CLARIFICATION

1. Due to the similarity in the requirements for dedication to safety, training and onsite location and the desire that the accident assessment function be performed by someone whose normal duties involve review of operating experiences, our preferred position is that the same people perform the accident and operating experience assessment functions. The performance of these two functions may be split if it can be demonstrated the persons assigned the accident assessment role are aware, on a current basis, of the work being done by those reviewing operating experience.

2. To provide assurance that the STA will be dedicated to concern for the safety of the plant, our position has been that STA's must have a clear measure of independence from duties associated with the commercial operation of the plant. This would minimize possible distractions from safety judgements by the demands of commercial operations. We have determined that, while desirable, independence from the operations staff of the plant is not necessary to provide this assurance. It is necessary, however, to clearly emphasize the dedication to safety associated with the STA position both in the STA job description and in the personnel filling this position. It is not acceptable to assign a person, who is normally the immediate supervisor of the shift supervisor to STA duties as defined herein.

3. It is our position that the STA should be available within 10 minutes of being summoned and therefore should be onsite. The onsite STA may be in a duty status for periods of time longer than one shift, and therefore asleep at some times, if the ten minute availability is assured. It is preferable to locate those doing the operating experience assessment onsite. The desired exposure to the operating plant and contact with the STA (if these functions are to be split) may be able to be accomplished by a group, normally stationed offsite, with frequent onsite presence. We do not intend, at this time, to specify or advocate a minimum time onsite.

4. The implementation schedule for the STA requirements is to have the STA on duty by January 1, 1980, and to have STAs, who have all completed training requirements, on duty by January 1, 1981. While minimum training requirements have not been specified for January 1, 1980, the STAs on duty by that time should enhance the accident and operating experience assessment function at the plant.

SHIFT AND RELIEF TURNOVER PROCEDURES (2.2.1.c)

POSITION

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist.

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console.

      (what to check and criteria for acceptable status shall be included on the checklist);

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and ongoing auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

CLARIFICATION

No clarification provided.

CONTROL ROOM ACCESS (2.2.2.a)

POSITION

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access, and

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

CLARIFICATION

No clarification provided.

ONSITE TECHNICAL SUPPORT CENTER (TSC) 2.2.2.b

POSITION

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center. Records that pertain to the as-built conditions and layout of structures, systems and components shall be readily available to personnel in the TSC.

CLARIFICATION

1. By January 1, 1980, each licensee should meet items A-G that follow. Each licensee is encouraged to provide additional upgrading of the TSC (items 2-10) as soon as practical, but no later than January 1, 1981.

   A. Establish a TSC and provide a complete description,

   B. Provide plans and procedures for engineering/management support and staffing of the TSC,

   C. Install dedicated communications between the TSC and the control room, near site emergency operations center, and the NRC,

   D. Provide monitoring (either portable or permanent) for both direct radiation and airborne radioactive contaminmants. The monitors should provide warning if the radiation levels in the support center are reaching potentially dangerous levels. The licensee should designate action levels to define when protective measures should be taken (such as using breathing apparatus and potassium iodide tablets, or evacuation to the control room),

   E. Assimilate or ensure access to Technical Data, including the licensee's best effort to have direct display of plant parameters, necessary for assessment in the TSC,

F. Develop procedures for performing this accident assessment function from the control room should the TSC become uninhabitable, and

G. Submit to the NRC a longer range plan for upgrading the TSC to meet all requirements.

2. Location

It is recommended that the TSC be located in close proximity to the control room to ease communications and access to technical information during an emergency. The center should be located onsite, i.e., within the plant security boundary. The greater the distance from the CR, the more sophisticated and complete should be the communications and availability of technical information. Consideration should be given to providing key TSC personnel with a means for gaining access to the control room.

3. Physical Size & Staffing

The TSC should be large enough to house 25 persons, necessary engineering data and information displays (TV monitors, recorders, etc.). Each licensee should specify staffing levels and disciplines reporting to the TSC for emergencies of varying severity.

4 Activation

The center should be activated in accordance with the "Alert" level as defined in the NRC document "Draft Emergency Action Level Guidelines, NUREG-0610" dated September, 1979, and currently out for public comment. Instrumentation in the TSC should be capable of providing displays of vital plant parameters from the time the accident began (t = 0 defined as either reactor or turbine trip). The Shift Technical Advisor should be consulted on the "Notification of Unusual Event" however, the activation of the TSC is discretionary for that class of event.

5. Instrumentation

The instrumentation to be located in the TSC need not meet safety-grade requirements but should be qualitatively comparable (as regards accuracy and reliability) to that in the control room. The TSC should have the capability to access and display plant parameters independent from actions in the control room. Careful consideration should be given to the design of the interface of the TSC instrumentation to assure that addition of the TSC will not result in any degradation of the control room or other plant functions.

6. Instrumentation Power Supply

The power supply to the TSC instrumentation need not meet safety-grade requirements, but should be reliable and of a quality compatible with the TSC instrumentation requirements. To insure continuity of information at the TSC, the power supply provided should be continuous once the TSC is activated. Consideration should be given to avoid loss of stored data (e.g., plant computer) due to momentary loss of power or switching transients. If the power supply is provided from a plant safety-related power source, careful attention should be give to assure that the capability and reliability of the safety-related power source is not degraded as a result of this modification.

7. Technical Data

Each licensee should establish the technical data requirements for the TSC, keeping in mind the accident assessment function that has been established for those persons reporting to the TSC during an emergency. As a minimum, data (historical in addition to current status) should be available to permit the assessment of:

Plant Safety Systems Parameters for:

- Reactor Coolant System
- Secondary System (PWRs)
- ECCS Systems
- Feedwater & Makeup Systems
- Containment

In-Plant Radiological Parameters for:

- Reactor Coolant System
- Containment
- Effluent Treatment
- Release Paths

Offsite Radiological

- Meteorology
- Offsite Radiation Levels

8. <u>Data Transmission</u>

In addition to providing a data transmission link between the TSC and the control room, each licensee should review current technology as regards transmission of those parameters identified for TSC display.

Although there is not a requirement at the present time, each licensee should investigate the capability to transmit plant data offsite to the Emergency Operations Center, the NRC, the reactor vendor, etc.

9. Structural Integrity

A. The TSC need not be designed to seismic Category I requirements. The center should be well built in accordance with sound engineering practice with due consideration to the effects of natural phenomena that may occur at the site.

B. Since the center need not be designed to the same stringent requirements as the Control Room, each licensee should prepare a backup plan for responding to an emergency from the control room.

10. Habitability

The licensee should provide protection for the technical support center personnel from radiological hazards including direct radiation and airborne contaminants as per General Design Criterion 19 and SRP 6.4.

A. Licensee should assure that personnel inside the technical support center (TSC) will not receive doses in excess of those specified in GDC 19 and SRP 6.4 (i.e., 5 Rem whole body and 30 Rem to the thyroid for the duration of the accident). Major sources of radiation should be considered.

B. Permanent monitoring systems should be provided to continuously indicate radiation dose rates and airborne radioactivity concentrations inside the TSC. The monitoring systems should include local alarms to warn personnel of adverse conditions. Procedures must be provided which will specify appropriate protective actions to be taken in the event that high dose rates or airborne radioactive concentrations exist.

C. Permanent ventilation systems which include particulate and charcoal filters should be provided. The ventilation systems need not be qualified as ESF systems. The design and testing guidance of Regulatory Guide 1.52 should be followed except that the systems do not have to be redundant, seismic, instrumented in the control room or automatically activated. In addition, the HEPA filters need not be tested as specified in Regulatory Guide 1.52 and the HEPA's do not have to meet the QA requirements of Appendix B to 10 CFR 50. However, spare parts should be readily available and procedures in place for replacing failed components during an accident. The systems should be designed to operate from the emergency power supply.

D. Dose reduction measures such as breathing apparatus and potassium iodide tablets can not be used as a design basis for the TSC in lieu of ventilation systems with charcoal filters. However, potassium iodide and breathing apparatus should be available.

ONSITE OPERATIONAL SUPPORT CENTER (SECTION 2.2.2.c)

POSITION

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

CLARIFICATION

No clarification provided.

IMPLEMENTATION SCHEDULE

SECTION NUMBER	TITLE	IMPLEM. CAT. (1)	PROPOSAL REVIEW	IMPLEMENTATION REVIEW
2.1.1	Emergency Power Supply			
	Pressurizer Heaters	A		X
	Pressurizer Level	A		X
	PORV and Block Valve	A		X
2.1.2	Relief and Safety Valve Test			
	Program and Schedule	A		X
	Complete Test	07/81		X
2.1.3 a	Direct Indication of Valve Position	A		X
2.1.3.b	Instrumentation for Inadequate Core Cooling			
	Procedures	A		X
	Design of New Instrumentation	A		X
	Subcooling Meter	A		X
	Installation of New Instr. (E.G., Level Meter)	B	X	

(1) CATEGORY A: IMPLEMENTATION COMPLETE BY JANUARY 1, 1980,
CATEGORY B: IMPLEMENTATION COMPLETE BY JANUARY 1, 1981.

Enclosure 2

<u>IMPLEMENTATION SCHEDULE</u>

SECTION NUMBER	TITLE	IMPLEM. CAT. (1)	PROPOSAL REVIEW	IMPLEMENTATION REVIEW
2.1.4	Containment Isolation	A		X
2.1.5	Dedicated H^2 Control Penetrations			
	Description and Schedule	A		X
	Installation	B		X
2.1.5.c	Recombiner Procedures	A		X
2.1.6.a	Systems Integrity for High Radioactivity			
	Leak Reduction Program	A		X
	Preventative Maintenance Program	A		X
2.1.6.b	Plant Shielding Review			
	Design Review	A		X
	Plant Modifications	B		X

(1) CATEGORY A: IMPLEMENTATION COMPLETE BY JANUARY 1, 1980,
CATEGORY B: IMPLEMENTATION COMPLETE BY JANUARY 1, 1981.

IMPLEMENTATION SCHEDULE

SECTION NUMBER	TITLE	IMPLEM. CAT. (1)	PROPOSAL REVIEW	IMPLEMENTATION REVIEW
2.1.7.a	Auto Initiation of AFW			
	Control Grade	A	X	
	Safety Grade	B	X	
2.1.7.b	AFW Flow			
	Control Grade	A		X
	Safety Grade	B		X
2.1.8.a	Post-Accident Sampling			
	Design Review	A		X
	Procedures	A		X
	Description of Plant Modifications	A		X
	Plant Modifications	B		X
2.1.8.b	High Range Radiation Monitors			
	In-Containment	B	X	
	Effluents - Procedures	A		X
	Implement	B	X	

(1) CATEGORY A: IMPLEMENTATION COMPLETE BY JANUARY 1, 1980.
CATEGORY B: IMPLEMENTATION COMPLETE BY JANUARY 1, 1981.

IMPLEMENTATION SCHEDULE

SECTION NUMBER	TITLE	IMPLEM. CAT. (1)	PROPOSAL REVIEW	IMPLEMENTATION REVIEW
2.1.8.c	Improved Iodine Instrumentation	A		X
2.1.9	Transient and Accident Analysis	(2)		X
	Containment Pressure Monitor	B		X
	Containment Water Level Mointor	B		X
	Containment Hydrogen Monitor	B		X
	RCS Venting			
	Design Complete	A		X
	Installation Complete	B	X	
2.2.1.a	Shift Supervisor Responsibilities	A		X

(1) CATEGORY A: IMPLEMENTATION COMPLETE BY JANUARY 1, 1980,
CATEGORY B: IMPLEMENTATION COMPLETE BY JANUARY 1, 1981.

(2) SEE NUREG-0578

IMPLEMENTATION SCHEDULE

SECTION NUMBER	TITLE	IMPLEM. CAT. (1)	PROPOSAL REVIEW	IMPLEMENTATION REVIEW
2.1.2.B	Shift Technical Advisor			
	Advisor on Duty	A		X
	Complete Training	B		X
2.2.1.C	Shift Turnover Procedure	A		X
2.2.2.A	Control Room Access	A		X
2.2.2.B	On Site Technical Support Center			
	Establish Center	A		X
	Upgrade to Meet All Requirements	B		X
2.2.2.C	On Site Operational Support Center	A		X

(1) CATEGORY A: IMPLEMENTATION COMPLETE BY JANUARY 1, 1980.
CATEGORY B: IMPLEMENTATION COMPLETE BY JANUARY 1, 1981.

ANALYSIS AND TRAINING SCHEDULE

Task Description	Completion Date
1. Small Break LOCA analysis and preparation of emergency procedure guidelines.	July-September 1979*
2. Implementation of small break LOCA emergency procedures and retraining of operators	December 31, 1979
3. Analysis of inadequate core cooling and preparation of emergency procedure guidelines	October 1979
4. Implementation of emergency procedures and retraining related to inadequate core cooling	January 1980
5. Analysis of accidents and transients and preparation of emergency procedure guidelines	Early 1980
6. Implementation of emergency procedures and retraining related to accidents and transients	3 months after guidelines established
7. Analysis of LOFT small break tests	Pretest (Mid-September 1979)

* Range covers completion dates for the four NSSS vendors

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 14 1979

Mr. G. E. Liebler, Chairman
Combustion Engineering Owners Group
Florida Power & Light Company
P. O. Box 013100
Miami, Florida 33101

Dear Mr. Liebler:

SUBJECT: EVALUATION OF OPERATOR GUIDELINES FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS IN C-E DESIGNED OPERATING PLANTS

Our letter of June 5, 1979 (Robert W. Reid to all operating Combustion Engineering plants) requested that operating plants with C-E designed reactors develop guidelines for the preparation of operating procedures to cope with small-break LOCA's. In response to this request, the C-E Owners Group submitted report CEN-114-P (Amendment 1P) which included said guidelines. In response to our requests for additional information and to issues raised during our meeting of October 30, 1979, the guidelines were subsequently modified. The modified guidelines were submitted by your letter to D. F. Ross dated November 8, 1979. We have completed our review of the modified guidelines, and are attaching hereto as Enclosure 1 a copy of our evaluation.

As stated in our evaluation, we have concluded that the guidelines submitted by your November 8, 1979 letter are acceptable for use in developing operating procedures to cope with small-break LOCA's in C-E operating plants having high-pressure safety injection pumps with shut-off heads less than 1600 psi. Although the guidelines were based on a reference plant having 200 psi safety injection tanks and 1300 psi high-pressure safety injection pumps, you have stated that they are applicable to all operating C-E plants, including those with 600 psi safety injection tanks and those with 2400 psi high-pressure safety injection pumps. However, we have not as yet determined that the guidelines are acceptable for a plant having high-pressure safety injection pumps with a 2400 psi shut-off head. Our concern is related to the potential events in which water could be discharged through the safety valves while the operator is attempting to reach a condition of at least 50° F below saturation. A copy of the approved guidelines, subject to acceptably incorporating those revisions required by Enclosure 1, is attached hereto as Enclosure 2.

Those licensees with C-E designed reactors for which these guidelines are approved may now proceed with the development of small-break LOCA emergency procedures and operator training. In developing these procedures, each licensee must account for the effects of specific design characteristics at its plant. As indicated on Page 5 of Enclosure 6 to the Darrell G. Eisenhut letter dated September 13, 1979 to all operating nuclear power plants, these procedures and related operator training are to be implemented by December 31, 1979.

NOV 14 1979

In implementing these procedures, each licensee shall provide:

(1) The instrument uncertainties involved with HPI termination criteria to indicate that the criteria will assure subcooled conditions.

(2) Adequate assurance that the HPSI pumps will not be run deadheaded in the recirculation mode and that minimum flow requirements will be met.

(3) An indication of the typicality of the analyses documented in CEN-114-P (Amendment 1P) and in the modified guidelines shown in Enclosure 2 relative to its own plant.

Licensees will also be required to implement emergency procedures covering the extended loss of all feedwater, (including pressure vessel integrity considerations), and to revise emergency procedures for initiating and monitoring natural circulation, including provisions for plant cooldown. These procedures will be based on guidelines which the C-E Owners Group are developing under "inadequate core cooling."

As part of our audit program, we expect to examine the procedures at a lead C-E operating plant initially, and at other C-E operating plants at a later date to assure that the procedures were developed in accordance with the approved guidelines. We also plan to check out some of the procedures at a C-E simulator on a schedule to be developed later. It should be noted however, that our audit program need not impede progress toward implementing the procedures and associated training by December 31, 1979.

Sincerely,

Original signed by:

D. F. Ross, Jr., Director
Bulletins & Orders Task Force

Enclosures:
As stated
cc: See attached lists

* See previous yellow for concurrences

OFFICE	B&OTF	B&OTF	B&OTF	B&OTF	B&OTF	B&OTF
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ENCLOSURE 1

Evaluation of Combustion Engineering Post-LOCA Operating Guidelines

Introduction

By letter dated June 5, 1979, the staff requested that all operating CE plants provide guidelines for the preparation of operational procedures for the recovery of plants following small LOCA's. The guidelines were to cover both short-term and long-term situations and follow through to a stable condition. Recognition of the event, precautions, actions, and prohibited actions were to be included also. CE submitted CEN-114-P-(NP), "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems" in July, 1979 and CEN-115-P(NP), "Response to NRC IE Bulletin 79-06C Items 2 and 3 for Combustion Engineering Nuclear Steam Supply Systems" in August, 1979. CEN-114-P(NP) was submitted in response to our request for information while CEN-115-P(NP) revised this response to account for the impact of RCP operating requirements.

Summary Description: CE Post-LOCA Operating Guidelines

The guideline submitted by CE is preceded by a bases section which supplies background material for the information presented in the guideline. The guideline itself is split into four sections: Symptoms, Immediate Actions, Follow-Up Actions, and Precautions.

The Symptoms are a list of indications which an operator is expected to utilize in confirming that a small break loss-of-coolant accident has occurred. Low pressurizer pressure, high containment sump level, high containment pressure or temperature, safety injection actuation, and high or low pressurizer level are among the symptoms provided to the operator to assist in the identification of this accident. A diagnostics chart has been appended to the LOCA guidelines to clarify symptoms and to channel the operator's actions into the correct procedure.

Immediate Actions are those actions which are required to place the plant in a safe condition. These steps are distinguished from subsequent procedural steps by a requirement for memorization. An operator must know these steps without reference to a procedure, thereby ensuring that there is no delay in achieving a safe condition. The guidelines require that the reactor be tripped; standard post-trip actions be carried out (plant specific); safety injection be initiated (if not automatically actuated); reactor coolant pumps be tripped after SIAS actuation on low RCS pressure; auxiliary feedwater flow be established if main feedwater is not available; verification that the CIAS and SIAS signals have properly actuated; the SIS be operated to maintain a 50°F subcooling margin and indicated pressurizer level; and the break be located and isolated if possible.

Follow-Up Actions are actions required to place the plant in a stable condition. The previous procedural steps (Immediate Actions) ensured that the reactor was in a safe condition, that the core remains covered by ECCS operation, and that escaping radioactivity is isolated by CIAS. The next steps are aimed at bringing the plant to a lower mode of operation, cold shutdown. The Follow-Up Actions require a plant cooldown within one-hour using the steam dumps or turbine bypass system. The cooldown is continued via a number of alternative paths such as long-term recirculation, initiation of shutdown cooling, continued use of the steam dumps and emergency feed, or, as a last resort, opening of the power operated relief valves.

The Precautions section lists warnings which the operator must observe to ensure plant safety. For example, the operator is warned that pressurizer level may not always be a true indicator of fluid inventory and that primary system temperature must be monitored when establishing auxiliary feedwater to prevent excessive cooldown rates. A total of eleven Precautions have been included for implementation by the licensees in the appropriate procedural locations.

Evaluation

The NRC staff reviewed the post-LOCA operating guidelines with respect to the following critical operator actions:

1. Reactor coolant pump trip
2. Safety injection termination criteria
3. Verification of safety systems actuation
4. Verification of a heat sink

During our review, the staff identified modifications to be made to the guidelines to enhance the directions to the operator. These modifications were subsequently incorporated in the guidelines via revisions issued on November 8, 1979.

The criteria for tripping the reactor coolant pumps are consistent with the requirements of IE Bulletin 79-06C. All operating reactor coolant pumps are stopped after an SIAS caused by low reactor coolant system pressure and after it has been verified that the reactor has been shutdown for at least five seconds. We conclude that this criterion is acceptable subject revising "Immediate Action" item 3 of the guidelines to be consistent with the above wording.

The criterion for terminating safety injection flow is based on the establishment and maintenance of a 50^{o}F subcooling margin along with an indication of pressurizer level. The staff concurs that these criteria are sufficient for ensuring that safety injection can be terminated without concern for detrimental voiding in the primary system. We conclude that this criterion is acceptable for those plants with low-head HPSI pumps (< 1600 psi).

As part of his immediate actions, the operator is directed to verify the reactor trip, safety injection actuation, adequate auxiliary feedwater flow (if main feedwater is not available), and containment isolation actuation. We concur that these actions are sufficient to ensure minimum safeguards and heat sink availability needed to mitigate small break LOCAs.

The staff noted that the guidelines are based on obtaining at least minimum safeguards operation to mitigate small break LOCAs. We require each licensee to extend the emergency procedures to cover the loss of all feedwater. Procedures for this degraded condition should also take into account pressure vessel integrity considerations. The Owners Group has committed to prepare guidelines for operational procedures regarding the loss of all feedwater as part of its effort on the issue of inadequate core cooling.

The staff also requires that the emergency procedures include instructions for monitoring and initiating (if lost) natural circulation for small break LOCAs where heat removal by the steam generators is required. A separate guideline has been received on natural circulation operation. The staff, upon completion of its evaluation, will require that the natural circulation guideline be appended to or referenced by the appropriate emergency procedures.

The staff requires that each licensee provide procedures for cooling down the plant under natural circulation conditions. These procedures should address boration control and monitoring, cooldown of the pressurizer, and adequate criteria for monitoring coolant system temperatures to ensure that voids do not form in the primary system which could inhibit adequate heat removal. As in the case of loss of all feedwater, the Combustion Engineering Owners Group has committed to prepare guidelines for operational procedures regarding cooldown under natural circulation conditions as part of its effort on inadequate core cooling.

Conclusions

Based on our review, we conclude that the small-break loss-of-coolant accident operating guidelines submitted by the Combustion Engineering Owners Group on November 8, 1979 are acceptable for C-E plants having high-pressure safety injection pumps with shut-off heads 1600 psi or less. Accordingly, said guidelines can be used for developing operating procedures for coping with small-break loss-of-coolant accidents for such plants, provided that the licensees implement the requirements noted above when developing their procedures. Our acceptance of these generic guidelines notwithstanding, each licensee must account for the effects of specific plant design parameters (e.g., differences in the shut-off pressures of high-pressure safety injection pumps, differences in the design pressure of the safety injection tanks), when translating these guidelines into plant specific operating procedures.

<u>E N C L O S U R E   2</u>

P.O. Box 529100
Miami, FL 33152

November 8, 1979

Dr. Denwood F. Ross, Jr.
Director
Bulletins and Orders Task Force
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Transmittal of Revised Post-LOCA Guidelines

Reference: (A) NRC letter from Dr. D. F. Ross, Jr. to Mr. G. E. Liebler, dated October 19, 1979

(B) IE Bulletin 79-06C, dated July 26, 1979

(C) NUREG-0578, July 1979

Dear Dr. Ross:

Reference A requested additional information regarding the guidelines presented in CEN-114 Revision A and CEN-115 for loss of coolant accidents (LOCA). Questions regarding those guidelines were further discussed in a meeting with the NRC staff on October 30, 1979, and a number of revisions were agreed upon. This letter transmits those revised Post-LOCA guidelines. These guidelines are being submitted for your approval on behalf of the Combustion Engineering Owners Group so that they may be incorporated into utility procedures in accordance with Reference B and the schedule presented in Reference C.

It should be noted that these guidelines do not necessarily reflect the preferred actions of our vendor, Combustion Engineering. Combustion Engineering's preferred actions remain as stated in CEN-115. The NRC staff has specifically requested that the guidelines for RCP operation be revised to incorporate the RCP operating requirements stated in IE Bulletin 79-06C (Reference A, Item I.6.E). Combustion Engineering has been unable to identify a transient analyzed in Chapter 6 or 15 of the FSAR that will result in violation of acceptance criteria, provided the RCP's are not tripped until the rods have been fully inserted for 5 seconds. The enclosed guidelines have therefore been revised to reflect the staff's request.

If you should have any questions regarding these guidelines, please feel free to contact me at (305) 552-3811.

Very truly yours,

C-E OWNERS GROUP

George E. Liebler
Chairman

Enclosure

POST LOCA GUIDELINES

Bases for Post-LOCA Operating Guidelines

Provided below is a general description of plant responses to large and small break LOCA's. This is intended to supply background material for the information presented in the guidelines.

A small break LOCA is characterized by:

a) A slow loss of RCS pressure during the short term (10 to 30 minutes) and equilibrium pressure above * 300 psia in the long term (30 to 480 minutes) resulting from matching safety injection flow and flow from the break.

b) A loss of RCS inventory during the short term followed by a refilling of the RCS during the long term.

c) Core cooling is initially by the steam generator(s) and flow from the break and later by the shutdown cooling system. The break does not always (depending on size) provide the necessary heat removal yet depletes RCS inventory. Breaks in RCS piping less than 2 inches in diameter fall into this category. The steam generators provide cooling for forced or natural circulation of the RCS, if inventory is depleted, in a boiloff and reflux mode. The shutdown cooling system is used after the RCS has been refilled and pressure control is provided by the HPSI pumps and the charging pumps.

A general description of small break LOCA operations follows:

Initially, the plant is hot and pressurized. A small break LOCA results in a slow loss of RCS inventory and a decrease in pressure. Low pressurizer pressure initiates a SIAS which automatically actuates the SIS. The reactor is tripped. The operator stops the reactor coolant pumps. Auxiliary feedwater is established to the steam generators. Steam dump is provided manually using atmospheric dump valves or turbine bypass valves, or automatically by the steam generator dump and bypass system or by steam generator relief valves.

*This value is typical, it may vary for specific designs.

For very small breaks, the steam generators are the main heat sink, and additional heat is removed with the coolant through the break. Continued reactor coolant pump operation during this period could aid heat removal by the steam generators. However, for small hot leg breaks, reactor coolant pump operation will result in a higher two-phase mixture level in the reactor vessel and hot leg piping. Consequently, for a break in the bottom of the hot leg, the break is covered longer by two-phase mixture, causing a larger loss of water inventory from the vessel. This eventually results in a lower coolant level in the reactor vessel. The result could be a higher clad temperature and a delay in refilling the vessel. The net effect of reactor coolant pump operation during the initial period may be to increase the severity of the accident. The NRC has therefore requested that the RCP operating requirements stated in IE Bulletin 79-06C be incorporated into the guidelines for operating plants following LOCA's (NRC letter from Dr. D.F. Ross to G. E. Liebler, dated October 19, 1979). Bulletin 79-06C directed to holders of operating licenses to: "Upon reactor trip and HPI initiation caused by low reactor coolant system pressure, immediately trip all operating RCP's." This action should not result in the violation of acceptance criteria for transients or accidents in chapter 6 or 15 of the FSAR, provided the RCP's are not tripped until rods have been fully inserted for 5 seconds. This delay is to allow for the decay of the heat flux following reactor trip before reducing forced flow.

The time necessary to refill the RCS and regain control of pressure and inventory depends on break size, break location, and the number of HPSI pumps and charging pumps actuated. With only one HPSI pump activated, and a break located on the bottom of the cold leg, it may take as long as 8 hours to refill the RCS. With all injection pumps operable, the time is about 1 hour. In the period of time it takes the RCS to refill some voiding in the RCS will occur. This, condition can be recognized by indication that RCS hot leg temperature or core thermocouple temperature is equal to the saturation temperature for the existing RCS pressure. In this mode,decay heat is removed by boiling in the core and condensation in the steam generator. In addition, heat is removed by flow from the break. The operator must ensure that the SIS is providing flow to the RCS, and the steam generators are removing heat. These actions will ensure adequate core cooling and eventually a subcooled condition will be achieved. Once RCS pressure and temperature are adequately reduced, the shutdown cooling system is placed in operation. In the event that the feedwater supply to the steam generator is exhausted and the shutdown cooling system is inoperable, the PORV's are opened to ensure that the flow from the injection system is sufficient to cool the core. The SIS will be realigned for cold leg injection only. Core flushing is from the cold legs through the core and out the PORV.

Simultaneous hot and cold leg injection is used for both small break and large break LOCA's so the operator does not have to distinguish between them at the time when simultaneous injection is required for large breaks. (For small breaks, the boron concentration remains low due to dispersal throughout the RCS, so hot and cold leg injection is not essential).

Reactor coolant system pressure is used to differentiate between small and large break LOCA's. However, the delineation between small and large breaks does not need to be precise since there is a range of intermediate breaks for which either response will produce satisfactory results. The guidelines take this into account with the decisions to be made after eight hours.

The large break LOCA is characterized by:

a) A rapid loss of RCS pressure in 10 seconds to 3 minutes with equilibrium pressures below* 300 psia and, in the case of the largest breaks, the RCS pressure nearly equal to containment pressure.
b) Core cooling is provided for by large flow from the injection system due to low RCS pressure. The flow from the break provides sufficient heat removal. Simultaneous hot and cold leg injection is required to prevent possible boric acid accumulation in the core.

A general description of large break LOCA operations follows:

Initially, the plant is hot and pressurized. A large break LOCA results in a rapid loss of inventory and pressure. Low pressurizer pressure initiates a SIAS which automatically actuates the SIS. The reactor is tripped. Auxiliary feedwater is established to the steam generators. Steam dump is provided manually using atmospheric steam dump valves or turbine bypass valves. The major mechanism for heat removal is the flow from the SIS

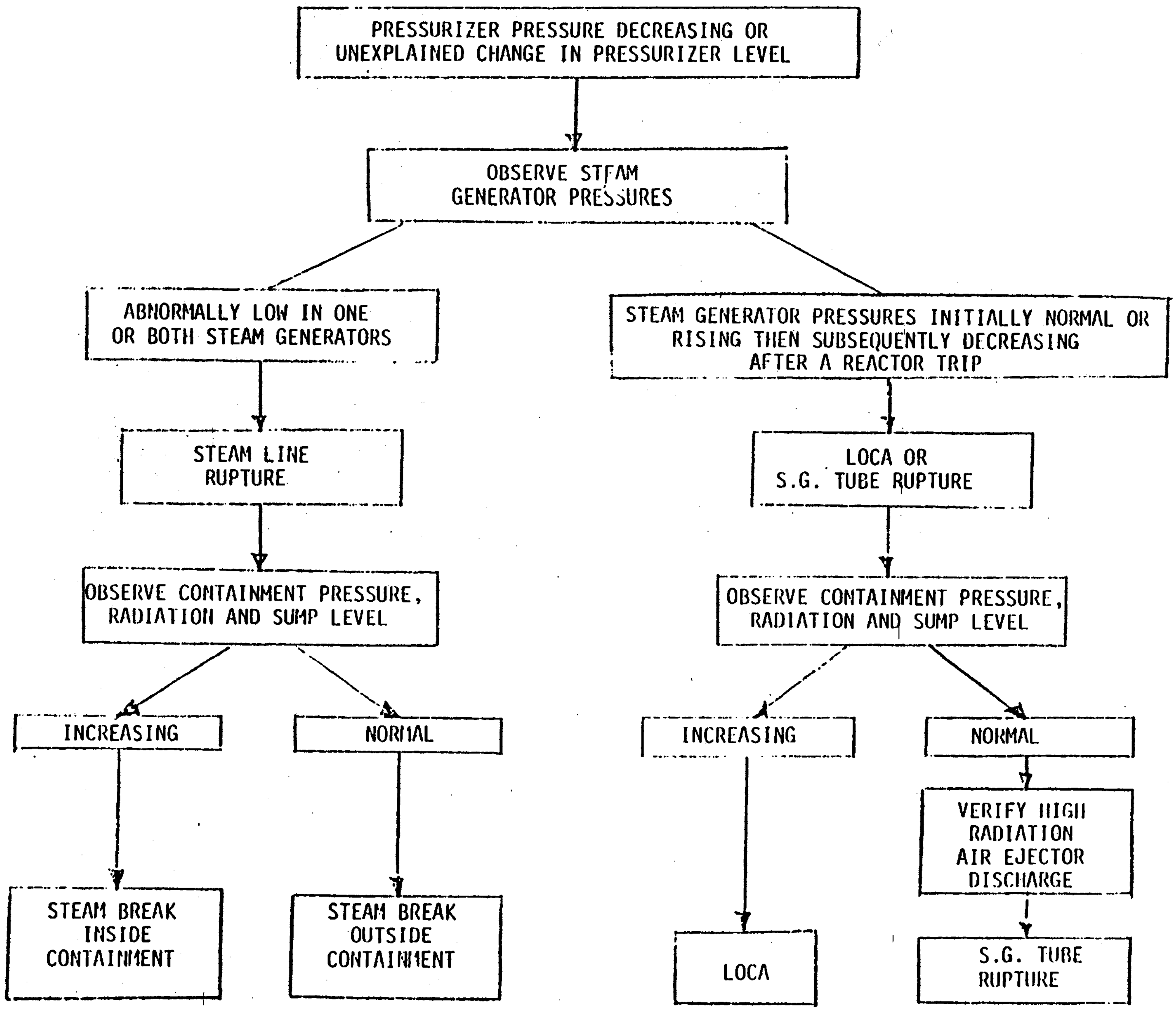
*This valve is typical, it may vary for specific designs.

through the core and out the break. Containment pressure may be high and containment isolation is likely. Containment spray may have been automatically activated.

The SIS is aligned to provide simultaneous hot and cold leg injection which is sufficient to cool the core and flush the reactor vessel indefinitely. For both large and small break LOCA's, continued monitoring of conditions in the RCS and performance of safety systems should be done. All available indications should be used to aid in diagnosing the event since the accident may cause irregularities in a particular instrument reading.

Regardless of the cause of actuation of a safety system, the automatic response should not be altered until it has been demonstrated that other systems and equipment are providing the functions that the safety system is intended to perform.

BREAK IDENTIFICATION



Guidelines for Operating Plants Following LOCA's

Symptoms

1. Reactor coolant system leak exceeds the capacity of the operable charging pumps.
2. A reactor trip may have occurred.
3. The Safety Injection System (SIS) may have automatically actuated.
4. Any one or more of the following indications or alarms may be present.
   a) Low pressurizer pressure
   b) High containment pressure or temperature
   c) High containment sump level
   d) High containment radiation
   e) High or low pressurizer level
   f) High quench tank level
   g) High quench tank temperature
   h) High quench tank pressure
   i) T_{av} decreasing or at saturation temperature for RCS pressure.

Immediate Actions

1. Trip the reactor if not already tripped and carry out standard post trip actions.
2. Initiate safety injection if it has not already been actuated by the safety injection actuation signal.
3. After an SIAS caused by low reactor coolant system pressure and after it has been verified that all rods have been fully inserted for 5 seconds, stop all operating reactor coolant pumps.
4. If main feedwater is not available, immediately establish or verify an auxiliary feedwater flow of *gpm.
5. If the containment isolation actuation signal (CIAS) is activated, ensure that the system has properly actuated.
6. Ensure that the systems receiving an SIAS are properly actuated and that CIAS is actuated.
7. After any SIAS, operate the SIS** until RCS hot and cold leg temperatures are at least 50°F below saturation temperature for the RCS pressure and a pressurizer level is indicated, unless the cause of the SIAS has been verified to be an inadvertent actuation. If 50°F subcooling cannot be maintained after the system has been stopped, the high pressure injection system must be restarted.

8. Attempt to locate and isolate the source of the leak. Possible leak locations include, but are not limited to the PORV's, the letdown line and sample lines.

<u>Follow-Up Actions</u>

1. Operate atmospheric steam dump valves (or turbine bypass valves if the condenser is available) to maintain or reduce plant temperature and reduce steam generator pressure below the steam generator relief valve setpoints. Begin a plant cooldown as soon as possible and in any case within 1 hour.
2. Manually align the safety injection and charging systems to provide flow to the RCS hot and cold legs* two hours after the LOCA**.
3. If the pressure and inventory control with the SIS cannot be established after* eight hours and RCS pressure is less than* 300 psig, continue the hot and cold leg injection.
4. If pressure and inventory control with the SIS are established after* eight hours and RCS pressure is greater than* 300 psig, conduct one of the following activities. The activities are listed in order of decreasing preference.

   a) RCS pressure above* 300 psig indicates that the system has refilled and subcooling has occurred. Verify this by checking the saturation pressure for the existing temperature. Realign the SIS for cold leg injection. Continue to maintain subcooling and reduce RCS pressure to the initiation pressure for shutdown cooling by reducing the flow delivered by the high pressure injection and charging pumps and by venting or isolating the safety injection tanks as necessary. While reducing pressure and after shutdown cooling is initiated, maintain RCS pressure with the charging pumps and/or the HPSI pumps to continue to maintain at least 50^{o} subcooling, or

   b) Continue to remove decay heat using emergency feed and steam dump if adequate condensate is available and (a) cannot be implemented, or
   c) Open pressurizer power operated relief valves and align the SIS for cold leg injection if (a) or (b) cannot be implemented.

* This value is typical, it may vary for specific designs.

** Includes stopping charging pumps on some plants

Precautions

1. Before restarting RCP's ensure that cooling water services to the pumps has been restored.

2. Pressurizer level may not always be a true indicator of RCS fluid inventory. Pressurizer steam space ruptures, reference leg failures, and reference leg flashing may cause indications which are contrary to true conditions.

3. All available indications should be used to aid in diagnosing the event since the accident may cause irregularities in a particular instrument reading. Critical parameters must be verified when one or more confirmatory indications are available.

4. When extablishing auxiliary feedwater flow to the steam generators, monitor primary system temperature and pressure to avoid exceeding a 100^{o}F/hour cooldown rate.

5. Feedwater is normally provided to both steam generators. Isolation of a single steam generator is mandatory if a steam generator tube rupture is detected in that generator to prevent lifting of the safety valves or reseat them if they have lifted. This action will also reduce the amount of radioactivity released. For small breaks in the RCS where steam generators are important for heat removal one steam generator must be used for this purpose even if primary to secondary leaks are detected.

6. Continued lengthy operation of the containment spray may jeopardize the operation of equipment which would be desirable or necessary to mitigate the consequences of the event. Early consideration should be given to termination of spray operation. If the containment pressure has returned to below the actuation setpoint, the system may be stopped. The system should be realigned for automatic actuation.

7. Observe all available indications to determine conditions within the RCS. Use RCS hot leg temperature, RCS cold leg temperature, core exit thermocouple temperature, and RCS pressure to determine if the RCS is subcooled or saturated. An increase in temperature above the saturation temperature for the existing pressure is an indication of voiding in the RCS. A decrease in operating RCP motor current or erratic pump ΔP is also an indication of voiding. If this occurs the operator must ensure that the RCP's are turned off, the SIS is providing makeup to the RCS, and that the steam generators are removing heat from the RCS.

8. Monitor refueling water tank level to verify the shift from injection to recirculation. If a recirculation actuation signal (RAS) occurs, the operator must prevent the HPSI pumps from operating at less than minimum flow conditions. If all HPSI pumps and charging pumps are operating and the HPSI pumps are delivering less than 30 gpm per pump, turn off the charging pumps one at a time and then HPSI pumps one at a time until only one HPSI pump remains operating. This will ensure that minimum flow requirements will be met by the flow through the pump to the RCS for the smallest break size that results in a SIAS.

9. Monitor the auxiliary building radiation levels and sump levels after an RAS to attempt to detect leakage from the SIS. Even if leaks are detected at least one high pressure safety injection pump must remain in operation to provide flow to the RCS.

10. If there is a high radioactivity level in the reactor coolant system, circulation of this fluid in the SCS may result in high area radioactivity readings in the auxiliary building. The activity level of the RCS should be determined prior to initiating SCS flow.

11. Minimum Pressure - Temperature operating restrictions take precedence over requirements for operation of the high pressure injection or charging system to achieve 50^{o} subcooling during operation of the shutdown cooling system.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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DEC 2 6 1979

Mr. G. E. Liebler, Chairman
Combustion Engineering Owners' Group
Florida Power & Light Company
P. O. Box 013100
Miami, Florida 33101

Dear Mr. Liebler:

SUBJECT: EVALUATION OF OPERATOR GUIDELINES FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS IN C-E DESIGNED OPERATING PLANTS

Our letter of June 5, 1979 (Robert W. Reid to all operating Combustion Engineering plants) requested that operating plants with C-E-designed reactors develop guidelines for the preparation of operating procedures to cope with small-break LOCA's. In response to this request, the C-E Owners' Group submitted report CEN-114-P (Amendment 1P) which included these guidelines. In response to our requests for additional information and to issues raised during our meeting of October 30, 1979, the guidelines were subsequently modified. The modified guidelines were submitted by your letter dated November 8, 1979. In my letter to you dated November 14, 1979, we approved the modified guidelines for all C-E operating plants except for a plant having high pressure safety injection pumps with a 2400 psi shutoff head.

In your letter of December 13, 1979 (see Enclosure 1) you provided modified guidelines for this class of plant. Subsequent to that, we held discussions with members of the C-E Owners' Group and C-E to clarify certain matters. We have now completed our review of the modified guidelines. Our supplemental evaluation is provided as Enclosure 2 to this letter. The supplemental evaluation in Enclosure 2, together with the evaluation provided in my letter to you dated November 14, 1979, comprise the bases for our approval of the guidelines for this class of plant. The November 14, 1979 letter is provided as Enclosure 3 to this letter.

The November 14, 1979 letter contains a number of provisions which licensees are required to meet in implementing the guidelines. These provisions are equally applicable to those licensees that develop procedures from the revised guidelines.

All licensees with C-E-designed reactors are expected to proceed with the development of small break LOCA emergency procedures and operator training. As indicated on Page 5 of Enclosure 6 to the Darrell G. Eisenhut letter dated September 13, 1979 to all operating nuclear power plants, these procedures and related operator training are to be implemented by December 31, 1979.

Sincerely,

D F Ross

D. F. Ross, Jr., Director
Bulletins and Orders Task Force

Enclosures:
As stated

cc: See attached lists

December 13, 1979

Dr. Denwood F. Ross, Jr.
Director
Bulletins and Orders Task Force
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

Subject: Additional Post-LOCA Guidance for Plants with High Pressure Safety Injection Pumps with a 2400 psi Shut-off Head

Reference: NRC letter from Dr. D. F. Ross, Jr. to Mr. G. E. Liebler, dated November 14, 1979

Dear Dr. Ross:

Your referenced letter forwarded the NRC evaluation of the LOCA guideline for CE designed plants. That evaluation concluded that the guidelines are acceptable for CE operating plants having high pressure safety injection pumps with shut-off heads less than 1600 psi. However, the NRC has not yet determined that the guidelines are acceptable for a plant (Maine Yankee) having high pressure safety injection pumps with shutoff heads of 2400 psi.

The NRC is concerned with potential events in which water could be discharged through the safety valves while the operator is attempting to achieve a RCS fluid condition of at least 50° below saturation. The question is whether or not 50° of subcooling can be achieved, at which point the operator is allowed to stop high pressure flow to the RCS, before the pressure in the RCS reaches the setpoint of the safety valves and water is discharged through these valves. The shutoff head of the Maine Yankee pump (2425 psi) is below the setpoint of the safety valves (2500 psi) but is above the setpoint of the PORV (2400 psi) and this needs to be addressed.

To evaluate the NRC concern, representative non LOCA events which depressurize the RCS were studied. The events chosen were a failure of the reactor coolant pressure regulating system and a steam line break. Non-LOCA events result in the highest RCS repressurization due to high pressure pump action. These events essentially represent a "zero break" LOCA case. In this study it was assumed that all four RCP's were tripped following SIS actuation and that the resulting flow coastdown causes loss of pressurizer sprays. The study confirmed that 50° of subcooling is achieved prior to reaching the setpoint of the PORV. If no operator action is taken the high pressure pumps could cause the PORV's to lift.

There is approximately 5 minutes between the point in time that 50^{0} subcooling is reached and the setpoint of the PORV is reached. Additionally, when the PORV setpoint (2400 psi) is reached, the pressurizer is not water solid. The study also indicates that there is an additional 5 minutes before the pressurizer is filled solid assuming two HPSI pumps are operating.

This is judged to be sufficient time frame for the operator to take action. The action the operator takes in this situation is as follows:

Preferred Action - Terminate high pressure pump flow to the reactor coolant system.

Alternative Action - Prevent operation of the PORV's by:

(a) shutting down stream block valves.

or

(b) position PORV's control switch to prevent automatic opening.

As a result of the above evaluation it has been concluded that an additional precaution should be added to the CE Post LOCA Guidelines in order to incorporate applicability to the Maine Yankee Plant. That precaution forms the attachment to this letter. The addition of this precaution should result in a determination that the LOCA guidelines are acceptable for developing operating procedures for Maine Yankee.

If you should have any questions regarding this guidance; please feel free to contact me or Mr. R. T. Harris of our Technical Advisary Committee at (203) 666-6911, extension 5519.

Very truly yours,

C-E OWNERS GROUP

Robert T. Harris for
George E. Liebler
Chairman

ADDITIONAL GUIDANCE FOR PLANTS WITH HIGH PRESSURE SAFETY INJECTION PUMPS WITH A 2400 PSI SHUT-OFF HEAD

Add the additional precaution to the Post LOCA Guidelines as follows:

12. An SIAS can be generated by events other than a LOCA, such as a failure of the reactor coolant pressure regulating system. Continued operation of high head-high pressure injection pumps can cause the RCS to repressurize to the setpoint of the PORV's. Opening of the PORV's should be prevented by:

    (a) operating the SIS to maintain RCS pressure below the PORV setpoint (2400 psi) while maintaining at least 50°F subcooling.

    <u>or</u>

    (b) position the PORV control switch to prevent automatic opening of the PORV's.

    <u>or</u>

    (c) shut the PORV block valves to negate consequences of PORV's opening.

EVALUATION OF SMALL-BREAK LOCA GUIDELINES FOR C-E OPERATING PLANTS HAVING SAFETY INJECTION PUMPS WITH SHUTOFF HEADS GREATER THAN 1600 PSI

INTRODUCTION

By letter dated November 14, 1979, we approved the guidelines for developing small-break LOCA procedures in C-E operating plants. This approval was limited to plants having safety injection (SI) pumps with shutoff heads less than 1600 psi; therefore, to support the implementation of these guidelines at plants having SI pumps with shutoff heads greater than 1600 psi, (i.e., Maine Yankee), the C-E Owners Group submitted additional information in a letter dated December 13, 1979. In addition, on December 13 and 14, 1979, we held discussions with C-E personnel regarding our concerns associated with plants having SI pumps with high shutoff pressures.

EVALUATION

To evaluate the potential of lifting the PORVs in C-E designed plants, all of which have a setpoint of 2400 psi, prior to satisfying the 50°F subcooling criterion at plants having SI pumps with a 2425 psi shutoff head (i.e., Maine Yankee), two non-LOCA events were analyzed: (i) failure of the pressure regulating system, and (ii) a steam line break. The maximum calculated hot leg temperature for these events was 540°F. Since this temperature is significantly below 612°F, the saturation temperature at 2400 psi with a 50° subcooling margin, our subcooling criterion should be satisfied prior to lifting the PORVs. This calculated temperature of 540°F was based on the dynamic conditions prevailing during the pressurizer refill portion of the transients.

On December 13 and 14, 1979, the staff discussed a more conservative steady state analysis with C-E personnel wherein no credit is taken for the dynamic effects of cold feedwater and SI flow. Such an analysis simply considers natural circulation in the primary system transferring heat to the steam generator (SG) whose temperature corresponds to the saturation pressure of the SG safety valves. C-E stated that under these conditions, their analyses show that the maximum hot leg temperature would be 580°F. Since this temperature is also below the 612°F cited above, the 50°F subcooling criterion would also be met without exceeding 2400 psi, the PORV setpoint.

CONCLUSIONS

Based on the results of the analyses that show that the 50°F criterion will be met prior to lifting the PORVs, we find that the LOCA guidelines are acceptable for C-E plants having SI pumps with shutoff heads of 2425 psi (i.e., Maine Yankee). This approval, however, is contingent upon receiving documentation from the C-E Owners Group of the analyses showing that the 50°F subcooling criterion can be met without exceeding 2400 psi.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 1 1979

Mr. G. E. Liebler, Chairman
Combustion Engineering Owners Group
Florida Power & Light Company
P. O. Box 013100
Miami, Florida 33101

Dear Mr. Liebler:

SUBJECT: EVALUATION OF OPERATOR GUIDELINES FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS IN C-E DESIGNED OPERATING PLANTS

Our letter of June 5, 1979 (Robert W. Reid to all operating Combustion Engineering plants) requested that operating plants with C-E designed reactors develop guidelines for the preparation of operating procedures to cope with small-break LOCA's. In response to this request, the C-E Owners Group submitted report CEN-114-P (Amendment 1P) which included said guidelines. In response to our requests for additional information and to issues raised during our meeting of October 30, 1979, the guidelines were subsequently modified. The modified guidelines were submitted by your letter to D. F. Ross dated November 8, 1979. We have completed our review of the modified guidelines, and are attaching hereto as Enclosure 1 a copy of our evaluation.

As stated in our evaluation, we have concluded that the guidelines submitted by your November 8, 1979 letter are acceptable for use in developing operating procedures to cope with small-break LOCA's in C-E operating plants having high-pressure safety injection pumps with shut-off heads less than 1600 psi. Although the guidelines were based on a reference plant having 200 psi safety injection tanks and 1300 psi high-pressure safety injection pumps, you have stated that they are applicable to all operating C-E plants, including those with 600 psi safety injection tanks and those with 2400 psi high-pressure safety injection pumps. However, we have not as yet determined that the guidelines are acceptable for a plant having high-pressure safety injection pumps with a 2400 psi shut-off head. Our concern is related to the potential events in which water could be discharged through the safety valves while the operator is attempting to reach a condition of at least 50° F below saturation. A copy of the approved guidelines, subject to acceptably incorporating those revisions required by Enclosure 1, is attached hereto as Enclosure 2.

Those licensees with C-E designed reactors for which these guidelines are approved may now proceed with the development of small-break LOCA emergency procedures and operator training. In developing these procedures, each licensee must account for the effects of specific design characteristics at its plant. As indicated on Page 5 of Enclosure 6 to the Darrell G. Eisenhut letter dated September 13, 1979 to all operating nuclear power plants, these procedures and related operator training are to be implemented by December 31, 1979.

In implementing these procedures, each licensee shall provide:

(1) The instrument uncertainties involved with HPI termination criteria to indicate that the criteria will assure subcooled conditions.

(2) Adequate assurance that the HPSI pumps will not be run deadheaded in the recirculation mode and that minimum flow requirements will be met.

(3) An indication of the typicality of the analyses documented in CEN-114-P (Amendment 1P) and in the modified guidelines shown in Enclosure 2 relative to its own plant.

Licensees will also be required to implement emergency procedures covering the extended loss of all feedwater, (including pressure vessel integrity considerations), and to revise emergency procedures for initiating and monitoring natural circulation, including provisions for plant cooldown. These procedures will be based on guidelines which the C-E Owners Group are developing under "inadequate core cooling."

As part of our audit program, we expect to examine the procedures at a lead C-E operating plant initially, and at other C-E operating plants at a later date to assure that the procedures were developed in accordance with the approved guidelines. We also plan to check out some of the procedures at a C-E simulator on a schedule to be developed later. It should be noted however, that our audit program need not impede progress toward implementing the procedures and associated training by December 31, 1979.

Sincerely,

D. F. Ross, Jr., Director
Bulletins & Orders Task Force

Enclosures:
As stated
cc: See attached lists

ENCLOSURE 1

Evaluation of Combustion Engineering Post-LOCA Operating Guidelines

Introduction

By letter dated June 5, 1979, the staff requested that all operating CE plants provide guidelines for the preparation of operational procedures for the recovery of plants following small LOCA's. The guidelines were to cover both short-term and long-term situations and follow through to a stable condition. Recognition of the event, precautions, actions, and prohibited actions were to be included also. CE submitted CEN-114-P-(NP), "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems" in July, 1979 and CEN-115-P(NP), "Response to NRC IE Bulletin 79-06C Items 2 and 3 for Combustion Engineering Nuclear Steam Supply Systems" in August, 1979. CEN-114-P(NP) was submitted in response to our request for information while CEN-115-P(NP) revised this response to account for the impact of RCP operating requirements.

Summary Description: CE Post-LOCA Operating Guidelines

The guideline submitted by CE is preceded by a bases section which supplies background material for the information presented in the guideline. The guideline itself is split into four sections: Symptoms, Immediate Actions, Follow-Up Actions, and Precautions.

The Symptoms are a list of indications which an operator is expected to utilize in confirming that a small break loss-of-coolant accident has occurred. Low pressurizer pressure, high containment sump level, high containment pressure or temperature, safety injection actuation, and high or low pressurizer level are among the symptoms provided to the operator to assist in the identification of this accident. A diagnostics chart has been appended to the LOCA guidelines to clarify symptoms and to channel the operator's actions into the correct procedure.

Immediate Actions are those actions which are required to place the plant in a safe condition. These steps are distinguished from subsequent procedural steps by a requirement for memorization. An operator must know these steps without reference to a procedure, thereby ensuring that there is no delay in achieving a safe condition. The guidelines require that the reactor be tripped; standard post-trip actions be carried out (plant specific); safety injection be initiated (if not automatically actuated); reactor coolant pumps be tripped after SIAS actuation on low RCS pressure; auxiliary feedwater flow be established if main feedwater is not available; verification that the CIAS and SIAS signals have properly actuated; the SIS be operated to maintain a 50°F subcooling margin and indicated pressurizer level; and the break be located and isolated if possible.

Follow-Up Actions are actions required to place the plant in a stable condition. The previous procedural steps (Immediate Actions) ensured that the reactor was in a safe condition, that the core remains covered by ECCS operation, and that escaping radioactivity is isolated by CIAS. The next steps are aimed at bringing the plant to a lower mode of operation, cold shutdown. The Follow-Up Actions require a plant cooldown within one-hour using the steam dumps or turbine bypass system. The cooldown is continued via a number of alternative paths such as long-term recirculation, initiation of shutdown cooling, continued use of the steam dumps and emergency feed, or, as a last resort, opening of the power operated relief valves.

The Precautions section lists warnings which the operator must observe to ensure plant safety. For example, the operator is warned that pressurizer level may not always be a true indicator of fluid inventory and that primary system temperature must be monitored when establishing auxiliary feedwater to prevent excessive cooldown rates. A total of eleven Precautions have been included for implementation by the licensees in the appropriate procedural locations.

Evaluation

The NRC staff reviewed the post-LOCA operating guidelines with respect to the following critical operator actions:

1. Reactor coolant pump trip
2. Safety injection termination criteria
3. Verification of safety systems actuation
4. Verification of a heat sink.

During our review, the staff identified modifications to be made to the guidelines to enhance the directions to the operator. These modifications were subsequently incorporated in the guidelines via revisions issued on November 8, 1979.

The criteria for tripping the reactor coolant pumps are consistent with the requirements of IE Bulletin 79-06C. All operating reactor coolant pumps are stopped after an SIAS caused by low reactor coolant system pressure and after it has been verified that the reactor has been shutdown for at least five seconds. We conclude that this criterion is acceptable subject revising "Immediate Action" item 3 of the guidelines to be consistent with the above wording.

The criterion for terminating safety injection flow is based on the establishment and maintenance of a 50°F subcooling margin along with an indication of pressurizer level. The staff concurs that these criteria are sufficient for ensuring that safety injection can be terminated without concern for detrimental voiding in the primary system. We conclude that this criterion is acceptable for those plants with low-head HPSI pumps (< 1600 psi).

As part of his immediate actions, the operator is directed to verify the reactor trip, safety injection actuation, adequate auxiliary feedwater flow (if main feedwater is not available), and containment isolation actuation. We concur that these actions are sufficient to ensure minimum safeguards and heat sink availability needed to mitigate small break LOCAs.

The staff noted that the guidelines are based on obtaining at least minimum safeguards operation to mitigate small break LOCAs. We require each licensee to extend the emergency procedures to cover the loss of all feedwater. Procedures for this degraded condition should also take into account pressure vessel integrity considerations. The Owners Group has committed to prepare guidelines for operational procedures regarding the loss of all feedwater as part of its effort on the issue of inadequate core cooling.

The staff also requires that the emergency procedures include instructions for monitoring and initiating (if lost) natural circulation for small break LOCAs where heat removal by the steam generators is required. A separate guideline has been received on natural circulation operation. The staff, upon completion of its evaluation, will require that the natural circulation guideline be appended to or referenced by the appropriate emergency procedures.

The staff requires that each licensee provide procedures for cooling down the plant under natural circulation conditions. These procedures should address boration control and monitoring, cooldown of the pressurizer, and adequate criteria for monitoring coolant system temperatures to ensure that voids do not form in the primary system which could inhibit adequate heat removal. As in the case of loss of all feedwater, the Combustion Engineering Owners Group has committed to prepare guidelines for operational procedures regarding cooldown under natural circulation conditions as part of its effort on inadequate core cooling.

Conclusions

Based on our review, we conclude that the small-break loss-of-coolant accident operating guidelines submitted by the Combustion Engineering Owners Group on November 8, 1979 are acceptable for C-E plants having high-pressure safety injection pumps with shut-off heads 1600 psi or less. Accordingly, said guidelines can be used for developing operating procedures for coping with small-break loss-of-coolant accidents for such plants, provided that the licensees implement the requirements noted above when developing their procedures. Our acceptance of these generic guidelines notwithstanding, each licensee must account for the effects of specific plant design parameters (e.g., differences in the shut-off pressures of high-pressure safety injection pumps, differences in the design pressure of the safety injection tanks), when translating these guidelines into plant specific operating procedures.

E N C L O S U R E 2

P.O. Box 529100
Miami, FL 33152

November 8, 1979

Dr. Denwood F. Ross, Jr.
Director
Bulletins and Orders Task Force
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Transmittal of Revised Post-LOCA Guidelines

Reference: (A) NRC letter from Dr. D. F. Ross, Jr. to Mr. G. E. Liebler, dated October 19, 1979

(B) IE Bulletin 79-06C, dated July 26, 1979

(C) NUREG-0578, July 1979

Dear Dr. Ross:

Reference A requested additional information regarding the guidelines presented in CEN-114 Revision A and CEN-115 for loss of coolant accidents (LOCA). Questions regarding those guidelines were further discussed in a meeting with the NRC staff on October 30, 1979, and a number of revisions were agreed upon. This letter transmits those revised Post-LOCA guidelines. These guidelines are being submitted for your approval on behalf of the Combustion Engineering Owners Group so that they may be incorporated into utility procedures in accordance with Reference B and the schedule presented in Reference C.

It should be noted that these guidelines do not necessarily reflect the preferred actions of our vendor, Combustion Engineering. Combustion Engineering's preferred actions remain as stated in CEN-115. The NRC staff has specifically requested that the guidelines for RCP operation be revised to incorporate the RCP operating requirements stated in IE Bulletin 79-06C (Reference A, Item I.6.E). Combustion Engineering has been unable to identify a transient analyzed in Chapter 6 or 15 of the FSAR that will result in violation of acceptance criteria, provided the RCP's are not tripped until the rods have been fully inserted for 5 seconds. The enclosed guidelines have therefore been revised to reflect the staff's request.

If you should have any questions regarding these guidelines, please feel free to contact me at (305) 552-3811.

Very truly yours,

C-E OWNERS GROUP

George E. Liebler
Chairman

Enclosure

POST LOCA GUIDELINES

Bases for Post-LOCA Operating Guidelines

Provided below is a general description of plant responses to large and small break LOCA's. This is intended to supply background material for the information presented in the guidelines.

A small break LOCA is characterized by:

a) A slow loss of RCS pressure during the short term (10 to 30 minutes) and equilibrium pressure above * 300 psia in the long term (30 to 480 minutes) resulting from matching safety injection flow and flow from the break.

b) A loss of RCS inventory during the short term followed by a refilling of the RCS during the long term.

c) Core cooling is initially by the steam generator(s) and flow from the break and later by the shutdown cooling system. The break does not always (depending on size) provide the necessary heat removal yet depletes RCS inventory. Breaks in RCS piping less than 2 inches in diameter fall into this category. The steam generators provide cooling for forced or natural circulation of the RCS, if inventory is depleted, in a boiloff and reflux mode. The shutdown cooling system is used after the RCS has been refilled and pressure control is provided by the HPSI pumps and the charging pumps.

A general description of small break LOCA operations follows:

Initially, the plant is hot and pressurized. A small break LOCA results in a slow loss of RCS inventory and a decrease in pressure. Low pressurizer pressure initiates a SIAS which automatically actuates the SIS. The reactor is tripped. The operator stops the reactor coolant pumps. Auxiliary feedwater is established to the steam generators. Steam dump is provided manually using atmospheric dump valves or turbine bypass valves, or automatically by the steam generator dump and bypass system or by steam generator relief valves.

*This value is typical, it may vary for specific designs.

For very small breaks, the steam generators are the main heat sink, and additional heat is removed with the coolant through the break. Continued reactor coolant pump operation during this period could aid heat removal by the steam generators. However, for small hot leg breaks, reactor coolant pump operation will result in a higher two-phase mixture level in the reactor vessel and hot leg piping. Consequently, for a break in the bottom of the hot leg, the break is covered longer by two-phase mixture, causing a larger loss of water inventory from the vessel. This eventually results in a lower coolant level in the reactor vessel. The result could be a higher clad temperature and a delay in refilling the vessel. The net effect of reactor coolant pump operation during the initial period may be to increase the severity of the accident. The NRC has therefore requested that the RCP operating requirements stated in IE Bulletin 79-06C be incorporated into the guidelines for operating plants following LOCA's (NRC letter from Dr. D.F. Ross to G. E. Liebler, dated October 19, 1979). Bulletin 79-06C directed to holders of operating licenses to: "Upon reactor trip and HPI initiation caused by low reactor coolant system pressure, immediately trip all operating RCP's." This action should not result in the violation of acceptance criteria for transients or accidents in chapter 6 or 15 of the FSAR, provided the RCP's are not tripped until rods have been fully inserted for 5 seconds. This delay is to allow for the decay of the heat flux following reactor trip before reducing forced flow.

The time necessary to refill the RCS and regain control of pressure and inventory depends on break size, break location, and the number of HPSI pumps and charging pumps actuated. With only one HPSI pump activated, and a break located on the bottom of the cold leg, it may take as long as 8 hours to refill the RCS. With all injection pumps operable, the time is about 1 hour. In the period of time it takes the RCS to refill some voiding in the RCS will occur. This, condition can be recognized by indication that RCS hot leg temperature or core thermocouple temperature is equal to the saturation temperature for the existing RCS pressure. In this mode,decay heat is removed by boiling in the core and condensation in the steam generator. In addition, heat is removed by flow from the break. The operator must ensure that the SIS is providing flow to the RCS, and the steam generators are removing heat. These actions will ensure adequate core cooling and eventually a subcooled condition will be achieved. Once RCS pressure and temperature are adequately reduced, the shutdown cooling system is placed in operation. In the event that the feedwater supply to the steam generator is exhausted and the shutdown cooling system is inoperable, the PORVs are opened to ensure that the flow from the injection system is sufficient to cool the core. The SIS will be realigned for cold leg injection only. Core flushing is from the cold legs through the core and out the PORV.

Simultaneous hot and cold leg injection is used for both small break and large break LOCA's so the operator does not have to distinguish between them at the time when simultaneous injection is required for large breaks. (For small breaks, the boron concentration remains low due to dispersal throughout the RCS, so hot and cold leg injection is not essential).

Reactor coolant system pressure is used to differentiate between small and large break LOCA's. However, the delineation between small and large breaks does not need to be precise since there is a range of intermediate breaks for which either response will produce satisfactory results. The guidelines take this into account with the decisions to be made after eight hours.

The large break LOCA is characterized by:

a) A rapid loss of RCS pressure in 10 seconds to 3 minutes with equilibrium pressures below* 300 psia and, in the case of the largest breaks, the RCS pressure nearly equal to containment pressure.
b) Core cooling is provided for by large flow from the injection system due to low RCS pressure. The flow from the break provides sufficient heat removal. Simultaneous hot and cold leg injection is required to prevent possible boric acid accumulation in the core.

A general description of large break LOCA operations follows:

Initially, the plant is hot and pressurized. A large break LOCA results in a rapid loss of inventory and pressure. Low pressurizer pressure initiates a SIAS which automatically actuates the SIS. The reactor is tripped. Auxiliary feedwater is established to the steam generators. Steam dump is provided manually using atmospheric steam dump valves or turbine bypass valves. The major mechanism for heat removal is the flow from the SIS

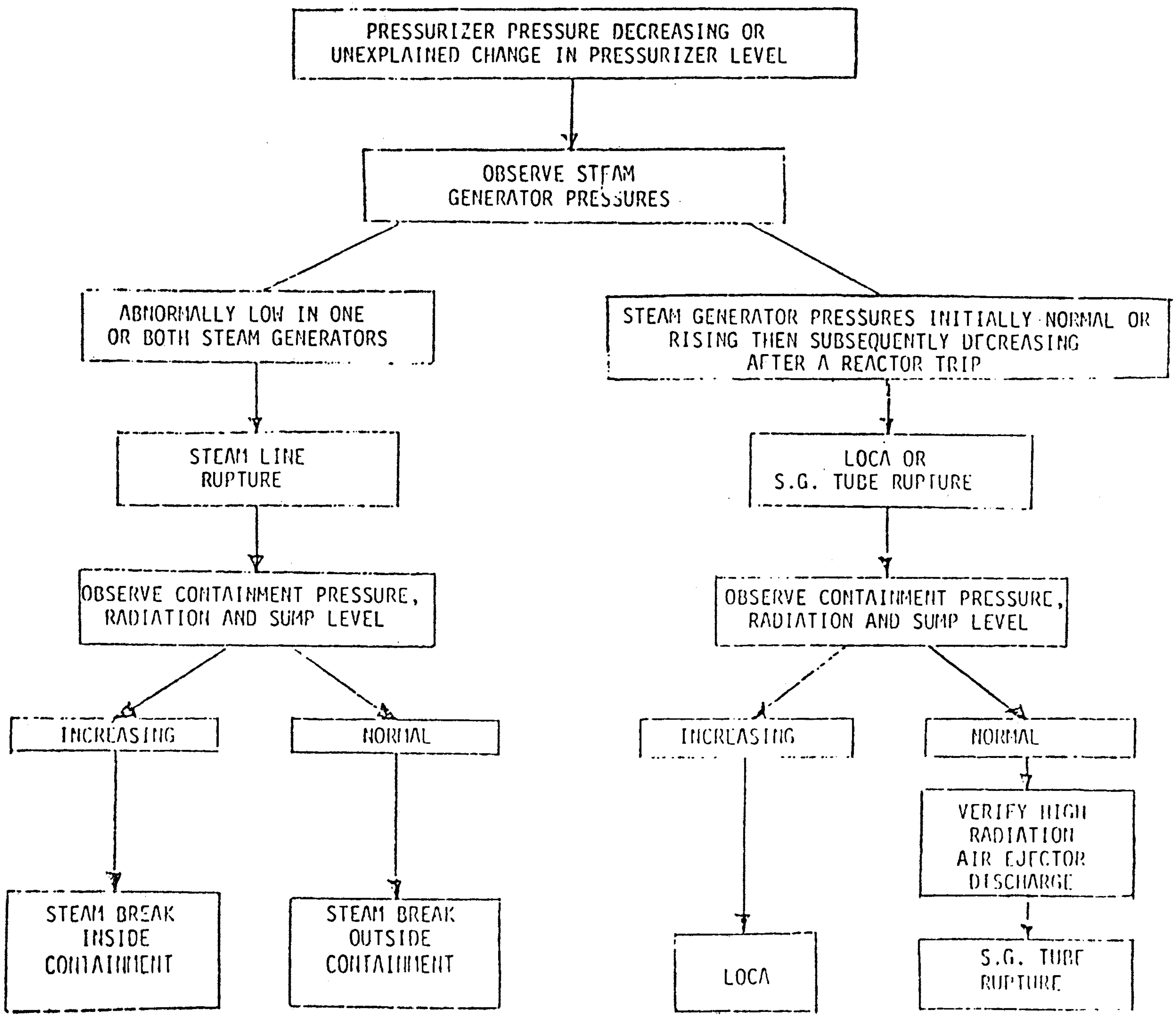
*This valve is typical, it may vary for specific designs.

through the core and out the break. Containment pressure may be high and containment isolation is likely. Containment spray may have been automatically activated.

The SIS is aligned to provide simultaneous hot and cold leg injection which is sufficient to cool the core and flush the reactor vessel indefinitely. For both large and small break LOCA's, continued monitoring of conditions in the RCS and performance of safety systems should be done. All available indications should be used to aid in diagnosing the event since the accident may cause irregularities in a particular instrument reading.

Regardless of the cause of actuation of a safety system, the automatic response should not be altered until it has been demonstrated that other systems and equipment are providing the functions that the safety system is intended to perform.

BREAK IDENTIFICATION



Guidelines for Operating Plants Following LOCA's

Symptoms

1. Reactor coolant system leak exceeds the capacity of the operable charging pumps.
2. A reactor trip may have occurred.
3. The Safety Injection System (SIS) may have automatically actuated.
4. Any one or more of the following indications or alarms may be present.
   a) Low pressurizer pressure
   b) High containment pressure or temperature
   c) High containment sump level
   d) High containment radiation
   e) High or low pressurizer level
   f) High quench tank level
   g) High quench tank temperature
   h) High quench tank pressure
   i) T_{av} decreasing or at saturation temperature for RCS pressure.

Immediate Actions

1. Trip the reactor if not already tripped and carry out standard post trip actions.
2. Initiate safety injection if it has not already been actuated by the safety injection actuation signal.
3. After an SIAS caused by low reactor coolant system pressure and after it has been verified that all rods have been fully inserted for 5 seconds, stop all operating reactor coolant pumps.
4. If main feedwater is not available, immediately establish or verify an auxiliary feedwater flow of *gpm.
5. If the containment isolation actuation signal (CIAS) is activated, ensure that the system has properly actuated.
6. Ensure that the systems receiving an SIAS are properly actuated and that CIAS is actuated.
7. After any SIAS, operate the SIS** until RCS hot and cold leg temperatures are at least 50^{o}F below saturation temperature for the RCS pressure and a pressurizer level is indicated, unless the cause of the SIAS has been verified to be an inadvertent actuation. If 50^{o}F subcooling cannot be maintained after the system has been stopped, the high pressure injection system must be restarted.

6. Attempt to locate and isolate the source of the leak. Possible leak locations include, but are not limited to the PORV's, the letdown line and sample lines.

Follow-Up Actions

1. Operate atmospheric steam dump valves (or turbine bypass valves if the condenser is available) to maintain or reduce plant temperature and reduce steam generator pressure below the steam generator relief valve setpoints. Begin a plant cooldown as soon as possible and in any case within 1 hour.
2. Manually align the safety injection and charging systems to provide flow to the RCS hot and cold legs* two hours after the LOCA**
3. If the pressure and inventory control with the SIS cannot be established after* eight hours and RCS pressure is less than* 300 psig, continue the hot and cold leg injection.
4. If pressure and inventory control with the SIS are established after* eight hours and RCS pressure is greater than* 300 psig, conduct one of the following activities. The activities are listed in order of decreasing preference.

   a) RCS pressure above* 300 psig indicates that the system has refilled and subcooling has occurred. Verify this by checking the saturation pressure for the existing temperature. Realign the SIS for cold leg injection. Continue to maintain subcooling and reduce RCS pressure to the initiation pressure for shutdown cooling by reducing the flow delivered by the high pressure injection and charging pumps and by venting or isolating the safety injection tanks as necessary. While reducing pressure and after shutdown cooling is initiated, maintain RCS pressure with the charging pumps and/or the HPSI pumps to continue to maintain at least 50^{0} subcooling, or

   b) Continue to remove decay heat using emergency feed and steam dump if adequate condensate is available and (a) cannot be implemented, or
   c) Open pressurizer power operated relief valves and align the SIS for cold leg injection if (a) or (b) cannot be implemented.

* This value is typical, it may vary for specific designs.

** Includes stopping charging pumps on some plants

Precautions

1. Before restarting RCP's ensure that cooling water services to the pumps has been restored.

2. Pressurizer level may not always be a true indicator of RCS fluid inventory. Pressurizer steam space ruptures, reference leg failures, and reference leg flashing may cause indications which are contrary to true conditions.

3. All available indications should be used to aid in diagnosing the event since the accident may cause irregularities in a particular instrument reading. Critical parameters must be verified when one or more confirmatory indications are available.

4. When extablishing auxiliary feedwater flow to the steam generators, monitor primary system temperature and pressure to avoid exceeding a 100°F/hour cooldown rate.

5. Feedwater is normally provided to both steam generators. Isolation of a single steam generator is mandatory if a steam generator tube rupture is detected in that generator to prevent lifting of the safety valves or reseat them if they have lifted. This action will also reduce the amount of radioactivity released. For small breaks in the RCS where steam generators are important for heat removal one steam generator must be used for this purpose even if primary to secondary leaks are detected.

6. Continued lengthy operation of the containment spray may jeopardize the operation of equipment which would be desirable or necessary to mitigate the consequences of the event. Early consideration should be given to termination of spray operation. If the containment pressure has returned to below the actuation setpoint, the system may be stopped. The system should be realigned for automatic actuation.

7. Observe all available indications to determine conditions within the RCS. Use RCS hot leg temperature, RCS cold leg temperature, core exit thermocouple temperature, and RCS pressure to determine if the RCS is subcooled or saturated. An increase in temperature above the saturation temperature for the existing pressure is an indication of voiding in the RCS. A decrease in operating RCP motor current or erratic pump ΔP is also an indication of voiding. If this occurs the operator must ensure that the RCP's are turned off, the SIS is providing makeup to the RCS, and that the steam generators are removing heat from the RCS.

8. Monitor refueling water tank level to verify the shift from injection to recirculation. If a recirculation actuation signal (RAS) occurs, the operator must prevent the HPSI pumps from operating at less than minimum flow conditions. If all HPSI pumps and charging pumps are operating and the HPSI pumps are delivering less than 30 gpm per pump, turn off the charging pumps one at a time and then HPSI pumps one at a time until only one HPSI pump remains operating. This will ensure that minimum flow requirements will be met by the flow through the pump to the RCS for the smallest break size that results in a SIAS.

9. Monitor the auxiliary building radiation levels and sump levels after an RAS to attempt to detect leakage from the SIS. Even if leaks are detected at least one high pressure safety injection pump must remain in operation to provide flow to the RCS.

10. If there is a high radioactivity level in the reactor coolant system, circulation of this fluid in the SCS may result in high area radioactivity readings in the auxiliary building. The activity level of the RCS should be determined prior to initiating SCS flow.

11. Minimum Pressure - Temperature operating restrictions take precedence over requirements for operation of the high pressure injection or charging system to achieve 50^{o} subcooling during operation of the shutdown cooling system.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 26 1979

Mr. Thomas D. Keenan, Chairman
General Electric Boiling Water Reactor Owners' Group
Vermont Yankee Nuclear Power Corporation
Seventy-Seven Grove Street
Rutland, Vermont 05701

Dear Mr. Keenan:

SUBJECT: EVALUATION OF SMALL-BREAK LOSS-OF-COOLANT ACCIDENT OPERATOR GUIDELINES

We have completed our review of the small-break loss-of-coolant accident operator guidelines contained in Section 3.1.1.2 of General Electric Company Report NEDO-24708. Our evaluation of these guidelines is documented in the enclosure.

As stated in our evaluation, we have concluded that these guidelines, as modified in accordance with your letters dated October 18, 1979 and October 23, 1979, are acceptable. We request, however, that you submit for our confirmation the modified guidelines by November 16, 1979. Following our receipt of these guidelines, we plan to acknowledge, by letter, that the guidelines have been modified in accordance with our agreements.

All licensees of General Electric Company boiling water reactor plants may now proceed with their development of small-break loss-of-coolant accident emergency procedures and operator training based on the modified guidelines. As indicated on Page 5 of Enclosure 6 to Darrell G. Eisenhut's letter to all operating nuclear power plants, implementation of these procedures and operator training are to be completed by December 31, 1979.

We understand that your present plans are to modify the procedures in two phases. The first, or short term, phase, will produce small-break loss-of-coolant procedures that ensure that all considerations that underlie the guidelines are in fact considered in the procedures. The second, long-term, phase will:

(i) incorporate a new procedure philosophy,

(ii) require restructuring and reformatting of many procedures;

(iii) start when guidelines for other, non-loss-of-coolant accident events are approved; and,

(iv) be in place in the spring-summer time span of 1980.

As this long-term approach involves implementing guidelines from the General Electric Company for the first time, we agree that more time is needed, and your schedule seems consistent with the schedule on Page 5 of Enclosure 6 to Darrell G. Eisenhut's letter mentioned above. It is important to provide in the phase I procedure revisions to all of the substantial considerations that have resulted from the analysis efforts of NEDO-24708. As long as this is done, it is acceptable to defer the restructuring and reformatting to phase II. We request that when you submit for our confirmation the modified guidelines, you also confirm the accuracy of our understanding.

As part of our audit program, we expect to examine the procedures of lead plants in several of the classes of boiling water reactors to assure that they have been developed in accordance with the approved guidelines. We also plan to check out some of the procedures at a boiling water reactor simulator, on a schedule to be developed later. It should be noted however, that our audit program need not impede progress towards implementation of approved procedures and associated training by December 31, 1979.

Sincerely,

Original signed by:

D. F. Ross, Jr., Director
Bulletins and Orders Task Force

Enclosure:
As stated

EVALUATION OF GE BWR OPERATOR GUIDELINES

FOR SMALL BREAK LOCA'S

Introduction

By letter dated July 13, 1979, we transmitted a formal request for additional information required by the Analysis Group of the Bulletins and Orders Task Force. Specifically, item (12) of Enclosure 2 to this request required the preparation of guidelines or revised emergency procedures for the recovery of plants following small LOCA's. This was to include both short-term and long-term situations with follow-through to a stable condition. The guidelines were to also include recognition of the event, precautions, actions, and prohibited actions. By letter dated August 17, 1979, the BWR Owners Group transmitted General Electric Company Report NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," which provided the information requested.

Summary Description - GE BWR Operator Guidelines

General Electric, on behalf of the BWR Owners Group, has developed a set of operator guidelines based on the design of various reactor systems and the analysis derived from Appendix K LOCA models. The guidelines define operator actions following a loss of coolant accident inside and outside the primary containment. Three guidelines have been provided with each guideline having four major headings:

1. Purpose--describe the intent or objective of the guideline.

2. Symptoms and Automatic Actions--The "Symptoms" are process variable indications or alarms which the operator is expected to see or hear in the control room. "Automatic Actions" are actions taken by the plant protective instrumentation and associated systems without assistance from the operator.

3. Immediate Operator Actions--"Immediate Operator Actions" are actions the operator takes as soon as possible to protect the core and to reduce the loss of primary inventory. This section also includes immediate verification

that automatic actions have been correctly performed by checking multiple indications which should change as a result of the automatic actions.

4. Subsequent Operator Actions--"Subsequent Operator Actions" are taken by the operator after Immediate Actions to bring the plant to a stable condition. These actions are sometimes called follow-up or long-term actions.

Within each heading, GE has made an effort to list the more important items first and the less important items last. Also, "Caution" statements have been added within some headings advising the operator not to take certain actions which might increase the severity of the consequences or to be aware of key indications during the course of the accident.

Small Break Accident Guideline #1 (SBA-1), "Pipe Break Diagnosis," provides the objectives an operator is to achieve in the event of a pipe rupture and the guidelines to be used in distinguishing between small pipe breaks inside the primary containment and outside the primary containment. SBA-1 also lists six Cautions which are applicable to the specific guidelines for inside and outside pipe breaks. The operator checks SBA-1 for an entry point in determining which guideline, SBA-2 or SBA-3, is used.

Small Break Accident Guideline #2 (SBA-2), "Pipe Break Inside Primary Containment," describes the events to be expected and the operator actions required to bring the reactor and containment to a controlled, stable condition after a pipe break inside containment. Symptoms provided in the guideline enable the operator to determine that a pipe break inside containment has occurred. Immediate and Subsequent Actions are followed by the operator to assure a prompt safety response and maintenance of stable conditions. A "Contingency" procedure is also provided in the event that vessel level cannot be maintained with the high pressure system.

Small Break Accident Guideline #3 (SBA-3), "Pipe Break Outside Primary Containment," is very similar to SBA-2 except its Symptoms and Actions are changed to reflect the difference in break location and two additional contingency procedures are provided.

Conclusion

The staff provided the Owners Group, via conference calls on October 17, 1979 and October 22, 1979, its comments on the Small Break LOCA Guidelines. Each of these comments was resolved to the mutual satisfaction of the BWR Owners Group and the staff. The resulting Owners Group commitments were documented in letters dated October 18, 1979 and October 23, 1979.

Based on our review of the information provided by the GE BWR Owners Group, we conclude that the GE BWR operator guidelines for small break LOCAs in NEDO-24708, as modified in accordance with the Owners Group's letters dated October 18, 1979 and October 23, 1979 are acceptable for use in developing operating procedures for GE BWRs.

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UNITED STATES
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NOV 2 8 1979

Mr. Thomas D. Keenan, Chairman
General Electric Boiling Water Reactors
Owner's Group
Vermont Yankee Nuclear Power Corporation
Seventy-Seven Grove Street
Rutland, Vermont 05701

Dear Mr. Keenan:

SUBJECT: EVALUATION OF SMALL-BREAK LOSS-OF-COOLANT ACCIDENT GUIDELINES

By letter dated October 26, 1979, subject as above, we advised you that the small-break loss-of-coolant accident operator guidelines contained in Section 3.1.1.2 of General Electric Company Report NEDO-24708 as modified in accordance with your letters dated October 18, 1979 and October 23, 1979 were acceptable. We requested, however, that you submit for our confirmation the modified guidelines by November 16, 1979. We further advised you that upon our receipt of these guidelines, we would acknowledge by letter that the guidelines had been modified in accordance with our agreements.

By letter dated November 16, 1979, you submitted for our confirmation the modified guidelines, a copy of which is enclosed. We have completed our review of these guidelines, have determined that they have been modified in accordance with our agreements and conclude, therefore, that they are acceptable. As indicated on Page 5 of Enclosure 6 to Darrell G. Eisenhut's September 13, 1979 letter to all operating nuclear power plants, implementation of the small-break loss-of-coolant accident emergency procedures and operator retraining based on these guidelines must be completed by December 31 1979.

Sincerely,
Original signed by
D. F. Ross

D. F. Ross, Jr., Director
Bulletins and Orders Task Force

Enclosure:
As stated

3.1.1.2 Operator Guidelines

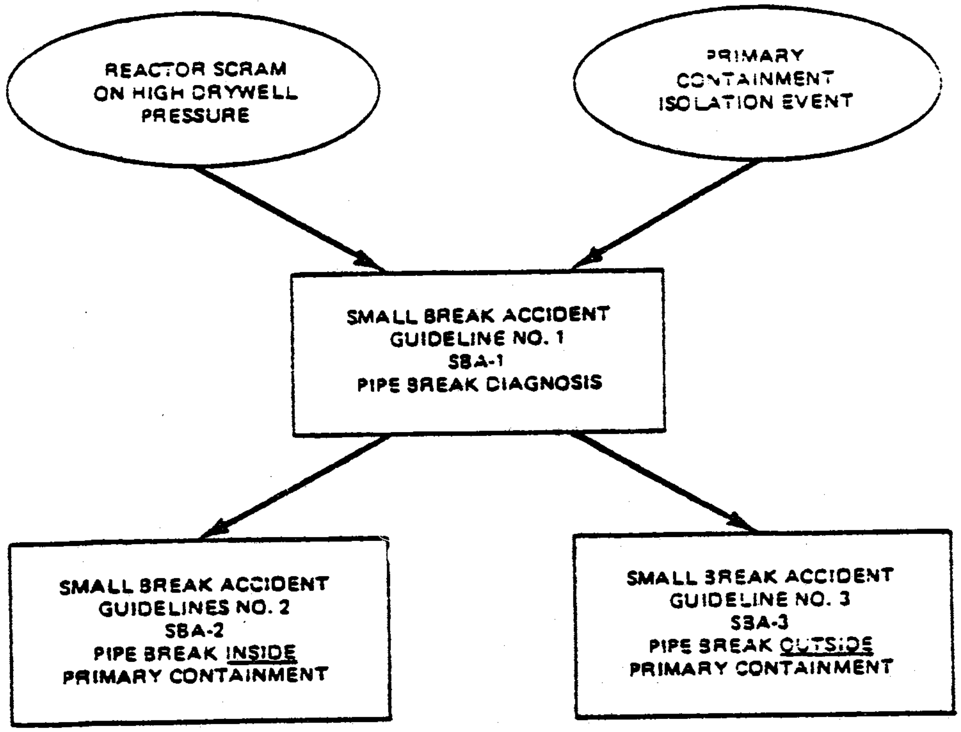
Introduction

Based on the Analyses discussed in Sections 3.1.1.1 and the design of the various reactor systems, a set of operator-guidelines has been developed. These guidelines define operator actions following a loss of coolant accident at rates large enough to cause one or both of two automatic actions: a) reactor scram on high drywell pressure, and/or b) initiation of the primary containment isolation system. It should be noted that a small pipe break inside the primary containment will cause a reactor scram from high drywell pressure. Losses of coolant at lesser rates are considered leaks, instead of breaks, and are not covered in these guidelines. Technical Specifications and existing utility procedures require the operator to shut down the plant whenever: a) total leakage exceeds [25]* gpm, and b) unidentified leakage exceeds [5] gpm. (A 25 gpm leak corresponds to a liquid break size of 0.0005 ft^2, or a circular hole of diameter 0.3 inch.)

After the break occurs, plant protective instrumentation will sense the break, and cause either a scram and/or an isolation. If the operator sees the break symptoms before automatic scram and/or isolation, and manually initiates scram and/or isolation, so much the better. These guidelines would not be written differently, except perhaps to note that the scram and/or isolation was manual instead of automatic.

* [ ] indicates plant specific value.

The following section has three guidelines, organized as follows:



The first guideline, SBA-1, is diagnostic and provides an entry point for the second and third guidelines, SBA-2 and SBA-3. Guidelines SBA-2 and SBA-3 contain specific recommendations. They each have 4 major headings:

1. <u>Purpose</u>

2. <u>Symptoms and Automatic Actions</u>
   "Symptoms" are process variable indications or alarms which the operator is expected to see or hear in the control room.

   "Automatic Actions" are actions taken by the plant protective instrumentation and associated systems without assistance from the operator.

3. Immediate Operator Actions

"Immediate Operator Actions" are actions the operator takes as soon as possible to protect the core. The goals of Immediate Operator Actions are to reduce the loss of primary inventory as quickly as possible, and to minimize the immediate release of radioactivity outside the containment. Immediate Operator Actions include, as first actions, the verification of Automatic Actions, and are done before any further manual actions are taken. Verification means the operator confirms that the Automatic Actions have been correctly performed by checking multiple indications which should change as a result of the Automatic Action. For example, the operator verifies reactor scram by noting all control rod position lights show rods fully inserted, and that the neutron flux indicators show decaying flux. As a second example, the operator verifies HPCI initiation by noting indications on flow, turbine rpm, valve positions, pump discharge pressure, etc., in addition to the annunciator that signals HPCI start.

As a third example, the operator manually initiates RCIC if RCIC does not automatically start on low vessel level.

4. Subsequent Operator Actions

"Subsequent Operator Actions" are actions the operator should take after the Immediate Operator Actions. The goal of the Subsequent Operator Actions is to bring the plant to a stable condition, where the vessel water level is steady or cycling within a satisfactory range, and containment cooling has been established.

Within these major headings, an effort has been made to list the more important items first and the less important items last. Plant specific values are enclosed in brackets [ ]. Utility-prepared procedures should contain the specific alarm window numbers, setpoint values, panel locations, notification instructions, valve numbers, and reference to other applicable accident or emergency procedures.

Also within some of the headings, caution statements have been added. "Cautions" are advice to the operator not to take certain actions which might increase the severity of the consequences, or to be aware of key indications during the course of the accident. Cautions are included in the guidelines where appropriate.

The guidelines contained within this report are generic to all GE-BWR's in that they include all systems which may be used to mitigate the consequences of a small break accident, (i.e. HPCI, RCIC, LPCS, LPCI, CRD, and ADS). Because any specific plant may not include all of the above systems, care must be exercised by the individual plant operator when applying these guidelines. The guidelines will be applied to individual plants by either not considering statements from the guidelines which are not applicable, or by substituting corresponding systems. For example, plants with no LPCI will not consider statements referring to LPCI, and plants with isolation condensers will substitute IC for RCIC. In this manner the guidelines apply to all plants.

All systems function normally, including feedwater and condensate systems, off-site power, instrument air, control rod drive pumps, and isolation valves. Degraded cases (such as loss of high pressure systems or failure of valves to close) are considered under contingencies. Contingencies for failed equipment are contained within the guidelines; separate guidelines are not provided. However, it is expected that the current utility procedures concerning equipment out of service will be referenced where applicable.

SMALL BREAK ACCIDENT GUIDELINE #1

SBA-1

PIPE BREAK DIAGNOSIS

PURPOSE

There are four basic objectives the operator is to achieve in the event of a pipe break, with respect to the core and containment:

a. Maintain core cooling to prevent excessive cladding heatup and oxidation;

b. Limit the release of offsite radiation by maintaining the integrity of the primary and secondary containments;

c. Place the reactor in a safe, stable condition;

d. Keep the pool bulk temperature below [°F] to prevent excessive loads to the pool boundary and structures during safety/relief valve discharges, and maintain peak allowable temperatures within cooling equipment and containment structure design limits.

This guideline, SBA-1, provides the entry point for the two following guidelines, which include specific recommendations depending on the break location:

Small Break Accident Guideline SBA-2: Pipe Break Inside Primary Containment
Small Break Accident Guideline SBA-3: Pipe Break Outside Primary Containment

Guidance is provided to the operator in diagnosing the symptoms displayed in the control room so that he may distinguish between small pipe breaks inside the primary containment and small pipe breaks outside the primary containment, and to select the appropriate guideline.

GENERAL OPERATOR AWARENESS ITEMS

This section provides "cautions" which are common to both guidelines SBA-1 and SBA-3. These items must be kept in mind at all times during the course of a pipe break accident.

CAUTION #1

Operators should be prepared to take immediate actions as necessary to protect the core and containment. Immediate Operator Actions include verification of Automatic Actions and taking manual action to initiate an automatic function whenever an Automatic Action did not occur that should have occurred.

CAUTION #2

Continuously monitor vessel level and pressure from multiple indications.

CAUTION #3

On any automatic initiation of a safety function, assume a true initiating event has occurred, until otherwise confirmed by two or more independent process indications.

CAUTION #4

Automatic controls should not be placed in MANUAL mode, unless 1) misoperation in AUTOMATIC mode is confirmed by at least two independent process parameter indications; or 2) core cooling is assured, and these guidelines state specifically to do otherwise. When manual operation is no longer needed, restore the system to AUTOMATIC/STANDBY mode, if possible.

CAUTION #5

Any emergency core cooling system should not be shut off unless there are multiple confirming process parameter indications (such as level indications from several instruments) that the core and containment are in a safe, stable condition.

CAUTION #6

If any system is switched from AUTOMATIC to MANUAL mode, then frequent checks of the controlled parameter must be made.

ENTRY POINT EVENTS

1. If a reactor scram occurs from a high drywell pressure signal, then go to SBA-2, "Pipe Break Inside Primary Containment".

2. If any one or more of the primary piping isolation valve groups isolates, then go to SBA-3, "Pipe Break Outside Primary Containment".

SMALL BREAK ACCIDENT GUIDELINE #2

SBA-2

PIPE BREAK INSIDE PRIMARY CONTAINMENT

1. PURPOSE

The intent of this guideline is to assure that the normal water makeup systems, the Emergency Core Cooling Systems (ECCS), and containment cooling systems operate as designed to protect the core and containment in the event of a Small Break Accident (SBA) inside the primary containment. In the guideline are described events to be expected and operator actions which are required to bring the reactor and containment to a controlled, stable condition. Also included in this guideline are the operator actions following loss of the high pressure water make-up systems.

2. SYMPTOMS AND AUTOMATIC ACTIONS

The symptoms and automatic actions which are displayed in the control room in the event of a pipe break inside the primary containment are grouped below. The symptoms observed will depend upon the severity of the accident. For smaller breaks, only a few symptoms may be observed. For a larger break, more symptoms would be expected.

2.1 Symptoms

2.1.1 Drywell atmosphere symptoms

- High drywell pressure alarm [1.5 psig]
- High temperature
- High humidity
- High radiation

2.1.2 Drywell Sump Symptoms

- High or high-high levels
- High integrator readings
- High sump temperature
- Excessive sump pump operation

2.1.3 Other Symptoms

- Generator load decrease
- Steam flow/feed flow mismatch

2.2 Automatic Actions

2.2.1 ECCS Actuations [2.0 psig]

- Emergency diesel-generators start
- HPCI starts and injects into the vessel [when vessel pressure is greater than the low pressure isolation setpoint of 100 psig]
- LPCI pumps start
- LPCS pumps start
- ADS high drywell pressure permissive

2.2.2 Other Automatic Actions

- Reactor scram [2.0 psig]
- Standby Gas Treatment System initiates [2.0 psig]
- Valve group [2] isolates [RHR shutdown cooling, drywell sumps, TIP system] [2.0 psig]
- Valve group [6] isolates [primary containment atmospheric control systems] [2.0 psig]
- Containment spray permissive [2.0 psig]

3. IMMEDIATE OPERATOR ACTIONS

3.1 Any Automatic Actions listed in Section 2. above which should have initiated must be verified, preferably by at least two independent indications, or manually initiated if the Automatic Action did not occur.

*	Scram is verified by noting that control rod position lights show rods fully inserted, and that neutron flux indicators show decaying flux.	*

3.2 Take the reactor MODE switch out of RUN to prevent MSIV from closing on low vessel pressure [850 psig].

3.3 Continuously monitor vessel water level using all available instrumentation.

3.3.1 Narrow range level control indicators [instrument type, #s]

3.3.2 Wide range safety trip indicators [instrument type, #s]

3.3.3 Fuel zone indicators [instrument type, #s]

3.3.4 Refueling zone indicators [instrument type, #s]

CAUTION #7

The indicated water level, where provided by Yarway instrumentation utilizing reference legs in the drywell, is dependent upon drywell temperature. Very large increases in drywell temperature (an increase from 135°F to 340°F) could result in a level inaccuracy (as much as [28] inches depending on drywell temperature and type of instrument) with indicated level being higher than actual level.

During rapid reactor depressurization (with ADS operation for example), and particularly below 500 psig, the operator should utilize the cold reference leg type of level indicators (such as operating range and fuel zone indication) to give backup information on vessel water level. The operator should not turn off any ECCS unless there is sufficient confirming information from cold reference leg level instruments that vessel water level has been restored. The operator should not rely on the Yarways if erratic behavior, indicative of reference leg flashing, has occurred until the Yarway readings are on scale and in reasonable agreement with other (cold reference leg) types of level instruments. The operator should verify that automatic ECCS actuations occur when the levels are at the trip points (adjusted in accordance with each plant's individually verified recommendations). The operator should be prepared to manually actuate ECCS during a suspected LOCA if automatic actuation is not achieved.

3.4 Control vessel level with available high pressure systems (feedwater, control rod drive pumps, HPCI, RCIC). When the level approaches the high level [8] trip [+58 inches] for feedwater, HPCI, and RCIC, take manual control (if possible) of the high pressure systems to maintain level, and to prevent trips of feedwater, HPCI, and RCIC. Restore to AUTOMATIC/STANDBY mode (if possible) the systems which are not needed to maintain level. Make frequent checks of level when systems are in the MANUAL mode.

* If high pressure systems are unable to maintain level, then go to CONTINGENCY #1. *

CAUTION #8

If signals of high pool level or low condensate storage tank level occur, then manually transfer RCIC suction from the condensate storage tank to the pool, and verify automatic transfer of suction for HPCI.

CAUTION #9

Do not throttle HPCI and RCIC systems below turbine speeds which yield acceptable continuous operation.

3.5 If the vessel pressure falls below the shutoff head of the low pressure systems (condensate [300 psig], LPCS [300 psig], and LPCI [300 psig]), confirm that these systems inject into the vessel, and that the water level responds accordingly.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 Continue to monitor and control vessel level. As vessel pressure decreases, maintain level with low pressure systems in the following order of preference: condensate, LPCS, then LPCI. Manually control flow to prevent water from flooding the steam lines. Shut off or direct to other cooling modes low pressure systems not needed to maintain level. One low pressure system must be dedicated to maintaining level.

CAUTION #10

If vessel pressure increases above the shutoff head of the low pressure systems being used to maintain level, then depressurize the vessel in the following order of preference: 1) condenser; 2) one or more SRVs to the pool (rotate use of SRVs to distribute heat uniformly to the pool); or 3) manual ADS initiation.

4.2 Continuously monitor and control pool temperature to keep the bulk pool temperature from exceeding [°F].

4.2.1 Re-establish main condenser as a heat sink, if possible.

CAUTION #11

Assure main steam lines are drained before opening main steam isolation valves.

4.2.2 As soon as the pool temperature exceeds the service water temperature, use the RHR pumps (if they are not needed in the LPCI mode for maintaining vessel level) in the normal pool cooling mode.

CAUTION #12

If vessel level cannot be maintained, then do not divert RHR pumps from the LPCI mode.

4.2.3 If the pool temperature reaches [°F], then manually depressurize the vessel to below [130 psig] using one or more SRVs. Rotate use of SRVs to distribute heat uniformly within the pool.

4.3 Monitor and control containment pressure to keep pressure below design limit [62 psig].

4.3.1 If possible spray the <u>pool</u> airspace when the <u>drywell</u> pressure exceeds [2.0 psig].

4.3.2 <u>If</u> the drywell pressure exceeds [35 psig] for [10 minutes], <u>then</u> spray the drywell until pressure is reduced to [25 psig].

<u>CAUTION #13</u>

Do <u>not</u> operate recirculation pumps when spraying the drywell. Drywell spraying may put recirculation pumps out of service.

4.4 Follow procedures for post-LOCA containment venting and hydrogen control [procedure #s].

CONTINGENCY #1

INABILITY TO MAINTAIN LEVEL WITH HIGH PRESSURE SYSTEMS*

If the operator determines that vessel level cannot be maintained by the high pressure systems, then the operator must verify automatic operation of the ADS on low level [1 plus 120 seconds preset time delay]. While waiting for the Automatic Actions to occur on decreasing vessel level, the operator should make all attempts to start the high pressure systems and regain level before the low level [1 plus 120 seconds preset time delay] is reached.

> CAUTION #14
>
> If the ADS does not initiate automatically on low level [1] [-146 inches], then manually initiate ADS (operator opens the same valves that the ADS logic would open automatically). Do not manually initiate ADS unless it is confirmed that at least one low pressure pump [condensate, LPCS, LPCI] is running. If the operator is unable to manually initiate ADS, then the operator must manually open other safety/relief valves. As many valves as possible, up to the number used for ADS, should be opened.

The operator should also verify those Automatic Actions which occur on decreasing level, and have not already occurred on high drywell pressure.

When the low level [1] [-146 inches] is reached, the operator should confirm that the ADS timer begins and that the proper valves open [120] seconds after the timer begins, if possible.

> CAUTION #15
>
> Do not block or defeat the ADS sequence by resetting the ADS timer, unless vessel level can be maintained.

After ADS actuates, return to Guideline SBA-2, Section 4, "Subsequent Operator Actions".

*Includes failure of high pressure systems

SMALL BREAK ACCIDENT GUIDELINE #3
SBA-3
PIPE BREAK OUTSIDE PRIMARY CONTAINMENT

1. PURPOSE

The intent of this guideline is to assure that the primary containment isolation system operates as designed in the event of a Small Break Accident (SBA) outside the primary containment. If the break occurs in a pipe which can be automatically isolated, neither reactor scram nor reactor isolation may be necessary. Technical Specifications address how long the plant can operate before the isolated system must be restored to service.

2. SYMPTOMS AND AUTOMATIC ACTIONS

The symptoms and automatic actions which are displayed in the control room in the event of a pipe break outside primary containment are listed below. The specific symptoms observed will depend upon the location and size of the break.

2.1 Primary Containment Isolation

There are [7] isolation valve groups, each associated with a system connected to the primary coolant outside of primary containment. One or more of these valve groups will isolate on system signals of high flow, low vessel water level, high radiation, high area temperature, high drywell pressure, low system pressure, etc., which are indicative of a pipe break outside of primary containment. [Utility may provide more specific information on the valve groups and Automatic Actions].

2.2 Other Symptoms and Automatic Actions

2.2.1 Symptoms

[Excess flow check valves actuation and alarm]

- Reactor building high radiation at exhaust vent [ ]
- Area radiation monitor alarm [ ]
- Decreasing hotwell level or condensate storage tank level
- Mismatch between steam and feed flow
- Decrease in generator output (MWe)
- Increase in reactor power (MWt)

2.2.2 Automatic Actions

- SBGTS initiation [ ]
- Reactor building isolation [ ]

3. IMMEDIATE OPERATOR ACTIONS

3.1 Any Automatic Actions listed in Section 2. above which should have been initiated must be verified, preferably by at least two independent indications, or manually initiated if the Automatic Action did not occur.

* If scram occurs, scram is to be verified by noting that all control rod position lights show rods fully inserted, and that neutron flux indicators show decaying flux. *

CAUTION #16

If an automatic isolation of a particular system has occurred, do not attempt to de-isolate or restore the system until all available indications have been checked and are found to be normal.

For automatic isolation of a particular valve group, verify all valves in the group are closed by valve position indication and by noting confirming process variables (such as zero flow). When all isolation valves in the suspected broken system are closed, note that area symptoms of high temperature and radiation decrease.

* If there is a failure to completely isolate a suspected broken system, then go to CONTINGENCY #2. *

* If a reactor scram occurs on high drywell pressure, then go to SEA-1 "Pipe Break Inside Primary Containment". *

3.2 Continuously monitor vessel water level using all available instrumentation (Repeat 3.3.1 through 3.3.4 in SBA-2.)

3.3 Control vessel level with available high pressure systems (feedwater, control rod drive pumps, HPCI, RCIC). When the level approaches the high level [8] trip [+58 inches] for feedwater, HPCI, and RCIC, take manual control (if possible) of the high pressure systems to maintain level, and to prevent trips of feedwater, HPCI, and RCIC. Restore to AUTOMATIC, STANDBY mode (if possible) the systems which are not needed to maintain level. Make frequent checks of level when systems are in the MANUAL mode.

* If high pressure systems are unable to maintain level, then go to CONTINGENCY #3. *

CAUTION #9

Do not throttle HPCI and RCIC systems below turbine speeds which yield acceptable continuous operation.

4. SUBSEQUENT OPERATOR ACTIONS

4.1 If the break is successfully isolated, and reactor operation is unaffected, then continue normal operation per the applicable Technical Specification and procedures for equipment out of service.

4.2 If the isolation resulted in a reactor isolation and scram, then follow applicable procedures for scram/isolation recovery [procedure #s].

4.3 If vessel depressurization is necessary, then go to guidelines in SBA-2, Section 4.2.

CONTINGENCY #1

FAILURE TO ISOLATE

If the isolation valves fail to isolate the break on either automatic or manual signal, then scram the reactor. Attempt to close the failed isolation valve manually, or to isolate the suspected broken pipe with other valves. Initiate the SBGTS, isolate and evacuate the reactor building. Go to Guideline SBA-3, Section 3.2.

CAUTION #17

Do not dispatch personnel to an area of a suspected pipe break without consideration of adverse environments.

CONTINGENCY #3

INABILITY TO MAINTAIN LEVEL WITH HIGH PRESSURE SYSTEMS*

If the operator determines that vessel level cannot be maintained by the high pressure systems, then the operator must manually initiate ADS (operator opens the same valves that the ADS logic would open automatically) when the level reaches level [1]**

The operator should make all attempts to start the high pressure systems and regain level before manually initiating ADS.

CAUTION #18
Do not manually initiate ADS unless it is confirmed that at least one low pressure pump [condensate, LPCS, LPCI] is running.

If the operator is unable to manually initiate ADS, then the operator must manually open other safety/relief valves. As many valves as possible, up to the number used for ADS, should be opened.

After ADS is manually initiated go to Guideline SBA-2, Section 4, "Subsequent Operator Actions".

* includes failure of high pressure systems

** The preset time delay should be included for plants whose ADS level permissive is at the same level as the high-pressure ECCS initiation trips.

Conclusions

The guidelines reflect the conclusions of the analysis Section 3.1.1, which are that there is only one immediate operator action needed to maintain core cooling, and it is needed only for a degraded condition. In almost all cases the operator actions are aimed at keeping the vessel from overfilling, or to minimize containment heat loads. The key operator actions are summarized below for breaks inside and outside the primary containment.

For Breaks Inside the Containment:

Immediate Operator Actions

1. Verify automatic actions
2. Take reactor MODE switch out of RUN
3. Monitor water level
4. Control high pressure systems to prevent overfilling the vessel
5. Verify low pressure system injection

Subsequent Operator Actions

1. Control low pressure systems to prevent overfilling the vessel
2. Keep the pool cool by using the main condenser or RHR as heat sinks, or if necessary, depressurize the vessel to prevent later SRV discharge into a hot pool
3. Control containment pressure
4. Follow post-LOCA hydrogen control procedures

For Breaks Outside the Primary Containment:

Immediate Operator Actions are the same as for breaks inside the primary containment except that 1) no action is required regarding the MODE switch, and 2) no low pressure systems are expected to initiate.

Subsequent Operator Actions are either normal operation, isolation recovery, depressurizing the vessel for pool cooling considerations and maintaining pool cooling.

In addition if no high pressure systems are available, then the operator must manually initiate ADS.

These guidelines clearly demonstrate that the small break accident does not present a severe challenge to the BWR and that it can be automatically mitigated. Even for severely degraded conditions (no high pressure systems available) the operator has only to manually initiate ADS to decrease vessel pressure so that the low pressure systems keep the core cooled.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 5 1979

Mr. Cordell Reed, Chairman
Westinghouse Owners' Group
Commonwealth Edison Company
P. O. Box 767
One First National Plaza
Chicago, Illinois 60690

Dear Mr. Reed:

SDBJECT: EVALUATION OF SMALL-BREAK LOSS-OF-COOLANT ACCIDENT OPERATOR GUIDELINES

We have completed our review of the small-break loss-of-coolant accident operator guidelines E-0 and E-1 contained in Appendix A of Westinghouse Electric Corporation Report WCAP-9600. Our evaluation of these guidelines is documented in Enclosure 1.

As stated in our evaluation, we have concluded that the guidelines E-0 and E-1, as modified in accordance with your letters dated October 16, 1979, October 31, 1979 and November 2, 1979 are acceptable for plants having high head safety injection pumps similar to the 412 standard plant. A copy of the approved guidelines is provided in Enclosure 2 for your information and use. For the case of 4-loop, 3-loop, and 2-loop plants with nominal 1400 psi range safety injection pumps, it is our understanding that revisions to the guidelines for these plants similar to those provided for the 412 standard plant in your October 31, 1979 and November 2, 1979 letters will be submitted shortly. Based on your commitment to provide these agreed-upon revisions to the guidelines in a timely manner, we find the guidelines for these plants acceptable, pending fulfillment of this commitment.

All licensees of Westinghouse Electric Corporation pressurized water reactor plants may now proceed with their development of small-break loss-of-coolant accident emergency procedures and operator training based on the modified guidelines. As indicated on Page 5 of Enclosure 6 to Darrell G. Eisenhut's September 13, 1979 letter to all operating nuclear power plants, implementation of these procedures and operator training are to be completed by December 31, 1979.

In the implementation of these procedures, each licensee shall provide:

(1) The basis for the pressure setpoint at which the operator is to trip the reactor coolant pumps. The basis should include defining the steam generator safety valve setpoints and instrument uncertainties.

(2) The instrument uncertainties involved with the HPI termination criteria to indicate the criteria will insure subcooled conditions.

(3) Justification that the procedures for switchover from inspection to recirculation will insure that the valve realignments can be accomplished before the RWST is emptied. This justification should include instrument uncertainties and show that the pumps will be protected against operating without adequate suction pressure.

(4) Licensees with 4-loop, 3-loop, or 2-loop plants with nominal 1400 psi range safety injection pumps, should show that the pumps will not be run deadheaded when in the recirculation mode.

(5) An indication of the typicality of the analyses documented in WCAP-9600 relative to its own plant.

Licensees will also be required to implement emergency procedures covering the extended loss of all feedwater (including pressure vessel integrity considerations), and to revise emergency procedures for initiating (if necessary) and monitoring natural circulation (including provisions for plant cooldown). Such procedures will be based on guidelines which you are developing under inadequate core cooling.

As part of our audit program, we expect to examine the procedures of lead plants in several of the classes of Westinghouse-designed pressurized water reactors to assure that they have been developed in accordance with the approved guidelines. We also plan to check out some of the procedures at a Westinghouse pressurized water reactor simulator, on a schedule to be developed later. It should be noted however, that our audit program need not impede progress towards implementation of approved procedures and associated training by December 31, 1979.

Sincerely,

Original signed by:

D. F. Ross, Jr., Director
Bulletins and Orders Task Force

Enclosures:
As stated

cc: See attached lists

OFFICE	B&OTF	B&OTF	B&OTF	B&OTF	B&OTF	B&OTF
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DATE	11/01/79	11/01/79	11/2/79	11/2/79	11/2/79	11/4/79

Enclosure 1

EVALUATION OF WESTINGHOUSE OPERATOR GUIDELINES FOR SMALL BREAK LOCA

Guidelines for Emergency Procedures

A Westinghouse Interdisciplinary Task Force was formed to prepare guidelines for operators for small break loss-of-coolant accidents (LOCA's). The Task Force consisted of safety analysts, systems analysts, training personnel and other disciplines. The guidelines which were developed were reviewed and approved by the Working Group on Procedures, which is a subgroup of the Westinghouse Owners' Group.

Preliminary guidelines were submitted to the NRC staff as part of the generic report WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System." The intent of the guidelines was for each of the utilities using a Westinghouse nuclear steam supply system to revise or develop its emergency procedures for the operators to use in diagnosing and responding to a loss of reactor coolant. The reference instructions developed by Westinghouse were expanded to include all emergency events in which the Emergency Core Cooling System (ECCS) was automatically actuated. The guidelines include Immediate Actions and Diagnostics (E-0), Loss of Reactor Coolant (E-1), Loss of Secondary Coolant (E-2), and Steam Generator Tube Rupture (E-3). Only E-0 and E-1 have been reviewed by the NRC staff at this time for both the 412 Standard Plant, which has high head safety injection pumps, and other plants with nominal 1400 psi safety injection (SI) pumps. The staff will not review E-2 and E-3 until after January 1, 1980.

The philosophy of the instructions was for the operator to respond to an event in which safety injection was initiated and, following the required immediate actions, to diagnose the event and perform the necessary subsequent actions. The immediate actions consist of verifying that the automatic actions did occur. Verification, in this context, includes performing the action manually if it did not occur automatically. These actions are intended to assure that the reactor is adequately shut down, that the safety injection system is performing its design function and that auxiliary feedwater is being delivered to the steam generators as a heat sink for the core decay heat.

In the diagnostic procedure E-0, the operator assesses the event using reactor coolant system pressure as a key parameter. If the pressure falls below a specified value, he must immediately trip all the reactor coolant pumps. The primary system pressure at which the pumps will be tripped will be determined based on the secondary system pressure in the following manner:

(1) Secondary System Pressure - Based on the number and size of the secondary system safety valves, the secondary pressure will be established by determining the pressure setpoint for that valve in which the calculated steam relief is less than 60% of the valve's relief rating. If the calculated relief is greater than 60% of the rated capacity, then the next highest pressure setpoint should be used.

(2) Primary to Secondary Pressure Difference - To account for the pressure gradient needed for heat removal, pressure drop between the steam generator and safety valves, pressure drop from steam generator to measurement location, etc., the primary pressure for RCP trip should be the secondary pressure as established by (1) above plus 100 psi if the adjustments calculated are 100 psi or less. If the adjustments are determined to be greater than 100 psi, the larger value should be used.

(3) Instrument inaccuracies appropriate for that time in the accident should be added to the primary system pressure value established in (2) above. The resulting pressure is the indicated primary system pressure at which the operator should trip the reactor coolant pumps.

The action regarding reactor coolant pump trip was deemed necessary by a Westinghouse analysis of delayed reactor coolant pump trip for a limited range of small break LOCAs (WCAP-9584). If, in addition to low pressure, the condenser air ejector radiation or steam generator blowdown radiation monitor readings are abnormally high, the operator is directed to E-3, the steam generator tube rupture procedure. If the steamline pressure is abnormally lower in one steam generator than in the others, he must assume a loss of secondary coolant. Abnormally high readings for containment pressure, containment high radiation, or containment recirculation sump levels are symptomatic of a loss of reactor coolant.

In the diagnostic procedure, the operator is permitted to terminate a spurious high pressure injection (HPI) actuation if the primary system pressure, pressurizer level, and subcooling are within acceptable limits and there is sufficient water level in at least one steam generator and no abnormal readings for containment atmosphere monitors.

If HPI actuation is not spurious, the operator would proceed to the emergency procedures for one of the depressurization accidents. The core is assured of adequate core cooling in the LOCA procedure (E-1) in that the operator is prevented from terminating high pressure injection unless certain criteria are met. These criteria include:

1. At least normal full power subcooling,
2. Primary system pressure of 2000 psig or greater and increasing, and
3. Pressurizer level at or above programmed no-load range, and
4. Sufficient water level in at least one steam generator to assure a heat sink.

Similar HPI termination criteria are included in the other emergency procedures.

Recognizing that, in most instances of safety injection, the primary system will be repressurized, these criteria are necessary to allow the operator to terminate safety injection to reduce the probability of lifting the pressurizer power operated relief or safety valves.

The criteria for the lower head SI plants are basically the same. If the safety injection is sufficient to repressurize the plant, flow will stop when the shutoff head of the pumps is reached. The normal charging pumps and pressurizer heaters can be used to bring pressure above 2000 psig, at which time the SI pumps can be stopped.

Subsequent actions in the E-1 guidelines are based on whether the plant can be repressurized. If the plant can be repressurized, the operator is directed to increase the subcooling to 50° F and proceed with plant cooldown while monitoring subcooling. If subcooling cannot be maintained, HPI is reinitiated. Subsequent actions include switchover from injection to recirculation when the level in the refueling water storage tank (RWST) is low and hot leg injection at about 20 hours. The NRC staff has not reviewed the guidelines for switchover and hot leg injection, because these will be plant-specific.

Evaluation

The NRC staff reviewed the guidelines with respect to critical operator actions, namely:

1. reactor coolant pump trip.
2. HPI termination criteria.
3. verification of safety systems actuation.
4. verification of a heat sink.
5. monitoring of important system parameters.

During our review, the staff identified modifications to be made to the guidelines to enhance the directions to the operator. These modifications were subsequently incorporated in the guidelines as defined by Revision 1 (October 16, 1979) and revisions dated October 31, 1979, and November 2, 1979.

The criteria for tripping the reactor coolant pumps are consistent with the analyses presented in WCAP-9584, which have been reviewed by the staff and found acceptable. In order to implement the criteria in individual plant procedures, each licensee must document the basis for the low pressure set point. This documentation should include defining the steam generator safety valve set points and system and instrument uncertainties associated with the plant. Based on our review of WCAP-9584 and the requirement for each licensee to justify the low pressure trip point described in the preceding section, we conclude that the reactor coolant pump trip criteria are acceptable.

Although we find that the reactor coolant pump trip criteria are acceptable, manual tripping of the pumps should be considered only a short-term solution. For the long-term, we will require that this trip be made automatic.

The criteria for terminating HPI flow is based on a combination of system pressure, subcooling, pressurizer level, and steam generator water level. The staff concurs that these criteria are sufficient for establishing subcooled conditions in the core so that HPI can be safely terminated without concern for detrimental voids being formed in the primary system. In implementing these criteria, each licensee is required to document the instrument uncertainties (even in an adverse environment) to show that the criteria in the guidelines will indeed insure subcooled conditions. Based on the above requirement, we find the HPI termination criteria acceptable.

As part of the immediate actions, the operator is directed to verify that the ECCS, auxiliary feedwater (AFW), and containment isolation systems have been actuated. We concur that these verifications are sufficient to insure minimum safeguards availability needed to mitigate small break LOCAs.

The operator is also directed to verify that he has established heat removal from the steam generator. We concur that this is a necessary instruction for mitigating small break LOCAs.

The operator is directed to monitor primary system pressure, pressurizer level, and coolant hot leg temperatures to insure that subcooling is maintained if HPI has been terminated. We concur that monitoring these system variables is sufficient to maintain adequate subcooling in the primary system.

The staff has not reviewed the guidelines for switchover from injection to recirculation or hot leg injection because these actions are mostly plant-specific instructions. The staff requires each licensee to justify the procedures for switchover to insure that the valve realignments can be accomplished before the RWST is emptied. This justification should include instrument uncertainties and show that the pumps will be protected against operating with inadequate suction. We will require that plants with nominal 1400 psi range SI pumps demonstrate that these pumps will not be deadheaded when in the recirculation phase.

The staff noted that the guidelines are based on obtaining at least minimum safeguards operation to mitigate small break LOCAs. We require each licensee to extend the emergency procedures to cover the loss of all feedwater. Procedures for this degraded condition should also take into account pressure vessel integrity considerations. The Owners' Group has committed to prepare guidelines for operational procedures regarding the loss of all feedwater as part of its effort on the issue of inadequate core cooling.

The staff also requires that the emergency procedures include instructions for monitoring and initiating (if lost) natural circulation for small break LOCAs where heat removal by the steam generators is required.

The guidelines for such procedures should direct the operator to initiate a controlled plant cooldown is stable system conditions can be maintained. The staff requires that each licensee provide procedures for cooling down the plant under natural circulation conditions. These procedures should address boration control and monitoring, cooldown of the pressurizer, and adequate criteria for monitoring coolant system temperatures to insure that voids do not form in the primary system which could inhibit adequate heat removal. As in the case of loss of all feedwater, the Owners' Group has committed to prepare guidelines for operational procedures regarding natural circulation and cooldown under natural circulation conditions as part of its effort on inadequate core cooling.

Conclusion

Based on our review, we conclude that the guidelines E-0 and E-1 as revised by the Owners' Group letters dated October 16, 1979, October 31, 1979, and November 2, 1979 are acceptable for plants having high head safety injection pumps similar to the 412 standard plant, provided that licensees implement the requirements noted above when they develop their procedures. For the case of 4-loop, 3-loop, and 2-loop plants with nominal 1400 psi range safety injection pumps, the Owners' Group has committed to submit revisions to the guidelines for these plants which are similar to those provided for the 412 standard plant in the Owners' Group letters dated October 31, 1979 and November 2, 1979. Based on this commitment, we find the guidelines for these plants acceptable, pending, submission of such revisions, subject to the requirements on individual licensees identified above.

412 STANDARD PLANT
REFERENCE EMERGENCY
OPERATING INSTRUCTIONS

Revision 1

September 26, 1979

with Revised Pages

dated October 15, 1979,

October 29, 1979,

and November 2, 1979

This document contains Emergency Instructions for the Model 412 Standard Plant and is intended to provide guidance in the preparation of Emergency Operating Procedures for individual plants. It is not likely that these instructions will apply in their entirety to any specific plant design and adaptation will be required.

ATTACHMENT A

(1) Secondary System Pressure - Based on the number and size of the secondary system safety valves, the secondary pressure will be established by determining the pressure setpoint for that valve in which the calculated steam relief is less than 60% of the valve's relief rating. If the calculated relief is greater than 60% of the rated capacity, then the next highest pressure setpoint should be used.

(2) Primary to Secondary Pressure Difference - To account for the pressure gradient needed for heat removal, pressure drop between the steam generator and safety valves, pressure drop from steam generator to measurement location, etc., the primary pressure for RCP trip should be the secondary pressure as established by (1) above plus 100 psi if the adjustments calculated are 100 psi or less. If the adjustments are determined to be greater than 100 psi, the larger value should be used.

(3) Instrument inaccuracies appropriate for that time in the loss of coolant accident should be added to the primary system pressure value established in (2) above. The resulting pressure is the indicated primary system pressure at which the operator should trip the reactor coolant pumps.

412 STANDARD PLANT

E-0

EMERGENCY INSTRUCTIONS

IMMEDIATE ACTIONS AND DIAGNOSTICS

A. PURPOSE

This instruction presents the automatic actions, the immediate operator actions and the diagnostic sequence which is to be followed in the identification of the following:

1. Spurious Actuation of Safety Injection
2. Loss of Reactor Coolant
3. Loss of Secondary Coolant
4. Steam Generator Tube Rupture

The reactor automatic protection equipment is designed to safely shut down the reactor in the event of any of the above emergencies. The safety injection system is designed to provide emergency core cooling and boration to maintain the safe reactor shutdown condition. These plant safeguards systems operate with offsite electrical power or from onsite emergency diesel-electric power should offsite power not be available.

In the subsequent documents in this series (E-1, E-2 and E-3), instructions for recovery from the event are presented for each particular accident.

B. SYMPTOMS

NOTE: The process variables referred to in this Instruction are typically monitored by more than one instrumentation channel. The redundant channels should be checked for consistency while performing the steps of this Instruction.

The following symptoms are typical of those which may arise in a plant which is undergoing a loss of reactor coolant, loss of secondary coolant or steam generator tube rupture (one or more symptoms may appear in any order):

Low Pressurizer Pressure
Low Pressurizer Water Level
High Pressurizer Water Level
High Containment Pressure
High Containment Radiation
High Air Ejector Radiation
High Steam Generator Blowdown Radiation
Steam Flow/Feedwater Flow Mismatch
Letdown Isolation/Pressurizer Heater Cutout
Low Low Reactor Coolant System Average Coolant Temperature

High Containment Recirculation Sump Water Level

Low Steamline Pressure (one or all Steamlines)

Low Steam Generator Water Level

Increasing Steam Generator Water Level

Rapidly Changing Reactor Coolant System Average Coolant Temperature

Increased Charging Flow

High Steam Flow (one or all Steam lines)

High Containment Humidity

High Containment Temperature

Low Feedwater Pump Discharge Pressure

NOTE: The pressurizer water level indication should always be used in conjunction with other specified reactor coolant system indications to evaluate system conditions and to initiate manual operator actions.

C. IMMEDIATE ACTIONS

1. Conditions warranting reactor trip or safety injection may be characterized by a number of anomalous situations or unusual instrument indications.

   a. If the plant is in a condition for which a reactor trip is warranted and an automatic reactor trip has not yet occurred, manually trip the reactor. Continue monitoring plant conditions as shown in Figure 1.

b. If the plant is in a condition for which safety injection is warranted and an automatic safety injection has not yet occurred, manually initiate safety injection.

2. Verify the following actions and system status:

a) Reactor trip and turbine trip have occurred.

b) Bus voltages indicate that the busses are energized and all intended loads are being powered.

c) Feedwater Isolation has occurred.

d) Containment Isolation Phase A has occurred.

e) Auxiliary Feedwater Pumps have started and the Auxiliary Feedwater System valves are in their proper Emergency Alignment and are fully open or fully closed as appropriate.

f) Safety Injection Pumps have started and the monitor lights indicate that the Safety Injection System valves are in the proper safeguards position.

g) Service and Component Cooling Water Pumps have started.

h) Containment Ventilation isolation has occurred.

i) Other essential equipment as required by the specific plant design has been put into service.

3. If any of the above automatic actions have not occurred and are required, they should be manually initiated.

Verify the following:

a) Safety Injection flow from at least one train is being delivered to the reactor coolant system when the Reactor Coolant System pressure is below the high head safety injection pump shutoff head. If not, attempt to operate equipment manually or locally.

b) Auxiliary Feedwater flow from at least one train is being delivered to the steam generators. If not, attempt to operate equipment manually or locally.

NOTE: Only after steam generator water level is established above the top of the U-Tubes, should the Auxiliary Feedwater System Flow be regulated to maintain required level.

c) Verify that heat is being removed from the reactor plant via the steam generators by noting the following:

a) Automatic steam dump to the condenser is occurring;

b) Reactor coolant average temperature is decreasing towards programmed no-load temperature.

NOTE: Atmospheric steam dump will be blocked by an existing "Turbine Tripped" condition. If condenser steam dump has been blocked due to a control malfunction or loss of the "Condenser Available" condition, decay heat removal will be effected by automatic actuation of the steam generator power-operated relief valves, or, if these prove ineffective, the steam generator code safety valves. In this event, steam pressure will be maintained at the set pressure of the controlling valve(s) and reactor coolant average temperature will stabilize at approximately the saturation temperature for the steam pressure being maintained.

4. Whenever the Containment Hi-2 pressure setpoint is reached, verify that the Main Steam Isolation Valves have closed. If not, manually close the Main Steam Isolation Valves from the Control Board.

5. Whenever the Containment Hi-3 pressure setpoint is reached, verify that the following have occurred:

   a) Containment Spray is initiated

   b) Containment Isolation Phase B is initiated

   If not, manually initiate Containment Spray and Containment Isolation Phase B.

D. ACCIDENT DIAGNOSTICS (Refer to Figure 2)

1. Evaluate reactor coolant pressure to determine if it is low or decreasing in an uncontrolled manner. If it is low or decreasing, verify that:

   a. all pressurizer spray line valves are closed and

   b. all pressurizer relief valves are closed.

If not, manually close the valves from the Control Board.

If the RCS pressure is above the low pressure reactor trip setpoint and is stable or increasing, go to STEP 7.

2. Stop ALL Reactor Coolant Pumps after the high head safety injection pump operation has been verified and when the wide range reactor coolant pressure is at (plant specific pressure derived from method in Attachment A of letter OG-17).

CAUTION: If component cooling water to the reactor coolant pumps is isolated on a containment pressure signal, all reactor coolant pumps should be stopped within 5 minutes because of loss of motor bearing cooling.

CAUTION: If the reactor coolant pumps are stopped, the seal injection flow should be maintained.

NOTE: The conditions given above for stopping reactor coolant pumps should be continuously monitored throughout this instruction.

*3. IF the condenser air ejector radiation or steam generator blowdown radiation monitor exhibit abnormally high readings, AND containment pressure, containment radiation and containment recirculation sump level exhibit normal readings, THEN go to E-3, "Steam Generator Tube Rupture."

*4. IF the steamline pressure is abnormally lower in one steam generator than in the other steam generators, THEN go to E-2, "Loss of Secondary Coolant."

5. IF containment pressure, OR containment radiation OR containment recirculation sump levels exhibit either abnormally high readings or increasing readings, THEN go to E-1, "Loss of Reactor Coolant".

NOTE: For very small breaks inside the containment building, the containment pressure increase will be very small and possibly not recognizable by the operator. For very small breaks the containment recirculation sump water level will increase very slowly and early in the transient may not indicate a level increase.

*These steps may be interchanged.

6. <u>IF</u> the containment pressure, containment radiation <u>AND</u> containment recirculation sump water level continue to exhibit stable readings in the normal pre-event range, <u>THEN</u> go to E-2, "Loss of Secondary Coolant".

7. In the event of a spurious safety injection signal, the sequence of reactor trip, turbine trip and safeguards actuation will occur.

   The operator must assume that the safety injection signal is non-spurious unless the following are exhibited:

   a. Normal readings for containment temperature, pressure, radiation and recirculation sump level <u>AND</u>

   b. Normal readings for auxiliary building radiation and ventilation monitoring <u>AND</u>

   c. Normal readings for steam generator blowdown and condenser air ejector radiation.

   <u>IF</u> all of the symptoms a through c above are met and <u>when the following</u> d through f <u>are exhibited</u>:

   d. Reactor coolant pressure is greater than 2000 psig and increasing <u>AND</u>

e. Pressurizer water level is greater than programmed no load water level* AND

f. The reactor coolant indicated subcooling is greater than (insert plant specific value of subcooling equal to full power normal operation). 10/2

g. Water level in at least one steam generator is in the narrow range span, or in the wide range span at a level sufficient to assure that the U-tubes are covered 10/29

*NOTE: Pressurizer water level should trend with reactor coolant system temperature. If the pressurizer water level is low enough to prohibit pressurizer heater operation, re-establish water level by operating the charging system. Energize the heaters.

THEN:

h. Reset safety injection and stop safety injection pumps not needed for normal charging and RCP seal injection flow. 10/29

CAUTION: Automatic reinitiation of safety injection will not occur since the reactor trip breakers are not reset. 10/19

CAUTION: Subsequent to this Step, should loss of offsite power occur, manual action (e.g., manual safety injection initiation) will be required to load the safeguards equipment onto the diesel powered emergency busses. 10/1

i. Place all safety injection pumps not needed to provide normal changing flow in standby mode and maintain operable safety injection flowpaths.

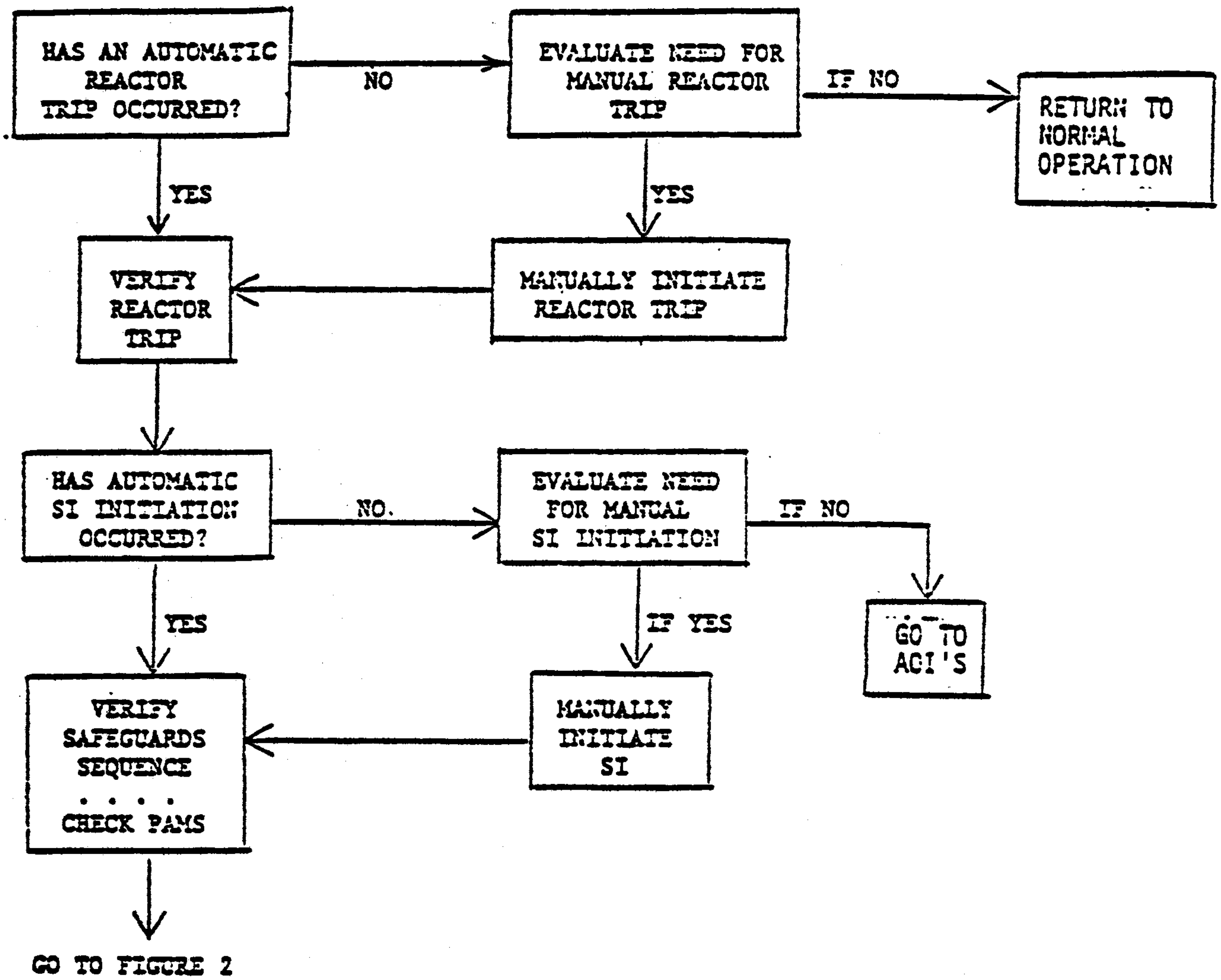
j. Isolate safety injection flow to RCS Cold Legs via Boron Injection Tank and establish normal charging flow.

k. Reestablish normal makeup and letdown (if letdown is unaffected) to maintain pressurizer water level in the normal operating range and to maintain reactor coolant pressure at values reached when safety injection is terminated. Ensure that water addition during this process does not result in dilution of the reactor coolant system boron concentration.

l. Reestablish operation of the pressurizer heaters. When reactor coolant pressure can be controlled by pressurizer heaters alone, return makeup and letdown to pressurizer water level control only.

NOTE: IF after securing safety injection and attempting to transfer to normal pressurizer pressure and level control, reactor coolant pressure drops below the low pressurizer pressure setpoint for safety injection actuation OR if pressurizer water level drops below 10% of span, OR the reactor coolant T_H > normal full power T_H, THEN SAFETY INJECTION MUST BE MANUALLY REINITIATED. The operator must rediagnose plant conditions and proceed to the appropriate emergency instruction.

NOTE: IF after securing safety injection and transferring the plant to normal pressurizer pressure and level control, the reactor coolant pressure does not drop below the low pressurizer pressure setpoint for safety injection actuation AND the pressurizer water level remains above 10% span, AND the reactor coolant indicated subcooling is greater than (insert plant specific value of subcooling based on full power normal operation), THEN go to the abnormal operating instructions.



IMMEDIATE ACTIONS

FIGURE 1

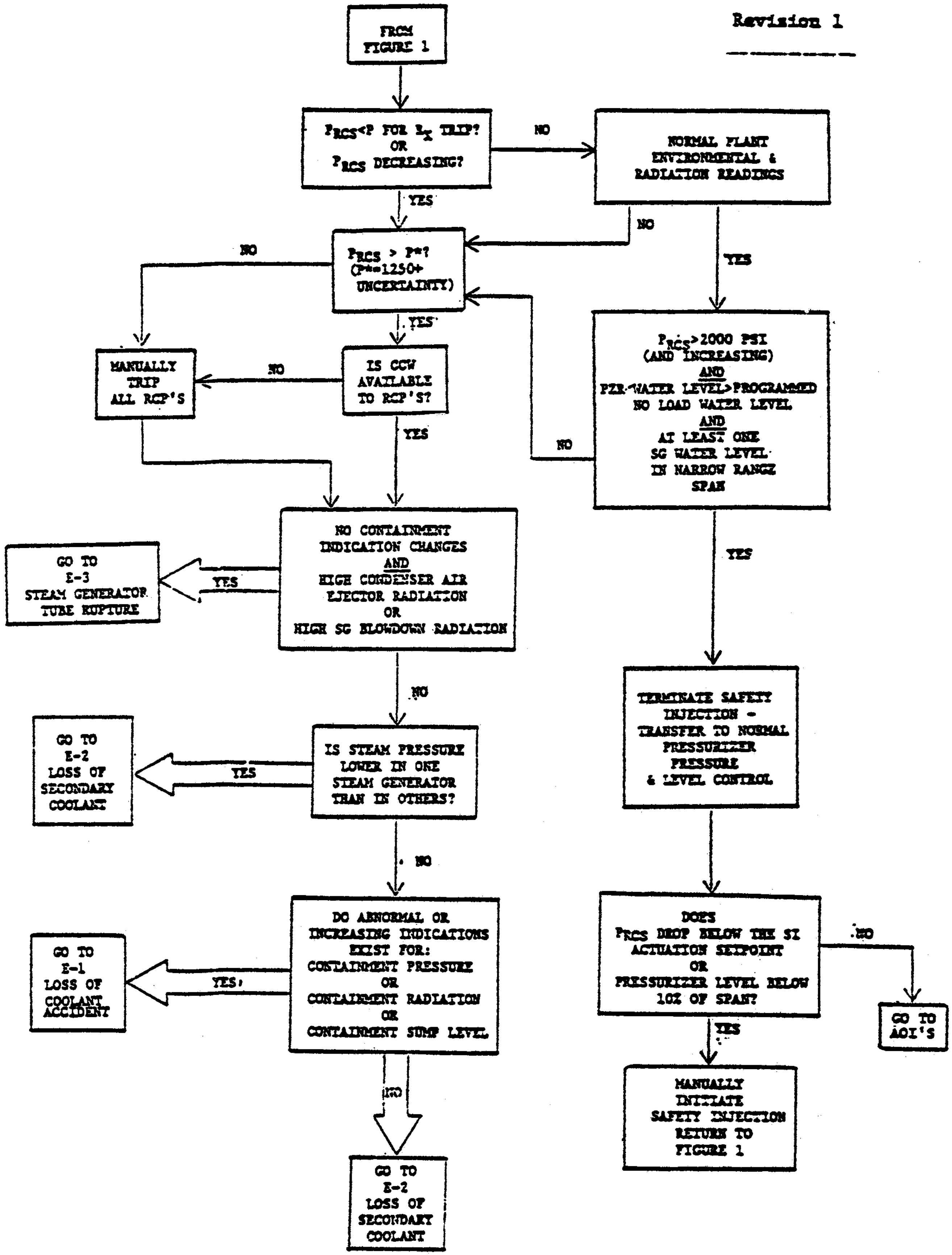


FIGURE 2
E-0(HP)-14

412 STANDARD PLANT

E-1

EMERGENCY INSTRUCTION

LOSS OF REACTOR COOLANT

A. PURPOSE

The objectives of these instructions are to specify required operator actions and precautions necessary to:

1. Verify and establish short term core cooling to prevent or minimize damage to the fuel cladding and release of excessive radioactivity.

2. Maintain long term shutdown and cooling of the reactor by recirculation of spilled reactor coolant, injected water and containment spray system drainage.

B. IMMEDIATE ACTIONS

Refer to section on Immediate Actions of E-0, Immediate Actions and Diagnostics, if not already performed.

C. SUBSEQUENT ACTIONS

CAUTION: Monitor RWST level closely. If RWST level decreases rapidly such that the RWST low level alarm appears imminent, go directly to step 5.

CAUTION: The diesels should not be operated at idle or minimum load for extended periods of time. If the diesels are shut down, they should be prepared for restart.

NOTE: The operator should verify that the Post Accident Monitoring (PAM) instruments are operating and recording. These instruments include wide range RCS temperature and pressure, steam pressure, steam generator water level, containment pressure, RWST water level, condensate storage tank water level, pressurizer water level, and boric acid storage tank water level.

NOTE: The process variables referred to in this Instruction are typically monitored by more than one instrumentation channel. The redundant channels should be checked for consistency while performing the steps of this Instruction.

NOTE: Reactor coolant system isolation valves (LSIV) are optional equipment on the Westinghouse Standard Plants. If a plant is so equipped, the use of LSIV's is not currently recommended during the course of this Instruction. Any use of LSIV's must be justified on a plant specific basis.

NOTE: The pressurizer water level indication should always be used in conjunction with other specified reactor coolant system indications to evaluate system conditions and to initiate manual operator actions.

1. As the water level (PAMS) in the refueling water storage tank decreases under the action of the safeguards pumps, check that the recirculation sump water level instrumentation indicates an increase in water level in the sump. If a sump water level increase is not evident then a re-evaluation of the symptoms in E-0 must be conducted.

   Regulate the auxiliary feedwater flow to the steam generators to restore and/or maintain an indicated narrow range water level (PAMS). If narrow range water level increases in an unexplained manner in one steam generator, go to E-3, Steam Generator Tube Rupture.

NOTE: Monitor the primary water supply (Condensate Storage Tank) for the auxiliary feedwater pumps and upon reaching a low level, switch over to an alternate water supply source.

2. Close all pressurizer power operated relief valves and backup isolation valves.

3. NOTE: The conditions given below for termination of safety injection should be continuously monitored throughout this instruction:

Ensure that containment isolation is maintained, i.e., not reset until such time as manual action is required on necessary process streams.

Safety Injection can be terminated IF:

(A) Reactor coolant pressure is greater than 2000 psig and increasing, AND

(B) Pressurizer water level is greater than 50% of span, AND

(C) The reactor coolant indicated subcooling is greater than (insert plant specific value of subcooling based on full power normal operation), AND

(D) Water level in at least one Steam Generator is in the narrow range span, or in the wide range span at a level sufficient to assure that the U-tubes are covered.

THEN:

(E) Reset safety injection and stop safety injection pumps not needed for normal charging and RCP seal injection flow.

CAUTION: Automatic reinitiation of safety injection will not occur since the reactor trip breakers are not reset.

CAUTION: Subsequent to this Step, should loss of offsite power occur, manual action (e.g., manual safety injection initiation) will be required to load the safeguards equipment onto the diesel powered emergency busses. 10/

(F) Place all safety injection pumps not needed to provide normal charging flow in standby mode and maintain operable safety injection flowpaths. 10/2

(G) Isolate safety injection flow to RCS Cold Legs via Boron Injection Tank and establish normal charging flow. 10/2

CAUTION: If reactor coolant pressure drops below the low pressurizer pressure setpoint for safety injection or pressurizer water level drops below 20% of span following termination of safety injection flow or the reactor coolant T_H >Normal Full Power T_H MANUALLY REINITIATE safety injection to establish reactor coolant pressure and pressurizer water level. Go to Section D of E-0 to reevaluate the event, unless this reevaluation has already been performed. 10/2

(H) Reestablish normal makeup and letdown (if letdown is unaffected) to maintain pressurizer water level in the normal operating range and to maintain reactor coolant pressure at values reached when safety injection is terminated. Ensure that water addition during this process does not result in dilution of the reactor coolant system boron concentration.

(I) Reestablish operation of the pressurizer heaters. When reactor coolant pressure can be controlled by pressurizer heaters alone, return makeup and letdown to pressurizer water level control only.

(J) Monitor either the average temperature indication of core exit thermocouples (if available) or all wide range reactor coolant temperature T_H (PAMS) to verify that RCS temperature is at least 50°F less than saturation temperature at RCS indicated pressure.

If 50°F indicated subcooling is not present, then attempt to establish 50°F indicated subcooling by steam dump from the steam generators to the condenser or the atmosphere.

CAUTION: If steam dump is necessary, reduce the steam generator pressure 200 psi below the lowest steam safety valve setpoint and maintain a reactor coolant cooldown rate of no more than 50°F/HR, consistent with plant make-up capability.

If 50°F indicated subcooling cannot be established or maintained, then manually reinitiate safety injection. Go to Section D of E-0 to re-evaluate the event, unless this re-evaluation has already been performed.

(K) Perform a controlled cooldown to cold shutdown conditions using Normal Cooldown Procedures if required to affect repairs. Maintain subcooled conditions (at least 50°F indicated subcooling) in the reactor coolant system. If subcooled conditions cannot be maintained, go to Step 4.

4. If the conditions for terminating safety injection in Step 3 are not met, maintain necessary safety injection pumps operating. If any safeguards equipment is not operating, attempt to operate the equipment from the control room or locally. Effect repairs if necessary. If reactor coolant pressure is above the low head safety injection pump shut-off head, manually reset safety injection so that safeguards equipment can be controlled by manual action. Stop the low head safety injection pumps and place in the standby mode.

<u>CAUTION</u>: Whenever the reactor coolant pressure decreases below the low head safety injection shutoff head, the low head safety injection pumps must be manually restarted to deliver fluid to the reactor coolant system.

5. Stop ALL Reactor Coolant Pumps after the high head safety injection pump operation has been verified and when the wide range reactor coolant pressure is at (plant specific pressure derived from method in Attachment A of letter OG-17).

CAUTION: If component cooling water to the reactor coolant pumps is isolated on a containment pressure signal, all reactor coolant pumps are to be stopped within 5 minutes because of loss of motor bearing cooling.

CAUTION: If reactor coolant pumps are stopped, the seal injection flow should be maintained.

NOTE: The conditions given above for stopping reactor coolant pumps should be continuously monitored throughout this instruction:

6. In the case of a break characterized by reactor coolant pressure quickly decreasing below steam generator pressure, go to step 7. In the case of a break characterized by a slowly decreasing reactor coolant pressure or stabilized reactor coolant system pressure above the lowest steam system safety valve setpoint, (plant specific) psig, the following additional manual actions should be taken to aid the cooldown and depressurization of the reactor coolant system:

a. If the main condenser is in service, open at least one main steamline isolation valves or bypass valves and transfer the steam dump control to steam header pressure control and dump steam to the condenser to lower the reactor coolant temperature (PAMS) and consequently the reactor coolant pressure.

b. If the main condenser is not in service, dump steam to the atmosphere with the steam relief valves to lower the reactor coolant temperature and consequently the reactor coolant pressure.

<u>CAUTION</u>: Reduce the steam generator pressure 200 psi below the lowest steam system safety valve setpoint and maintain a reactor coolant cooldown rate of no more than 50°F/HR, consistent with plant make-up capability.

7. Go to the Cold Leg Recirculation Instruction presented in Table E-1.1. Note, if the reactor coolant system pressure is above the shut-off head of the high head safety injection pumps, stop these pumps and place them in a standby mode prior to transfer to cold leg recirculation.

   CAUTION: The cold leg recirculation procedures are different for each plant ECCS design. The plant specific procedures should be incorporated in Table E-1.1.

   NOTE: If RWST low level alarm is not imminent, then consideration should be given to performing a preliminary evaluation of the plant status in Steps 9 and 10.

8. If containment spray has been actuated, and if the containment pressure is reduced to nominal operation containment pressure, reset containment spray. Spray pumps should be shut-off and placed in the standby mode with operable flow paths.

9. Periodically check auxiliary building area radiation monitors for detection of leakage from ECCS during recirculation. If significant leakage has been identified in the ECCS, attempt to isolate the leakage. The operator <u>must</u> maintain recirculation flow to the RCS at all times.

10. While the plant is in cold leg recirculation mode, plant operators should make provision for an evaluation of equipment in the plant. This evaluation should include the primary safeguards equipment e.g., RCS pumps and valves, emergency diesels, containment fan coolers, etc. and support equipment e.g., ECCS HVAC equipment, diesel fuel supply, diesel start air supply, sampling of RCS for boron concentration and fuel damage, sampling of containment atmosphere, sampling of recirculation sump, etc. Adjust recirculation sump pH, if required.

11. Prior to the time specified for the plant for the switchover to the hot leg recirculation mode, the operator in the control room should:

a. Ensure that control room valve switches are aligned in the proper positions for cold leg recirculation mode.

b. Re-energize the breakers, as required, for valves needed to effect switchover to the hot leg recirculation mode.

12. At (plant specific) hours after the accident, realign the safety injection systems for hot leg recirculation. Go to Table E-1.2.

CAUTION: The hot leg recirculation switchover procedures are different for each plant ECCS design. The plant specific procedures should be incorporated in Table E-1.2.

13. Continue to implement the hot leg recirculation mode of cooling.

14. Recovery procedures for the particular event must be developed and implemented to effect plant return to service.

TABLE E-1.1

COLD LEG RECIRCULATION SWITCHOVER INSTRUCTIONS

PREREQUISITES AND PRECAUTIONS:

A. Prior to receipt of the Refueling Water Storage Tank (RWST) Low Level Alarm restart any safety injection pump not operating and reset/defeat the safety injection signal. Also open component cooling water (CCW) valves to Residual Heat Removal (RHR) heat exchangers if these valves are not interlocked to open automatically.

B. The Refueling Water Storage Tank (RWST) Low Level Alarm signifies automatic initiation of cold leg recirculation. The containment recirculation sump valves will immediately start to open automatically.

C. IMMEDIATELY perform all steps given below when the recirculation sump isolation valve position lights indicate that the valve is fully open.

D. Do not close a RWST/RHR pump suction valve unless the corresponding recirculation sump valve is open.

E. All operator actions must be performed expeditiously, in a precise, orderly sequence. Do not interrupt this operation until all actions are completed. When both trains are initially available and a valve

TABLE E-1.1 (Continued)

fails to respond or to complete its demanded operation, postpone any corrective action until the subsequent operational steps are performed.

F. IMMEDIATELY stop any pumps taking suction from the RWST on receipt of a RWST empty alarm. Complete the switchover steps listed below, then restart required pumps.

OPERATIONAL STEPS (NO SINGLE FAILURES):

STEP 1

a) Close the RWST to low head safety injection pump suction isolation valves

b) Close the high head safety injection pump miniflow valves

c) Close the low head safety injection crossover isolation valves

STEP 2

a) Open parallel valves in the high head safety injection and charging safety injection pump common suction header.

TABLE E-1.1 (Continued)

b) Open the low head safety injection to high head safety injection and charging/safety injection pump suction isolation valves

c) After completion of the above steps VERIFY that the two high head safety injection pumps and the two charging/safety injection pumps are receiving suction flow from the low head safety injection pumps.

CAUTION: Do not perform the following steps until the above verification is made.

STEP 3

a) Close the RWST to high head safety injection pump suction valves

b) Close the RWST to charging/safety injection pump suction isolation valves

STEP 4

The utility should provide spray system switchover procedures and integrate them into this instruction.

TABLE E-1.1 (Continued)

NOTE: For plant designs which utilize only the spray system heat exchanger to remove energy from the containment recirculation sump the spray system must be operated during the long term even if it was not automatically actuated.

VERIFICATION:

STEP 5

After completing the preceding steps, verify that the safety injection system is aligned for cold leg recirculation as follows:

a) One low head safety injection pump is delivering from the containment recirculation sump directly to two reactor coolant system cold legs and to the suction of two charging/safety injection pumps.

b) The other low head safety injection pump is delivering from the containment recirculation sump directly to two reactor coolant system cold legs and to the suction of two high head safety injection pumps.

c) The two high head safety injection and two charging/safety injection pumps are taking suction from the low head safety injection pumps and are delivering to four reactor coolant system cold legs.

TABLE E-1.1 (Continued)

d) The suction paths from the RWST to all safety injection pumps have been isolated.

e) If containment spray is required, verify that flow is being delivered.

STEP 6

If the system alignment has been verified go to E-1 Step 9. If any failures have occurred, proceed to contingency actions.

CONTINGENCY ACTIONS

1. CONTAINMENT RECIRCULATION SUMP VALVE FAILS TO OPEN

If a containment recirculation sump valve cannot be opened, stop the corresponding low head safety injection pump and verify that:

a) One low head safety injection pump is delivering flow to two reactor coolant system cold legs and to the suction of the two high head safety injection and two charging/safety injection pumps.

b) The two high head safety injection and the two charging/safety injection pumps are delivering to four reactor coolant system cold legs.

TABLE E-1.1 (Continued)

2. LOSS OF ONE TRAIN OF ELECTRICAL POWER

NOTE: If the single active failure is the failure of one of the emergency diesel generators to start in conjunction with a LOCA and a loss of offsite power, electrical power would not be available to one of the vital safeguard busses. As a consequence, all engineered safeguards equipment assigned to that corresponding electrical power train would not be available for operation until power could be restored to that bus. The instruction for switchover to cold leg recirculation, assuming a train failure, is essentially the same as the instruction above, which assumed no single failures. The operator could follow the above instruction with the understanding that those valves, without power, do not have to be repositioned.

The following instruction is provided to illustrate the similarity between the instruction which assumes no single failures, and an instruction which assumes one complete electrical power train failure. For this instruction, it is assumed that Train B failed simultaneously with the loss of reactor coolant. It should be noted that if a train failed subsequent to the initiation of the "S" signal additional steps may be required. For example, if no failure is assumed, the parallel suction valves in the line

TABLE E-1.1 (Continued)

from the RWST to the charging/safety injection pump suction header would open on an "S" signal. Should a subsequent failure of one of the electrical trains occur, one of the parallel suction valves could not be closed from the main control board. Therefore, positive isolation of the RWST to charging/safety injection pump suction path would have to be accomplished locally.

OPERATIONAL STEPS: (Assume only Train A available)

STEP 1

a) Close the RWST to low head safety injection pump suction isolation valve

b) Close the high head safety injection pump miniflow valves

c) Close the low head safety injection crossover isolation valve

STEP 2

a) Open one of the parallel valves in the high head safety injection and charging/safety injection pump common suction header

TABLE E-1.1 (Continued)

b) Open the low head safety injection to charging/safety injection pump suction isolation valve

After completing the above steps, verify that the one high head safety injection pump and one charging/safety injection pump are receiving suction flow from the one operating low head safety injection pump.

Caution: Do not perform the following steps unless the above verification is absolute.

STEP 3

a) Close the RWST to high head safety injection pump suction valve

b) Close the RWST to charging/safety injection pump suction valve

VERIFICATION:

STEP 4

After completing the above step, verify that the safety injection system is aligned for cold leg recirculation as follows:

TABLE E-1.1 (Continued)

a) One low head safety injection pump is delivering from the containment recirculation sump to two reactor coolant system cold legs and to the suction of one high head safety injection and one charging/safety injection pump.

b) The one high head safety injection and one charging/safety injection pump are taking suction from the low head safety injection pumps and are delivering to four reactor coolant system cold legs.

c) The suction paths from the RWST to all safety injection pumps have been isolated.

d) If containment spray is required, verify that flow is being delivered.

STEP 5

If the system alignment in Step 4 has been verified, go to E-1 Step 9. If any failures have occurred, attempt to operate the equipment manually and locally.

TABLE E-1.2

HOT LEG RECIRCULATION SWITCHOVER INSTRUCTIONS

NOTE: Hot Leg Recirculation Phase - At approximately 24 hours after the accident, hot leg recirculation shall be initiated. The following manual operator actions are required to complete the switchover operation from the cold leg recirculation mode to the hot leg recirculation mode. In this instruction it is assumed that both electrical power trains A and B are available and that all safety injection pumps are operating. (No single failure has occurred). If failures have occurred continue through the instruction to contingency actions.

OPERATIONAL STEPS BASED ON NO SINGLE FAILURE

Step 1: Terminate low head safety injection pump flow to reactor coolant system cold legs and establish low head safety injection flow to reactor coolant system hot leg by performing the following actions:

a) Close the low head safety injection cold leg header isolation valves

b) Open the low head safety injection crossover isolation valves

c) Open the low head safety injection leg header isolation valve

TABLE E-1.2 (Continued)

Step 2: Terminate high head safety injection pump flow to reactor coolant system cold legs and establish high head safety injection flow to reactor coolant system hot legs by performing the following steps:

a) Stop high head safety injection pump no. 1

b) Close the corresponding high head safety injection crossover header isolation valve

c) Open the corresponding hot leg header isolation valve

d) Restart the high head safety injection pump no. 1

e) Stop high head safety injection pump no. 2

f) Close the corresponding high head safety injection crossover isolation valve

g) Close the corresponding high head safety injection cold leg header isolation valve

h) Open the corresponding high head safety injection hot leg header isolation valve

i) Restart the high head safety injection pump no. 2

TABLE E-1.2 (Continued)

VERIFICATION:

STEP 3

After completing the above steps, verify that the safety injection system is aligned to hot leg recirculation as follows:

a) Both low head safety injection pumps are aligned to deliver flow directly to the two reactor coolant system hot legs via the single low head safety injection hot leg header while each high head safety injection pump is aligned to deliver flow to the two reactor coolant system hot legs via two separate and redun- dant high head safety injection hot leg headers.

b) The low head safety injection pumps continue to provide suction flow to the high head safety injection and charging pumps.

c) The charging pumps continue to provide flow directly to the four reactor coolant system cold legs.

d) If containment spray is required, verify flow is being delivered.

STEP 4

If the system alignment has been verified go to E-1 Step 13. If any failures have occurred, proceed to contingency actions.

TABLE E-1.2 (Continued)

CONTINGENCY ACTIONS

1. LOSS OF ONE TRAIN OF ELECTRICAL POWER:

In the event that a single failure had resulted in a complete loss of power to one of the electrical power trains in conjunction with a LOCA and a loss of offsite power, the hot leg switchover procedures would require some operations to be performed outside the main control room, unless power could be restored to the failed train during the 24 hour cold leg recirculation phase. These operations, outside the main control room, would be necessary to open a hot leg isolation valve and to close a cold leg isolation valve. In both cases this can be accomplished either by manually operating the valve or by disconnecting the power to the valve from the failed train and temporarily connecting it to the available power.

In the following steps, it is assumed that train B failed simultaneously with the accident.

OPERATIONAL STEPS (Assume only Train A Available)

STEP 1

Terminate low head safety injection pump flow to reactor coolant system cold legs and establish low head safety injection flow to reactor coolant system hot legs.

TABLE E-1.2 (Continued)

a) Close the low head safety injection cold leg header isolation valves

NOTE: Since it is assumed in this case that train B has failed, power to close one isolation valve may not be available. This valve could be closed manually or it could be closed remotely by disconnecting it from train B and temporarily connecting it to train A.

b) Open the low head safety injection crossover isolation valve

c) Open the low head safety injection hot leg header isolation valve

NOTE: Since it is assumed in this case that train B has failed, power to open this valve may not be available. This valve could be opened manually or it could be opened remotely be disconnecting it from train B and temporarily connecting it to train A.

STEP 2

Terminate high head safety injection pump flow to reactor coolant system cold legs and establish high head safety injection flow to reactor coolant system hot legs:

TABLE E.1-2 (Continued)

a) Stop the Train A high head safety injection pump

b) Close the corresponding high head safety injection crossover header isolation valve

c) Open the corresponding high head safety injection hot leg header isolation valve

d) Restart the Train A high head safety injection pump.

STEP 3

Go to E-1 Step 13. If any failures have occurred, attempt to operate the equipment manually or locally.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 6 1979

Mr. Cordell Reed, Chairman
Westinghouse Operating Plants Owners' Group
Commonwealth Edison Company
P. O. Box 767
One First National Plaza
Chicago, Illinois 60690

Dear Mr. Reed:

SUBJECT: EVALUATION OF SMALL-BREAK LOSS-OF-COOLANT ACCIDENT OPERATOR GUIDELINES FOR PLANTS WITH NOMINAL 1400 PSI RANGE SAFETY INJECTION PUMPS

Our letter of November 5, 1979 approved the guidelines for emergency operational procedures E-0 and E-1 which you have developed for Westinghouse-designed plants with high head safety injection pumps similar to the 412 standard plant. By that letter, we also approved the corresponding guidelines for 2-loop, 3-loop, and 4-loop plants with nominal 1400 psi range safety injection pumps pending fulfillment of your commitment to provide certain agreed-upon revisions to these guidelines in a timely manner.

In your letter dated November 5, 1979, you provided revised pages for the guidelines for emergency operating procedures for nominal 1400 psi range safety injection pump plants which are similar to the October 31, 1979 and November 2, 1979 revisions to the guidelines for the 412 standard plant.

We have reviewed the revised pages submitted with the November 5, 1979 letter and have confirmed that they contain the previously agreed-upon revisions. Therefore, we reaffirm our previous acceptance of these guidelines as stated in our letter of November 5, 1979. A copy of the approved guidelines for plants with nominal 1400 psi range safety injection pumps is enclosed for your information and use.

Sincerely,

D. F. Ross, Jr., Director
Bulletins and Orders Task Force

Enclosure:
As stated

cc: see attached lists

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4-LOOP, 3-LOOP and 2-LOOP PLANTS WITH NOMINAL 1400 PSI RANGE SI PUMPS

REFERENCE EMERGENCY OPERATING INSTRUCTIONS

Revision 1

September 26, 1979

with Revised Pages

dated October 15, 1979

and October 29, 1979

and November 2, 1979

This document contains typical Emergency Instructions for 2, 3 and 4 loop plants with nominal 1400 psi range safety injection pumps and is intended to provide guidance in the preparation of Emergency Operating Procedures for individual plants in this category. It is not likely that these instructions will apply in their entirety to any specific plant design and adaptation will be required.

EQUIPMENT PARAMETERS

The following parameters are typical of plants with nominal 1400 psi range safety injection systems:

	2 Loop	3 Loop	4 Loop
High Head Safety Injection Pump			
Number	2	3	3
Maximum Flow Rate, gpm/pump	1100	600	650
Shutoff Head, psig	1520	1500	1515
Low Head Safety Injection/ Residual Heat Removal Pump			
Number	2	2	2
Maximum Flow Rate, gpm/pump	1520	4500	5500
Shutoff Head, psig	140	140	140

E-0

EMERGENCY INSTRUCTIONS for 2, 3 &4 LOOP PLANTS WITH NOMINAL 1400 PSI RANGE SAFETY INJECTION PUMPS

IMMEDIATE ACTIONS AND DIAGNOSTICS

A. PURPOSE

This instruction presents the automatic actions, the immediate operator actions and the diagnostic sequence which is to be followed in the identification of the following:

1. Spurious Actuation of Safety Injection
2. Loss of Reactor Coolant
3. Loss of Secondary Coolant
4. Steam Generator Tube Rupture

The reactor automatic protection equipment is designed to safely shut down the reactor in the event of any of the above emergencies. The safety injection system is designed to provide emergency core cooling and boration to maintain the safe reactor shutdown condition. These plant safeguards systems operate with offsite electrical power or from onsite emergency diesel-electric power should offsite power not be available.

In the subsequent documents in this series (E-1, E-2 and E-3), instructions for recovery from the event are presented for each particular accident.

B. SYMPTOMS

NOTE: The process variables referred to in this Instruction are typically monitored by more than one instrumentation channel. The redundant channels should be checked for consistency while performing the steps of this Instruction.

The following symptoms are typical of those which may arise in a plant which is undergoing a loss of reactor coolant, loss of secondary coolant or steam generator tube rupture (one or more symptoms may appear in any order):

Low Pressurizer Pressure
Low Pressurizer Water Level
High Pressurizer Water Level
High Containment Pressure
High Containment Radiation
High Air Ejector Radiation
High Steam Generator Blowdown Radiation
Steam Flow/Feedwater Flow Mismatch
Letdown Isolation/Pressurizer Heater Cutout
Low Low Reactor Coolant System Average Coolant Temperature

High Containment Recirculation Sump Water Level

Low Steamline Pressure (one or all Steamlines)

Low Steam Generator Water Level

Increasing Steam Generator Water Level

Rapidly Changing Reactor Coolant System Average Coolant Tem-

Temperature

Increased Charging Flow

High Steam Flow (one or all Steam lines)

High Containment Humidity

High Containment Temperature

Low Feedwater Pump Discharge Pressure

NOTE: The pressurizer water level indication should always be used in conjunction with other specified reactor coolant system indications to evaluate system conditions and to initiate manual operator actions.

C. IMMEDIATE ACTIONS

1. Conditions warranting reactor trip or safety injection may be characterized by a number of anomalous situations or unusual instrument indications.

   a. If the plant is in a condition for which a reactor trip is warranted and an automatic reactor trip has not yet occurred, manually trip the reactor. Continue monitoring plant conditions as shown in Figure 1.

b. If the plant is in a condition for which safety injection is warranted and an automatic safety injection has not yet occurred, manually initiate safety injection.

2. Verify the following actions and system status:

   a) Reactor trip and turbine trip have occurred.

   b) Bus voltages indicate that the busses are energized and all intended loads are being powered.

   c) Feedwater Isolation has occurred.

   d) Containment Isolation Phase A has occurred.

   e) Auxiliary Feedwater Pumps have started and the Auxiliary Feedwater System valves are in their proper Emergency Alignment and are fully open or fully closed as appropriate.

   f) Safety Injection Pumps have started and the monitor lights indicate that the Safety Injection System valves are in the proper safeguards position.

   g) Service and Component Cooling Water Pumps have started.

   h) Containment Ventilation isolation has occurred.

i) Other essential equipment as required by the specific plant design has been put into service.

3. If any of the above automatic actions have not occurred and are required, they should be manually initiated.

Verify the following:

a) Safety Injection flow from at least one train is being delivered to the reactor coolant system when the Reactor Coolant System pressure is below the high head safety injection pump shutoff head. If not, attempt to operate equipment manually or locally.

b) Auxiliary Feedwater flow from at least one train is being delivered to the steam generators. If not, attempt to operate equipment manually or locally.

NOTE: Only after steam generator water level is established above the top of the U-Tubes, should the Auxiliary Feedwater System Flow be regulated to maintain required level.

4. Whenever the Containment Hi-2 pressure setpoint is reached, verify that the Main Steam Isolation Valves have closed. If not, manually close the Main Steam Isolation Valves from the Control Board.

i) Other essential equipment as required by the specific plant design has been put into service.

3. If any of the above automatic actions have not occurred and are required, they should be manually initiated.

Verify the following:

a) Safety Injection flow from at least one train is being delivered to the reactor coolant system when the Reactor Coolant System pressure is below the high head safety injection pump shutoff head. If not, attempt to operate equipment manually or locally.

b) Auxiliary Feedwater flow from at least one train is being delivered to the steam generators. If not, attempt to operate equipment manually or locally.

NOTE: Only after steam generator water level is established above the top of the U-Tubes, should the Auxiliary Feedwater System Flow be regulated to maintain required level.

c) Verify that heat is being removed from the reactor plant via the steam generators by noting the following:

a) Automatic steam dump to the condenser is occurring;

b) Reactor coolant average temperature is decreasing towards programmed no-load temperature.

NOTE: Atmospheric steam dump will be blocked by an existing "Turbine Tripped" condition. If condenser steam dump has been blocked due to a control malfunction or loss of the "Condenser Available" condition, decay heat removal will be effected by automatic actuation of the steam generator power-operated relief valves, or, if these prove ineffective, the steam generator code safety valves. In this event, steam pressure will be maintained at the set pressure of the controlling valve(s) and reactor coolant average temperature will stabilize at approximately the saturation temperature for the steam pressure being maintained.

4. Whenever the Containment Hi-2 pressure setpoint is reached, verify that the Main Steam Isolation Valves have closed. If not, manually close the Main Steam Isolation Valves from the Control Board.

5. Whenever the Containment Hi-3 pressure setpoint is reached, verify that the following have occurred:

   a) Containment Spray is initiated

   b) Containment Isolation Phase B is initiated

   If not, manually initiate Containment Spray and Containment Isolation Phase B.

D. <u>ACCIDENT DIAGNOSTICS</u> (Refer to Figure 2)

1. Evaluate reactor coolant pressure to determine if it is low or decreasing in an uncontrolled manner. If it is low or decreasing, verify that:

   a. all pressurizer spray line valves are closed and

   b. all pressurizer relief valves are closed.

   If not, manually close the valves from the Control Board.

   If the RCS pressure is above the low pressure reactor trip setpoint and is stable or increasing, go to STEP 7.

2. Stop <u>ALL</u> Reactor Coolant Pumps after the high head safety injection pump operation has been verified and when the wide

range reactor coolant pressure is at (plant specific pressure derived from method in Attachment A of letter OG-19).

CAUTION: If component cooling water to the reactor coolant pumps is isolated on a containment pressure signal, all reactor coolant pumps should be stopped within 5 minutes because of loss of motor bearing cooling.

CAUTION: If the reactor coolant pumps are stopped, the seal injection flow should be maintained.

NOTE: The conditions given above for stopping reactor coolant pumps should be continuously monitored throughout this instruction.

*3. IF the condenser air ejector radiation or steam generator blowdown radiation monitor exhibit abnormally high readings, AND containment pressure, containment radiation and containment recirculation sump level exhibit normal readings, THEN go to E-3, "Steam Generator Tube Rupture."

*4. IF the steamline pressure is abnormally lower in one steam generator than in the other steam generators, THEN go to E-2, "Loss of Secondary Coolant."

*These steps may be interchanged.

5. IF containment pressure, OR containment radiation OR containment recirculation sump levels exhibit either abnormally high readings or increasing readings, THEN go to E-1, "Loss of Reactor Coolant".

   NOTE: For very small breaks inside the containment building, the containment pressure increase will be very small and possibly not recognizable by the operator. For very small breaks the containment recirculation sump water level will increase very slowly and early in the transient may not indicate a level increase.

6. IF the containment pressure, containment radiation AND containment recirculation sump water level continue to exhibit stable readings in the normal pre-event range, THEN go to E-2, "Loss of Secondary Coolant".

7. In the event of a spurious safety injection signal, the sequence of reactor trip, turbine trip and safeguards actuation will occur.

   The operator must assume that the safety injection signal is non-spurious unless the following are exhibited:

   a. Normal readings for containment temperature, pressure, radiation and recirculation sump level AND

b. Normal readings for auxiliary building radiation and ventilation monitoring AND

c. Normal readings for steam generator blowdown and condenser air ejector radiation.

IF all of the symptoms a through c above are met and when the following d through g are exhibited: 10/29

d. Reactor coolant pressure is greater than 2000 psig and increasing AND

e. Pressurizer water level is greater than programmed no load water level* AND

f. The reactor coolant indicated subcooling is greater than (insert plant specific value of subcooling equal to full power normal operation). 10/29

g. Water level in at least one steam generator is in the narrow range span, or in the wide range span at a level sufficient to assure that the U-tubes are covered 10/29

*NOTE: Pressurizer water level should trend with reactor coolant system temperature. If the pressurizer water level is low enough to prohibit pressurizer heater operation, re-establish water level by operating the charging system. Energize the heaters.

THEN:

h. Reset safety injection and stop the safety injection pumps. 10/29

CAUTION: Automatic reinitiation of safety injection will not occur since the reactor trip breakers are not reset.

CAUTION: Subsequent to this Step, should loss of offsite power occur, manual action (e.g., manual safety injection initiation) will be required to load the safeguards equipment onto the diesel powered emergency busses. 10/15

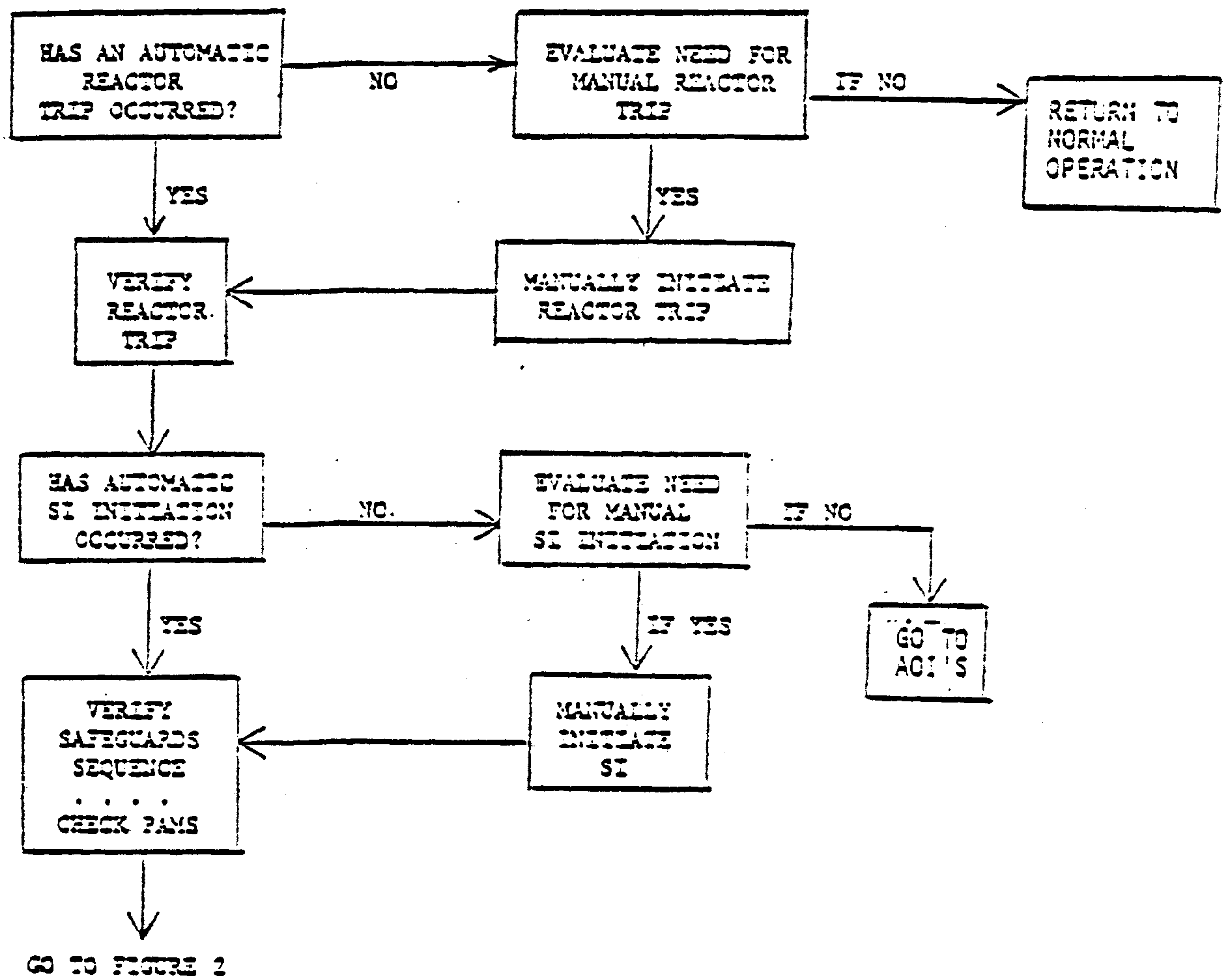
i. Following termination of safety injection place all safety injection pumps in standby mode and maintain operable safety injection flowpaths.

j. Reestablish normal makeup and letdown (if letdown is unaffected) to maintain pressurizer water level in the normal operating range and to maintain reactor coolant pressure at values reached when safety injection is terminated. Ensure that water addition during this process does not result in dilution of the reactor coolant system boron concentration.

k. Reestablish operation of the pressurizer heaters. When reactor coolant pressure can be controlled by pressurizer heaters alone, return makeup and letdown to pressurizer water level control only.

NOTE: IF after securing safety injection and attempting to transfer to normal pressurizer pressure and level control, reactor coolant pressure drops below the low pressurizer pressure setpoint for safety injection actuation OR if pressurizer water level drops below 10% of span, OR the reactor coolant T_H > normal full power T_H, THEN SAFETY INJECTION MUST BE MANUALLY REINITIATED. The operator must rediagnose plant conditions and proceed to the appropriate emergency instruction.

NOTE: IF after securing safety injection and transferring the plant to normal pressurizer pressure and level control, the reactor coolant pressure does not drop below the low pressurizer pressure setpoint for safety injection actuation AND the pressurizer water level remains above 10% span, AND the reactor coolant indicated subcooling is greater than (insert plant specific value of subcooling based on full power normal operation), THEN go to the abnormal operating instructions.



IMMEDIATE ACTIONS

FIGURE 1

Revision 1

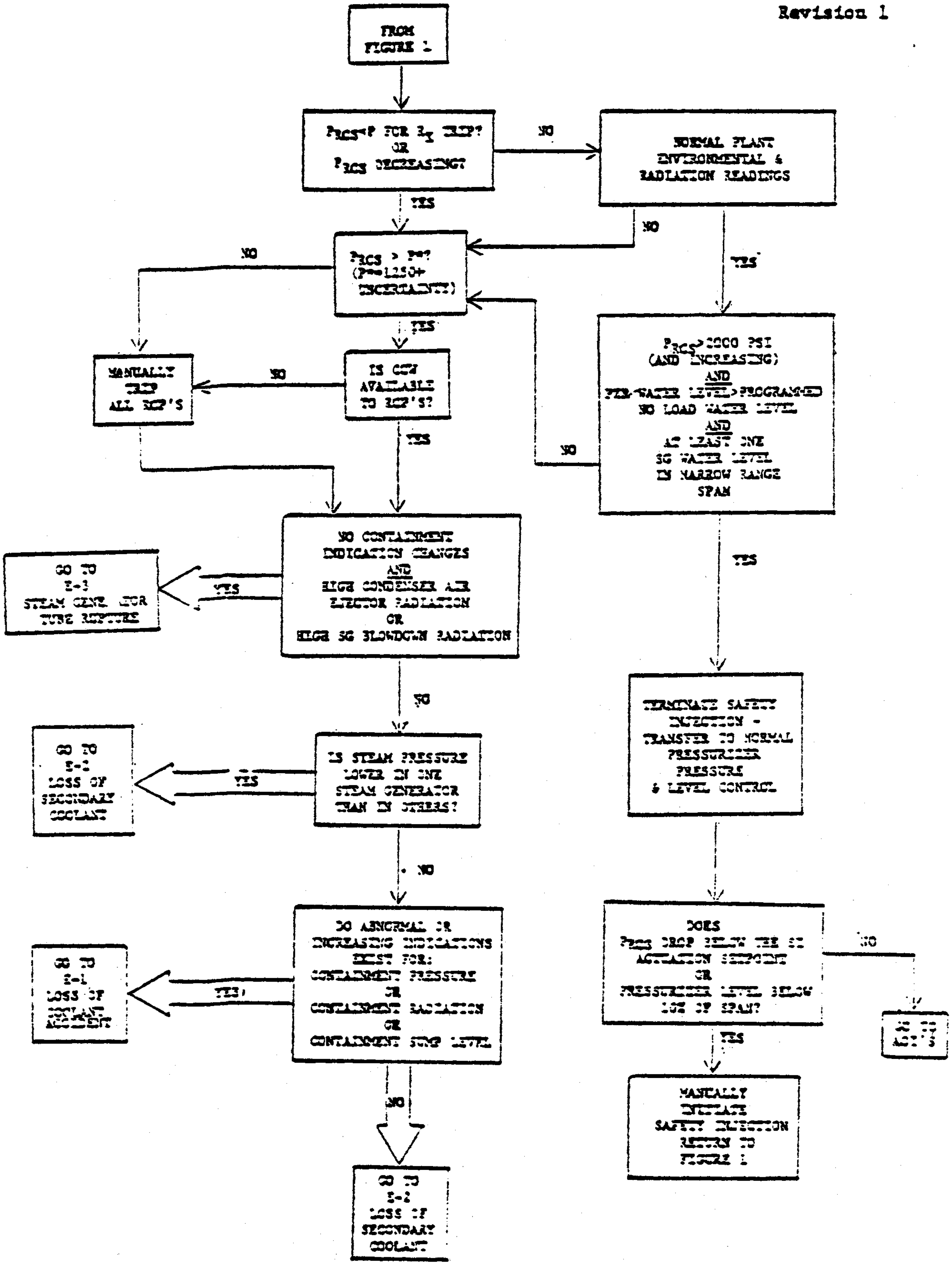


FIGURE 2

E-1
EMERGENCY INSTRUCTION FOR 2, 3 & 4 LOOP PLANTS WITH
NOMINAL 1400 PSI RANGE SAFETY INJECTION PUMPS

LOSS OF REACTOR COOLANT

A. PURPOSE

The objectives of these instructions are to specify required operator actions and precautions necessary to:

1. Verify and establish short term core cooling to prevent or minimize damage to the fuel cladding and release of excessive radioactivity.

2. Maintain long term shutdown and cooling of the reactor by recirculation of spilled reactor coolant, injected water and containment spray system drainage.

B. IMMEDIATE ACTIONS

Refer to section on Immediate Actions of E-0, Immediate Actions and Diagnostics, if not already performed.

C. SUBSEQUENT ACTIONS

CAUTION: Monitor RWST level closely. If RWST level decreases rapidly such that the RWST low level alarm appears imminent, go directly to step 5.

CAUTION: The diesels should not be operated at idle or minimum load for extended periods of time. If the diesels are shut down, they should be prepared for restart.

NOTE: The operator should verify that the Post Accident Monitoring (PAM) instruments are operating and recording. These instruments include wide range RCS temperature and pressure, steam pressure, steam generator water level, containment pressure, RWST water level, condensate storage tank water level, pressurizer water level, and boric acid storage tank water level.

NOTE: The process variables referred to in this Instruction are typically monitored by more than one instrumentation channel. The redundant channels should be checked for consistency while performing the steps of this Instruction.

NOTE: Reactor coolant system isolation valves (LSIV) are optional equipment on the Westinghouse Standard Plants. If a plant is so equipped, the use of LSIV's is not currently recommended during the course of this Instruction. Any use of LSIV's must be justified on a plant specific basis.

NOTE: The pressurizer water level indication should always be used in conjunction with other specified reactor coolant system indications to evaluate system conditions and to initiate manual operator actions.

1. As the water level (PAMS) in the refueling water storage tank decreases under the action of the safeguards pumps, check that the recirculation sump water level instrumentation indicates an increase in water level in the sump. If a sump water level increase is not evident then a re-evaluation of the symptoms in E-0 must be conducted.

   Regulate the auxiliary feedwater flow to the steam generators to restore and/or maintain an indicated narrow range water level (PAMS). If narrow range water level increases in an unexplained manner in one steam generator, go to E-3, Steam Generator Tube Rupture.

NOTE: Monitor the primary water supply (Condensate Storage Tank) for the auxiliary feedwater pumps and upon reaching a low level, switch over to an alternate water supply source.

2. Close all pressurizer power operated relief valves and backup isolation valves.

3. NOTE: The conditions given below for termination of safety injection should be continuously monitored throughout this instruction.

   Ensure that containment isolation is maintained, i.e., not reset until such time as manual action is required on necessary process streams.

<u>IF</u> reactor coolant pressure is above the shutoff head of the safety injection pumps

<u>AND</u> safety injection flow to the Reactor Coolant System is zero

<u>THEN</u> attempt to reestablish the reactor coolant pressure to greater than 2000 psig and pressurizer water level to greater than 50% of span

<u>BY</u> a) Resetting safety injection, and

b) Establishing full charging flow.

CAUTION: Automatic reinitiation of safety injection will not occur since the reactor trip breakers are not reset.

<u>CAUTION:</u> Subsequent to this Step, should loss of offsite power occur, manual action (e.g., manual safety injection initiation) will be required to load the safeguards equipment onto the diesel powered emergency busses.

Safety Injection can be terminated <u>IF</u>:

(A) Reactor coolant pressure is greater than 2000 psig and increasing, <u>AND</u>

(B) Pressurizer water level is greater than 50% of span, AND

(C) The reactor coolant indicated subcooling is greater than (insert plant specific value of subcooling based on full power normal operation), AND

(D) Water level in at least one Steam Generator is in the narrow range span, or in the wide range span at a level sufficient to assure that the U-tubes are covered.

THEN:

(E) Following termination of safety injection place all safety injection pumps in standby mode and maintain operable safety injection flowpaths.

CAUTION: If reactor coolant pressure drops below the low pressurizer pressure setpoint for safety injection or pressurizer water level drops below 20% of span following termination of safety injection flow or the reactor coolant T_H > Normal Full Power T_H, MANUALLY REINITIATE safety injection to ensure core cooling. Go to Section D of E-0 to reevaluate the event, unless this reevaluation has already been performed.

(F) Reestablish normal makeup and letdown (if letdown is unaffected) to maintain pressurizer water level in the normal operating range and to maintain reactor coolant pressure at values reached when safety injection is terminated. Ensure that water addition during this process does not result in dilution of the reactor coolant system boron concentration.

(G) Reestablish operation of the pressurizer heaters. When reactor coolant pressure can be controlled by pressurizer heaters alone, return makeup and letdown to pressurizer water level control only.

(H) Monitor either the average temperature indication of core exit thermocouples (if available) or all wide range reactor coolant temperature T_H (PAMS) to verify that RCS temperature is at least 50°F less than saturation temperature at RCS indicated pressure.

If 50°F indicated subcooling is not present, then attempt to establish 50°F indicated subcooling by steam dump from the steam generators to the condenser or the atmosphere.

CAUTION: If steam dump is necessary, reduce the steam generator pressure 200 psi below the lowest steam safety valve setpoint and maintain a reactor coolant cooldown rate of no more than 50°F/HR, consistent with plant make-up capability.

If 50°F indicated subcooling cannot be established or maintained, then manually reinitiate safety injection. Go to Section D of E-0 to re-evaluate the event, unless this re-evaluation has already been performed.

(I) Perform a controlled cooldown to cold shutdown conditions using Normal Cooldown Procedures if required to affect repairs. Maintain subcooled conditions (at least 50°F indicated subcooling) in the reactor coolant system. If subcooled conditions cannot be maintained, go to Step 4.

4. If the conditions for terminating safety injection in Step 3 are not met, maintain necessary safety injection pumps operating. If any safeguards equipment is not operating, attempt to operate the equipment from the control room or locally. Effect repairs if necessary. If reactor coolant pressure is above the low head safety injection pump shut-off head, manually reset safety injection so that safeguards equipment can be controlled by manual action. Stop the low head safety injection pumps and place in the standby mode.

<u>CAUTION</u>: Whenever the reactor coolant pressure decreases below the low head safety injection shutoff head, the low head safety injection pumps must be manually restarted to deliver fluid to the reactor coolant system.

5. Stop <u>ALL</u> Reactor Coolant Pumps after the high head safety injection pump operation has been verified and when the wide range reactor coolant pressure is at (plant specific pressure derived from method in Attachment A of letter OG-19).

<u>CAUTION</u>: If component cooling water to the reactor coolant pumps is isolated on a containment pressure signal, all reactor coolant pumps are to be stopped within 5 minutes because of loss of motor bearing cooling.

CAUTION: If reactor coolant pumps are stopped, the seal injection flow should be maintained.

NOTE: The conditions given above for stopping reactor coolant pumps should be continuously monitored throughout this instruction:

6. In the case of a break characterized by reactor coolant pressure quickly decreasing below steam generator pressure, go to step 7. In the case of a break characterized by a slowly decreasing reactor coolant pressure or stabilized reactor coolant system pressure above the lowest steam system safety valve setpoint, (plant specific) psig, the following additional manual actions should be taken to aid the cooldown and depressurization of the reactor coolant system:

   a. If the main condenser is in service, open at least one main steamline isolation valves or bypass valves and transfer the steam dump control to steam header pressure control and dump steam to the condenser to lower the reactor coolant temperature (PAMS) and consequently the reactor coolant pressure.

   b. If the main condenser is not in service, dump steam to the atmosphere with the steam relief valves to lower the reactor coolant temperature and consequently the reactor coolant pressure.

CAUTION: Reduce the steam generator pressure 200 psi below the lowest steam system safety valve setpoint and maintain a reactor coolant cooldown rate of no more than 50°F/HR, consistent with plant make-up capability.

7. Go to the Cold Leg Recirculation Instruction presented in Table E-1.1. Note, if the reactor coolant system pressure is above the shut-off head of the high head safety injection pumps, stop these pumps and place them in a standby mode prior to transfer to cold leg recirculation.

   CAUTION: The cold leg recirculation procedures are different for each plant ECCS design. The plant specific procedures should be incorporated in Table E-1.1.

   NOTE: If RWST low level alarm is not imminent, then consideration should be given to performing a preliminary evaluation of the plant status in Steps 9 and 10.

8. If containment spray has been actuated, and if the containment pressure is reduced to nominal operation containment pressure, reset containment spray. Spray pumps should be shut-off and placed in the standby mode with operable flow paths.

9. Periodically check auxiliary building area radiation monitors for detection of leakage from ECCS during recirculation. If significant leakage has been identified in the ECCS, attempt to isolate the leakage. The operator must maintain recirculation flow to the RCS at all times.

10. While the plant is in cold leg recirculation mode, plant operators should make provision for an evaluation of equipment in the plant. This evaluation should include the primary safeguards equipment e.g., RCS pumps and valves, emergency diesels, containment fan coolers, etc. and support equipment e.g., ECCS HVAC equipment, diesel fuel supply, diesel start air supply, sampling of RCS for boron concentration and fuel damage, sampling of containment atmosphere, sampling of recirculation sump, etc. Adjust recirculation sump pH, if required.

11. Prior to the time specified for the plant for the switchover to the hot leg recirculation mode, the operator in the control room should:

    a. Ensure that control room valve switches are aligned in the proper positions for cold leg recirculation mode.

    b. Re-energize the breakers, as required, for valves needed to effect switchover to the hot leg recirculation mode.

12. At (plant specific) hours after the accident, realign the safety injection systems for hot leg recirculation. Go to Table E-1.2.

CAUTION: The hot leg recirculation switchover procedures are different for each plant ECCS design. The plant specific procedures should be incorporated in Table E-1.2.

13. Continue to implement the hot leg recirculation mode of cooling.

14. Recovery procedures for the particular event must be developed and implemented to effect plant return to service.

TABLE E-1.1

COLD LEG RECIRCULATION SWITCHOVER INSTRUCTIONS

See the Cold Leg Recirculation Procedure which is currently incorporated in the plant Emergency Procedures.

TABLE E-1.2

HOT LEG RECIRCULATION SWITCHOVER INSTRUCTIONS

See the Hot Leg Recirculation Procedure which is currently incorporated in the plant Emergency Procedures.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 27 1979

Mr. Cordell Reed, Chairman
Westinghouse Owner's Group
Commonwealth Edison Company
P. O. Box 767
One First National Plaza
Chicago, Illinois 60690

SUBJECT: MODIFICATION OF SMALL-BREAK LOSS-OF-COOLANT ACCIDENT OPERATOR GUIDELINES

Dear Mr. Reed:

We have reviewed the proposed modifications of the small-break loss-of-coolant accident operator guidelines that were transmitted to me by Mr. R. Newton of your group on December 21, 1979. These revisions are concerned with the HPI termination criteria and reflect feedback from several utilities that encountered difficulties in implementing the guidelines we approved on November 5, 1979. The revisions are presented in Enclosure 1.

Item 1(b) and 2(b) of the enclosure refer to conditions in at least one steam generator to ensure an adequate heat sink. The revised criteria are based on an increasing level in a steam generator as verified by auxiliary feedwater flow sufficient to remove decay heat after 20 minutes following reactor trip. We concur that these criteria are sufficient for ensuring an adequate heat sink, however, each utility must document that this criterion is consistent with any other restrictions that may be in force on AFW/steam generator operation such as water hammer considerations.

The remaining items in the revision refer to the primary system subcooling criterion for HPI termination and reinitiation. As noted in our evaluation of the guidelines on November 5, 1979, we are concerned that the criteria be sufficient for establishing subcooled conditions in the core while allowing the operator to terminate HPI to reduce the probability of lifting the pressurizer power operated relief or safety valves. Based on our discussions with Mr. Newton on December 21, 1979, we interpret the "sum of the errors..." used to establish the subcooling criterion at each plant to mean that the errors referred to in 1(a) and 2(a) that are statistically defined will be on a basis of 95% probability and combined by the square root of the sum of the squares method. Other errors will be defined by an envelope of the uncertainty data and the two types of errors will be summed arithmetically. In no event should the subcooling criterion be less than 30°F. We believe in the short term that this criterion is sufficient to establish subcooling because it is used in conjunction with criteria on primary system pressure and heat sink availability. We require that the utilities provide documentation within 21 days showing the errors (and their source) used to develop their subcooling criterion and how the errors were combined. If the calculated subcooling criterion does not permit timely termination of HPI for non-LOCA events, the utilities must document and justify any proposed revisions to their procedures within 21 days.

In the longer term, we require the utilities to install instrumentation and readout devices that will ensure 20°F of actual subcooling based on the criterion (may be modified) in the procedure. This instrumentation must be installed by January 1981. The utilities should acknowledge this commitment within 21 days.

In conjunction with I&E Bulletin No. 79-27, the utilities should confirm that sufficient instrumentation is available on redundant emergency power supplies to permit the operator to perform the actions defined in the proeedure.

Sincerely,

Original signed by:

Denwood F. Ross, Jr., Acting Director
Division of Project Management
Office of Nuclear Reactor Rec :ation

Enclosure:
As stated

E-0 AND E-1 PROCEDURE REVISIONS

The following revisions to the 412 reference emergency operating instructions were agreed to at the WOG procedures subcommittee meeting held in Pittsburgh on December 17, 1979.

1. Revise E-0 SI Termination Criteria as follows:

(a) Change the words in paragraph f on page E-0(HP)-10 from "insert plant specific value of sub-cooling equal to full power normal operation" to "insert plant specific value which is the sum of the errors for the temperature measurement system used, and the pressure measurement system translated into temperature using the saturation tables".

(b) Change paragraph g on page E-0(HP)-10 to read as follows: "Water level in at least one steam generator is stable and increasing as verified by auxiliary feedwater flow to that unit. Auxiliary feedwater flow to the unaffected steam generators should be greater than (a value sufficient to remove core decay heat after 20 minutes following reactor trip) gpm until indicated level is returned to within the narrow range level instrument."

(c) Replace the words "the reactor coolant T_H > normal full power T_H" in the note at the bottom of page E-0(HP)-11 with the following words "reactor coolant sub-cooling drops below the value for SI termination".

(d) Add the following caution after the note at the bottom of page E-0(HP)-11

CAUTION: Stopping and starting of the high head safety injection pumps and the charging/safety injection pumps can cause pump motor overheating or reduced motor life. Hence, if the pumps are restarted once after termination, an additional 15°F of sub-cooling should be added to the required sub-cooling prior to the second termination of the high head pumps.

(e) Change the words in Note at the top of page E-0(HP)-11 from "insert plant specific value of sub-cooling based on full power normal operation" to "insert plant specific value which is the sum of the errors for the temperature measurement system used, and the pressure measurement system translated into temperature using the saturation tables".

2. Revise E-1 SI Termination Criteria as follows:

(a) Change the words in paragraph (C) on page E-1(HP)-4 from "insert plant specific value of sub-cooling equal to full power normal operation" to "insert plant specific value which is the sum of the errors for the temperature measurement system used and the pressure measurement system translated into temperature using the saturation tables".

(b) Change paragraph (D) on page E-1(HP)-4 to read as follows: "Water level in at least one steam generator is stable and increasing as verified by auxiliary feedwater flow to that unit. Auxiliary feedwater flow to the unaffected steam generators should be greater than (a value sufficient to remove core decay heat after 20 minutes following reactor trip) gpm until indicated level is returned to within the narrow range level instrument.

(c) Replace the words "The reactor coolant T_H > Normal Full Power T_H" in the caution at the bottom of page E-1(HP)-5 with the following words "reactor coolant sub-cooling drops below the value for SI termination".

(d) Add the following caution after the caution at the bottom of page E-1(HP)-5.

CAUTION: Stopping and starting of the high head safety injection pumps and the charging/safety injection pumps can cause pump motor overheating or reduced motor life. Hence, if pumps are restarted once after termination, an additional 15°F of sub-cooling should be added to the required sub-cooling prior to the second termination of the high head pumps.

EXPLANATION OF ERRORS:

The errors referred to in 1.a. and 2.a. that are statistically defined will be on a basis of 95% probability and combined by the square root of the sum of the squares (SRSS) method. Other errors will be defined by an envelope of the uncertainty data. The two types of errors will be summed arithmetically. In no event will the total error be less than 30°F.

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The results of the Bulletins & Orders review of the Office of Inspection and Enforcement bulletins, Commission Orders, and the Office of Nuclear Reactor Regulation generic evaluation of feedwater transients, small-break loss-of-coolant accidents, and other Three Mile Island Unit 2 related events in operating plants to confirm or establish the bases for their continued safe operation are summarized. As a result of its review, the Bulletins & Orders Task Force has concluded that (1) the continued operation of the operating plants is acceptable provided that certain actions related to the plants' designs and operation, and training of operators are implemented consistent with the recommended implementation schedules, and (2) the actions taken by the licensees with operating plants in response to the IE bulletins (including the actions specified in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors") provide added assurance for the protection of the health and safety of the public. In addition, the Bulletins & Orders Task Force has independently confirmed the safety significance of those related short-term and long-term actions recommended by other Office of Nuclear Reactor Regulation task forces.

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