Washington Public Power Supply System
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Docket No. 50-508

January 4, 1983
G03-83-08

Mr. G. W. Knighton, Chief
Licensing Branch #3
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: NUCLEAR PROJECT 3
SUPPLEMENTAL INFORMATION ON CONFORMANCE
OF WNP-3 TO STANDARD REVIEW PLAN

Reference: a) Letter #G03-82-1015, G. D. Bouchey to
J. D. Kerrigan, dated October 6, 1982.

Reference a) transmitted Amendment #1 to the WNP-3 FSAR. This amendment contained the initial phase of the WNP-3 review for conformance with the Standard Review Plan (SRP) NUREG-0800, required by 10CFR50.34(g).

In those cases where differences between the WNP-3 design criteria and the SRP acceptance criteria were identified in the initial Supply System review, a schedule was provided detailing when the bases would be presented for concluding that the WNP-3 design criteria are in compliance with the Commission Regulations.

Presented herewith is the material for which commitments were made for the month of December. Included are marked up FSAR pages to show the changes which will be incorporated into a subsequent amendment. In those cases where exception is taken to the SRP acceptance criteria a reference is provided to the FSAR section where further information is provided. Where necessary, additional information will be added to the appropriate FSAR section as indicated on the marked up FSAR pages.

B001

SUPPLEMENTAL INFORMATION ON CONFORMANCE OF WNP-3 TO STANDARD REVIEW PLAN

In certain instances, following a detailed review, we have been able to conclude based on information presented in the FSAR that the WNP-3 design criteria do, in fact, conform to the SRP acceptance criteria. For these cases, with the exception of a change to the FSAR conformance review table (Table 1.8-3), no further change will be necessary.

The detailed comparisons of the WNP-3 design with the guidance contained in Regulatory Guide 1.143 and Branch Technical Position PSB-1, which were scheduled to be submitted at this time, are not yet complete. It has become apparent that much more effort will be required to provide an accurate response than was initially thought. For these cases the individual FSAR pages from Table 1.8-3 have been marked up as shown with revised schedules.

If you require further information of clarification, the Supply System point of contact for this matter is Mr. K. W. Cook, Licensing Project Manager (206/482-4428 ext. 5436).

Sincerely,

For G. D. Bouchey, Manager
Nuclear Safety and Regulatory Programs

AJM:ss

Attachments:

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
2.1.2 Exclusion Area Authority And Control Rev. 2 - July 1981				

ACCEPTANCE CRITERIA

SAR acceptance criteria are based on meeting the relevant requirements of 10 CFR Part 100 with respect to the applicant's legal authority with the designated exclusion area. 10 CFR Part 100 (Ref. 1) in Section 100.3(a) states as follows:

"Exclusion area" means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result."

To meet the requirements of 10 CFR Part 100 the applicant must demonstrate, prior to issuance of a construction permit or limited work authorization, that it has the authority within the exclusion area as required by Section 100.3(a), or must provide reasonable assurance that it will have such authority prior to start of construction. Absolute ownership of all lands within the exclusion area, including mineral rights, is considered to carry with it the required authority to determine all activities on this land and is acceptable.

Where the required authority is contingent upon future procurement of ownership (e.g., by eminent domain proceedings), or by lease, easement, contract, or other means, the exclusion area may be acceptable if OELD can determine that the information provided by the applicant provides reasonable assurance that

	X see remark (1)			(1) Not all exclusion area controls (i.e. easements) are yet in place. ~~Acquisition is anticipated by December 1981.~~ Condemnation proceedings have been concluded. The Supply System will notify the staff when an agreement has been executed for the remaining parcel within the exclusion area.

TABLE 1.8-3

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
3.7.4 Seismic Instrumentation Rev. 1 - July 1981 (Cont'd)				
In addition, the triaxial time history accelerograph located in the containment foundation or in the free field should be connected to the control room, so that peak acceleration level experienced in the basement of the reactor containment structure or in the free field is indicated to the control room operator. The response spectrum recorder in the reactor containment foundation or in the free field is also connected to the control room to indicate if the design response spectra values for discrete frequencies are exceeded during an earthquake.				
4. Comparison of Measured and Predicted Responses				
In the event of an earthquake, the control room operator should be immediately informed through the event indicators. If the instrumentation shows that the peak acceleration or the response spectra experienced at the foundation of the containment building or in the free field exceed the OBE acceleration level or response spectra, the plant should be shut down (Ref. 3) pending permission to resume operations. To help predict the capability of the plant for resuming operations, field inspection of safety-related items should be implemented and the measured responses from both the peak-recording and strong motion accelerographs should be compared with those assumed in the design.	X			
The procedures for comparison of measured and predicted responses are acceptable if a commitment is made to provide detailed comparisons, as outlined below, between measured seismic responses of Category I structures and equipment with calculated responses determined from dynamic analysis. First, the time history records are digitized and corrected for time signal variations and baseline variations. The time history records from the triaxial sensors located in the free field or at the foundation of the containment building are used to calculate response spectra at appropriate critical damping values. The response spectra thus obtained, or the response spectra from the response spectrum recorder, are compared with the design response spectra. In addition, the time history records from the free field triaxial sensor are used as input ground motion for the reactor building dynamic model, including soil where applicable. Amplified response spectra are then calculated at the locations of the other sensors in the reactor building for comparison and correlation with the response spectra directly measured. Structural responses and amplified response spectra are calculated using the free field time history records with the dynamic model for comparison with the original design and analysis parameters. This comparison permits evaluation of seismic effects on structures and equipment and forms the basis for remodeling, detailed analyses, and physical inspection.	X			
5 Inservice Surveillance				
Each of the seismic instruments shall be demonstrated operable by the performance of the channel check, channel calibration, and channel functional test operations at the intervals specified in Table 3.7.4-2.	X		X See Remark (1)	

(1) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the basis for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982.

SRP ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
3.10 Seismic And Dynamic Qualification Of Mechanical And Electrical Equipment (Cont'd) Rev. 2 July 1981				
(4) The criteria of subsections II.1.a(7), (8), and (13) and II.2.a(1), (2), and (3), above, are applicable when tests are conducted on the equipment supports.				
3. GDC 1 of Appendix A and paragraph XVII of Appendix B to 10 CFR 50 establish requirements for records concerning the qualification of equipment. In order to satisfy these requirements, complete and auditable records must be available and maintained by the applicant, for the life of the plant, at a central location. Their files should describe the qualification method used for all equipment in sufficient detail to document the degree of compliance with the criteria of this SRP section. These records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.			X ~~See Remark (1)~~	Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982.
The equipment qualification file should contain a list of all systems equipment and the equipment support structures, as defined in paragraph 2 of subsection I. The equipment list should identify which equipment is NSSS supplied and which equipment is BOP supplied. The equipment qualification file should also include qualification summary data sheets for each piece of equipment, i.e., each mechanical and electrical component of each system, which summarize the component's qualification. These data sheets should include the following information:				
a. Identification of equipment, including vendor, model number and location within each building. Valves that are part of the reactor coolant pressure boundary should be so identified.				
b. Physical description, including dimensions, weight and field mounting condition. Identification of whether the equipment is pipe, floor, or wall supported.				
c. A description of the equipment's function within the system.				
d. Identification of all design (functional) specifications and qualification reports, and their locations. Functional specifications for active valve assemblies should confirm to the Regulatory Position of Regulatory Guide 1.148.				
e. Description of the required loads and their intensities for which the equipment must be qualified.				
f. If qualification by test, identification of the test methods and procedures, important test parameters and a summary of the test results.				
g. If qualification by analysis, identification of the analysis methods and assumptions and comparisons between the calculated and allowable stresses and deflections for critical elements.				

TABLE 1.8-3

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
8.1 Electric Power - Introduction Rev. 2 - July 1981				

ACCEPTANCE CRITERIA

Table 8-1, "Acceptance Criteria and Guidelines for Electric Power Systems," lists the acceptance criteria currently applied by the staff to electric power systems. Implementation of these criteria in accordance with applicable guidelines of regulatory guides and Branch Technical Positions will provide assurance that systems will perform their design safety functions when required.

X See remark (1)

~~X See Remark (1)~~

(1) Implementation of criteria list on Table 8-1 of NUREG-0800 but not presently listed in FSAR Table 8.1-2 will be addressed by December 82.

(1) The WNP-3 design complies with the acceptance criteria Listed in Table 8-1 of NUREG 0800, with the exception of Branch Technical Position "PSB-1, "Adequacy of Station Electric Distribution System Voltages". A detailed comparison of these requirements against the WNP-3 design will be provided by April 1983.

TABLE 1.8-3

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
9.4.5 Engineered Safety Feature Ventilation System Rev. 2 - July 1981				
ACCEPTANCE CRITERIA				
Acceptability of the ESFVS design, as described in the applicant's Safety Analysis Report (SAR), is based on specific general design criteria and regulatory guides.				
The design of safety-related portions of the ESFVS is acceptable if the integrated design of the systems is in accordance with the following criteria:				
1. General Design Criterion 2, as related to the system being capable of withstanding the effects of earthquakes. Acceptance is based on meeting the guidance of Regulatory Guide 1.29, position C.1 for safety-related portions and C.2 for nonsafety-related portions.	X			
2. General Design Criterion 4, with respect to maintaining environmental conditions in essential areas compatible with the design limits of the essential equipment located therein during normal, transient, and accident conditions.	X			
3. General Design Criterion 5, as related to shared systems and components important to safety.		X	See Remark (1)	(1) There are no shared systems and components at WNP 3.
4. General Design Criterion 17, as related to assuring proper functioning of the essential electric power system. Acceptance is based on meeting the guidance of item 2 under subsection A and item 1 under subsection C of the section on "Recommendations" of NUREG-CR/0660 (Ref. 9) relating to the protection of essential electrical components from failure due to the accumulation of dust and particulate materials.		X	See Remark (2)	(2) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982.
5. General Design Criterion 60, as related to the systems capability to suitably control release of gaseous radioactive effluents to the environment. Acceptance is based on meeting the guidance of Regulatory Guides 1.52 and 1.140, as related to design, testing, and maintenance criteria for atmosphere cleanup system, and normal ventilation exhaust system air filtration and adsorption units of light-water-cooled nuclear power plants, Position C.2 and Positions C.1 and C.2, respectively.	X			The WNP-3 design complies with the recommendations of NUREG/CR-0660 with exceptions as noted in Subsection 8.3.1.1.1.7.

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Amendment No. 11 (10/82)

TABLE 1.8-3

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
11.2 Liquid Waste Management Systems Rev. 2 - July 1981				
ACCEPTANCE CRITERIA				
EISB acceptance criteria are based on meeting the relevant requirements of the following regulations:				
1. 10 CFR Part 20, § 20.106 as it relates to radioactivity in effluents to unrestricted areas.	X			
2. 10 CFR Part 50, § 50.34a as it relates to sufficient design information being provided to demonstrate that design objectives for equipment necessary to control releases of radioactive effluents to the environment have been met.	X			
3. General Design Criterion 60 as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment.	X			
4. General Design Criterion 61 as it relates to radioactive waste systems to be designed to assure adequate safety under normal and postulated accident conditions.	X			
The relevant requirements of the Commission regulations identified above are met by using the regulatory positions contained in the following regulatory guides listed below.				
a. Regulatory Guide 1.110 as it relates to performing a cost-benefit analysis for reducing cumulative dose to the population by using available technology.	X See Remark (1)			(1) Refer to FSAR subsections 11.2.3 and 11.3.3.
b. Regulatory Guide 1.143 as it relates to the seismic design and quality group classification of components used in the liquid waste treatment system and structures housing systems and the provisions used to control leakages.	X			
5. 10 CFR Part 50, Appendix I, Sections II.A and II.D as it relates to the numerical guides for dose design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion.	X ~~X See Remark (2)~~			(2) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982.
The liquid radwaste treatment system should have the capability to meet the requirements specified in 10 CFR Part 20, § 20.106 and 10 CFR Part 50, § 50.34a, and General Design Criteria 60 and 61 of Appendix A of 10 CFR Part 50 and the dose design objectives specified in Sections II.A and II.D of Appendix I to 10 CFR Part 50, including provisions to treat liquid radioactive waste.				

(CP/01) 1.8-23

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
11.2 Liquid Waste Management Systems Rev. 2 - July 1981 (Cont'd)				
1. Specific criteria necessary to meet the relevant requirements of the Commission regulations are as follows:				
a. The calculated annual total quantity of all radioactive material released from ~~each~~ reactor at the site to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ.	X			
b. In addition to 1.a above, the liquid radwaste treatment systems should include all items of reasonably demonstrated technology that when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. Regulatory Guide 1.110 provides an acceptable method for performing this analysis.	X See Remark (1)			(1) Refer to FSAR subsections 11.2.1.2 and ~~11.2.3 and 11.3.1~~ 11.2.3
c. The concentrations of radioactive materials in liquid effluents released to an unrestricted area should not exceed the limits in 10 CFR Part 20, Appendix B, Table II, Column 2.	X			
2. The liquid radwaste treatment system should be designed to meet the anticipated processing requirements of the station. Adequate capacity should be provided to process liquid wastes during periods when major processing equipment may be down for maintenance (single failures) and during periods of excessive waste generation. ETSB will accept systems that have adequate capacity to process the anticipated wastes and that are capable of operating within the design objectives during normal operation, including anticipated operational occurrences. To meet these processing demands, ETSB will consider interconnections between subsystems, redundant equipment, and reserve storage capacity.	X			
3. The seismic design of structures housing liquid radwaste systems, the quality group classification of liquid radwaste treatment equipment, and provisions to prevent and collect spills from indoor and outdoor storage tanks should conform to the guidelines of Regulatory Guide 1.143.	(X) X see Remark (2)			
4. ETSB will accept system designs that contain provisions to control leakage and facilitate operation and maintenance in accordance with the guidelines of Regulatory Guide 1.143.	X See Remark (2)			(2) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982. A detailed comparison of the requirements of Regulatory Guide 1.143 with the WNP-3 design will be complete by June 1983

TABLE 1.8-3

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
11.3 Gaseous Waste Management Systems Rev. 2 - July 1981				
ACCEPTANCE CRITERIA				
A. ETSB acceptance criteria are based on meeting the relevant requirements of the following regulations:				
1. 10 CFR Part 20, §20.106, as it relates to radioactivity in effluents to unrestricted areas.	X			
2. 10 CFR Part 50, §50.34a, as it relates to sufficient design information being provided to demonstrate that design objectives for equipment necessary to control releases of radioactive efffluents to the environment have been met.	X			
3. General Design Criterion 3 as it relates to providing protection to gaseous waste handling and treatment systems from the effects of an explosive mixture of hydrogen and oxygen.	X			1
4. General Design Criterion 60 as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment.	X			
5. General Design Criterion 61 as it relates to radioactivity control in gaseous waste management systems and ventilation systems associated with fuel storage and handling areas.	X			
6. 10 CFR Part 50 Appendix I, Sections II.B., II.C., and II.D., as it relates to the numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion.	X			
The requirements of the Commission regulations identified above are met by using the regulatory positions contained in the following regulatory guides:				
a. Regulatory Guide 1.140 as it relates to the design testing and maintenance of normal ventilation exhaust systems at nuclear power plants.	X			
b. Regulatory Guide 1.143 as it relates to the seismic design and quality group classification of components used in the gaseous waste treatment system and structures housing the systems and the provisions used to control leakages.		X See Remark (1)		(1) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982. A detailed comparison of the requirements of Regulatory Guide 1.143 with the WNP-3 design will be complete by June 1983

TABLE 1.8-3

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
11.3 Gaseous Waste Management Systems Rev. 2 - July 1981 (Cont'd)				
3. The seismic design and quality group classification of components used in the gaseous waste management systems and structures housing these systems should conform to the guidelines of Regulatory Guide 1.143. The design should include precautions to stop continuous leakage paths, i.e., to provide liquid seals downstream of rupture discs and to prevent permanent loss of the liquid seals in the event of an explosion.		X See Remark (1)		(1) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982. A detailed comparison of the requirements of Regulatory Guide 1.143 with the WNP-3 design will be complete by June 1983
4. ETSB will accept system designs that contain provisions to control leakage and to facilitate operation and maintenance in accordance with the guidelines of Regulatory Guide 1.143.		X See Remark (1)		
5. ETSB will use the guidelines in Regulatory Guide 1.140 (Ref. 9) for the design, testing and maintenance of HEPA filters and charcoal adsorbers installed in normal ventilation exhaust systems.	X			
If decontamination factors for iodine different from those specified in Regulatory Guide 1.140 are used for design purposes, they should be supported by test data under operating or simulated operating conditions (temperature, pressure, humidity, expected iodine concentrations, and flow rate). The effects of aging and poisoning by airborne contaminants should also be supported by test data.	X			
6. If the potential for an explosive mixture of hydrogen and oxygen exists, the gaseous waste management systems should either be designed to withstand the effects of a hydrogen explosion, or be provided with dual gas analyzers with automatic control functions to preclude the formation or buildup of explosive mixtures.		X See Remark (2)		(2) Refer to FSAR subsection 11.3.1 for a discussion on the design of the gaseous waste management system (GWMS).
a. For a system designed to withstand the effects of a hydrogen explosion, the design pressure of the system should be approximately 20 times the operating absolute pressure (including the intermediate stage condenser for BWR offgas systems).		X See remark (2)		
Small allowances should be made to conform to standard design pressures for off-the-shelf components; i.e., if the system operating pressure is nominally 15 psia but could approach 20 psia by design, piping could be designed to 350 psia, since the next higher standard pressure rating is 600 psia.				
The process gas stream should be analyzed for potentially-explosive mixtures and annunciated both locally and in the control room.				

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
11.3 Gaseous Waste Management Systems Rev. 2 - July 1981 (Cont'd)				
b. For systems not designed to withstand a hydrogen explosion, dual gas analyzers (with dual being defined as two independent gas analyzers continuously operating and providing two independent measurements verifying that hydrogen and/or oxygen are not present in potentially-explosive concentrations) with automatic control functions are required to preclude the formation or buildup of explosive hydrogen/ oxygen mixtures. Gas analyzers should annunciate alarms both locally and in the control room. "High alarm" should be set approximately 2% and "High-high alarm" should be set at a maximum of 4% hydrogen or oxygen.	X See Remarks (1) & (2)			(1) Refer to FSAR subsection 11.3.1 for a discussion of the GWMS design bases and instrumentation provisions. (2) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982.
Control features to reduce potential for explosion should be automatically initiated at "High-high alarm" setting. The automatic control features should be as follows: (1) for systems designed to preclude explosions by maintaining either hydrogen or oxygen below 4%, the source of hydrogen or oxygen (as appropriate) should be automatically isolated from the system (valve should fail in closed position); (2) for systems using recombiners, if the downstream hydrogen and/or oxygen concentration exceeds 4% (as appropriate), acceptable control features include automatically switching to an alternate recombiner train; and (3) injection of diluents to reduce concentrations below the limits specified herein.	(X) X see remark (1)			
Systems designed to operate below 4% hydrogen and below 4% oxygen may be analyzed for either hydrogen or oxygen; systems designed to operate below 4% hydrogen only (no oxygen restrictions), should be analyzed for hydrogen; and systems designed to operate above 4% hydrogen, should be analyzed for oxygen.	X			
For BWR systems with steam dilution upstream of the recombiners, analysis for hydrogen (oxygen is not an acceptable alternative) should be downstream of the recombiners and upstream of the delay portions of the system (analysis upstream of the recombiners is not required if the system is designed to assure the availability of dilution steam during operation). For PWR systems using recombiners, analysis for hydrogen and/or oxygen should be downstream of the recombiners. In addition, unless the system design features preclude explosive mixtures of hydrogen and oxygen upstream of the recombiners, analysis for hydrogen and/or oxygen (as appropriate) should be upstream of the recombiners as well. The number of gas	X See Remark (2)			(3) Compliance is to PWR systems.

TABLE 1.8-3

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
11.4 Solid Waste Management Systems Rev. 2 - July 1981 (Cont'd)				
C. General Design Criterion 60 as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment.	X			
D. General Design Criterion 63 and 64 as it relates to the radioactive waste system being designed for monitoring radiation levels and leakage.	X			
E. 10 CFR Part 71 as it relates to radioactive material packaging.	X			
The relevant requirements of the Commission regulations identified above are met by using the regulatory positions contained in Regulatory Guide 1.143 as it relates to the seismic design and quality group classification of components used in the gaseous waste treatment system and structures housing the systems and the provisions used to control leakages.		X See Remark (1)		(1) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982. A detailed comparison of the requirements of Regulatory Guide 1.143 with the WNP-3 design will be complete by June 1983
Specific criteria necessary to meet the relevant requirements of the Commission's regulations are as follows:				
1. The system design parameters are based on radionuclide concentrations and volumes consistent with reactor operating experience for similar designs.	X			
2. All liquid wet wastes will be solidified in accordance with a process control program prior to shipment offsite or there will be provisions to verify the absence of free liquid in each container and to reprocess containers in which free liquid is detected in accordance with Branch Technical Position (BTP) ETSB 11-3 (Ref. 1).	X			
3. Other wet wastes will be solidified or dewatered (subject to receiving burial site acceptance) in accordance with a process control program or there will be provisions to verify the absence of free liquid in each container and to reprocess containers in which excess water is detected, in accordance with Branch Technical Position (BTP) ETSB 11-3 (Ref. 1).	X			
4. Solid waste containers, shipping casks, and methods of packaging meet applicable Federal regulations, e.g., 10 CFR Part 71 (Ref. 7), and wastes are to be shipped to a licensed burial site in accordance with applicable Commission, Department of Transportation, and State regulations.	X			
5. Processing equipment is sized to handle the design SWS inputs, i.e., the solid waste generation rates reviewed under item I.1 of this SRP section.	X			
6. Onsite waste storage facilities provide sufficient storage capacity to allow time for short-lived radionuclides to decay prior to shipping in accordance with Branch Technical Position (BTP) ETSB 11-3 (Ref. 1). (The bases for the storage time chosen should be given in the safety analysis report).	X			

TABLE 1.8-3

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
11.4 Solid Waste Management Systems Rev. 2 - July 1981 (Cont'd)				
7. SWS components and piping systems, and structures housing SWS components, are designed in accordance with the provisions of Regulatory Guide 1.143 (Ref. 2), and Branch Technical Position (BTP) ETSB 11-3 (Ref. 1).	X See Remark (1)			(1) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982.
8. The SWS contains provisions to reduce leakage and facilitate operations and maintenance in accordance with the provisions of Regulatory Guide 1.143 (Ref. 2) and Branch Technical Position (BTP) ETSB 11-3 (Ref. 1).	X See Remark (1)			
9. For longer term onsite storage (several years, but significantly less than the life of the plant) the storage facility should be designed to the guidelines of Appendix A to this SRP Section (Ref. 3).	~~X See Remark (1)~~ X See remark (2)			A detailed comparison of the requirements of Regulatory Guide 1.143 with the WNP-3 design will be complete by June 1983

(2) The supply System currently has no plans for long term on site storage at WNP-3.

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
11.5 Process And Effluent Radiological Monitoring Instrumentation And Sampling Systems Rev. 3 - July 1981				
ACCEPTANCE CRITERIA				
ETSB acceptance criteria for the process and effluent radiological monitoring instrumentation and sampling systems are based on meeting the relevant requirements of the following regulations:				
A. 10 CFR Part 20, §20.106 as it relates to radioactivity monitoring of effluents to unrestricted areas.	X			
B. General Design Criterion 60 as it relates to the radioactive waste management systems being designed to control release of radioactive materials to the environment.	X			
C. General Design Criteria 63 and 64 as they relate to the radioactive waste management systems being designed to monitor radiation levels and leakage.	X			
Specific criteria necessary to meet the relevant requirements of the Commission regulations identified above are:				
1. Provisions should be made for the instrumented monitoring or for the sampling and analyses of all normal and potential effluent pathways for release of radioactive materials to the environment to meet General Design Criterion 64.	X			
To meet Criterion 64, the design of systems should meet the provisions of Regulatory Guide 1.21 (Position C and Appendix A) (Ref. 2), Regulatory Guide 1.97 (Position C and Table 1 or Table 2, as applicable) (Ref. 3), and Regulatory Guide 4.15 (Position C) (Ref. 4).		X	See Remarks (1) ~~and (2)~~	(1) Conformance to Regulatory Guide 1.97 is being implemented at this time. Existing instrumentation ranges are being expanded and new instruments and monitors are being purchased to meet the requirements of the subject Regulatory Guide. (2) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982.
a. The gaseous and liquid process streams or effluent release points should be monitored and sampled according to Tables 1 and 2	X			
b. For both BWRs and PWRs, liquid wastes and confined volumes of gaseous waste should be sampled batchwise prior to release, in accordance with Regulatory Guide 1.21. Continuous gaseous effluent monitors are not required for open structures, such as PWR turbine buildings or atmospheric vents for liquid waste tanks containing treated or processed liquid waste and located outside of buildings. For liquid and gaseous effluents that cannot be practicably monitored or sampled batchwise, one of the following methods of representative sampling should be provided:	X			
(1) A continuous proportioning sampling system with at least two sample collection tanks. The system should be designed to provide a fixed or measured flow ratio of the sample collected to the sampled stream discharge, or, alternatively		X	See Remark ~~(3)~~ (2)	(2) ~~(3)~~ Radiological monitoring and sampling of liquid and gaseous effluents is provided batchwise.
(2) A periodic automatic grab sampling system with at least two sample collection tanks. The system should be designed to		X	See Remark ~~(3)~~ (2)	

WNP-3 will comply with Regulatory Guide 1.97 recommendations, with exceptions as noted in sections 6.2.2, 11.5.2.4.1 & 12.3.4.2.3.1.1

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SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
11.5 Process And Effluent Radiological Monitoring Instrumentation And Sampling Systems (Cont'd) Rev. 3 - July 1981				
collect a fixed volume of sample at a rate proportional to the measured flow in the sampled stream discharge.				
(3) Radioactive materials other than noble gases in gaseous effluents. A continuous sampling system with replaceable particulate filter and radioiodine adsorber. The system should be designed to automatically take samples at a fixed or measured flow ratio of the sample throughput to the sampled stream discharge flow.		X See Remark (1)		(1) Radiological monitoring and sampling of liquid and gaseous effluents is provided batchwise.
For intermittently operating effluent release points, the system should be designed to automatically take samples whenever there is flow in the effluent stream.	X			
For all of the above samples, a periodic analysis frequency for the collected samples should be specified in the technical specifications.	X			
2. Provisions should be made for the instrumented monitoring of, or the periodic or continuous sampling and analysis of, radioactive waste process systems. To meet Criteria 60 and 63, as they relate to radioactive waste systems and detection of excessive radiation levels and initiation of appropriate safety actions, the design of systems should meet the guidelines of Appendix 11.5-A (this SRP section), Regulatory Guide 1.21 (Position C, as applicable), Regulatory Guide 1.97 (Position C and Table 1 or Table 2, as applicable), and Regulatory Guide 4.15 (Position C).		X See Remarks (1) and (2)		(2) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1987. WNP-3 will comply with the recommendations of Regulatory Guide 1.97, with exceptions as noted in sections 6.2.2, 11.5.2.4.1(e) and 12.3.4.2.3.1.1
a. Provisions should be made to assure representative sampling from radioactive process streams and tank contents. Recirculation pumps for liquid waste tanks (collection or sample test tanks) should be capable of recirculating at a rate of not less than two tank volumes in eight hours. For gaseous liquid process stream samples, provisions should be made for purging sample lines and for reducing plateout in sample lines. Provisions for gaseous sampling from ducts and stacks should be in agreement with ANSI N13.1. (Ref. 5)	X			
b. Where practicable, provisions should be made to collect samples from process waste streams at central sample stations to reduce leakage, spillage, and radiation exposures to operating personnel in accordance with SRP Section 9.3.2.	X			
c. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system.	X			
3. Provisions should be made for administrative and procedural control, for necessary auxiliary or ancillary equipment, and for special features				

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NRC STANDARD REVIEW PLAN

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
11.5 Process And Effluent Radiological Monitoring Instrumentation And Sampling Systems (Cont'd) Rev. 3 - July 1981				
for the instrumented radiological monitoring, sampling, and analysis of process and effluent streams. To meet Criterion 63 and Criterion 64, as they relate to radioactive waste process systems and effluent discharge paths, the design of systems and the implementation of administrative and procedural controls should meet the guidelines of Appendix 11.5-A (this SRP section), Regulatory Guide 1.21 (Position C) and Regulatory Guide 4.15 (Position C).	X	~~X See Remark (1)~~		(1) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by December 1982.
Instrumentation, sampling, and monitoring provisions should conform to the following:				
a. Sampling frequencies, required analyses, instrument alarm/trip setpoints, caliboration and sensitivities, and provisions for preparing composite samples for low-level analyses should be in conformance with Regulatory Guides 1.21 and 4.15. Sampling frequencies and required analyses should be given in the plant technical specifications; these provisions will be reviewed at the OL stage.	X	~~X See Remark (1)~~		
b. Provisions should be made for the necessary instrumentation and facilities to perform gross beta-gamma and gross alpha measurements, isotopic analyses, and other routine analyses in conformance with Regulatory Guide 1.21.	X			
c. Provisions should be made to perform routine instrument calibration, maintenance, and inspections in conformance with guidelines of Regulatory Guide 4.15. The frequencies of such actions should be given in the plant technical specifications. The provisions will be reviewed at the OL stage. Provisions should also be made to replace or decontaminate monitors without opening the process system or losing the capability to isolate the effluent stream.	X	~~X See Remark (1)~~		
d. Isolation valves, dampers, or diversion valves with automatic control features should fail in the closed or safe position. Setpoints for actuation of automatic control features initiating actuation of isolation valves, dampers, or diversion valves should be established in the plant technical specifications. Non-ESF instrumentation provisions for automatic termination or diversion of releases should conform to the design guidance contained in Appendix 11.5-A (this SRP section). ESF instrumentation provisions for automatic termination or diversion of releases are reviewed in SRP Section 7.6 by ICSB.	X			
4. Provisions should be made for the instrumented monitoring or sampling and analysis of identified gaseous effluent paths in the event of postulated accident releases. To meet Criterion 64, as it relates				

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SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
12.3 - 12.4 Radiation Protection Design Features Rev. 2 - July 1981 (Cont'd)				
b. General Design Criterion 61 - Fuel Storage and Handling and Radioactivity Control as it relates to occupational radiation protection aspects of fuel storage, handling, radioactive waste, and other systems designed to assure adequate safety during normal and postulated accident conditions, with suitable shielding and appropriate containment and filtering systems.	X			
8. 10 CFR Part 70, Section 70.24, "Criticality Accident Requirements" as it relates to procedures and criteria for monitoring for criticality accidents involving special nuclear material.		X	See Remark (1)	(1) Specific exemption ~~will be~~ has been requested for WNP-3. ~~Pursuant information will be available by December 1982.~~ See letter # G03-82-1324.
The following regulatory guides, NUREGs and industry standards provide information, recommendations and guidance and in general describe a basis acceptable to the staff for implementing the requirements of the regulations identified above:				
1. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Waters Reactors," as it relates to assumptions used in evaluating gaseous concentrations of radionuclides in containment and plant systems, following a loss-of-coolant accident for BWRs.		X	See Remark (2)	(2) Not applicable to PWRs.
2. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," as it relates to assumptions used in evaluating gaseous concentrations of radionuclides in containment and plant systems, following a loss-of-coolant accident for PWRs.	X			
3. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," as it relates to methods for determining gaseous concentrations of radionuclides in containment following an accident.	X			

TABLE 1.8-3

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SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
12.3 - 12.4 Radiation Protection Design Features Rev. 2 - July 1981 (Cont'd)				
10. Regulatory Guide 8.12, "Criticality Accident Alarm Systems," as it relates to a system acceptable to the staff for meeting the Commission's requirements for a criticality accident alarm system.		X	See Remark (1)	(1) Specific exemption ~~will be~~ has been requested for WNP-3 ~~Further information will be available by December 1982.~~ See letter # G03-82-1324
11. Regulatory Guide 8.19, "Occupational Dose Assessment in Light-Water-Reactor Power Plants Design Stage Man-Rem Estimates," as it relates to a method acceptable to the staff for performing an assessment of collective occupational radiation dose as part of the ongoing design review process so that such exposures will be ALARA.	X			
12. NUREG-0103 "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors," as it relates to radiation protection considerations in the applicability, format, and implementation of the Babcock and Wilcox Technical Specification package.		X	See Remark (2)	(2) Not applicable to WNP-3 design which has a CE NSSS
13. NUREG-0123, "Standard Technical Specifications for General Electirc Boiling Water Reactors (BWR's)," as it relates to radiation protection considerations in the applicability, format, and implementation of the General Electric Technical Specification package.		X	See Remark (2)	
14. NUREG-0212, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," as it relates to radiation protection considerations in the applicability, format, and implementation of the Combustion Engineering Technical Specification package.		X	See Remark (3)	(3) Technical specifications will be completed prior to issuance of OL
15. NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," as it relates to radiation protection considerations in the applicability, format, and implementation of the Westinghouse Technical Specification package.		X	See Remark (2)	
16. NUREG-0718 and NUREG-0737 as they relate to implementing task Action Plan Items II.B.2 and II.F.1(3) for construction permit and operating license applications.		X	See Remark (4)	(4) II.F.1 will be met prior to full power operation and issuance of OL respectively
17. ANSI/ANS-HPSSC-6.8.1-1981, "Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors," as it relates to criteria for establishment of locations for fixed continuous area gamma radiation monitors, and for design features and ranges of measurement.			See Remark (5)	(5) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by June 1983.
18. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," as it relates to the principles which apply in obtaining valid samples of airborne radioactive materials, and acceptable methods and materials for gas and particle sampling.	X			

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
12.3 - 12.4 Radiation Protection Design Features . Rev. 2 - July 1981 (Cont'd)				
19. ANSI N16.2-1969, "Criticality Accident Alarm Systems," as it relates to guidance for the prevention of criticality accidents in the handling, storing, processing, and transporting of fissionable materials.		X	See Remark (1)	(1) Specific exemption ~~will be~~ has been requested for WNP-3. ~~Further information will be available by December 1982~~ See letter # G03-82-1324
10. ANSI N101.6-1972, "Concrete Radiation Shields," as it relates to requirements and recommended practices for construction of concrete radiation shielding structures.	X			
21. "Reactor Shielding for Nuclear Engineers," N. M. Schaeffer, Editor; Published by USAEC-OTS, 1973," as it relates to the shield designing process including physics, radiation transport, shielding calculations, special problems, and materials.	X			
22. "Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants," Stone and Webster Topical Report, RP-8, 1974," as it relates to the approach and objectives of the shield design and the methods of analysis employed in determining specific shielding requirements.			See Remark (2)	(2) Where differences exist between the WNP-3 design criteria and the acceptance criteria identified in this SRP, the bases for concluding that the WNP-3 design criteria are in compliance with the Commission's regulations will be provided by June 1983.
1. FACILITY DESIGN FEATURES				
Acceptability of the facility design features will be based on evidence that the applicant has fulfilled the dose limiting requirements of 10 CFR Part 20.101, 20.103, and 20.104, as well as the radiation protection aspects of General				
Design Criteria 19 and 61, and 10 CFR 50.34. This includes evidence that major exposure accumulating functions (maintenance, refueling, radioactive material handling, processing, etc., in-service inspection, calibration, decommissioning, and recovery from accidents) have been considered in plant design and that potential radiation exposure from these activities will be kept ALARA in accordance with 10 CFR Part 20.1(c) and Regulatory Guides 8.8 and 8.10 by radiation protection features incorporated in the design. Such features may include (1) ease of accessibility to work and inspection and sampling areas, (2) the ability to reduce source intensity, (3) design measures to reduce the production, distribution, and retention of activated corrosion products, (4) the ability to reduce time required in radiation fields, and (5) provision for portable shielding and remote handling tools. Access control will be judged for acceptability in accordance with the requirements of 10 CFR 20.203 or access control alternatives in Standard Technical Specifications (NUREG- 0103, 0123, 0212, and 0452).	X			

SRP/ACCEPTANCE CRITERIA	COMPLIANCE YES	NO	N/A	REMARKS
12.3 - 12.4 Radiation Protection Design Features Rev. 2 - July 1981 (Cont'd)				
c. The in-plant accident radiation monitoring systems will be acceptable if they meet the following criteria:				
1. Personnel have the capability to assess the radiation hazard in areas which may be accessed during the course of an accident, in accordance with the criteria of Item II.F.1 of NUREG-0718 and 0737, and Regulatory Guide 1.97.	X		See Remark (1)	(1) Task Action Plan Item II.F.1 of NUREG 0737 shall be met prior to full power operation.
2. Portable instruments to be used in the event of an accident should be placed so as to be readily available to personnel responding to an emergency.	X			
3. Emergency power should be provided for installed accident monitoring systems.	X			
4. The accident monitoring systems should have usable ranges which include the maximum calculated accident levels; and should be designed to operate properly in the environment caused by the accident.	X			
5. Applicants for CPs and OLs should provide two high-range radiation monitor systems in containment which are documented to meet the requirements of Table II.F.1 of NUREG-0718 and 0737.	X		See Remark (2)	(2) This item shall be met prior to full power operation.
d. Regulatory Guide 1.21, Appendix A, provides useful guidance about effluent monitoring, that is applicable to the acceptability of airborne radioactivity monitoring in-plant. Regulatory Guide 8.2 includes guidance on surveys to evaluate radiation hazards. American National Standard ANSI N13.1-1969 provides detailed guidance on sampling airborne radioactive materials in nuclear facilities and may be used for acceptance criteria on the actual sampling process and certain techniques involved. Regulatory Guide 8.8 provides further guidance or monitoring systems.	X			
e. Instrumentation to monitor for accidental criticality will be acceptable if it meets the criteria of 10 CFR Part 70.24 (a)(1), Regulatory Guide 8.12, and ANSI Standard N16.2.		X	See Remark (3)	(3) Specific Exemption from the requirements of 10CFR 70.24 ~~was~~ has been ~~to~~ requested for WNP-3. ~~Further information will be available by December 1982~~ See letter # 603-82-1324
5. DOSE ASSESSMENT				
The dose assessment will be acceptable if it documents in appropriate detail the assumptions made, calculations used, the results for each radiation zone, including numbers and types of workers involved in each, expected and design dose rates, and projected person-rem doses, in accordance with Regulatory Criteria 8.19.	X			

(11.3.1 cont)

the operator to manually open the gas decay tank isolation valve and activate the discharge line isolation valve. Should activity in the gas being discharged exceed a predetermined level, the discharge line radiation monitor signals the isolation valve to close. Radiation monitor failure also closes the isolation valve. The isolation valve itself fails closed. There are also manually locked closed operated valves available to cut off flow.

The GWMS design provides for the reduction of operator radiation exposures to levels that are as low as reasonably achievable (ALARA) within the design objectives provided in 10CFR Part 50. The designed techniques incorporated to reduce operator radiation exposure include features such as remote valve operators, complete system and component purging capability, and provisions for shielding on package components.

In addition, all instruments in the GWMS which must be frequently monitored or actuated by operators during normal operation are provided with indication, ~~recording~~, control or alarm on the WMS control panel. Discharge monitors and discharge flow indicators have readouts in the Control Room. The waste gas discharge monitor alarms in the Control Room. The gas analyzer and gas recombiner have local alarms on local panels which activate general alarms ~~on the WMS control panel.~~ in the control room

Design features to reduce maintenance, equipment downtime, leakage, or otherwise improve ALARA operation are addressed in Section 12.1. The design activity inventories for shielding of GWMS components and the bases for the values are provided in Section 12.2. Further exposure reduction features are addressed in Section 12.3.

The Gas analyzer package contains two hydrogen analyzers and one oxygen analyzer. None of the GWMS components are designed to withstand a hydrogen explosion. Hydrogen and oxygen concentrations in the gas decay tanks, nitrogen recycle tank and gas surge tank are periodically monitored by the gas analyzer during periods when they are in use. The gas analyzer also is connected to the spent resin tank, CVCS holdup tanks, nitrogen recycle header, nitrogen recycle tank, volume control tank and the gas stripper to allow determination of hydrogen and oxygen concentration of gases entering GWMS components. The analyzer will alarm on high oxygen in nitrogen or hydrogen cover gas tanks, and on high hydrogen in aerated cover gas tanks. The analyzer indicates when and where mixtures are approximately 50 percent of explosive mixtures to allow time to take corrective action. This action would consist of terminating the flow of improperly mixed gas and initiating flow of nitrogen cover gas to the affected components where appropriate. The gas recombiner can also be used to remove any hydrogen or oxygen accumulation from the nitrogen recycle tank.

The gas analyzer package is provided to monitor hydrogen and oxygen concentrations in various plant components where potentially explosive mixtures could develop. The gas analyzer operates continuously by monitoring through a programmed sequence of sample sources. The gas analyzer is also capable of monitoring a single source for as long as desired by manually overriding the sequence selector. Each sample source is purged, analyzed and recorded. Continuous recording of sample concentrations allows for the detection and observation of trends which may be developing. When the analysis indicates that the hydrogen or oxygen concentration of a sample exceeds a pre-determined set point an alarm is annunciated.

This monitor is located on the 362.5 ft. level of the Reactor Auxiliary Building. It takes a sample of the water discharged from the gas stripper and returns it to the same line. Physically it is a fluid stream monitor as described in Subsection 11.5.2.3.2, and it requires a sample pump. In this case the microprocessor has been removed from the skid and placed on a nearby wall to protect it from possibly high radiation fields present near the skid.

The measured activity levels are automatically transmitted to the system computer where they are recorded and available for display through the system CRTs. If the activity exceeds pre-established setpoints an annunciation is made through the system CRTs and event typer. The receipt of these alarms will alert the operator so that additional radiation surveys, sampling and analysis can be effected in order to determine the cause of the problem. The alarm setpoints are to be set between the measured activity levels of the degasified reactor coolant and the maximum level of contamination permissible in this system. The setpoints may be adjusted continuously over the entire range of the monitor. The range of the monitor is from 10^{-4} to 10 μCi/cc which is the practical range of interest for normal power operation using a simple single detector radiation monitor.

e) CVCS Letdown Radiation Monitor

The CVCS letdown radiation monitor will alert operations personnel to an increase in the radioactive contamination of reactor coolant as quickly as possible.

insert 1

This monitor is located on the 373.5 ft. level of the Reactor Auxiliary Building. It receives a continuous sample of the CVCS letdown, in parallel with the boronometer and is upstream of the purification filter. System process travel time delays the sample for approximately two minutes delay to allow activation products with short half-lives, particularly N-16 to decay. Physically, it is a fluid stream monitor as described in Subsection 11.5.2.3.2, which does not require a sample pump. In addition, two modifications have been made, first the microprocessor has been removed from the skid and placed on a nearby wall to protect it from possibly high radiation fields present near the skid, and second a removable attenuator has been provided which may be manually placed between the sample volume and the detector. 2

The measured activity levels are automatically transmitted to the system computer where they are recorded and available for display through the system CRTs. If the activity exceeds pre-established setpoints, an annunciation is made through the system CRTs and event typer. The receipt of these alarms will alert the operator so that additional radiation surveys, sampling, and analysis can be performed in order to detemine whether the increase of contamination is due to damage to the fuel cladding or due to some other cause such as a crud burst or iodine spiking. The level of the setpoints will have to be adjusted periodically during operation to allow for the gradual buildup of contamination in the reactor coolant. The setpoints may be adjusted continuously over the entire range of the monitor. The range of this 2

Insert 1:

This monitor will be used in lieu of the High Range Circulating Primary Coolant Monitor required by Regulatory Guide 1.97 Rev. 2 for the detection of fuel cladding failure. Regulatory Guide 1.97 recommends that Category 1 redundant detection systems be installed to measure this parameter. The recommended range is from 1/2 the Technical Specification limit up to 100 times the Technical Specificeion limit in Rads per hour. There is no instrument available to accomplish this, additionally instrumentation in this range would be saturated by N_{16}. Thus its capability would be unavailable until after shutdown.

There are four of these monitors. The sample points for two of them are located on the 362.5 ft. level of the Reactor Auxiliary Building. One of these samples air being exhausted from the north side and the other samples air being exhausted from the south side of the 335 ft. level. The other two sample points are located on the 390 ft. level of the Reactor Auxiliary Building. One of these samples air being exhausted from the north side and the other from the south side of the 335 ft. and 362.5 ft. levels. The air samples are withdrawn from the exhaust ducts through a multipoint isokinetic sampling nozzle array (per ANSI 13.1) from a point downstream of the confluence of the ducts carrying exhaust air from the area being monitored. In each case the sample is routed to a skid mounted radiation monitor that is located within a few feet of the sample point. The sample pump is mounted on the monitor skid. Physically this monitor is a two stage airborne monitor as described in Subsection 11.5.2.3.3.2 which uses a moving particulate filter.

The measured activity levels for both the particulate and gas channels are automatically transmitted to the system computer where they are recorded and available for display through the system's CRTs as described in Subsection 11.5.2. If the activity exceeds pre-established setpoints an annunciation is made through the system's CRTs and event logger.

The receipt of these alarms will alert the operator to the presence of an unusual level of airborne radioactivity coming from a paricular area so that additional surveys, sampling and equipment isolation can be effected in order to locate and control the source of the contamination. The setpoints may be adjusted over the entire range of the monitor. Additional information about this monitor may be found in Table 12.3.4-2.

insert 1

12.3.4.2.3.1.2 Control Room Air Intake Radiation Monitor

The Control Room air intake radiation monitors provide plant operations personnel with measurements and records of the radioactive contamination of air entering the Control Room ventilation system. If the measured contamination exceeds pre-established limits these monitors produce a signal to automatically isolate the Control Room air intake and thereby protect the habitability of the Control Room.

These monitors are located in both the Control Room air intake plenums which are on the 442 ft. level of the Reactor Auxiliary Building. Refer to Figure 12.3-5a. There are a total of four monitors, two in each of the two plenums. The detector assembly for each monitor is mounted from the plenum ceiling and observes air downstream of the tornado missile protection but upstream of the intakes isolation valves louvers. This location permits the continued observation of the contamination of the outside air after the intake has been isolated. 1

Each Control Room air intake radiation monitor is similar to a refueling pool area radiation monitor (Subsection 11.5.2.4.1i) except for its detector assembly and some changes in its microprocessors software. A schematic diagram of one of these monitors is shown on Figure 12.3-32. These monitors are seismically qualified, Class IE and powered in pairs from the A or B safety busses using the same bus as their respective air intake. 1

Insert 2

These monitors, which can detect airborne radioactive material leaking from containment will be used in lieu of the intermediate range Area Radiation Monitors recommended by Regulatory Guide 1.97 Rev. 2 for detection of a Containment Breach in areas with penetrations and hatched (Type C variable). An increase in radiation levels in these areas would be due primarily to streaming through the penetration or direct shine from the containment caused by elevated exposure rates inside containment. Any additional increase due to airborne radioactivity leaking from the containment would not be discriminated by the area radiation monitors suggested by Regulatory Guide 1.97 Rev. 2 from streaming or direct shine.

SIS recirculation sump to the suction of the engineered safety feature pumps. The stainless steel pipes are welded to the carbon steel pipes at the SIS recirculation sump so that water cannot enter the annulus formed by the concentric pipes. Outside the containment the stainless steel pipes are sealed to the guard pipes by means of a stainless steel bellows. The bellows seal allows for anticipated differential movement due to thermal or seismic forces.

The SIS recirculation sump isolation valves, Tag Nos. 2CS-B001SA and 2CS-B002SB, used in the recirculation lines are of the Butterfly type. To ensure containment integrity in the event of stem leakage the valves are provided with two full sets of packing separated by a lantern ring and a leakoff connection. Each set of packing is capable of withstanding the system design conditions without visible leakage. The leakoff piping is 3/4 inch schedule 40S. Isolation valves, Tag Nos. 2CS VS330SA and 2CS VS331SB are provided in the leakoff lines. A temperature element is provided in the leakoff line which will provide an alarm in the Control Room if there is stem leakage. The leakoff piping up to the leakoff isolation valves including the associated instrumentation is designed in accordance with ASME Section III, Code Class 2 requirements. The leakoff isolation valves are normally open. When stem leakage occurs through the first set of packing the temperature rise in the leakoff line will activate an alarm in the Control Room to alert the operator. Administrative procedures will require the operator to isolate the leakage by closing the leakoff isolation valves. Extension stems are provided in the isolation valves to permit remote manual operation from a shielded area for protection of operating personnel. Isolation of the leakoff valve and the second set of packing in the main valve will help maintain the containment integrity with no leakage of sump water in the Reactor Auxiliary Building. The containment sump isolation valves are conservatively designed to prevent any steam packing assembly blowoff effects due to excessive pressures. The valves are designed in accordance with class 150 pressure temperature rating requirements. The system design pressure is 50 psig. To ensure packing integrity the compression of the permanent packing will be checked every refueling cycle.

The primary source of post-accident debris that could be generated inside the containment and which could potentially clog the SIS sump screens, would be the disintegration of thermal insulation that is provided for equipment and piping. No other type of debris is expected to clog the screens or prevent in any way the flow of water. The WNP-3/5 design utilizes two types of insulation inside the containment. All piping and equipment that are part of the Reactor Coolant Pressure Boundary, including the secondary side of the steam generators, use metallic reflective insulation. Table 6.2.2-3 summarizes the type of insulation and the quantities for all equipment and piping inside the containment. The materials of construction of each type of insulation are listed in Table 6.2.2-4.

Permanent insulation assemblies are attached by stainless steel straps and fasteners of the expansion type which prevent overstressing of the bands or damage to the insulation coverings due to thermal expansion of the equipment

Insert 3

These temperature detectors will be used in lieu of the intermediate range Area Radiation Monitors required by Regulatory Guide 1.97 Rev. 2 for detection of Containment Sump Leakage past the packing of valves 2CS-B001SA and 2CS-B002SB (Type C variable). In an accident resulting in failed fuel, and when the Containment Spray pump and LPSI pump are used for shutdown cooling, extremely high radiation fields could exist in this area. Any additional exposure from the 3/4" leakoff line would not be detectable, even with lead collimation.