DOE/SF/20218--T3

AECL Technologies Inc.

Optimization and Implementation Study
of Plutonium Disposition using Existing
CANDU Reactors

Final Report
September, 1996

Work performed for the U.S. Department of Energy
Contract DE-AC03-94SF20218
by

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1. INTRODUCTION

Since early 1994, the Department of Energy has been sponsoring studies aimed at evaluating the merits of disposing of surplus U.S weapons plutonium as Mixed Oxide (MOX) fuel in existing commercial Canadian Pressurized Heavy Water reactors, known as CANDU's. These studies, along with studies of other disposition options, will form the basis for a Programmatic Environmental Impact Statement and a Record of Decision scheduled later this year. The CANDU studies have been conducted by AECL Technologies, the U.S. Office of Atomic Energy of Canada, utilizing a team of specialists from AECL, Ontario Hydro Nuclear, and other U.S. and Canadian firms.

The first report, submitted to DOE in July, 1994 (the 1994 Executive Summary is attached hereto), identified practical and safe options for the consumption of 50 to 100 tonnes of plutonium in 25 years in some of the existing CANDU reactors operating at the Bruce A generating station, on Lake Huron, about 300 km north east of Detroit. By designing the fuel and nuclear performance to operate within existing experience and operating/performance envelope, and by utilizing existing fuel fabrication and transportation facilities and methods, a low cost, low risk method for long term plutonium disposition was developed. The integrated system was predicted to require about 4 years before initial MOX fuel operations could begin. No changes to the existing reactor system, other than provision for safe and secure storage of new fuel, was required.

Since the 1994 study, DOE has sponsored additional studies aimed at determining the feasibility, cost and schedule for producing the CANDU MOX fuel in a private facility in Barwell South Carolina. The 1994 study was based on the assumption that an existing Government Facility, the FMEF, in Hanford, Washington, would be used for CANDU MOX fabrication. AECL Technologies submitted a report to DOE in August, 1995, evaluating the feasibility and cost of converting the existing, private facility in Barwell South Carolina for use in fabricating CANDU MOX Fuel.

In December, 1995, in response to evolving Mission Requirements, the DOE requested a further study of the CANDU option with emphasis on more rapid disposition of the plutonium, and retaining the early start and low risk features of the earlier work. This report is the result of that additional work.

1.1 OBJECTIVES OF 1996 STUDY

The primary objective of the study, as in the 1994 study, was to dispose of 50 tonnes of surplus weapons plutonium in the most efficient manner possible, giving due consideration to safety, environmental protection, safeguards and security, economics, and non-proliferation concerns. DOE requested the development of an optimized core design which closely matches programmatic needs. In particular:

1. The core design should permit maximum plutonium throughput without significantly impacting the existing safety envelope for the Bruce Reactors, and without requiring a protracted fuel development program.
2. To minimize overall cost by minimizing the duration of the MOX campaign, and keeping the heavy metal throughput through the MOX facility low, AECL was asked to consider increasing the plutonium concentration in the MOX fuel, either by using the CANFLEX design, or by other means.
3. AECL was asked to describe the required fuel qualification plan (for use by Oak Ridge in the development of overall plans for fuel qualification). Fuel qualification was not to be on the critical path. Plans should focus on identifying the sequence of operations and facilities to

prepare test specimens, conduct test irradiations, and ultimately, qualification in the Bruce reactors.

4. The cores should be designed for an overall economic basis to the Government, considering all phases of the fuel cycle including oxide preparation, fuel fabrication, and reactor operations.
5. *Spent fuel acceptability in the Canadian repository is to be evaluated.*
6. The core designs should consider the feedstock of materials expected to be declared surplus, including the isotopic concentrations and the presence of Gallium.
7. Revised cost (and schedule) estimates should be performed to determine where significant cost and schedule benefits might be realized including the impact of using existing facilities for making the MOX fuel.

1.2 RESULTS: COMPARISON WITH 1994 STUDY

This supplemental study was initiated in early January, 1996, with a conceptual evaluation of various fuel designs which could meet the revised mission objectives of DOE. Some 12 different fuel designs were identified, and underwent preliminary evaluation. Two preferred designs were selected for more detailed evaluation in March and form the basis for this report. A preliminary report indicating the rough parameters of these preferred designs was submitted to ORNL on March 6, 1996 and reviewed at a mid-term project review meeting with DOE and the National Laboratories on May 8, 1996.

In summary, AECL and Ontario Hydro have arrived at enhanced fuel designs which meet the new mission requirements of DOE. The enhanced fuel designs permit more than twice the plutonium throughput than the earlier 1994 designs, whilst remaining at or below the performance limits of the existing natural uranium fuel. As a result, the mission duration (from startup on MOX fuel to insertion of the last MOX fuel into the reactor) is now under 12.5 years as compared to 25 years in the 1994 study. Consequently, the overall mission cost is substantially reduced. Lead time from commercial contracts to startup of the first Bruce reactor on a full MOX core remains favorable at about 5 years. The additional year of preparatory time reflects a more detailed evaluation of the requirement for transitioning to the first full MOX core, which is described in more detail in section 3.0, CANDU MOX Fuel Qualification Program.

A section by section comparison of the current study results, as compared with those in our 1994 study, is as follows:

Reference Reactor - Ontario Hydro's 4 x 825 MW(e) Bruce A generating station remains as the reference site for this study. Bruce A is particularly suited for this mission because of the details of its core design, its base load operating mode, its proximity to the U.S. border, and its existing infrastructure. For example, this new study analyzes operation of the Bruce A reactors with 97 % purity of the heavy water moderator and coolant, as compared to the standard 99.75 %. The study concludes this is acceptable from a safety and licensing standpoint. The small increase of about 3 % in light water concentration within the moderator and coolant allows for a significant increase in plutonium concentration in the MOX fuel bundles, consistent with the revised mission objectives requested by DOE.

Nuclear and Fuel design Reference Option: As in the 1994 study, the reference fuel bundle design consists of a 37 element design of the type that is currently the standard natural uranium design for all operating CANDU reactors. By increasing the dysprosium concentration in the inner 7 elements from 7 % to 15 %, and reducing the heavy water purity in the moderator as described above, the plutonium metal loading in a standard fuel bundle has been increased from 230 grams in 1994 to 330 grams in this

1996 design. As described in section 2, the MOX fuel operates under these conditions well within the same burnup and power rating envelope as standard CANDU fuel, and the overall core nuclear parameters allow operation within the existing license envelope. As part of this 1996 study, AECL and Ontario Hydro presented the core conceptual design and preliminary analysis to the Staff of the Canadian Atomic Energy Control Board. AECB staff indicated they would be prepared to consider a license application from Ontario Hydro for operation of Bruce A reactors with MOX fuel when the utility is in a position to proceed, and communicated their need for advance notice for planning and resourcing such an activity.

Fuel Supply - Since the 1994 study, AECL and Ontario Hydro have evaluated alternative U.S sites for conversion to CANDU MOX fuel fabrication facilities. For example, in a report submitted to DOE in August, 1995, it was concluded that the Barnwell Nuclear Fuel Plant, an existing never used fuel reprocessing plant partially licensed by the Nuclear Regulatory Commission, could be used for this function, and could be converted at a cost of about $172 million. (This compares with a revised estimate for the FMEF facility at Hanford of about $112.) It should be noted that AECL and Ontario Hydro plan in the near future to reevaluate the capital and operating costs of CANDU MOX facility conversion in the U.S., using European style process equipment as requested by DOE, and this may change our cost estimates. With regard to the facility throughput, the current 1996 enhanced core design has a slightly higher fuel burnup, and this leads to a slightly smaller throughput, as shown on the following table:

CHARACTERISTICS WITH TWO BRUCE REACTORS USING STANDARD MOX FUEL

	1994 STUDY	1996 STUDY
BURNUP (mwd/t)	9,700	10,000
bundles per year	9,050	8,760
plutonium /bundle (kg)	.23	.33
plutonium metal per year (MT)	2.1	2.9
MOX FACILITY THROUGHPUT (30 MOX pins per bundle, dysprosia pins ignored)		
Mixed Oxide output (MT/YR)	156.6	151.5
Metal Output (MTHM/YR)	138.1	133.6

Although these throughput requirements would lead to a slight reduction in the cost of MOX facility operation, the changes are relatively small, as compared to uncertainties in overall cost estimates, and have therefore not been considered in overall cost estimates included in section 6.

Nuclear and Fuel Design - CANFLEX Option - As in the 1994 study, an alternate fuel design utilizing the 43 element CANFLEX fuel bundle has also been analyzed. The enhanced CANFLEX design also utilizes higher plutonium concentrations as compared to the 1994 CANFLEX design, leading to a plutonium disposition rate of 4.8 metric tonnes of plutonium metal per year using 4 Bruce A reactors. As requested by DOE, AECL and Ontario Hydro have reviewed the possibility of going directly to the CANFLEX design because of its capability for higher throughput and lower overall program costs. However, as noted in the Fuel Qualification section 3.0, the experience base with CANFLEX is extremely limited at this time, and therefore, a more extensive qualification program, with greater risk, would be required. Even though the CANFLEX design is being qualified for other CANDU 6 type reactors, the mechanical configuration of the bundle ends required for refueling is different with the BRUCE reactors, and a special Natural Uranium CANFLEX fuel qualification program is required in advance of the MOX CANFLEX fuel qualification at BRUCE. This study therefore retains the basic deployment approach of starting out with the more proven standard 37 element fuel design, and in parallel, conducting the more extensive CANFLEX qualification program. Following the successful qualification of CANFLEX in BRUCE, estimated about 3 years after startup on standard 37 element MOX fuel, the BRUCE reactors could be switched over to the CANFLEX design. This approach offers the best balance between starting the program as soon as possible with least risk and greatest experience base, and reducing the overall cost and schedule of the full program cycle.

CHARACTERISTICS WITH FOUR BRUCE REACTORS USING CANFLEX MOX FUEL

	1994 STUDY	1996 STUDY
BURNUP (mwd/t)	17,100	17,000
bundles per year	10,512	10,280
plutonium /bundle (KG)	0.37	0.47
plutonium metal per year (MT)	3.9	4.8
MOX FACILITY THROUGH PUT (35 MOX pins per bundle, dysprosia pins ignored)		
Mixed Oxide output (MT/YR)	170.0	166.2
Metal Output (MTHM/YR)	150.0	146.6

Safeguards, Security, and Transportation - As stated in the 1994 study, a primary factor in assuring maximum safeguards and security is the minimization of the number of sites for MOX production and for MOX disposition. For the CANDU option this is achieved with only one site for MOX production, and one site for MOX disposition.

Additional evaluation of new fuel storage requirements at the Bruce Site have confirmed the 1994 study results regarding the scope of Category 1 building additions required for storage of new fuel. To be conservative, the current study assumes that at least a three month supply of MOX fuel should be kept on site for supply reliability, compared to the 1 month supply assumed in the 1994 study. Despite this larger storage requirement, preliminary analysis of the footprint required indicates that a Category 1 Annex can be designed and constructed at the North side of the Powerhouse for costs which are similar to those projected in 1994. Supporting details are in Section 4.

With regard to transportation, no new analyses have been conducted. Based on analyses conducted by AECL with the cooperation of the USDOE Transportation Safeguards Division at Sandia, one shipment per month, consisting of three SST's, will be required. Since the 1994 study, it has been noted that some of the SST's in the U.S. fleet have lower weight limits, and therefore more trucks may be required than estimated at that time. This will depend on how many of the higher capacity trucks are available for this mission. It is recommended that this information be factored into any future studies.

Transition from Natural Uranium to MOX Fuel at Bruce A - As requested by DOE, further study of the transition process from Natural Uranium to MOX fuel has been made. For the first core starting up on MOX fuel, it has been concluded that from a safety and licensing point of view, it is best to defuel the reactor completely and load MOX fuel during a period of physics startup tests, as is traditional for any new core design. Once the first core testing is completed, and power operations begin, subsequent Bruce A reactors can transition from natural uranium to MOX fuel while at power.

This approach has now been factored into the fuel qualification schedule, and the Bruce reactor conversion schedule as shown in section 6.0. The proposed plan is that after completion of the MOX facility checkout, the first 1000 MOX fuel bundles produced by the MOX facility will be shipped to the Bruce Site for on power loading into an operating Bruce reactor, thus providing Final Verification of the quality of the new MOX fuel from the production facility. After about 9 months of operation of this new MOX fuel, there is expected to be sufficient confidence in its quality to allow the full MOX loading of a previously defueled calandria and the performance of physics testing. The MOX fuel for this initial core loading will have been produced and shipped to the Bruce site during the 9 month fuel production verification program. Allowing about 3 months for initial core loading and physics testing, the first full MOX core would start up about one year after the MOX production facility begins to ship new MOX fuel. The second Bruce A reactor to receive MOX fuel would begin transitioning to MOX fuel during the second year, as sufficient MOX fuel stocks are built up at the Bruce site. The details of this on-power transition require further analyses which are beyond the scope of this study.

Environment Safety and Health - Results are similar to those in 1994, with the enhanced core design providing somewhat greater benefits relative to natural uranium than the 1994 core design. (see Figure 1.2-2 for Reference fuel and Figure 1.2-3 for CANFLEX fuel) Instead of mining and refining over 6300 tonnes of uranium ore per year (see Figure 1.2-1), the MOX fuel cycle with standard fuel consumes 3 tonnes of plutonium and over 230 tonnes of depleted UF6 waste per year. Although the quantity of spent fuel is about 10% less, this study evaluated the effects of the MOX fuel on the repository as compared to natural uranium spent fuel. Some changes to the repository loading may be required to handle the higher decay heat loads projected for MOX fuel, and this could offset all or part of the savings which would result from the lower volume of spent fuel. Further study of this is required.

Figure 1.2-1
2 Bruce A Units using Natural Uranium

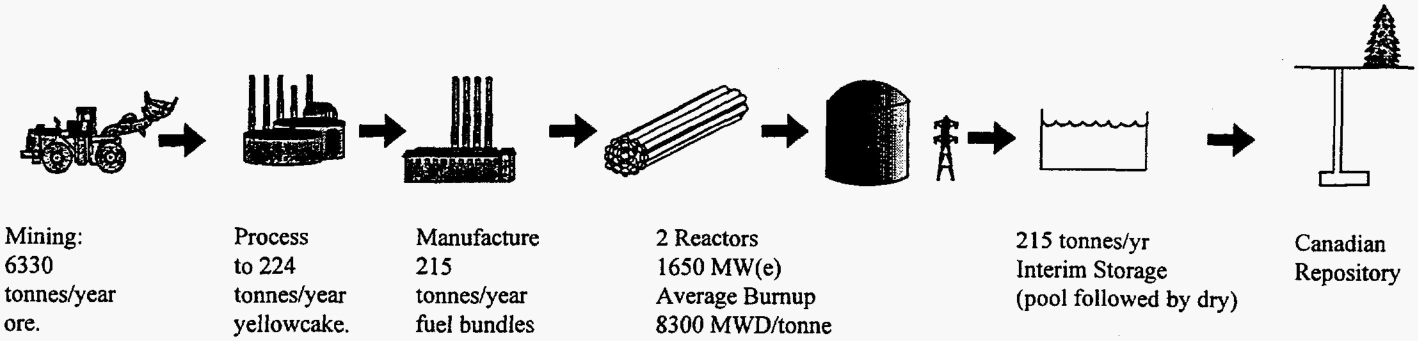


Figure 1.2-2
MOX Fuel Cycle During Dispositioning of 3 tonnes/yr of Plutonium using Standard 37 Element Fuel on 2 Bruce A Units

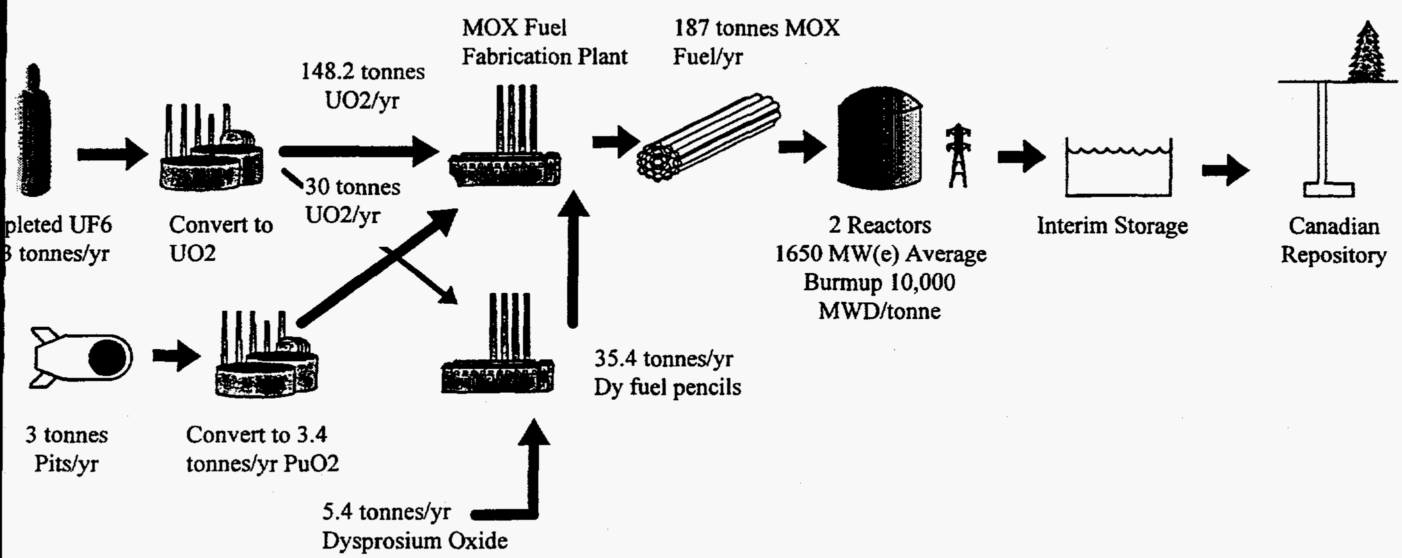
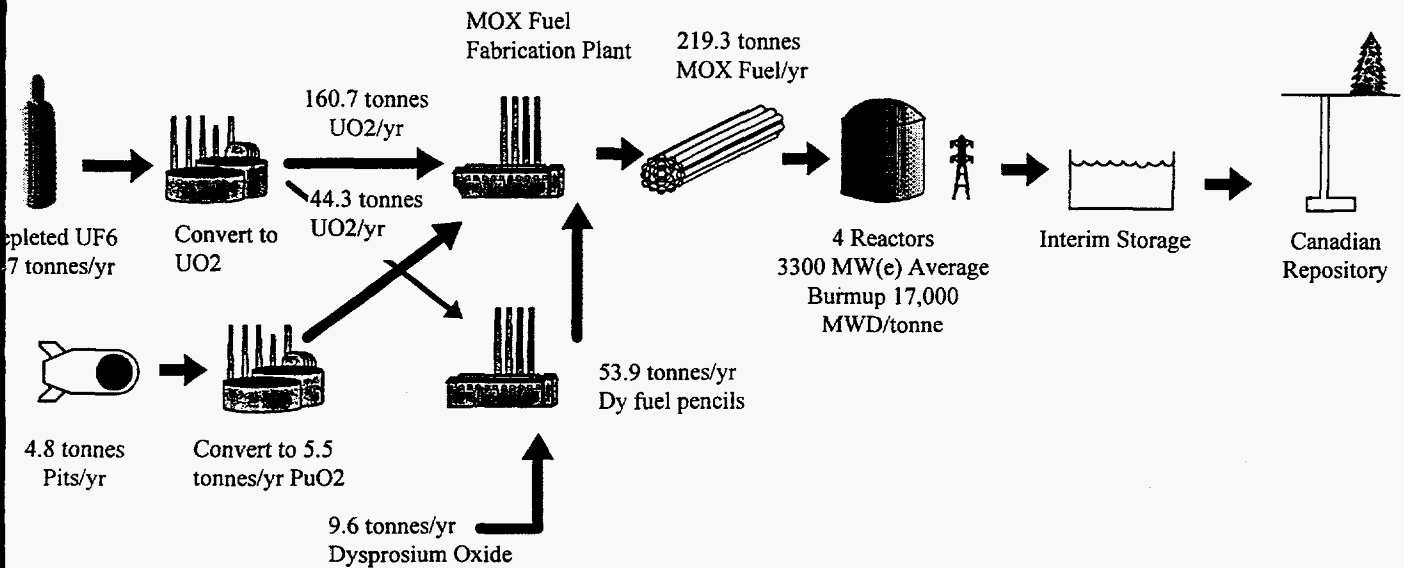


Table 1.2-3
Fuel Cycle During Dispositioning of 4.8 tonnes/yr of Plutonium using CANFLEX Fuel on 4 Bruce A Units



With respect to public acceptance, a variety of interactions were held with the public and the Atomic Energy Control Board. Reaction to the concept of using MOX fuel at Bruce A has been supportive including resolutions of support for further study from Bruce A's six (6) surrounding municipalities and the host, Bruce County. In addition both Huron and Lambton Counties passed similar resolutions which completed one transportation corridor to the United States via Sarnia Ontario. Further communication and consultation is planned. Completion of a comprehensive safety and environmental review would be required in Canada as a prerequisite to license the transport and use of MOX fuel at Bruce A.

Economics, Schedule and Risk - The objective was to develop a low cost, low risk method for plutonium disposition which, in response to the "clear and present danger" indicated by the National Academy of Sciences, could be implemented as quickly possible. As in the 1994 study, this has been achieved by designing the fuel and nuclear performance to operate within current experience including operating/performance envelopes, by limiting the changes to the existing reactors to provide for secure storage and handling of new MOX fuel, and by utilizing present infrastructure for MOX fuel fabrication and transportation.

Revised cost estimates for some elements of the program have been made, since the most recent estimates submitted to ORNL by letter dated March 6, 1996 and are presented in Section 6.0 The major up-front cost elements are: Fuel Qualification program for both Reference MOX Fuel and CANFLEX MOX fuel at Bruce ($78 million 1996 US dollars); modifications to Bruce site for storage and handling of MOX fuel, licensing of MOX fuel operations, ($34 million); design and qualification of packaging and transportation systems (5 million); modifications and licensing of a privatized MOX plant in SouthEast USA for production of up to 147 MTHM of CANDU MOX fuel per year ($183 million exclusive of finance costs).

The major operational cost elements obviously depend on the mission duration. For the base case of 50 MTHM of plutonium, the reference case is 3 years operation of reference fuel, and about 9 years of operation on CANFLEX fuel. Annual costs include about $42 million per year for utility fees including incremental costs of Bruce operations, $2 million per year for transportation, $80 million per year for MOX fuel production which are offset by about $30 million per year for natural uranium fuel production costs that will not be required for the length of the mission. The net annual costs not including front end are thus about $55 million per year for the twelve year operational period.

Deployment Options - This study was limited to the 50 tonne US surplus plutonium disposition mission as requested by DOE. Parallel studies funded by the Canadian Government are ongoing amongst AECL/Ontario Hydro and the Ministry of Atomic Energy of the Russian Federation for the disposition of similar quantities of Russian surplus weapons plutonium at the Bruce site. Results of these studies will be reported separately. Should there be interest by both the US and Russia in performing disposition of plutonium at the Bruce Site during the same time period, then there would obviously be some cost savings to both countries in terms of shared costs. These savings are addressed in Section 6.4.4.

In recent discussions with DOE and ORNL, interest in disposing of a lesser quantity (e.g. 34 tonnes) of US weapons plutonium has been expressed. In such case, a smaller sized MOX plant might be appropriate. AECL/Ontario Hydro have not evaluated this option at this time.

2. FUEL DESIGN AND NUCLEAR ANALYSIS

The feasibility of burning MOX fuel in Bruce A was established in the study performed by AECL in 1994 for the US Department of Energy. The objective of the current study is to optimize the MOX fuel design and the reactor operating parameters in order to significantly increase the plutonium disposition rate over the values achieved in the 1994 study. A major requirement in the 1994 study is that the Bruce A reactor using MOX fuel will operate within the safety and licensing parameters established for the existing natural uranium reactor. Detailed reactor physics calculations are carried out in this study to demonstrate that this requirement is also satisfied in Bruce A using the optimized MOX fuel designs.

An increase in the plutonium disposition rate requires an increase in the plutonium content in the MOX bundle without increasing the discharge fuel burnup. The excess reactivity due to the increased plutonium content can be compensated in two ways ,

- increasing the amount of the burnable poison, dysprosium, in the two inner fuel rings, and
- decreasing the isotope purity of the heavy water moderator and coolant.

The scope of the physics calculations are as follows:

- perform a parametric study of the MOX fuel design using the WIMS-AECL lattice code with various combinations of plutonium content, dysprosium content and heavy water purity,
- select the most promising design based on the experience of the 1994 study, and
- perform 3-D fuel management calculations with the RFSP code to fine-tune the optimized MOX fuel designs and to confirm that the reactor and the fuel operate within existing limits.

2.1 FUEL DESIGN

The geometries of the MOX fuel bundle designs used in the current study are the same as the two MOX fuel bundle designs documented in the 1994 study, i.e. the standard 37-element design which is currently used in Bruce A , and the advanced 43-element CANFLEX design, which is currently being qualified for irradiation in existing reactors. However, the amount of plutonium and dysprosium in the current fuel bundle designs are significant higher than those reported in the 1994 study in order to achieve a higher plutonium disposition rate.

A large number of WIMS-AECL calculations were performed for various combinations of plutonium content, dysprosium content and D_2O purity in the coolant and moderator. The 89-group ENDF/B-V cross-section library is used in the WIMS-AECL calculations. The objective is to maximize the plutonium content in the fuel bundle and to keep the fuel and the reactor operating within the limits established in the 1994 study using the following guide lines ,

- average fuel burnup of about 9,700 MWd/te for the reference 37-element design and 17,100 MWd/te for the advanced CANFLEX design as established in the 1994 study,
- full core coolant void reactivity of about -5 mk at mid-burnup,
- maximum dysprosium content not to exceed 20%,
- and minimum D_2O purity not to be lower than 95%.

Table 2.1.1-1 summarizes the characteristics of the MOX fuel bundle designs which satisfied the design guidelines. The characteristics of the reference 37-element MOX design and the advanced CANFLEX design recommended in the 1994 study are also included in this table for comparison purposes.

It should be noted that these estimates of fuel burnup and coolant void reactivity are based on WIMS-AECL calculations only. Based on the experience gained from the 1994 study, the lattice k-infinity at mid-burnup should be about 1.040. The excess reactivity of 40 mk compensates for neutron leakage, parasitic absorptions in the safety and control device assemblies and the effect of the downgraded heavy water in the reflector . The coolant void reactivity is also evaluated at mid-burnup based on the change in the WIMS-AECL lattice k-infinity.

The WIMS-AECL calculations should establish the plutonium content to within about 0.1%. The final specifications of the optimized MOX designs are based on iterations between the lattice code WIMS-AECL and the 3-D fuel management code RFSP.

Because of the large amount of dysprosium and plutonium in the fuel bundle, downgrading the purity of the coolant alone has no significant impact on either the fuel burnup or coolant void reactivity. Moderator purity is the dominating factor. Therefore, the coolant and moderator are assumed to have the same D_2O isotopic purity for all the cases in Table 2.1.1-1

Case 5 and case 11 were chosen as the starting points for the optimized 37-element design and CANFLEX design, respectively. The choice of these cases was based on the overall merit according to independent assessments from fuel fabrication, fuel performance, reactor operation, and plutonium disposition rate.

2.1.1 Reference MOX CANDU Optimized Design

The RFSP calculations indicated that some minor adjustments to the specifications in Case 5 in Table 2.1.1-1 are needed in order to achieve the desired fuel burnup for the reference MOX design. The final specifications for the optimized reference MOX CANDU are,
existing 37-element fuel bundle design,

- 15% dysprosium in rings 1 and 2
- 3.1 % Pu in ring 3
- 1.6% Pu in ring 4
- base material for all fuel pellets is 0.2% depleted uranium.

The D_2O purity is 97% for both the coolant and the moderator. This gives a core-averaged discharge fuel burnup of 10,000 MWd/te, which is equivalent to a fuelling rate of 15.0 bundles per full power day. The core-averaged void reactivity is -5.0 mk .

Assuming a capacity factor of 0.80, the number of MOX fuel bundles consumed per year in Bruce A is equal to 15.0 x 365 x 0.80 = 4380. The plutonium disposition rate is 1.5 te (tonne) of Pu metal per year per reactor. The MOX fuel fabrication capacity requirement is 78 te/year/reactor.

Table 2.1.1-1
Summary of DOE 96 MOX Fuel Bundle Optimization Study

Case	D2O Purity (%) Moderator & Coolant	Rings 1&2 Dy % in DU	Ring 3 Pu %HE in MOX	Ring 4 Pu %HE in MOX	Void Reactivity (mk)	Core-Avg. Burnup MWd/te	Pu Disposition te/yr/reactor	MOX Fabrication te/yr/reactor
1	99.75	5.0	2.0	1.2	-4.7	9700	1.05	78
2	99	4.7	2.1	1.2	-5.0	9700	1.08	78
3	98	6.6	2.5	1.3	-5.0	9700	1.23	78
4	97	13	2.9	1.5	-5.0	9700	1.42	78
5	97	15	3.0	1.5	-5.1	9700	1.45	78
6	97	20	3.0	1.6	-5.9	9700	1.49	78
7	96	15	3.1	1.6	-2.9	9700	1.52	78
8	96	20	3.3	1.6	-2.8	9700	1.58	78
9 *	99.75	6	3.6	2.1	-1.6	17100	0.95	41
10 *	98	8.9	4.1	2.4	-5.7	17100	1.08	41
11 *	97	15	4.6	2.6	-4.5	17100	1.2	41
12 *	97	20	4.8	2.6	-4.6	17100	1.22	41

2.1.2 MOX CANFLEX Optimized Design

The RFSP calculations indicated that no adjustment to the specifications in Case 11 in Table 2.1.1-1 is needed. The final specifications for the optimized MOX CANFLEX CANDU are,

- 43-element CANFLEX fuel bundle design,
- 15% dysprosium in rings 1 and 2
- 4.6 % Pu in ring 3
- 2.6% Pu in ring 4
- base material for all fuel pellets is 0.2% depleted uranium

The D_2O purity is 97% for both the coolant and the moderator. This gives a core-averaged discharge fuel burnup of 17,000 MWd/te, which is equivalent to a fuelling rate of 8.8 bundles per full power day. The core-averaged void reactivity is -4.5 mk.

Assuming a capacity factor of 0.80, the number of MOX fuel bundles consumed per year in Bruce A is equal to 8.8 x 365 x 0.80 = 2570. The plutonium disposition rate is 1.2 te of Pu metal per year per reactor. The MOX fuel fabrication capacity requirement is 41 te/year/reactor.

2.2 NUCLEAR ANALYSIS FOR REFERENCE MOX CORE

The procedures developed for the nuclear analysis of the reference MOX core in the 1994 study are used in the current study of the optimized reference MOX core. The fuel design and the reactor operating parameters are optimized to ensure that both the fuel and the reactor operate within the fuel performance and safety envelopes established for existing natural uranium CANDU reactors.

Although the current reference optimized 37-element MOX fuel design contains 49% more plutonium per bundle than the reference MOX fuel design established in the 1994 study, the flexibility of the CANDU design allows adjustments to be made in the fuel bundle design parameters and in the coolant and moderator systems to compensate for the excess reactivity.

The excess lattice reactivity is mainly compensated by downgrading the D_2O isotopic purity in the moderator and coolant from the existing nominal value of 99.75% to 97.0%. The downgrading of coolant and moderator purity would normally result in an increase in the coolant void reactivity. This effect is suppressed by increasing the amount of dysprosium from the 1994 level of 5% to the current level of 15%, which is within the range of current experience for dysprosium content. As a result of these design changes, the nuclear characteristics of Bruce A using the optimized MOX fuel are very similar to those established in the 1994 MOX study.

2.2.1 Reactor Operating Parameters

Table 2.2.1-1 compares the major nuclear characteristics of Bruce A using the reference MOX fuel with those for the existing Bruce A using the natural uranium fuel cycle.

Bi-directional on-power fuelling is used in Bruce A for both fuel cycles. A mixed 2, 4 and 8-bundle-shift fuelling scheme is used in the natural uranium Bruce A reactor. The 2-bundle-shift-scheme is used in high power channels in the inner region of the core. The 8-bundle-shift scheme is used in the low power channels in the outer region of the core. The 4-bundle-shift scheme is used in the intermediate channels. This mixed fuelling scheme is designed to minimize the power ripple, i.e. the increase in channel and bundle power due to fuelling, and to optimize the usage of the fuelling machine. The natural uranium Bruce A reactor requires the fuelling of 18 fuel bundles per full power day. The MOX Bruce reactor A requires the fuelling of 15 fuel bundles per full power day.

Table 2.2.1-1
Comparison of Natural Fuel and MOX Fuel Characteristics

	Exisiting Bruce-A Station	**Ref. MOX Bruce A Station**
Fuel Bundle Dimension	102 mm (4.02") dia. 495 mm (19.5") long	Same as existing NUO_2 fuel bundle.
Pellet Material Composition (% based on weight of heavy metal in fuel)	Natural UO_2 in all rings (37 pins)	Ring 4 (18 pins) 1.6% Pu Ring 3 (12 pins) 3.1% Pu Ring 2 (6 pins) 15.0% Dy Ring 1 (1 pins) 15.0% Dy All rings have 0.2% U^{235}
Bundle Material Composition (Fuel Materials Only)	U^{235} 0.13 kg U^{238} 18.67 kg U^{235} 2.53 kg Total : 21.33 kg	Pu 0.33 kg U^{235} 0.04 kg U^{238} 17.92 kg Dy 0.51 kg O_2 2.53 kg Total : 21.33 kg
Average Burnup	8,300 MWd/te	10,000 MWd/te
Maximum Burnup	15,000 MWd/te	15,500 MWd/te
Bundle/FPD	18	15 -
Fuel Management Scheme	2, 4, 8 Mixed Bundle Shift	2 Bundle Shift
Maximum Channel Power	7,200 KW	7,000 KW
Maximum Bundle Power	950 KW	780 KW
LOCA Void Reactivity	11 mk	-5.0 mk
Moderator Coolant Purity	99.75 %	97%

The lower fuelling requirement in the MOX core allows the use of a 2-bundle-shift fuelling scheme for all the channels without exceeding the capacity of the existing fuelling machine. This simple 2-bundle-shift scheme further flattens the reactor power distribution and introduces minimum perturbation into the reactor due to fuelling. Although the initial fissile content in the MOX bundle is much higher than that in a natural uranium bundle, the peak instantaneous channel and bundle powers in the MOX core are lower than those in the natural uranium core.

The peak channel power and peak bundle power in the natural Bruce A reactor are 7.2 MW and 950 kW respectively. The corresponding values for Bruce A using the optimized MOX fuel are 7.0 MW and 780 kW, respectively. The lower peak channel and bundle powers in the MOX Bruce A core allow the MOX reactor to operate at full power with improved safety margins relative to fuel channel thermal power limits.

The average burnup in the MOX fuel is 10,000 MWd/te, which is slightly higher than the average burnup of 8,300 MWd/te achieved by natural uranium fuel in Bruce A. However, the peak element burnup in the MOX fuel is estimated to be about 15,500 MWd/te, which is essentially the same as the peak element burnup of 15,000 MWd/te achieved by natural uranium fuel. The inner MOX fuel ring, i.e. ring 3, has a plutonium content of 3.1 wt%, which is higher than the plutonium content of 1.6 wt% in the outer MOX fuel ring, i.e. ring 4. This enrichment grading scheme has been carefully designed to equalize the power, hence burnup, of the MOX elements in both rings over the lifetime of the MOX fuel.

The major difference between the natural uranium Bruce A reactor and the MOX Bruce A reactor is the reactivity effect during a postulated Loss of Coolant Accident (LOCA) scenario. Full core LOCA reactivity in Bruce A is calculated to be +11 mk for the natural uranium reactor and -5.0 mk for the MOX reactor. It will be seen in later sections that this inherent negative coolant void reactivity in the MOX core precludes the need for major modifications to the control and safety systems in the existing Bruce A reactor.

Table 2.2.1-2 compares the uranium and plutonium content in the natural fuel with that in the reference MOX fuel. Fresh natural fuel contains 133 g of U-235 per bundle. At discharge, each NU bundle contains 38.7 g of U-235 and 51.1 g of fissile plutonium (Pu-239 and Pu-241). Each fresh MOX fuel bundle contains 36.0 g of U-235 and 315.0 g of fissile plutonium. At discharge, each MOX bundle contains 22.2 g of U-235 and 186.0 g of fissile plutonium. The fissile plutonium content drops from 94% in the fresh fuel to 72% in the discharged MOX bundle mainly due to the formation of the isotope Pu-240. The high Pu-240 content makes the plutonium in the discharged MOX bundle undesirable for weapons purposes. Furthermore, as has been previously established, the discharged irradiated fuel bundles loaded in spent fuel storage modules meet the spent fuel standard.

Table 2.2.1-2
Actinide Inventory for Natural and Reference MOX Fuel Bundle (g/bundle)

	New		Exit Burnup	
	Natural	Reference MOX	Natural 8300 MWd/teHE	Reference MOX 10,000 MWd/teHE
^{235}U	133.0	36.0	38.7	22.2
^{238}U	18670.0	17923.6	18534.0	17831.1
^{239}Pu		313.8	46.8	172.9
^{240}Pu		19.4	20.6	69.3
^{241}Pu		1.2	4.3	13.1
^{242}Pu		0.1	1.3	2.9

2.2.2 Reactor Power Distribution

Figure 2.2.2-1 shows the time-average channel power distribution in the Bruce A core with natural uranium fuel. The time-average channel power represents the average power over successive fuelling intervals in each channel. The power increase upon fuelling, i.e. power ripple, depends on the location of the fuel channel, the burnup of the fuel in that channel, and the number of fresh bundles inserted into that channel. Typical power ripple for a channel in the inner and intermediate regions of a natural uranium fuelled Bruce A core is about 10%. This means that a freshly fuelled channel typically has a channel power 10% higher than the time-average value. The power of the freshly fuelled channel starts to drop as fuel burnup in the channel increases. Eventually, the power drops significantly below the time-average value and the channel becomes eligible for fuelling again.

Figure 2.2.2-2 shows the time-average channel power distribution in the reference MOX Bruce A core. Typical power ripple in the reference MOX Bruce A is about 8% using uniform 2-bundle-shift scheme. Figure 2.2.2-3 compares radial power distributions in the MOX Bruce A core with that in the natural Bruce A core. The radial power shape in the MOX core is designed to be slightly flatter than that in the natural uranium core. A combination of lower power ripple and flatter power distribution enable the MOX Bruce A core to operate at full power with increased operating margin.

Figure 2.2.2-4 compares the axial power distribution in a MOX fuelled channel with the axial power distribution in a natural uranium fuelled channel. The MOX channel has a flatter bundle power profile than that of the natural uranium fuel channel. The MOX axial power shape is skewed towards the channel inlet end. The combination of lower peak bundle power and skewed axial power shape give the MOX channel a higher Critical Channel Power (CCP), which is the channel power at which dryout occurs at the surface of a fuel element. A higher CCP increases the reactor operating margin by increasing the maximum allowable channel power.

Figure 2.2.2-1

Time-Average Channel Powers in MW (TH) Normalized to Reactor Power of 3729 MW(TH) in Bruce A Natural Uranium Core

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
A								3.45	3.70	4.00	4.21	4.22	4.22	4.21	4.00	3.70	3.45							
B						3.40	4.01	4.30	4.65	4.95	5.05	5.07	5.07	5.05	4.95	4.65	4.30	4.01	3.40					
C					3.80	4.46	4.98	5.32	6.15	6.12	5.98	5.82	5.82	5.98	6.12	6.15	5.32	4.98	4.46	3.80				
D				3.77	4.48	5.17	5.91	6.02	6.42	6.41	6.25	6.05	6.05	6.25	6.41	6.42	6.02	5.91	5.17	4.48	3.77			
E			3.78	4.75	5.38	5.66	6.03	6.27	6.36	6.40	6.27	6.07	6.07	6.27	6.40	6.36	6.27	6.03	5.66	5.38	4.75	3.78		
F		3.50	4.38	5.40	5.69	5.86	5.94	6.21	6.32	6.40	6.32	6.13	6.13	6.32	6.40	6.32	6.21	5.94	5.86	5.69	5.40	4.38	3.50	
G		4.10	5.01	5.72	5.97	5.85	5.93	6.17	6.29	6.42	6.37	6.26	6.26	6.37	6.42	6.29	6.17	5.93	5.85	5.97	5.72	5.01	4.10	
H		4.63	5.64	6.01	6.01	5.88	5.96	6.17	6.29	6.41	6.40	6.34	6.34	6.40	6.41	6.29	6.17	5.96	5.88	6.01	6.01	5.64	4.63	
J	3.94	4.85	5.59	5.96	6.07	5.96	6.03	6.18	6.30	6.40	6.44	6.41	6.41	6.44	6.40	6.30	6.18	6.03	5.96	6.07	5.96	5.59	4.85	3.94
K	4.14	5.12	5.55	5.93	6.07	6.05	6.09	6.22	6.32	6.41	6.45	6.47	6.47	6.45	6.41	6.32	6.22	6.09	6.05	6.07	5.93	5.55	5.12	4.14
L	4.30	5.32	5.54	5.91	6.07	6.14	6.17	6.27	6.36	6.43	6.46	6.31	6.31	6.46	6.43	6.36	6.27	6.17	6.14	6.07	5.91	5.54	5.32	4.30
M	4.46	5.47	5.53	5.91	6.07	6.24	6.28	6.34	6.40	6.45	6.40	6.22	6.22	6.40	6.45	6.40	6.34	6.28	6.24	6.07	5.91	5.53	5.47	4.46
N	4.44	5.47	5.54	5.92	6.10	6.36	6.38	6.41	6.44	6.47	6.40	6.21	6.21	6.40	6.47	6.44	6.41	6.38	6.36	6.10	5.92	5.54	5.47	4.44
O	4.29	5.32	5.55	5.93	6.10	6.28	6.35	6.40	6.46	6.50	6.44	6.27	6.27	6.44	6.50	6.46	6.40	6.35	6.28	6.10	5.93	5.55	5.32	4.29
P	4.15	5.17	5.56	5.94	6.10	6.16	6.20	6.31	6.39	6.45	6.45	6.40	6.40	6.45	6.45	6.39	6.31	6.20	6.16	6.10	5.94	5.56	5.17	4.15
Q	3.99	4.94	5.57	5.95	6.10	6.07	6.10	6.23	6.32	6.41	6.39	6.35	6.35	6.39	6.41	6.32	6.23	6.10	6.07	6.10	5.95	5.57	4.94	3.99
R		4.64	5.57	5.95	6.09	5.99	6.05	6.18	6.24	6.28	6.31	6.24	6.24	6.31	6.28	6.24	6.18	6.05	5.99	6.09	5.95	5.57	4.64	
S		4.00	5.05	5.69	6.05	5.97	6.05	6.08	6.10	6.13	6.17	6.16	6.16	6.17	6.13	6.10	6.08	6.05	5.97	6.05	5.69	5.05	4.00	
T		3.40	4.44	5.32	5.70	5.97	5.97	5.98	6.00	6.01	6.04	6.06	6.06	6.04	6.01	6.00	5.98	5.97	5.97	5.70	5.32	4.44	3.40	
U			3.76	4.68	5.29	5.63	5.90	5.91	5.91	5.92	5.94	5.95	5.95	5.94	5.92	5.91	5.91	5.90	5.63	5.29	4.68	3.76		
V				3.76	4.62	5.17	5.58	5.58	5.85	5.86	5.87	5.88	5.88	5.87	5.86	5.85	5.58	5.58	5.17	4.62	3.76			
W					3.50	4.14	4.88	5.23	5.55	5.55	5.56	5.58	5.58	5.56	5.55	5.55	5.23	4.88	4.14	3.50				
X						3.28	3.92	4.25	4.56	4.77	5.05	4.86	4.86	5.05	4.77	4.56	4.25	3.92	3.28					
Y								3.20	3.60	3.95	4.00	4.15	4.15	4.00	3.95	3.60	3.20							

Figure 2.2.2-2
Time-Average Channel Powers in KW(TH) Normalized to Reactor Power of 3729 MW(TH) in Bruce A Reference MOX Core

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
A								3495	3958	4320	4520	4581	4579	4522	4318	3959	3494							
B						3636	3914	4412	4848	5149	5327	5364	5367	5324	5152	4845	4414	3912	3637					
C					4050	4824	4930	5316	5630	5863	5988	6003	6000	5992	5860	5633	5313	4933	4822	4052				
D				4074	4461	5057	5448	5615	5818	5956	6025	5989	5993	6021	5960	5814	5619	5445	5060	4459	4076			
E			4012	4466	5079	5516	5795	5885	6011	6096	6092	6002	5998	6097	6091	6015	5881	5799	5513	5083	4464	4014		
F		3519	4824	5073	5509	5765	5801	5955	6069	6113	6081	5943	5948	6077	6117	6065	5959	5797	5769	5506	5077	4823	3521	
G		3742	4875	5407	5709	5702	5823	6007	6120	6168	6116	5978	5974	6121	6164	6125	6003	5827	5699	5712	5404	4878	3741	
H		4238	5222	5479	5667	5686	5808	6035	6184	6232	6200	6069	6074	6195	6236	6179	6040	5804	5690	5664	5483	5220	4240	
J	3658	4715	5581	5687	5765	5727	5818	6065	6215	6282	6242	6170	6165	6247	6278	6219	6061	5823	5723	5769	5684	5584	4713	3659
K	4120	5142	5909	5894	5907	5810	5884	6111	6264	6303	6320	6267	6272	6316	6308	6259	6116	5881	5814	5904	5898	5906	5144	4118
L	4441	5442	6157	6084	6051	5934	5977	6190	6291	6366	6423	6387	6382	6428	6361	6295	6186	5981	5930	6055	6080	6160	5440	4441
M	4601	5597	6305	6203	6176	6057	6095	6263	6344	6392	6443	6376	6381	6438	6397	6339	6268	6091	6061	6172	6206	6301	5599	4598
N	4614	5621	6334	6256	6260	6203	6232	6351	6381	6410	6412	6300	6295	6418	6404	6386	6347	6236	6198	6264	6252	6337	5618	4614
O	4482	5500	6250	6210	6267	6246	6295	6389	6411	6409	6395	6249	6255	6390	6414	6406	6394	6291	6251	6263	6213	6246	5501	4479
P	4173	5226	6028	6069	6169	6193	6259	6377	6424	6403	6338	6209	6204	6343	6398	6429	6372	6264	6189	6173	6064	6030	5222	4173
Q	3716	4797	5707	5851	6015	6065	6165	6312	6402	6422	6371	6280	6285	6366	6428	6396	6317	6161	6068	6010	5854	5703	4798	3713
R		4314	5325	5619	5836	5901	6018	6226	6352	6420	6420	6369	6364	6425	6415	6358	6221	6023	5896	5839	5615	5327	4310	
S		3794	4956	5519	5813	5733	5865	6102	6273	6359	6388	6337	6342	6383	6364	6269	6107	5860	5737	5808	5522	4952	3795	
T		3538	4850	5144	5537	5722	5739	5997	6178	6287	6308	6254	6249	6314	6282	6183	5993	5744	5718	5540	5140	4850	3535	
U			4015	4495	5084	5455	5753	5905	6103	6217	6239	6137	6142	6235	6221	6098	5909	5749	5458	5080	4496	4012		
V				4062	4467	5080	5490	5674	5902	6060	6093	6001	5997	6098	6055	5907	5670	5494	5076	4469	4059			
W					4034	4837	5016	5414	5757	5984	6078	6014	6018	6074	5988	5754	5417	5013	4839	4031				
X						3649	3986	4512	4964	5273	5427	5458	5455	5430	5270	4967	4509	3988	3647					
Y								3567	4057	4423	4619	4677	4679	4617	4425	4056	3568							

MAXIMUM = 6443 AT M11

Figure 2.2.2-3
Radial Power Distribution Reference MOX Fuel vs. Natural Uranium

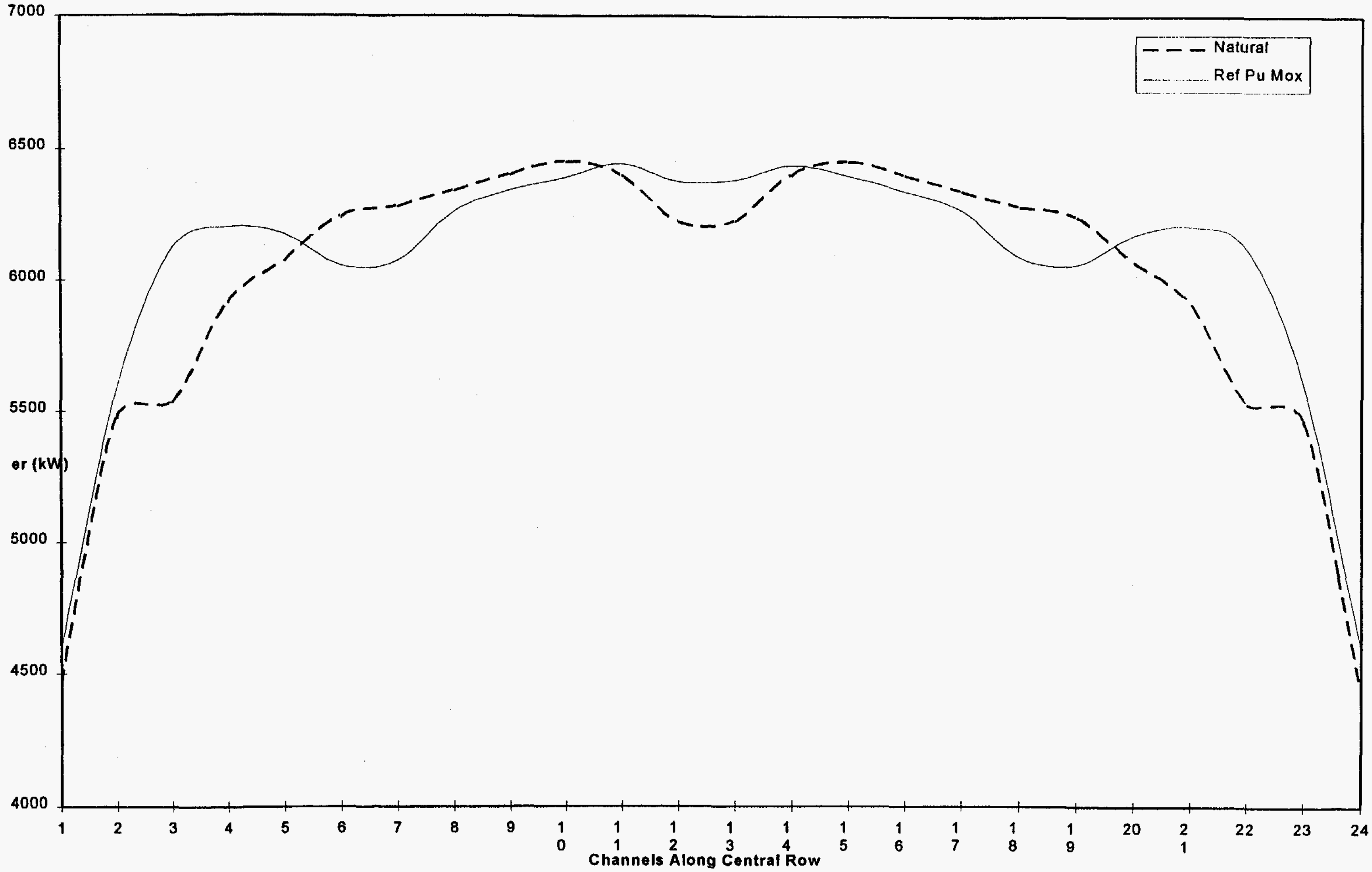


Figure 2.2.2-4
Axial Power Distribution Reference MOX Fuel vs. Natural Uranium
Normalized to 7.0 MW Channel Power

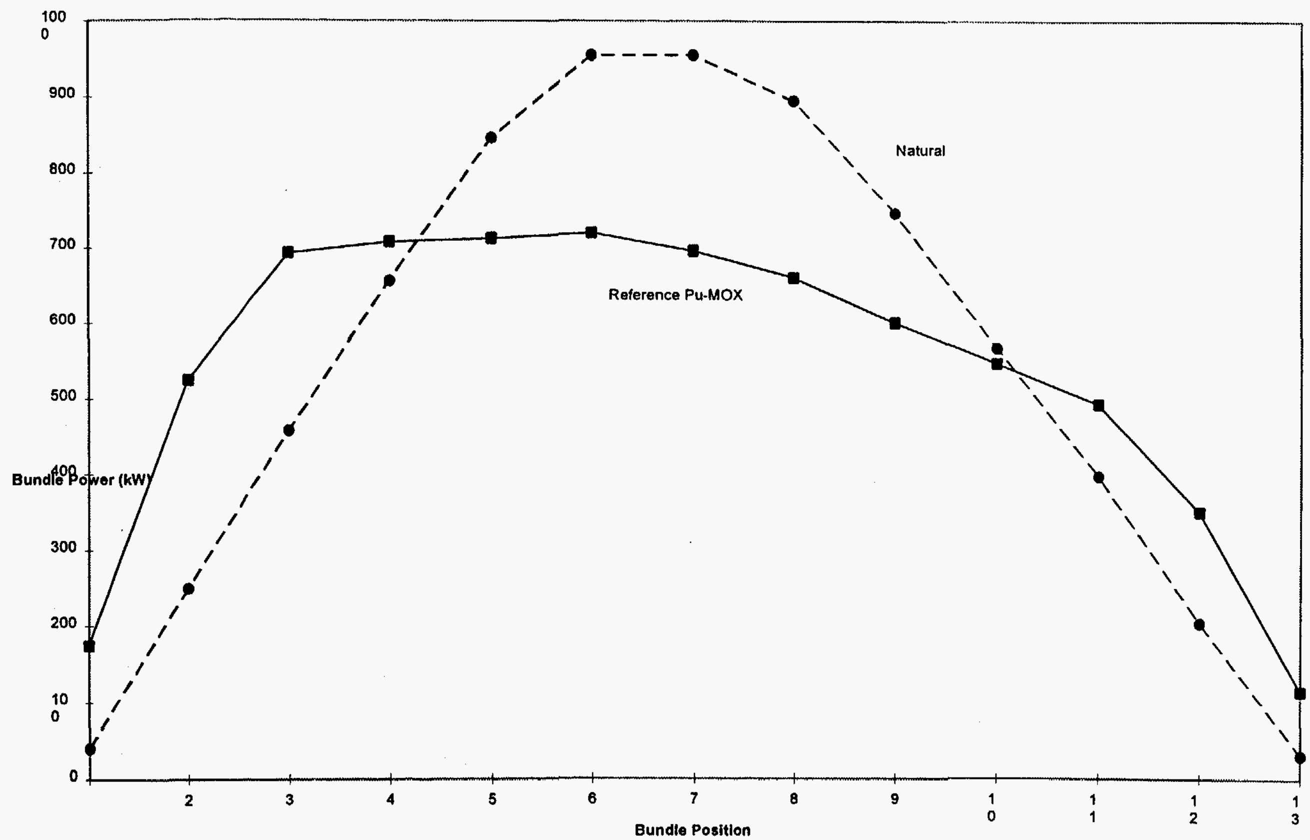


Table 2.2.2-1 gives the detailed element burnup and element power rating in each fuel ring for each bundle in a high power MOX channel. The bundle power shape is obtained from a time-average calculation. The bundle powers have been normalized to give a channel power of 7.0 MW, which is the highest expected in the MOX core. The axial bundle and element power shapes are used for CCP evaluations.

The RFSP code was used to simulate the fuelling operation of the MOX Bruce A reactor for a period of 100 full power days (FPDs), starting from an equilibrium core with a random fuel burnup distribution. Fuel performance data, in the form of maximum element power rating and maximum power boost due to fuelling, are extracted from the results of the RFSP simulations. Detailed analysis of the MOX fuel operating parameters is given in section 2.2.5.

Table 2.2.2-1
Axial Burnup and Power Distribution in a 7.0 MW Channel and using Reference MOX Fuel

Bundle	Bundle Burnup	Burnup in				Element Rating (kW/m)				Bundle Power
Position	MWd/te	Ring 1	Ring 2	Ring 3	Ring 4	Ring 1	Ring 2	Ring 3	Ring 4	kW
1	575	24	28	588	750	0.4	0.5	10.7	13.4	175
2	1728	72	84	1771	2248	1.2	1.5	33.3	39.3	525
3	2858	120	140	2932	3718	1.8	2.1	45.6	50.9	693
4	4060	177	208	4253	5222	1.9	2.4	48.2	50.8	708
5	5207	236	280	5565	6620	2.1	2.6	50.1	50.1	713
6	6432	303	364	7011	8085	2.2	2.9	52.1	49.5	720
7	7498	367	448	8322	9321	2.3	3.1	51.5	47.0	696
8	8608	438	541	9712	10590	2.4	3.2	49.9	43.9	661
9	9479	514	640	11199	11959	2.3	3.2	46.3	39.3	602
10	10411	558	701	12005	12624	2.2	3.0	42.4	35.5	547
11	11103	614	778	12915	13381	2.1	2.9	38.3	31.8	493
12	11568	651	830	13527	13890	1.5	2.2	27.4	22.6	351
13	11480	644	820	13411	13794	0.5	0.7	8.9	7.4	115

2.2.3 Reactor Safety and Control

Table 2.2.3-1 give the reactivity worth of the control and safety devices in Bruce A with natural fuel and with the reference MOX fuel. Equilibrium xenon load, average reactivity increase due to fuelling a single channel, and full core void reactivity are also given for both cases. As expected, all the reactivity devices in the MOX core are worth significantly less than those in the natural core because the absorption cross section of the basic MOX fuel lattice with 97% purity D_2O is much higher than that of the nominal natural uranium fuel lattice. However, the reactivity worth of -12.1 mk for the 28 shut-off rods in the Bruce A MOX core is more than adequate to safely terminate a full core large LOCA, the most severe accident scenario in a CANDU, because of the negative void reactivity of -5.0 mk in the MOX core. Detailed results of the LOCA analysis are given in section 4.0. The worth of -5.5 mk for the four MCAs in the MOX core is adequate for fast power reduction by the power stepback function of the reactor regulating system.

The worth of the Liquid Zone control system is -3.5 mk in the MOX Bruce A core compared to -6.0 mk in the natural core. This is adequate for both bulk and partial control purposes because fuelling a MOX channel with 2-bundle-shift is worth about 0.1 mk, which is about half the reactivity increase of fuelling a channel in a natural core using mixed 2, 4 and 8-bundle-shift fuelling. The equilibrium xenon load in the MOX core is about -20 mk, which is significantly lower than the -28 mk in the natural core.

The effectiveness of the control system for spatial control purpose is demonstrated by a detailed fuelling simulation on the 18th day of the 100 Full Power Days (FPDs) RFSP simulation (referred to subsequently as FPD18). Fuelling operations at eight channels were simulated individually, with both bulk and spatial reactor power controls operational. After the eight channels were fuelled a one day burnup step was simulated. The results are summarized in Table 2.2.3-2.

The average level of the zone control system increased from 41.9% to 64.7% after fuelling eight channels. The average increase of the zone control system level per channel fuelled is 2.9%, which is equivalent to 0.10 mk. For the 8 channels fuelled, the level is slightly more than the reactivity consumed in the MOX core per full power day because the average zone control level returns to 43.3%, i.e. slightly higher than the level before fuelling, after operating for one full power day. This is consistent with the estimated fuelling requirement of 15 bundles, i.e. 7.5 channels, per full power day in the MOX core.

It can be seen in Table 2.2.3-2 that the spatial control system is able to keep the maximum channel power below 7.0 MW and the maximum bundle below 760 kW throughout the fuelling operations. Channel power distribution at FPD18 is shown in Figure 2.2.3-1 and the channel power ripple map is shown in Figure 2.2.3-2. A value of 1100 represents a channel power ripple of 1.10. It can be seen that the channel power ripple in the high power region is below 10%. These results demonstrate that the bulk and spatial reactor power control systems in the MOX Bruce A core are adequate for regulating power distribution.

Table 2.2.3-1
Reactivity Effects in Bruce A Natural and Reference MOX Cores

	Natural Core (mk)	Reference MOX Core (mk)
14 Zone Controllers Inserted	-6.0	-3.5
30 SORs Inserted	-40.2	-13.3
28 SORs Inserted (Two Best Rods Missing)	-31.2	-12.1
4 MCAs Inserted	-7.2	-5.5
One Channel Refuelled	+0.2	+0.1
Full Core Voided	+11.0	-5.0
Equilibrium Xenon Load	-28.0	-20.0

Table 2.2.3-2
Summary of Detailed Refuelling Simulations at FPD18

Zone Controller	Zone Controller levels (%) After Channel Refuelled									
	none #	R22	J01	R08	W16	O12	G11	M21	L02	Burnup*
1	74.9	78.5	75.7	74.5	73.8	74.6	76.0	84.4	84.8	77.1
2	41.1	66.7	67.6	68.2	76.5	77.5	75.6	84.7	84.8	70.5
3	57.3	58.1	57.7	56.4	56.7	59.4	70.5	69.5	62.9	43.5
4	20.0	10.8	10.3	10.7	10.3	18.6	25.5	34.0	35.0	10.3
5	38.1	42.6	40.1	51.7	65.6	70.7	67.9	73.7	73.0	39.2
6	52.3	52.6	58.0	60.2	57.3	60.4	65.5	63.7	82.0	62.6
7	27.5	30.0	32.9	57.2	55.9	59.1	58.0	57.8	72.4	43.0
8	41.7	43.1	50.6	51.0	48.7	53.1	60.3	57.6	71.3	52.3
9	47.8	52.5	56.2	73.2	71.5	76.6	76.0	75.1	82.2	56.7
10	20.0	18.0	16.4	13.6	13.7	17.6	38.1	35.8	29.6	13.4
11	20.0	10.0	10.3	10.0	10.3	17.4	24.1	26.3	24.9	10.3
12	25.0	27.3	24.3	32.9	53.6	62.4	60.7	64.6	63.9	32.3
13	38.2	42.9	41.6	41.3	41.3	42.9	44.3	58.6	56.9	37.8
14	37.0	53.9	52.8	53.0	65.1	68.0	66.1	82.3	82.6	56.2
Average Level	38.6	41.9	42.5	46.7	50.0	54.2	57.7	62.0	64.7	43.3
Max Channel (kW)	6780	6777	6779	6769	6731	6819	6816	6951	6937	6793
Max Bundle (kW)	732	727	727	725	720	729	729	757	754	732

beginning of refuelling
* 1 FPD burnup applied to the core after all 8 channels have been refuelled

Figure 2.2.3-1
Channel Powers in KW (TH) in Reference MOX Core at FPD18

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
A								3509	4058	4283	4598	4524	4450	4597	4437	4223	3481							
B						3686	3935	4310	5018	4800	5322	5241	5548	5364	5465	4854	4542	4212	3862					
C					3989	4600	4799	4967	5501	5625	5712	6174	5957	5877	5911	5809	5271	4971	4636	4135				
D				3926	4221	4968	5504	5613	5530	5970	6051	6030	6285	6192	5818	5981	5403	5391	4821	4427	4048			
E			4004	4304	4724	5554	5666	6218	5995	6296	6024	6384	6062	6482	6226	6498	5638	6071	5217	5147	4326	3805		
F		3576	4669	5107	5239	5699	5480	5902	6394	6019	6156	5828	6193	6174	5959	6325	5858	5952	5535	5534	4798	4671	3587	
G		3760	4695	5368	5786	5504	5555	6177	6295	6247	6476	5991	5875	6032	6303	6042	5781	5798	5899	5602	5222	5011	3835	
H		4515	5203	5178	5507	5327	5745	5982	6042	6540	6362	6494	6073	6509	6319	5914	5946	5513	5544	5375	5531	5266	4199	
J	4064	4759	5722	5758	5573	5815	5626	5630	5920	6212	5964	6031	5895	6360	6276	6656	6281	5850	5488	5865	5608	5819	4626	3704
K	4461	5327	5859	5655	6113	5683	5627	6021	6316	6126	6316	6410	6040	6324	6651	6361	5994	5564	5373	5692	5671	5860	5310	4136
L	4667	5881	6279	6129	6023	6116	6026	6486	6407	6076	6366	6158	6389	6621	6429	6053	6270	6035	5871	6235	6155	5943	5571	4747
M	4872	5582	6487	6047	6433	5878	5704	6156	6176	6652	6368	6122	6697	6336	6145	6249	6552	6603	6419	6100	6612	6308	5544	4833
N	4549	5326	6045	6258	6575	6309	6114	6037	6598	6364	6715	6327	6580	6251	6324	6716	6144	6224	5885	6407	6319	6700	5419	4941
O	4536	5463	6106	6311	6077	5944	6083	6480	6376	6274	6575	6748	6430	6317	6667	6442	6239	6253	5999	5986	6148	6424	5563	4474
P	4154	5051	6101	6000	5835	6060	6493	6508	6793	6449	6144	6273	6054	6496	6400	6149	5766	5987	5758	6585	6423	6224	5582	4300
Q	3811	4594	5569	5462	5845	6338	6337	6271	6290	6624	6531	6189	6549	6318	6651	6047	6045	5549	5667	5617	5994	5713	4836	3909
R		4350	5102	5575	5935	5743	6073	6612	6430	6314	6192	6645	6257	6174	6289	6439	5949	5867	5811	5539	5612	5681	4518	
S		3808	4667	5402	5593	5769	6161	6275	6083	6432	6457	6332	6524	6311	5996	6291	6404	5768	5449	5717	5698	5130	4149	
T		3415	4815	5017	5266	5972	5778	6465	6153	6518	6225	6646	6098	5896	6073	5930	6108	5608	5373	5709	5082	4789	3593	
U			3942	4649	5031	5621	5646	5869	5851	6016	6334	6540	6349	6182	6412	5673	5706	5799	5305	5181	4388	4228		
V				4192	4805	5200	5474	5966	6058	6088	6282	5910	5761	5922	6119	5790	5434	5501	5292	4513	3999			
W					4390	4854	5187	5537	5765	6246	6089	5753	5881	6223	6040	5995	5399	4833	4797	4157				
X						3825	4049	4435	5300	5337	5190	5319	5651	5509	5213	4897	4682	4109	3614					
Y								3618	4172	4372	4573	4669	4536	4578	4525	4138	3818							

AXIMUM = 6793 AT P 9

Figure 2.2.3-2
Instantaneous Channel Overpower Dist (X100) at FPD18

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
A								1004	1025	991	1017	988	972	1017	1027	1067	996							
B						1014	1005	977	1035	932	999	977	1034	1007	1061	1002	1029	1077	1062					
C					985	954	973	935	977	959	954	1028	993	981	1009	1031	992	1008	961	1020				
D				964	946	983	1010	1000	950	1002	1004	1007	1049	1028	976	1029	961	990	953	993	993			
E			998	964	930	1007	978	1057	997	1033	989	1064	1011	1063	1022	1080	959	1047	946	1013	969	948		
F		1016	968	1007	951	989	945	991	1053	985	1012	981	1041	1016	974	1043	983	1027	960	1005	945	968	1019	
G		1005	963	993	1014	965	954	1028	1029	1013	1059	1002	983	986	1023	986	963	995	1035	981	966	1027	1025	
H		1065	996	945	972	937	989	991	977	1049	1026	1070	1000	1051	1013	957	985	950	974	949	1009	1009	990	
J	1111	1009	1025	1013	967	1015	967	928	953	989	956	978	956	1018	1000	1070	1036	1005	959	1017	987	1042	982	1012
K	1083	1036	992	959	1035	978	956	985	1008	972	999	1023	963	1001	1054	1016	980	946	924	964	962	992	1032	1004
L	1051	1081	1020	1007	995	1031	1008	1048	1019	954	991	964	1001	1030	1011	962	1014	1009	990	1030	1012	965	1024	1069
M	1059	997	1029	975	1042	971	936	983	974	1041	988	960	1050	984	961	986	1045	1084	1059	988	1066	1001	990	1051
N	986	947	954	1000	1050	1017	981	951	1034	993	1047	1004	1045	974	987	1052	968	998	949	1023	1011	1057	965	1071
O	1012	993	977	1016	970	952	966	1014	995	979	1028	1080	1028	988	1039	1006	976	994	960	956	990	1029	1011	999
P	995	967	1012	989	946	978	1037	1021	1057	1007	969	1010	976	1024	1000	956	905	956	930	1067	1059	1032	1069	1030
Q	1026	958	976	934	972	1045	1028	993	982	1031	1025	985	1042	992	1035	945	957	901	934	935	1024	1002	1008	1053
R		1009	958	992	1017	973	1009	1062	1012	983	965	1043	983	961	980	1013	956	974	985	949	999	1067	1048	
S		1003	942	979	962	1006	1051	1028	970	1011	1011	999	1029	989	942	1004	1049	984	950	984	1032	1036	1093	
T		965	993	975	951	1044	1007	1078	996	1037	987	1063	976	934	967	959	1019	976	940	1030	989	988	1016	
U			982	1034	990	1030	981	994	959	968	1015	1066	1034	992	1031	930	966	1009	972	1020	976	1054		
V				1032	1076	1024	997	1051	1026	1005	1031	985	961	971	1011	980	958	1001	1042	1010	985			
W					1088	1003	1034	1023	1001	1044	1002	957	977	1025	1009	1042	996	964	991	1031				
X						1048	1016	983	1068	1012	956	974	1036	1014	989	986	1038	1030	991					
Y								1014	1028	988	990	998	969	991	1023	1020	1070							

MAXIMUM CHANNEL POWER PEAKING FACTOR = 1.111 AT CHANNEL J-1

2.2.4 Reactor Kinetics Parameters

Table 2.2.4-1 gives the kinetics parameters for natural uranium and MOX fuelled Bruce A cores at equilibrium conditions. The kinetics parameters for the current reference MOX Bruce A core are not significantly different from those used in the 1994 MOX study. The results of the LOCA analysis, as described in section 4.2.2, using the current kinetics parameters are very similar to the results obtained in the 1994 study. The beneficial effect due to the negative void reactivity in the MOX lattice more than compensates for the reduced neutron lifetime, lower delayed neutron fraction (β), and smaller negative fuel temperature feedback.

Table 2.2.4-1
Delayed Neutron Data for Equilibrium Reference MOX Fuel and Natural Fuel

Group	**Reference MOX Fuel**		**Natural Fuel**	
	Delayed Neutron Fraction (b)	Time Constant (s^{-1})	Delayed Neutron Fraction (b)	Time Constant (s^{-1})
1	0.207E-03	0.470E-03	0.295E-03	0.612E-03
2	0.788E-03	0.311E-01	1.165E-03	0.316E-01
3	0.662E-03	0.132E-00	1.033E-03	0.122E+00
4	0.139E-02	0.327E-00	2.350E-03	0.318E+00
5	0.474E-03	0.135E+01	0.780E-03	0.139E+01
6	0.144E-03	0.357E+01	0.197E-03	0.378E+01
Total	0.366E-02		5.819E-03	

Prompt Neutron Lifetime	=	0.00047 sec (Reference)	=	0.0009 sec (Natural)
Fuel Temperature Coefficient	=	-2.9 mk/°C (Reference)	=	-6.0 mk/°C (Natural)
Full Core Void Reactivity	=	-5.0 mk (Reference)	=	+11.0 mk (Natural))

2.2.5 MOX Fuel Operating Parameters

A detailed assessment of the MOX fuel performance and defect probabilities was carried out in the 1994 MOX study. It was concluded that no detrimental impacts on fuel performance are expected due to the use of MOX fuel because,

- the maximum element rating envelope for the MOX fuel is within the envelope for the current Bruce A natural uranium fuel,
- the maximum element power boost envelope for the MOX fuel is also within the envelope for the current Bruce A natural uranium fuel, and
- the maximum MOX element burnup is less than 16,000 MWd/te, which has been achieved by the natural uranium fuel in Bruce A.

Figure 2.2.5-1 shows the maximum element rating envelope for the current reference MOX fuel based on the results of the 100-FPD simulation. The power boost envelope is shown in Figure 2.2.5-2. A snapshot of the MOX element power versus burnup at FPD18 of the 100-FPD simulation is shown in Figure 2.2.5-3. The distribution of MOX fuel element burnup in the MOX bundles discharged in the simulation period is shown in Figure 2.2.5-4. These results are very similar to those obtained for the 1994 reference MOX fuel. Consequently, acceptable fuel performance from the use of the current MOX fuel design is expected.

Figure 2.2.5-1
SCC Ramped-Power Assessment for Reference MOX Fuel

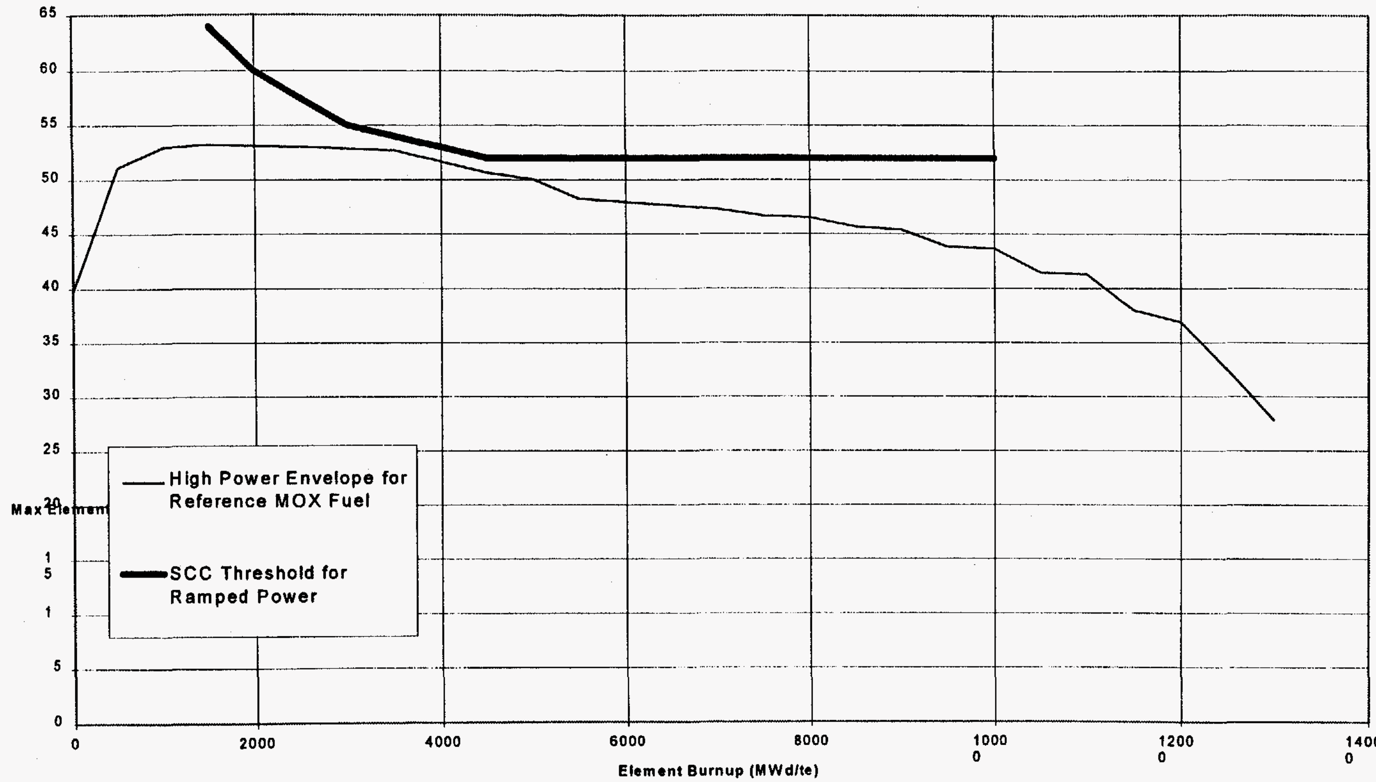


Figure 2.2.5-2
SCC Power Boost Assessment for Reference MOX Fuel

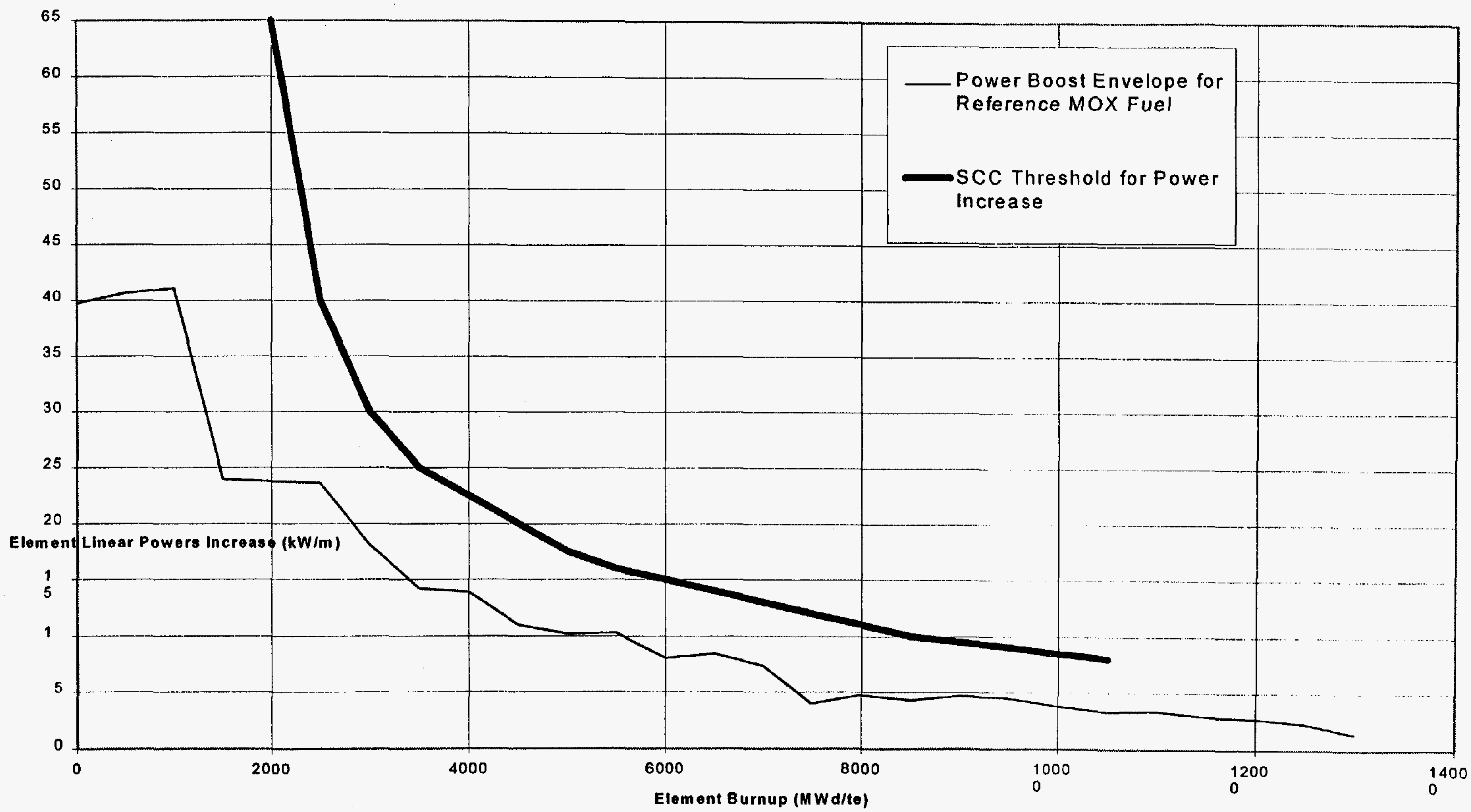


Figure 2.2.5-3
Snapshot of MOX Element Powers and Burnups for Reference MOX Fuel in Bruce A

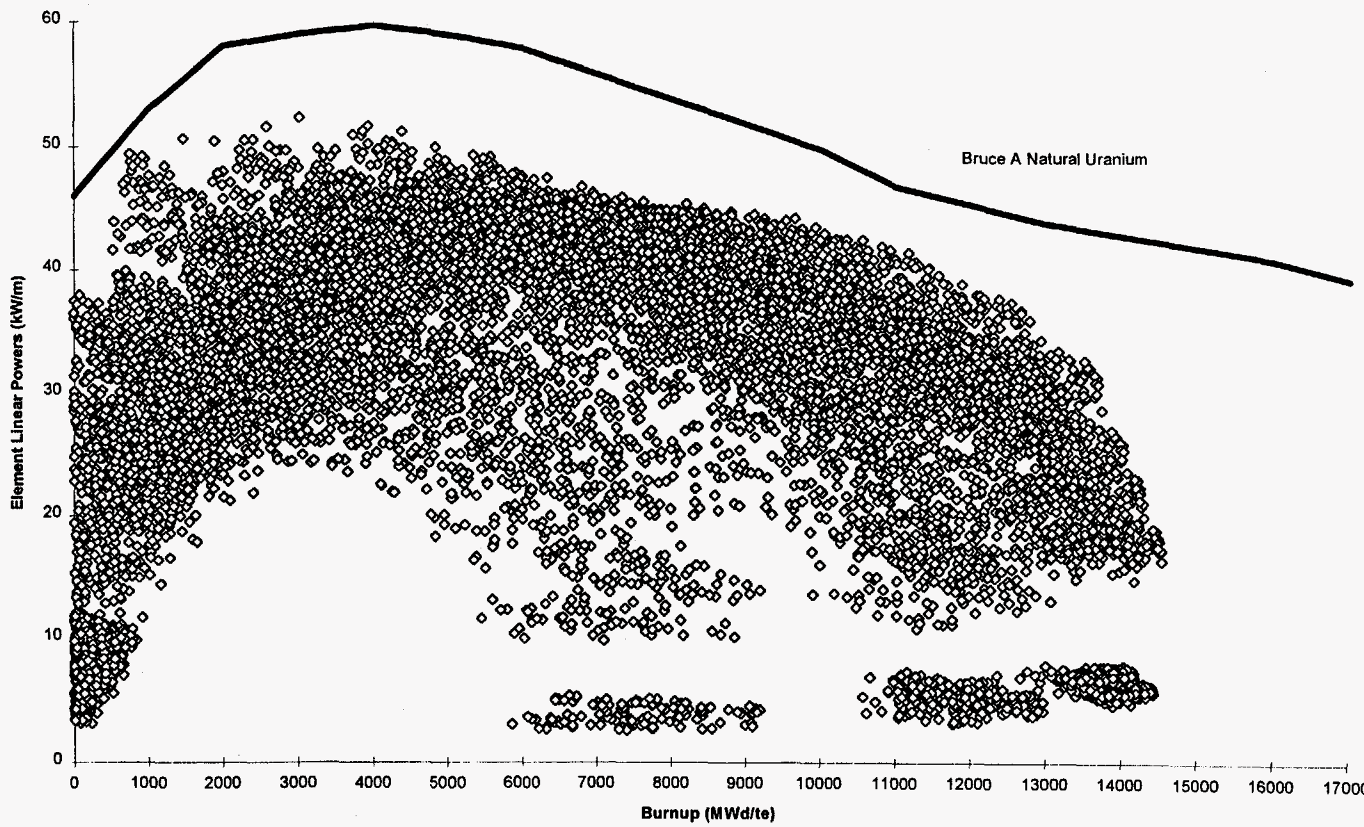
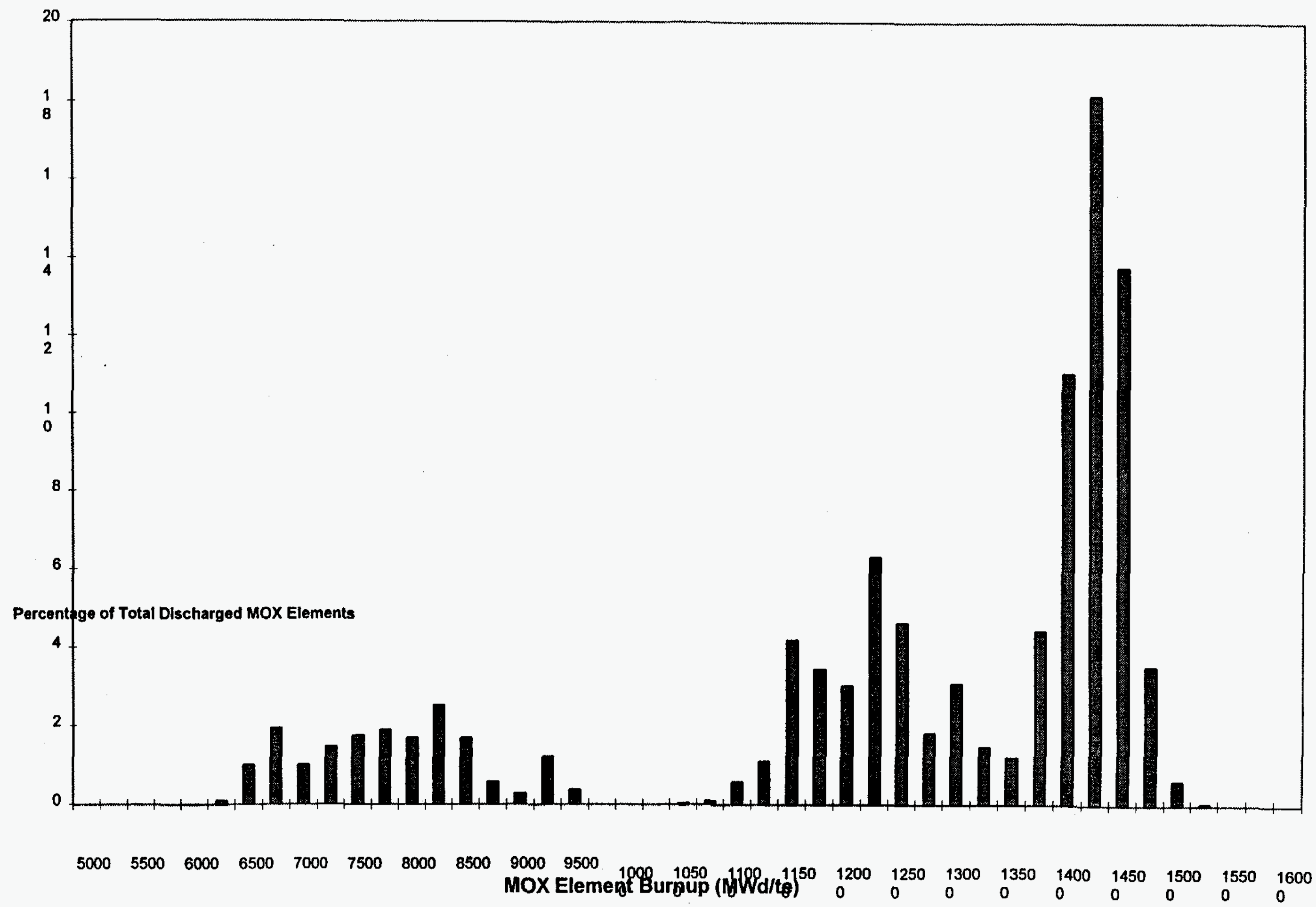


Figure 2.2.5-4
Distribution of MOX Element Burnup in Discharged MOX Fuel



2.3 NUCLEAR ANALYSIS FOR CANFLEX MOX CORE

The major difference between the reference MOX design and the CANFLEX MOX design is the MOX fuel fabrication requirement. From Table 2.1.1-1, it can be seen that a MOX fuel fabrication facility having the capacity of 160 te of MOX fuel per year will be able to supply two Bruce A reactors using the reference MOX fuel, or four Bruce A reactors using the CANFLEX MOX fuel. The disposition rate is 3.0 te of plutonium metal per year for 2 reactors using the reference MOX fuel, or 4.8 te of plutonium metal per year for 4 reactors using the CANFLEX MOX fuel.

The reduction in MOX fuel fabrication requirement is due to the increase in the maximum allowable fuel burnup. The larger number of fuel elements in the CANFLEX fuel design reduces the element rating, thus allowing the fuel elements to reach higher burnup without undue degradation of fuel performance.

2.3.1 Reactor Operating Parameters

Table 2.3.1-1 compares the major nuclear characteristics of Bruce A using the reference MOX fuel with those for Bruce A using CANFLEX MOX fuel. The plutonium content is increased from 334.5 g in the reference MOX fuel bundle to 472.6 g in the CANFLEX MOX fuel bundle. The amount of dysprosium is increased from 0.51 kg to 0.63 kg. The D_2O purity is 97% for both cases. This gives a core-average fuel burnup of 17,000 MWd/te and a full core coolant void reactivity of -4.5 mk. The fuelling rate is 8.8 bundles per full power day using uniform 2-bundle-shift scheme.

The maximum MOX fuel element burnup is about 28,000 MWd/te, which is higher than the maximum MOX element burnup of 15,500 MWd/te in the reference MOX fuel design. Because of the higher initial plutonium content, the maximum bundle power in the CANFLEX MOX Bruce A core is expected to be about 800 kW, i.e. slightly higher than the expected maximum bundle power of 780 kW for the reference MOX Bruce A. Maximum channel power in both MOX reactors is expected to be about 7.0 MW.

Table 2.3.1-2 compares the uranium and plutonium content in the CANFLEX MOX fuel with that in the reference MOX fuel. Each fresh CANFLEX MOX fuel bundle contains 445.1 g of fissile plutonium. At discharge, each CANFLEX MOX bundle contains 194.1 g of fissile plutonium. The fissile plutonium content as a percentage of the total plutonium content in the fuel, drops from 94% in the fresh fuel to 64% in the discharged MOX bundle mainly due to the formation of the isotope Pu-240. The high Pu-240 content makes the plutonium in the discharged MOX bundle undesirable for weapons purposes.

Table 2.3.1-1
Comparison of Advanced and Reference MOX Fuel Characteristics

	CANFLEX MOX Bruce-A Station	**Reference MOX Bruce-A Station**
Fuel Bundle Geometry	Canflex 43-element Design	Existing Bruce 37-element design
Pellet Material Composition (% based on weight of heavy metal in fuel)	Ring 4 (18 pins) 2.6% Pu Ring 3 (12 pins) 4.6% Pu Ring 2 (6 pins) 15.0% Dy Ring 1 (1 pins) 15.0% Dy All rings have 0.2% U^{235}	Ring 4 (18 pins) 1.6% Pu Ring 3 (12 pins) 3.1% Pu Ring 2 (6 pins) 15.0% Dy Ring 1 (1 pins) 15.0% Dy All rings have 0.2% U^{235}
Bundle Material Composition (Fuel Materials Only)	Pu 0.47 kg U^{235} 0.04 kg U^{238} 17.22 kg Dy 0.63 kg O_2 2.97 kg Total : 21.33 kg	Pu 0.33 kg U^{235} 0.04 kg U^{238} 17.92 kg Dy 0.51 kg O_2 2.53 kg Total : 21.33 kg
Average Burnup	17,000 MWd/te	10,000 MWd/te
Maximum Burnup	28,000 MWd/te	15,500 MWd/te
Bundle/FPD	8.8	15
Fuel Management Scheme	2 Bundle Shift	2 Bundle Shift
Maximum Channel Power	7,000 KW	7,000 KW
Maximum Bundle Power	800 KW	780 KW
LOCA Void Reactivity	-4.5 mk	-5.0 mk
Moderator Coolant Purity	97 %	97%

Table 2.3.1-2
Actinide Inventory for CANFLEX and Reference MOX Fuel Bundle (g/bundle)

	New		Exit Burnup	
	CANFLEX MOX	Reference MOX	CANFLEX MOX 17,000 MWd/teHE	Reference MOX 10,000 MWd/teHE
^{235}U	34.7	36.0	18.8	22.2
^{238}U	17218.8	17923.6	17088.1	17831.1
^{239}Pu	443.4	313.8	173.2	172.9
^{240}Pu	27.4	19.4	101.8	69.3
^{241}Pu	1.7	1.2	20.9	13.1
^{242}Pu	0.1	0.1	6.7	2.9

2.3.2 Reactor Power Distribution

Table 2.3.2-1 gives the detailed element burnup and element power rating in each fuel ring for each bundle in a high power MOX channel. The bundle power shape is obtained from a time-average calculation. The bundle powers have been normalized to give a channel power of 7.0 MW, which is the highest expected in the MOX core. it should be noted that further fine-tuning of the plutonium loading in rings 3 and 4 is possible to achieve a more uniform distribution of power rating in the elements of these two rings.

Figure 2.3.2-1 shows the time-average channel power distribution in the CANFLEX MOX Bruce A core. Figure 2.3.2-2 shows that the radial power distribution of the CANFLEX MOX core is very similar to that of the reference MOX core. A comparison of the axial bundle power distribution between a CANFLEX MOX channel and a reference MOX channel is shown in Figure 2.3.2-3.

For the same channel power, the peak bundle power in the CANFLEX MOX channel is higher than that in the reference MOX channel. The axial power profile is more skewed towards the inlet end of the channel. These effects are to be expected because of the higher initial plutonium content in the CANFLEX MOX bundle. Similar effects were observed in the 1994 MOX study. Thermalhydraulic analyses performed in the 1994 study indicated no significant difference in the CCPs between the reference MOX and the CANFLEX MOX channels, without taking any credit for CHF enhancement associated with turbulence promoting appendages on CANFLEX elements.

An instantaneous channel power distribution of the CANFLEX MOX Bruce A core is shown in Figure 2.3.2-4 and the channel power ripple map is shown in Figure 2.3.2-5. The maximum rippled channel power is less than 7.0 MW. The maximum bundle power is less than 800 kW and typical channel power ripple is about 10% in the high power region.

Table 2.3.2-1
Axial Burnup and Power Distribution in a 7.0 MW Channel using CANFLEX MOX Fuel

Bundle Position	Bundle Burnup MWd/te	Burnup in				Element Rating (kW/m)				Bundle Power kW
		Ring 1	Ring 2	Ring 3	Ring 4	Ring 1	Ring 2	Ring 3	Ring 4	
1	1110	39	45	1076	1633	0.4	0.5	9.8	14.3	212
2	3299	117	136	3214	4842	1.4	1.7	31.2	41.2	632
3	5259	196	232	5318	7582	2.0	2.4	41.7	50.1	794
4	7183	281	335	7505	10189	2.0	2.5	41.3	45.4	743
5	8812	358	432	9455	12324	2.0	2.5	39.4	40.3	680
6	10551	448	548	11643	14527	2.1	2.7	39.0	37.0	645
7	11977	527	651	13497	16290	2.1	2.8	37.8	33.9	606
8	13589	624	780	15661	18233	2.2	3.0	37.4	31.7	582
9	14867	811	1036	19613	21561	2.4	3.4	37.0	29.0	553
10	16406	813	1041	19595	21511	2.3	3.3	36.0	28.3	539
11	17550	899	1162	21232	22809	2.3	3.4	34.5	26.7	513
12	18375	964	1256	22425	23733	1.8	2.7	25.5	19.4	377
13	18417	967	1261	22486	23780	0.6	0.9	8.4	6.4	124

Figure 2.3.2-1
Time-Average Channel Powers in KW (TH) Normalized to Reactor Power of 3729 MW (TH) in Bruce A CANFLEX MOX Core

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
A								3563	4013	4370	4564	4622	4621	4566	4370	4014	3563							
B						3789	3968	4416	4826	5105	5273	5305	5307	5272	5107	4826	4418	3969	3790					
C					4247	5006	4966	5297	5567	5779	5888	5898	5897	5890	5778	5570	5297	4969	5007	4249				
D				4266	4566	5113	5465	5575	5750	5861	5920	5882	5884	5918	5864	5749	5578	5465	5116	4568	4267			
E			4208	4571	5138	5554	5796	5863	5959	6032	6019	5931	5929	6022	6030	5962	5863	5799	5556	5140	4573	4209		
F		3672	5017	5134	5553	5789	5859	5999	6108	6139	6105	5972	5974	6103	6141	6107	6002	5859	5791	5554	5134	5018	3672	
G		3792	4915	5427	5718	5764	5893	6085	6190	6235	6181	6048	6046	6184	6234	6192	6085	5895	5765	5720	5429	4915	3794	
H		4236	5210	5442	5656	5737	5890	6109	6259	6303	6271	6146	6148	6270	6305	6259	6111	5891	5738	5658	5442	5212	4235	
J	3667	4684	5522	5620	5725	5772	5894	6143	6290	6357	6316	6242	6241	6318	6356	6292	6143	5895	5774	5725	5623	5521	4686	3666
K	4117	5083	5824	5798	5850	5840	5955	6183	6336	6373	6385	6333	6335	6384	6375	6336	6184	5956	5840	5853	5797	5827	5082	4119
L	4429	5368	6051	5968	5977	5952	6039	6253	6358	6426	6468	6426	6425	6470	6425	6359	6254	6039	5953	5976	5970	6048	5370	4427
M	4581	5515	6183	6077	6087	6067	6145	6322	6401	6446	6476	6409	6411	6475	6448	6400	6322	6146	6066	6089	6075	6185	5512	4581
N	4595	5531	6214	6119	6169	6196	6275	6398	6436	6455	6445	6331	6329	6447	6453	6437	6397	6275	6197	6167	6120	6211	5532	4592
O	4461	5420	6126	6085	6169	6245	6327	6437	6456	6456	6424	6285	6287	6422	6457	6455	6438	6327	6243	6169	6082	6126	5417	4460
P	4164	5150	5929	5945	6090	6186	6299	6415	6472	6443	6381	6252	6250	6383	6440	6473	6413	6298	6185	6088	5944	5926	5149	4161
Q	3710	4752	5621	5763	5938	6076	6196	6355	6435	6460	6402	6316	6318	6399	6461	6433	6354	6195	6074	5936	5760	5620	4749	3709
R		4289	5291	5547	5795	5910	6063	6252	6383	6436	6438	6382	6380	6440	6433	6384	6250	6062	5908	5793	5546	5288	4288	
S		3828	4966	5510	5784	5765	5892	6134	6283	6366	6383	6334	6337	6380	6368	6280	6134	5889	5764	5782	5508	4964	3826	
T		3669	5015	5170	5549	5708	5759	5983	6152	6236	6254	6195	6192	6257	6232	6153	5980	5759	5706	5547	5167	5013	3667	
U			4183	4574	5105	5459	5705	5814	5963	6060	6066	5971	5974	6062	6062	5960	5814	5701	5458	5102	4572	4181		
V				4222	4541	5096	5459	5562	5753	5872	5897	5804	5802	5899	5868	5754	5558	5459	5093	4540	4220			
W					4198	4985	5016	5351	5636	5835	5908	5849	5851	5905	5837	5632	5352	5013	4984	4195				
X						3772	4011	4472	4888	5164	5306	5331	5328	5308	5161	4889	4469	4011	3770					
Y								3570	4030	4379	4561	4617	4618	4559	4379	4027	3570							

MAXIMUM = 6476 AT M11

Figure 2.3.2-2
Radial Power Distribution: CANFLEX Fuel Design vs. Reference MOX Fuel

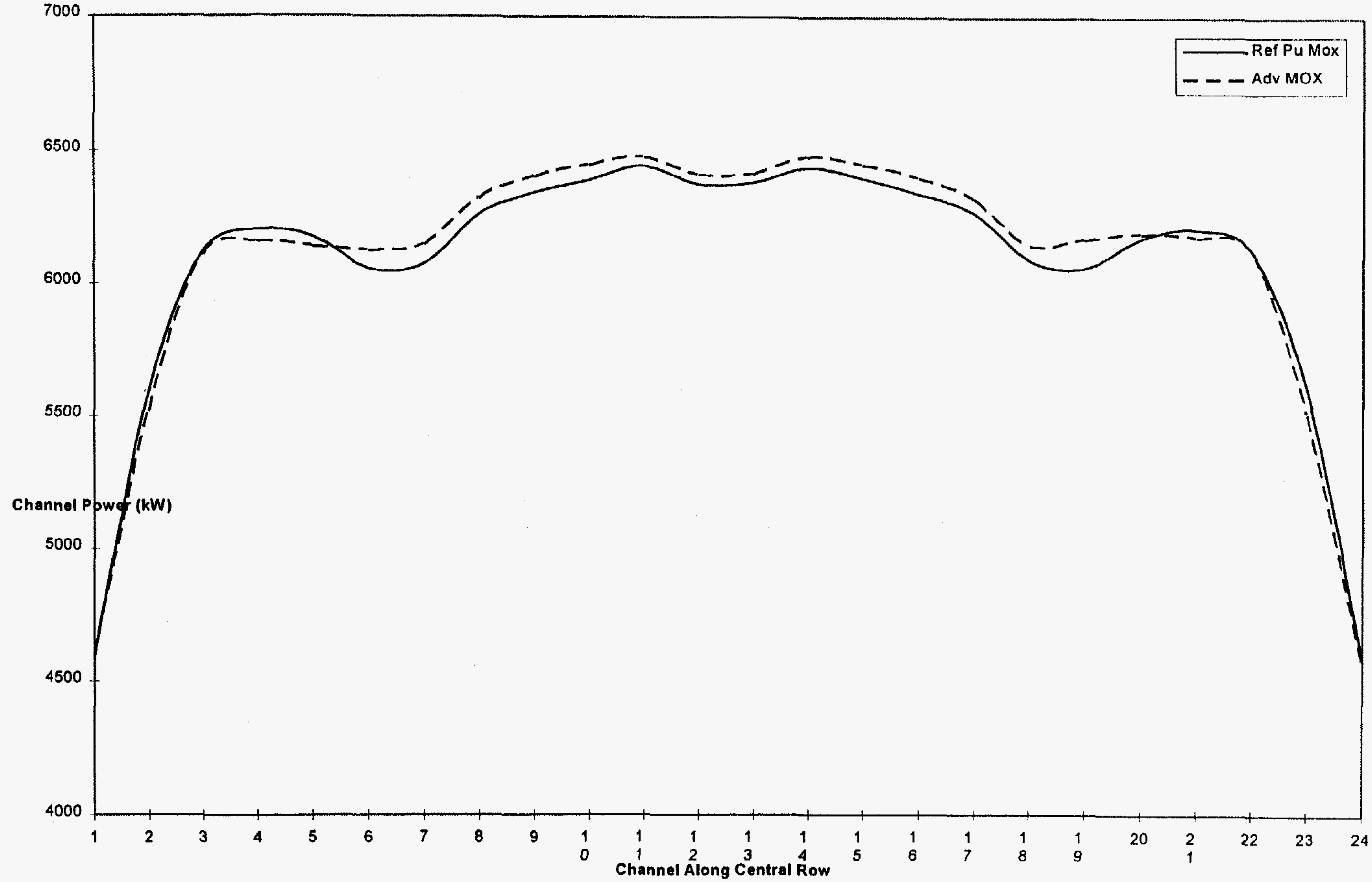


Figure 2.3.2-3
Axial Power Distribution Advanced Design vs. Reference MOX
Normalized to 7.0 MW Channel Power Fuel

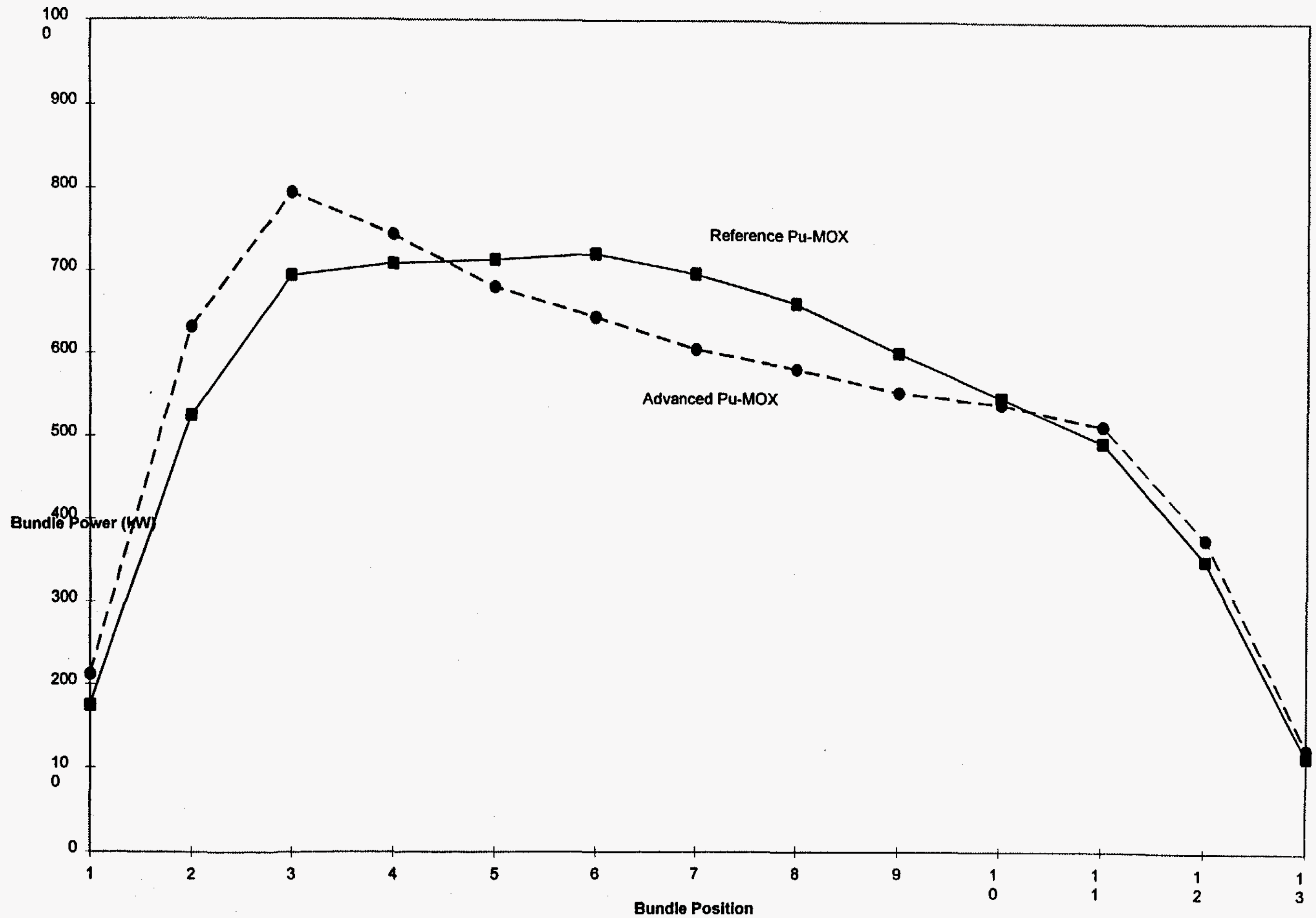


Figure 2.3.2-4
Instantaneous Channel Powers in KW (TH) Normalized to Reactor Power of 3729 MW (TH) in Bruce A CANFLEX MOX Core

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
A								3556	4214	4444	4734	4537	4501	4762	4417	4330	3602							
B						3683	4045	4787	5013	4989	5583	5301	5382	5718	5489	5043	4476	4301	3701					
C					4142	4776	5202	5270	5591	5950	6319	6083	5827	5850	5846	5446	5417	5241	5045	3988				
D				3945	4318	4998	5633	5775	6297	6237	5837	5625	6008	6327	6206	6138	6005	5338	4937	4424	4173			
E			4090	4185	4902	5266	5438	5776	5662	6175	5906	5931	5627	6008	5811	5899	5969	5505	5405	4840	4196	4028		
F		3670	4872	5058	5155	5480	5587	6073	5932	5760	6093	5785	5801	6095	5854	6098	5906	5910	5446	5275	4813	5002	3561	
G		4147	4812	5766	5672	5842	6202	6170	6289	6647	6377	5757	5757	6484	6524	6740	6104	6361	5840	5767	5692	4902	3852	
H		4551	5289	5707	5509	5526	6248	6058	6830	6225	6438	6021	6205	6223	6200	6687	6312	6337	5703	5452	5701	5141	4618	
J	4018	4928	5370	6043	5702	5926	6449	6626	6496	6342	6697	5822	5891	6307	6860	6555	6067	6421	5903	5839	6149	5885	4868	3681
K	4167	5124	6002	6260	6043	5746	5922	6188	6088	6361	6565	6183	5735	6539	6218	6319	6381	6504	6090	5776	5706	5856	4908	4179
L	4620	5885	6398	5832	5629	6001	6355	6517	6626	6719	6057	5909	6021	6441	6514	6858	6609	5920	5646	6006	6322	6281	5647	4729
M	4612	5266	6315	5870	5942	5600	6002	5960	6174	6333	5865	5899	5777	5847	6166	5969	6357	5928	5893	5615	5945	5861	5382	4661
N	4681	5372	5825	6050	5840	5862	6135	5968	6250	6122	6199	5642	5746	5898	6352	6098	5898	6121	5841	5797	6001	5787	5433	4453
O	4486	5480	6474	6270	5804	5885	6549	6629	6855	6240	6596	6042	6083	6512	6333	6430	6740	6440	5833	5760	6269	6237	5738	4328
P	4104	5596	5857	6161	6010	6273	6271	6261	6793	6422	6511	5872	5782	6468	6224	6931	6291	6435	6050	6106	5838	5706	5369	4092
Q	3920	4917	5639	6290	5657	5904	6298	6921	6644	6226	6670	6161	6237	6814	6813	6607	6350	6651	5755	5659	5729	5975	4825	3517
R		4161	5400	5902	5948	5627	6486	6227	6415	6573	6880	6480	6188	6267	6405	6139	6341	6419	6001	5390	5778	5122	4193	
S		4053	5296	5369	5644	5817	6174	6451	6884	6789	6248	6008	6371	6739	6648	6628	6557	5743	5588	5646	5570	5010	4033	
T		3637	5049	4821	5419	5468	5491	5923	5905	6387	6154	6139	5837	6179	5955	6014	6008	5390	5480	5179	4727	4800	3451	
U			4028	4500	4693	5189	5427	5930	5797	5746	6093	5870	5832	6069	5722	5861	5588	5576	4970	4679	4139	4050		
V				4266	4437	5057	5709	5603	5900	6315	6226	5654	5667	6227	6119	6143	5397	5606	4896	4331	4107			
W					3992	4813	5205	5339	6147	5906	6202	5942	6056	5962	5739	5895	5314	5064	4685	3827				
X						3816	4389	4831	5154	5286	5815	5209	5259	5403	5565	4998	4256	4076	3600					
Y								3671	4013	4542	4881	4803	4496	4767	4338	3975	3521							

MAXIMUM = 6931 AT P16

Figure 2.3.2-5
Instantaneous Channel Overpower Dist (X1000) in CANFLEX MOX Core

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
A								998	1050	1017	1037	982	974	1043	1011	1079	1011							
B						972	1019	1084	1039	977	1059	999	1014	1085	1075	1045	1013	1084	976					
C					975	954	1047	995	1004	1030	1073	1031	988	993	1012	978	1023	1055	1008	939				
D				925	946	978	1031	1036	1095	1064	986	956	1021	1069	1058	1068	1076	977	965	969	978			
E			972	915	954	948	938	985	950	1024	981	1000	949	998	964	989	1018	949	973	942	918	957		
F		999	971	985	928	947	954	1012	971	938	998	969	971	999	953	998	984	1009	940	950	937	997	970	
G		1094	979	1062	992	1013	1053	1014	1016	1066	1032	952	952	1048	1046	1088	1003	1079	1013	1008	1048	997	1015	
H		1074	1015	1049	974	963	1061	992	1091	988	1027	980	1009	993	983	1068	1033	1076	994	963	1048	986	1090	
J	1096	1052	972	1075	996	1027	1094	1079	1033	998	1060	933	944	998	1079	1042	988	1089	1023	1020	1093	1066	1039	1004
K	1012	1008	1031	1080	1033	984	995	1001	961	998	1028	976	905	1024	975	997	1032	1092	1043	987	984	1005	966	1015
L	1043	1096	1057	977	942	1008	1052	1042	1042	1046	936	920	937	996	1014	1078	1057	980	948	1005	1059	1038	1052	1068
M	1007	955	1021	966	976	923	977	943	965	982	906	920	901	903	956	933	1006	965	971	922	979	948	976	1017
N	1019	971	937	989	947	946	978	933	971	948	962	891	908	915	984	947	922	975	943	940	981	932	982	970
O	1006	1011	1057	1031	941	942	1035	1030	1062	967	1027	961	968	1014	981	996	1047	1018	934	934	1031	1018	1059	970
P	986	1087	988	1036	987	1014	995	976	1050	997	1020	939	925	1013	966	1071	981	1022	978	1003	982	963	1043	983
Q	1057	1035	1003	1092	953	972	1016	1089	1032	964	1042	975	987	1065	1054	1027	999	1074	947	953	995	1063	1016	948
R		970	1021	1064	1026	952	1070	996	1005	1021	1069	1015	970	973	996	962	1015	1059	1016	930	1042	969	978	
S		1059	1067	974	976	1009	1048	1052	1096	1066	979	948	1005	1056	1044	1056	1069	975	969	977	1011	1009	1054	
T		991	1007	933	977	958	953	990	960	1024	984	991	943	988	955	977	1005	936	960	934	915	958	941	
U			963	984	919	950	951	1020	972	948	1005	983	976	1001	944	983	961	978	910	917	905	969		
V				1010	977	992	1046	1007	1026	1075	1056	974	977	1056	1043	1068	971	1027	961	954	973			
W					951	966	1038	998	1091	1012	1050	1016	1035	1010	983	1047	993	1010	940	912				
X						1012	1094	1080	1054	1024	1096	977	987	1018	1078	1022	952	1016	955					
Y								1028	996	1037	1070	1040	974	1046	990	987	986							

MAXIMUM CHANNEL POWER PEAKING FACTOR = 1.096 IN CHANNEL L- 2

2.3.3 Reactor Safety and Control

Table 2.3.3-1 gives the reactivity worth of the control and safety devices in Bruce A with the CANFLEX MOX fuel and with the reference MOX fuel. Equilibrium xenon load, average reactivity increase due to fuelling a single channel, and full core void reactivity are also given for both cases. As expected, all the reactivity devices in the CANFLEX MOX core are worth less than those in the reference MOX core because of the higher absorption cross section in the CANFLEX MOX fuel lattice due to the higher plutonium content.

Detailed analyses of the performance of the safety and control systems in the CANFLEX MOX core have not been carried out in the current study. Appropriate analyses will be needed at a later stage to confirm the adequacy of the existing safety and control system for the CANFLEX MOX core.

Table 2.3.3-1
Reactivity Effects in Bruce A CANFLEX and Reference MOX Cores

	CANFLEX MOX Core (mk)	**Reference MOX Core (mk)**
14 Zone Controllers Inserted	-3.4	-3.5
30 SORs Inserted	-12.4	-13.3
28 SORs Inserted (Two Best Rods Missing)	-10.9	-12.1
4 MCAs Inserted	-2.8	-5.5
One Channel Refuelled	+0.2	+0.1
Full Core Voided	-4.5	-5.0
Equilibrium Xenon Load	-19.0	-20.0

2.3.4 Reactor Kinetics Parameters

Table 2.3.4-1 gives the kinetics parameters for reference MOX and CANFLEX MOX Bruce A at equilibrium conditions. The kinetics parameters for the CANFLEX MOX core are not significantly different from those in the reference MOX core. They are not expected to significantly affect the performance of the control and safety systems. However, suitable analyses should be carried out at a later stage to confirm this conclusion.

Table 2.3.4-1
Delayed Neutron Data for Equilibrium CANFLEX and Reference MOX Fuel

Group	CANFLEX MOX Fuel		REFERENCE MOX Fuel	
	Delayed Neutron Fraction (b)	Time Constant (s^{-1})	Delayed Neutron Fraction (b)	Time Constant (s^{-1})
1	0.202E-03	0.462E-03	0.207E-03	0.470E-03
2	0.779E-03	0.312E-01	0.788E-03	0.311E-01
3	0.649E-03	0.132E-00	0.662E-03	0.132E-00
4	0.136E-02	0.327E-00	0.139E-02	0.327E-00
5	0.464E-03	0.136E+01	0.474E-03	0.135E+01
6	0.138E-03	0.355E+01	0.144E-03	0.357E+01
Total	0.359E-02		0.366E-02	

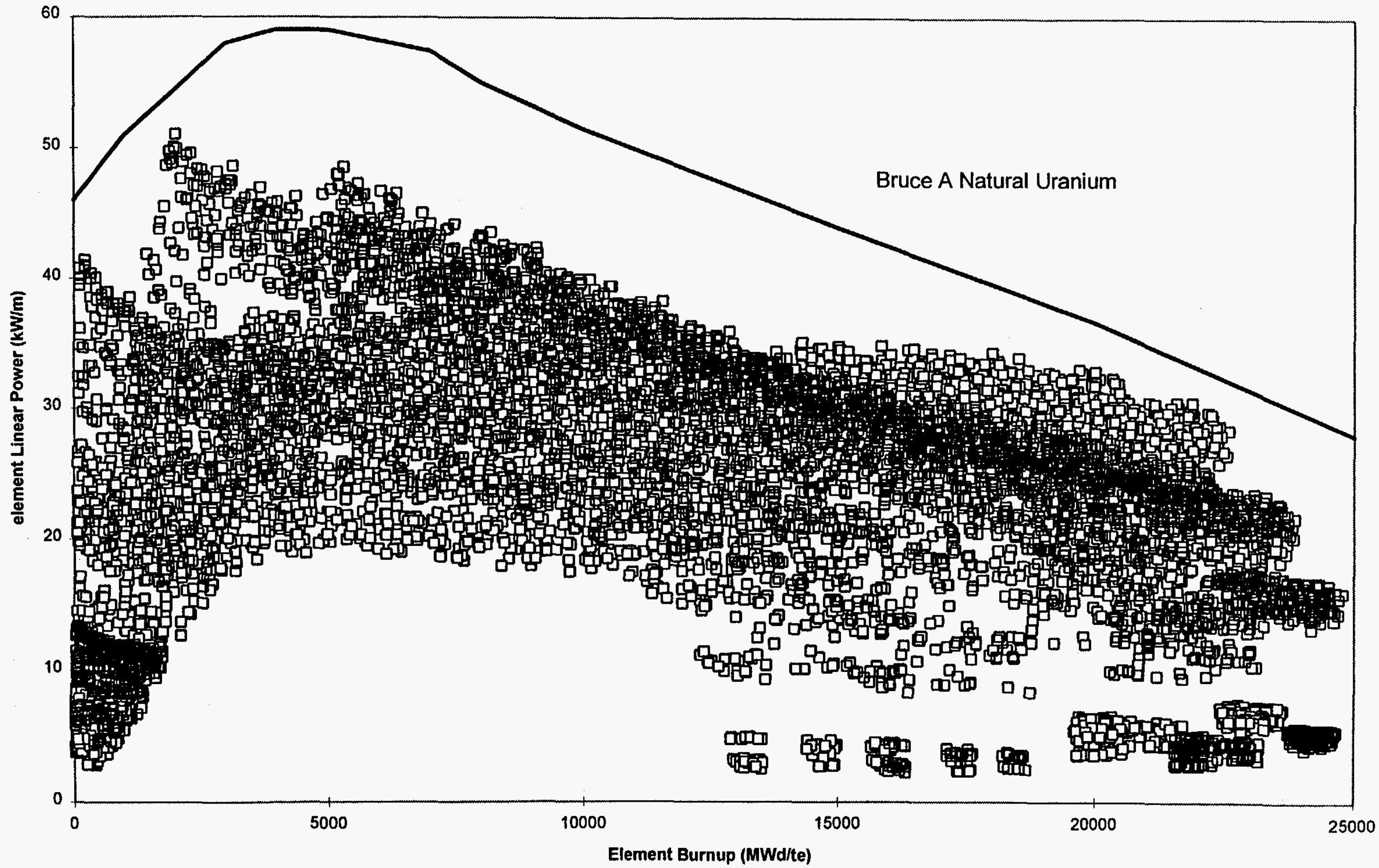
2.3.5 MOX Fuel Operating Parameters

A snapshot of the CANFLEX MOX element power versus burnup based on an instantaneous core calculation with a random fuel burnup distribution is shown in Figure 2.3.5-1. The maximum element rating envelope is very similar to the one obtained in the 1994 study. It is below the envelope established for the current natural uranium fuel in Bruce A. Therefore, no additional risk in fuel failure is expected for the CANFLEX MOX fuel.

Because no time-dependent fuelling simulation is carried out for the CANFLEX MOX core, the maximum power boost envelope for the CANFLEX MOX fuel cannot be established in the current study. However, fuel performance analyses carried out in 1994 concluded that the power boosting effect for the CANFLEX MOX fuel should be less than that for the reference MOX fuel because of the greater number of MOX fuel elements in the CANFLEX design.

Fuel performance is not expected to be a problem for the CANFLEX MOX core. However, this conclusion is based on limited information. It should be confirmed by detailed fuelling simulations at a later stage.

Figure 2.3.5-1
Snapshot of Intermediate - and Outer-Element Powers and Burnups for Advanced MOX Fuel in Bruce A



3. CANDU MOX FUEL QUALIFICATION PROGRAM

3.1 INTRODUCTION

This section outlines the scope of work, cost estimates and schedule for a MOX Fuel Qualification program. The program is designed to provide assurance to the utility (Ontario Hydro) and the Canadian regulator (AECB) that the new MOX fuel bundle can be successfully irradiated at the Bruce A reactor units.

The scope of work includes: all tests done at AECL and its subcontractors in support of the fuel element development and qualification, fuel bundle qualification; and production bundle verification; all post-irradiation-examinations; all engineering effort at AECL for fuel element and bundle design work, for the procurement and fabrication of test elements, bundles and related components at AECL, for the preparation of test specifications, for the evaluation of test results, for the participation at design reviews by experts from AECL, Ontario Hydro and their subcontractors, as requested by the Ontario Hydro design authority for fuel used at Bruce A.

The resources and costs for Ontario Hydro engineering and technical support of these qualification efforts, including the site effort in support of bundle demonstration efforts at Bruce A NGS, are included in the overall cost estimates in Section 6. The resources and costs for generic efforts, such as those to qualify CANFLEX fuel for Natural uranium, and to qualify dysprosia fuel are not included. Also the production of MOX facility fuel verification bundles (see Section 3.5) are to be included in the fuel supply costs, and are not included in this section.

With these exceptions, the cost estimates include all activities associated with the fuel qualification program.

The proposed schedule qualifies the MOX 37-element fuel bundles first and then the 43-element bundles. The MOX 37-element standard bundles are targeted to be fully qualified for use in Bruce-A reactors by early 2002, provided some engineering tasks begin before the project approval date of end of 1997 and preparations for Critical Heat Flux (CHF) testing start before the end of 1997. The schedule shows that the MOX CANFLEX bundles are expected to be qualified about two years after the MOX standard bundles.

Although most of the MOX fuel element development effort will be done on 37-element size pins, the tests results should be useful for both bundle designs. At this time, fuel element development effort for the CANFLEX bundle is expected to be less than that for the 37-element bundle.

The qualification program for the MOX 37-element bundle design will focus on the nuclear design system aspects. Fuel bundle interactions with the heat transport system (or primary circuit), the fuel channel and fuel handling systems are generally considered to be acceptable because bundle design is essentially identical to the standard design for natural uranium.

The qualification program for the MOX CANFLEX bundle design will focus on the differences in the bundle designs and the impact on all reactor systems. In addition to demonstrating compliance with the nuclear design, the qualification program for the MOX CANFLEX bundle design must demonstrate that the bundle is dimensionally compatible with the fuel handling, fuel channel and heat transport systems. Where applicable, the qualification program for the MOX CANFLEX design will make use of information available from the AECL/KAERI qualification program for natural uranium CANFLEX bundles designed for the CANDU 6 reactors. It should be noted that there are common features between the MOX CANFLEX bundle design for

Bruce A and the natural uranium CANFLEX bundle design for CANDU 6 reactors. The two CANFLEX bundle designs reflect the differences in fuel channel and fuel handling systems for the two reactor designs.

3.2 CONCEPTUAL FUEL DESIGN REQUIREMENTS

The design requirements for Bruce A fuel, listed in the subsections below, include the original ones imposed on the 37-element standard natural uranium fuel bundle. They also include new ones to reflect recent changes in the refuelling operations (i.e., "fuelling-with-flow" instead of "fuelling-against-flow") and to address safety issues related to void reactivity for MOX fuel. It should also be noted that the information generated by the Parallex program, is an important part of the development of the MOX fuel element. This program is currently underway and is described separately.

3.2.1 Functional Requirements

3.2.1.1 Bruce A Conditions

A Bruce A reactor core contains 480 horizontal fuel channels, each with 13 fuel bundle columns, held in position against the coolant hydraulic drag by four latch fingers. Latches are located near the core boundary at the downstream end of each fuel channel. Core reactivity is maintained by on-power refuelling. Two fuelling machines are needed to refuel one channel: one loads new fuel bundles into one end, while the other accepts old fuel bundles from the other end.

In the 1970s, the Bruce units were originally designed for refuelling against the direction of coolant flow. The refuelling concept was developed and fully proven earlier for the NPD and KANUPP reactors, built a decade earlier in Canada and Pakistan, respectively.

In the 1990s, a new requirement was introduced in the Bruce A reactors for having a lower reactivity change during postulated breaks in the inlet headers of the heat transport system. A break at this location would cause coolant flow reversal and the fuel columns in many channels to be displaced away from the fuel latch. With the fuelling-against-flow concept, the new fuel located at the downstream end would be shifted toward the centre region of the channel during an inlet header break. The resulting sudden movement of the fuel column would lead to a positive reactivity change. To reduce the reactivity increase, a decision was made by the Bruce A station to reverse the fuelling direction in all reactor units. By relocating new fuel at the upstream end of the channel, the reactivity change during the break would become negative.

Since the duty cycle for the fuel in the Bruce A units has changed in recent years, the qualification program being developed for MOX fuel must reflect these changes, as described below:

a) All downstream shield plugs have been replaced with a new design that supports the fuel column away from the fuel latch. The fuel string supporting shield plug (F3SP) extends past the latch and supports the fuel column against the coolant hydraulic drag by contacting the three endplate rings of the downstream bundle (a new design requirement introduced by the F3SP designers). With latch fingers, the reaction force was applied *eccentrically* to about 14 outer fuel elements of the downstream bundle. With the F3SP, it is applied *concentrically* to 36 fuel elements of the three rings of the endplate. The change in axial loading changes the bowing characteristics of the outer elements in the downstream bundles that come into direct contact with the pressure tube. Therefore, there may be some changes in the material loss characteristics due to fretting between the fuel bundle bearing pad and the pressure tube surfaces. The axial support between the F3SP and the MOX 37-element bundle, which is dimensionally identical to the current Bruce bundle, is considered to be acceptable because of the F3SP qualification program. The MOX qualification program needs to show that the fretting characteristics of the MOX CANFLEX bundle due to support with the F3SP are acceptable.

b) After the downstream shield plug is removed during refuelling, the entire fuel column will be pushed by the coolant flow until the downstream bundle comes into contact with the fuel latches. Latches are comprised of four separate fingers that are spring loaded and move in the radial direction only. They are designed to permit bundle passage against the flow and not with the flow. Latch fingers support the fuel column against the coolant drag by penetrating radially inward at the junctions between bundles and contacting the flat surfaces of the endcaps of the bundle. The radial penetration of the latch fingers has been designed to avoid contact with the endplates of the standard 37-element bundle. The outer elements of a CANFLEX bundle have smaller diameters than those of the standard 37 element bundle. Since endcaps are concentrically welded to the endplates, the CANFLEX bundle will have larger endplates than the 37 element bundle. Therefore, the dimensional compatibility of the CANFLEX endplates with the fuel latches must be demonstrated.

c) Because the F3SP is in contact with the fuel column, and because shield plugs are rotated to disengage with the liner lugs, the flat face surface of the F3SP applies a torque to the rings of the endplate of the downstream bundle. As part of the F3SP qualification program, the torque applied to the 37-element bundle is now considered acceptable. The mechanical response and the plastic deformation of the MOX 37-element bundle, are likely to be acceptable because the current bundle has already been qualified for these conditions. The MOX qualification program needs to show that the mechanical response and deformation characteristics of the MOX CANFLEX bundle due to the F3SP are acceptable.

d) In the fuelling-with-flow operation, the most upstream bundle is no longer needed and the utility has decided to eliminate this bundle from all fuel channels in the next few years. The conversion from 13 to 12 bundle fuel columns in all channels will likely take place before MOX fuel bundles become available. Fuelling-with-flow in channels having 12 bundle fuel columns will result in refuelling impacts as each pair of new bundles are loaded into the channel. The leading bundle will be swept into the channel as it enters the axial flow region of the pressure tube and will impact the stationary bundles in the channel. These refuelling impacts will be similar to those routinely experienced in the CANDU 6 reactors, which are also fuelled-with-flow. Due to the similarity between the Bruce and CANDU 6 37-element bundles, refuelling impacts at Bruce A are considered to be acceptable, based on the qualification tests done for CANDU 6 fuel. Similarly, refuelling impact tests are being done for the CANDU-6 CANFLEX bundle as part of the AECL/KAERI program. Depending on the results of these tests, refuelling impact tests for the MOX CANFLEX bundle may not be needed.

3.2.1.2 Interfacing System Requirements

The functional requirements are imposed on the fuel bundle by the interfacing systems. These systems include the heat transport system, the fuel channel, the fuel handling system design, and nuclear (physics) design.

HEAT TRANSPORT SYSTEM REQUIREMENTS

[1] The pressure drop across the fuel column shall be compatible with the heat transport system (HTS) design allowance.

[2] Over the life of the plant, fuel shall withstand the entire range of flows and pressure pulsations that may be present in the HTS.

[3] The fuel element surfaces shall remain wet during all normal operating conditions.

[4] The fuel element shall withstand the coolant pressure during normal operation and be able to withstand hydrostatic testing of the HTS.

[5] The fuel bundles shall provide adequate thermal performance in fuel channels with the maximum predicted radial creep of high power channels.

[6] To maintain the sensitivity of the delayed neutron system, the uranium (or fissile material) surface contamination of new fuel shall be less than the specified limit.

FUEL CHANNEL REQUIREMENTS

[7] The pressure tube wall reduction due to sliding wear, fretting and crevice corrosion shall not exceed the specified limits.

[8] The fuel bundle shall have the strength and flexibility to maintain its structural integrity under steady and fluctuating loads applied to it thermally, mechanically, hydraulically, and by refuelling operations (also a fuel handling requirement). Bundle ends must be dimensionally compatible with the fuel latch and downstream shield plug.

[9] For normal refuelling operations, fuel bundles shall accommodate dimensional changes due to irradiation effects and shall not jam in the fuel channel under all conditions of pressure tube sag, radial and axial creep, and misalignments between the pressure tube, spacer sleeves and fuel carriers.

FUEL HANDLING REQUIREMENTS

[10] The fuel bundle shall not become jammed in the channel during normal operation.

[11] The bundle ends must be compatible with the fuel carriers and the irradiated fuel discharge mechanism components.

[12] The fuel bundle must be able to withstand the combined axial loads imposed by the coolant drag and the fuelling machine rams without significant bundle deformation nor degradation of performance.

[13] During refuelling, fuel bundles shall be able to withstand the loads caused by the crossflow in the radial flow region of the fuel carrier for short periods of time.

[14] The fuel bundle must be designed to withstand all changes in power levels associated with refuelling (see also Nuclear Design Requirements).

NUCLEAR (PHYSICS) DESIGN REQUIREMENTS

[15] The fuel bundle shall be able to operate at high powers continuously.

[16] The fuel bundle shall withstand power changes caused by refuelling, movement of reactivity control devices, and normal reactor power manoeuvres.

[17] Fuel bundles shall contain the required levels of fissile material. A small number of bundles with depleted levels of fissile material are required for first core loading and occasionally during normal operation to suppress local power increases.

[18] Fuel bundles shall withstand power flux peaks that occur at the bundle ends.

[19] The fuel bundle design shall provide a margin on the critical channel power ratio for the expected range of axial flux shapes, element linear power distributions, flows, and flow area increases due to radial pressure tube creep up to the end of pressure tube life.

3.2.2 Seismic Requirements

[20] For a Design Basis Earthquake, the fuel bundle must retain a coolable geometry and not cause a loss in integrity of the pressure tube nor of the fuel elements.

3.2.3 Safety Requirements

[21] The nuclear design characteristics of a MOX fuel bundle and its lattice cell must not reduce the effectiveness of the current safety related systems and reactivity control systems. (This is one of the groundrules for selecting Bruce A for the disposition of plutonium.)

3.2.4 Interfacing Requirements

To meet the station requirement for low fission product emissions and occupational exposures, the fuel design imposes only one requirement on the reactor systems, as described below

[22] The reactor systems shall not cause systematic fuel failures during normal operation.

3.3 MOX 37-ELEMENT FUEL QUALIFICATION PROGRAM

The purpose of a fuel qualification program is to provide the "proof" that a proposed fuel design change will meet all design requirements. The MOX fuel qualification program outlined below includes fuel element development and qualification, fuel bundle qualification, design evaluations, design reviews with the utility, and the preparation of the fuel design manual. This section describes the program for the MOX 37-element bundle.

3.3.1 FUEL ELEMENT DEVELOPMENT AND QUALIFICATION

The external features of the MOX fuel element design will be identical to those of the Bruce fuel element, whereas the internal features of the element will be different. These differences may result in some changes in fuel element performance. Therefore, the first step in the qualification is to develop a MOX fuel element design and then show that it meets the nuclear design requirements listed in Section 3.2.1.2. The fuel element development and qualification testing to meet the nuclear design aspects will be done at CRL using the experimental loops of the NRU reactor and the ZED-2 reactor.

FUEL ELEMENT DEVELOPMENT IN NRU (PHASE 1)

The primary objective of Phase 1 testing in the NRU loops is to develop a MOX fuel element. The product is to develop and verify new AECL design specifications for the fabrication of MOX pellets and fuel elements. The results of the PIE work will also be used to help modify the existing fuel codes that predict fuel performance.

Subject to the results of the initial test element irradiations (Parallex Project), the key fuel performance aspects that are considered important for MOX fuel element development include:

- fission gas release and internal gas pressure,
- pellet thermal conductivity of irradiated pellets,
- elevated temperatures for standard and end pellets,
- sheath circumferential ridging at pellet interfaces and strains at the mid-pellet,
- incipient cracking of the fuel cladding,
- Canlub graphite coating adhesion to the inner surfaces of the cladding,
- element ovality,
- fission product migration within the pellets,
- hydride distribution within the cladding, and
- element axial strain.

Considering previous irradiation experience at AECL with CANDU type fuel elements containing enriched uranium, thoria, and MOX fuel materials, the above performance aspects can be controlled to some extent with the specific design features of the element. Characterization of pellets and non irradiated fuel elements, and the PIE of test elements will provide information to investigate various aspects of the design features. The features that are considered important include:

- pellet surface finish,
- degree of Pu homogenization,
- Pu particle size,
- fissile material enrichment,
- impurity content,

- gallium content,
- hydrogen gas content,
- oxygen-to-metal ratio,
- pellet-to-sheath clearance,
- Canlub graphite coatings,
- pellet density,
- endcap weld geometry,
- "standard" pellet design, and
- "end" pellet design.

Fuel element development irradiations will be done using existing hardware, i.e., demountable bundles sized for 37-element fuel irradiations for testing enrichments up to about 2%, and "fixed bundles" with Pu enrichments greater than 2%.

The test matrix will be prepared and prioritized on the basis of careful evaluation of the current database and the data provided by the Parallex Project.

Phase 1 tests will require 1-2 demountable bundles with MOX pins and 2-4 "fixed bundles" with non-outer elements having various Pu enrichments. Irradiation tests should test the capability of the MOX elements for withstanding power ramps and continuous high powers.

Tests should start Mid 98.

FUEL ELEMENT QUALIFICATION IN NRU (PHASE 2)

The primary objective of Phase 2 testing in the NRU loops is to demonstrate that the MOX fuel element design and the dysprosia doped inner elements, built to AECL specifications developed for MOX fuel under Phase 1 and for dysprosia doped fuel, meets the nuclear design requirements given in Section 3.2.1.2. The design of the test fuel elements should be as representative as possible of the final design for the Bruce A MOX fuel elements.

The extent of deterioration for most systematic fuel failures observed among CANDU reactors tend to be somewhat dependent on element powers (or temperatures) and power increases. Therefore, two types of irradiations are planned for qualification purposes in Phase 2:

a) continuous high power irradiation, and

b) a power ramp (or boost) test.

For the first test, the linear powers and final burnups of the experimental elements should be at least 10% higher than the maximum powers and twice the average discharge burnup given in Section 2.2.2. For the second test, the fuel elements must be subjected to power ramp conditions that cause defects due to stress corrosion cracking for natural uranium fuel in CANDU reactors.

It should be noted that the Pu enrichments of the MOX elements will be set at the maximum predicted by the fuel management simulations (4.6%) and the contents of the dysprosia doped elements set at about 15%.

To irradiate elements with Pu enrichments exceeding about 2% without overpowering the element in NRU, the test section may require special adjustments.

Phase 2 tests may require 1-2 demountable bundles with MOX pins or 2-4 "fixed bundles" with non-outer elements having various Pu enrichments.

Tests should start Mid 99.

3.3.2 FUEL BUNDLE QUALIFICATION

The external features of the MOX 37-element fuel bundle design will be essentially identical to the current standard Bruce fuel bundle, whereas the internal features of the elements will be different. These differences will result in some changes in the radial power profile for the bundle where most of the fission energy comes from the outer two rings of elements. A different power profile may change the bowing characteristics of the outer elements and the dryout characteristics of the bundle. Therefore, three types of tests are needed:

a) NRU prototype bundle qualification (Phase 3) having the same power profile expected for the MOX fuel bundle,

b) ZED-2 tests to confirm that the nuclear characteristics of the MOX bundle and its lattice cell are acceptable (Requirement [21]), and

c) Critical heat flux (CHF) tests to demonstrate that the bundle has an acceptable CCP margin (Requirement [19]).

NRU PROTOTYPE BUNDLE QUALIFICATION (PHASE 3)

The MOX fuel elements for the prototype fuel bundles are to be built using the AECL specifications for MOX pellets and fuel elements. These specifications are the ones developed from Phase 1 and 2 testing. The Pu enrichments for the outer two rings of elements and the dysprosia doping concentrations for the inner ring will be adjusted to give the correct radial power profile across the bundle. Bundles will be assembled to meet the existing specification for NU 37-element fuel.

The purpose of this irradiation is to show that the plastic deformation of the bundle due to the power profile is acceptable (Requirements [8 and 9]). To show compliance, the dimensions of the unirradiated bundles must be characterized, the axial loading of the fuel bundles must be representative of the steady state loads in the Bruce A channel, and the irradiated bundle characterized in the hot cells. The PIE plan will also include an examination of the welds.

Phase 3 tests may require 1-2 "fixed bundles" with outer and intermediate elements having Pu and inner ring with dysprosia. The bundles will be tested for continuous high power only without power ramping at bundle powers equal to that defined by the maximum power envelope (described in Section 2) and burnups to the predicted average discharge burnup.

Tests should start Jan 00.

ZED-2 TESTS

The purpose of the Zero Energy Demonstration (ZED-2) reactor at CRL is to measure the lattice pitch properties of the MOX bundles, specifically:

- reactivity of the MOX fuel lattice pitch compared to the NU fuel lattice,
- void reactivity of the MOX fuel lattice pitch,
- flux distributions across the fuel bundle, before and after coolant voiding, and

- coolant and fuel temperature reactivity effect of the MOX fuel lattice.

The results will help verify the physics calculations of the MOX design, and to confirm that the radial power profiles for the CHF tests are representative. They will form the basis for modifying the fuel management codes at Bruce. See Requirements [15,17,19,21]

The test setup will require 35 fixed bundles and one demountable bundle for fine structure measurements.

Tests should start on or before Jan 99.

CHF TESTS IN FREON

The purpose of the CHF tests in freon at CRL is to generate preliminary CHF characteristics of the MOX bundle to support the safety and licensing process to obtain approval for loading MOX prototype bundles into Bruce A. The CHF characteristics of the MOX bundle needs to be determined for various bundle radial power profiles corresponding to the fissile material enrichments, pressure tube geometries due to creep/aging effects, see Requirement [19]. These tests provide the basis for extrapolating the results obtained from the CHF water tests to predicted CHF characteristics for off normal operations.

The test bundles are comprised of electrical heaters and will be built in CRL. The heaters will reflect the power distribution shown in Table 2.2.2-1.

Tests should start Jan 98.

CHF TESTS IN WATER

The purpose of the CHF tests in water at STERN Labs in Hamilton Ontario is to generate CHF characteristics of the power distribution associated with the MOX bundle (Table 2.2.2-1) under representative in-reactor conditions. The results of these tests will support the approval process for loading MOX production bundles into the Bruce A.

The Bruce A reactor with a core containing MOX 37-element bundles is expected to have better critical channel power ratios than with NU 37-element bundles for three reasons:

a) the MOX core will have lower maximum channel powers under normal operating conditions than those for the NU core,

b) the power distribution along the channel containing MOX fuel is flatter than one with NU fuel, and

c) the bundle radial power profile for the MOX bundle is steeper than for the NU bundle resulting in lower powers for the inner elements that tend to dryout first.

The critical path for these CHF tests is the ordering and procurement of the heater elements that must have the correct electrical resistance to achieve the desired radial power distribution. To meet the target date for qualifying MOX 37-element bundles, it is proposed that the heater elements be ordered before the ZED-2 tests are completed using the predicted radial power profiles given in Table 2.2.2-1. The anticipated CPR improvements associated with the MOX fuel design are expected to offset any errors or uncertainties.

Heater elements should be ordered Jan 98.

POWER REACTOR DEMONSTRATION TESTS

The purpose of the power reactor demonstration test at Bruce A is to provide the final proof that the new MOX fuel element and bundle designs will successfully meet the specific requirements established by Ontario Hydro.

A small quantity of pre-production bundles are needed as a final check that the new bundle design can be successfully irradiated in the reactor. For the MOX 37-element bundle design, a 100 MOX bundle irradiation test has been suggested by OH. Prior to this test, the results of all preceding tests and the irradiation plan must be assessed and reviewed by the utility and the regulator. An irradiation plan will be prepared and approved by the utility.

After these bundles have been discharged from the core and inspected in the fuel bays, a small number of selected bundles and/or elements will transported by irradiated shipping flask to AECL for hot cell examinations.

The MOX 37-element reference bundle can be fabricated using the existing Bruce 37-element bundle assembly drawings, and as such it should be dimensionally compatible for the reactor systems in Bruce A, i.e., Requirements [1 to 5, and 7 to 14].

It should be assumed for costing purposes, that all MOX elements for the 100 bundles will be built at LANL, dysprosia doped elements at one of the Canadian fuel manufacturers, and bundle assembly at either CRL or LANL.

Bundle loading should start mid 2000.

3.3.3 FUEL ELEMENTS AND BUNDLES REQUIRED FOR QUALIFICATION PROGRAM

Considering the bundle requirements for each test site, as shown in Table 3.3.3-1, the total numbers of fuel elements and bundles are summarized below:

Fuel Element Fabrication:

4250 MOX fuel elements (to be fabricated in the USA); and

1220 Dy doped/ depleted uranium fuel elements (to be fabricated in Canada).

Fuel Bundle Assembly (at CRL):

4 Demountable bundles;

8 36-element MOX bundles; and

136 37-element MOX bundles.

Table 3.3.3-1
Number of Bundles Required for 37-MOX Fuel Qualification Program

	No. of BUNDLES	FUEL COMPOSITION IN EACH RING				Fuel Element requirements from:	
		OUTER	INTER	INNER	CENTRE	ZPI or GEC	LANL or CRL
IN-REACTOR TESTS							
37 ELEMENT							
PHASE 1	2 DME	~2% Pu	NU or 15% Dy/DU		none	38 = 2x19 (2 DME cores)	36 = 2 x 18 (outers)
	4 BDL	NU or 15% Dy/DU		~4% Pu	none	120 = 4x30 (outer/inter)	24 = 4x6 (inners)
PHASE 2	2 DME	~2% Pu	NU or 15% Dy/DU		none	38 = 2x19 (2 DME cores)	38 = 2 x 18 (outers)
	2 BDL	NU or 15% Dy/DU		~4% Pu	none	60 = 2x30 (outer/inter)	12 = 2x6 (inners)
PHASE 3	2 BDL	~1.5 % Pu	~0.8 % Pu	Dy/DU	none	12 = 2x6 (inners)	60 = 2x30 (outer/inter)
ZED-2 TESTS							
37-ELEMENT	36 MOX	1.6% Pu	3.1% Pu	15% Dy/DU		252 = 36x7 (inners/centre)	1080 = 36x30 (outer/inter)
BRUCE POWER REACTOR DEMONSTRATION							
37-ELEMENT	100 MOX	1.6% Pu	3.1% Pu	15% Dy/DU		700 = 100x7	3000 = 100x30

3.3.4 DESIGN EVALUATIONS

The purpose of the design evaluations, listed below, is to show analytically, that specific design requirements are met. In each case, tests are not needed.

1. Surface Contamination: An assessment is needed to calculate a new limit for surface contamination for MOX and dysprosia doped uranium. See Requirement [6]

2. Bent Tube Gauge: The currently used bent tube gauge used by the fuel manufacturers to ensure that new MOX bundles can pass through the fuel channel. The tube design must be reevaluated because of possible changes in fuel element swelling rates due to MOX. See Requirement [9]

3. Stress Corrosion Cracking Performance: The fuel failure defect thresholds should be evaluated and modified (if needed) for use with MOX fuel. The PIE results from all MOX fuel elements input will provide some indication if this is needed. See Requirement [14]

4. ASSERT Evaluations: The results of the CHF freon tests will be used to validate ASSERT code predictions of CHF performance of MOX fuel bundles. See Requirement [19]

5. Crevice Corrosion: A brief assessment is needed to show that the effect of crevice corrosion which occurs between the pressure tube and the fuel bearing pads is not a concern for MOX CANFLEX bundles. See Requirement [7].

3.3.5 DESIGN REVIEWS

As part of Canadian practice, any design change proposed for a component that is introduced into an operating reactor must undergo an extensive design review. Two design reviews are envisaged: a conceptual design review to be undertaken at the start of the MOX program, and a final design review to be undertaken near the end of the program. The main purpose of these reviews is to provide assurance to the appropriate utility design authority and to the AECB that the proposed design changes will not adversely affect the operation of the reactors. Each design review involves a series of meetings between two teams comprised of experts from several disciplines:

a) Resource Team - responsible for the design verification and the associated analyses of the test results and engineering assessments, and

b) Review Team - responsible for reviewing the documents produced by the resource team.

This process typically involves 30-40 professionals, about 3 meetings covering a 3 month period for the conceptual review and a 6 month period for the final review. These reviews will be conducted in parallel with the test program to the maximum extent possible.

Conceptual and final design reviews should take place in Jan 97 and Mid 01, respectively.

3.3.6 FUEL DESIGN MANUAL

As a final step, a Bruce A Fuel Design Manual needs to be prepared for MOX fuel bundles. This document should be prepared about 6 months before the verification program for MOX production bundles into Bruce A - see Section 3.5. (Design Manual issued by Jan 02)

3.3.7 SCHEDULE

Table 3.3.7-1 summarizes the schedule for all deliverables to be produced as a part of MOX 37-element bundle qualification.

Table 3.3.7-1
37-MOX Fuel Qualification Schedule

TEST/ ACTIVITY	START DATE	FINISH DATE
DESIGN ENGINEERING	MID-97	MID-00
NRU -PHASE 1 NRU -PHASE 2 NRU -PHASE 3	MID-98 MID-99 JAN-00	JAN-00 JAN-02 MID-02 (end of irradiations)
ZED-2	JAN-99	MID-02
CHF FREON	JAN-98	JAN-00
CHF WATER	JAN-98	MID-01
BRUCE DEMO	MID-00	MID-02 (first bundle out)
DESIGN REVIEWS		
CONCEPTUAL	JAN-97	JAN-98
FINAL	MID-01	MID-01
DESIGN MANUAL	JAN-01	JAN-02

3.4 MOX CANFLEX FUEL BUNDLE QUALIFICATION

The MOX 43-element bundle can be fabricated using a modified version of the existing CANFLEX bundle assembly drawings for the CANFLEX bundle intended for use in the CANDU 6 reactors. The CANDU 6/CANFLEX bundle drawing needs to be revised to make a Bruce specific design drawing. Some changes include:

a) the end bearing pads must be relocated according to the Bruce design to ensure proper interfacing with fuel carriers,

b) the endcap profiles must be changed to ensure appropriate contact with fuel latches,

c) the bearing pad heights may need to be changed to compensate for bundle droop and to avoid endplate interference with latches, and

d) endplate diameters may need to be reduced to ensure against interference with latches.

With respect to the last point, a CANFLEX bundle design with outer elements having smaller diameters requires endplates with a larger diameter than the reference 37-element bundle design. Hence, some engineering effort is needed to ensure that the bundle is dimensionally compatible with the fuel latches, installed in each fuel channel (unlike CANDU 6 reactors that have no latches). To ensure dimensional compatibility with latches, one of three changes needs to be pursued:

a) the endcaps of the outer elements need to be designed to permit off centre welds to permit CANFLEX bundle assembly with endplates having the same diameter as the 37 element bundle,

b) the fuel latch fingers that penetrate past the endcaps need to have less penetration to accommodate bundles with larger endplates (An option for consideration if CANFLEX bundles are introduced AFTER the Bruce A units are retubed.), or

c) the larger endplate associated with concentric welding needs to be qualified for existing fuel latches, i.e., a demonstration that endplate contact with latches is acceptable.

For the MOX CANFLEX, some engineering effort is required by AECL and the fuel handling test site to address the issues relating to the sizing of the CANFLEX bundle.

3.4.1 FUEL ELEMENT DEVELOPMENT AND QUALIFICATION

The information generated by the Parallex project and Phase 1 of the NRU tests will likely contribute to the development of the MOX fuel elements for the CANFLEX bundle. Nevertheless, some development may still be needed. However, prototype bundle qualification tests would still be required to demonstrate that the nuclear design requirements are met.

FUEL ELEMENT DEVELOPMENT IN NRU (PHASE 1)

No development is specified at this time provided a test program is underway as described in Section 3.3.1 for MOX 37-element bundles.

FUEL ELEMENT QUALIFICATION IN NRU (PHASE 2)

For the MOX 43-element qualification, tests are required as described in Section 3.3.2. Tests should start by Jan 02.

3.4.2 FUEL BUNDLE QUALIFICATION

The MOX 43-element bundle represents two major changes from the reference NU 37-element bundle: a change in fuel material and a change in bundle design. In addition to the in-reactor (NRU and ZED-2) tests and CHF tests that would be required, out-reactor tests are needed to address the design requirements for the heat transport system, fuel channel and fuel handling systems.

NRU PROTOTYPE BUNDLE QUALIFICATION (PHASE 3)

A repeat of tests described in Section 3.3.2 for MOX 37-element fuel would be needed. Tests should start by mid 02.

ZED-2 TESTS

A repeat of tests described in Section 3.3.2 for MOX 37-element fuel would be needed. Tests should start by Jan 01.

CHF TESTS IN FREON

A repeat of tests described in Section 3.3.2 for MOX 37-element fuel would be needed. Tests should start by mid 00.

CHF TESTS IN WATER

A repeat of tests described in Section 3.3.2 for MOX 37-element fuel would be needed. Heater elements should be ordered mid 00.

BUNDLE CHARACTERIZATION MEASUREMENTS

To ensure dimensional compatibility with the fuel channel and fuel handling system, the external dimensions of the CANFLEX bundle needs to be determined. The Metrology Lab at AECL has developed a procedure for characterizing 37- and 43-element bundles as part of the AECL/KAERI program. The characterization provides direct measurements of bundle droop, element sag, element bow, heights and profiles of bearing pads, endplate waviness, endplate diameters, pitch circle diameters, pressure tube-to-bearing pad interactions, etc.

This information will be used during the engineering stage for developing assembly drawings and for the evaluation of bundle deformation that occurs in out-reactor and in-reactor tests. For deformation measurements, bundles need to be characterized before and after testing. Post-test measurements can be done in the Metrology Lab for unirradiated bundles and in the hot cells for irradiated bundles.

Bundle characterization should start mid 98.

SPACER INTERLOCKING MEASUREMENTS

To ensure that bundles do not jam in the channel, new bundles are gauged prior to loading with a C-clamp having a pre-set circumference. Bundles having a circumference equal to or less than the circumference of the C-clamp will not have interlocked spacers, based on previous tests done for 37-element fuel. The basis for this Go-no-Go test needs to be established for the 43-element fuel bundles.

For this test, the circumference of the 43-element bundle will be measured with and without interlocked spacers. These measurements are needed to determine if the C-ring gauge, currently used for inspecting bundles, is appropriate for the new MOX bundle design.

Tests should start mid 98.

ENDURANCE/PRESSURE DROP TESTS

To demonstrate compliance with design requirements [1,7,8,12,19], out-reactor tests are needed. A representative Bruce test fuel channel will be set up at SPEL with prototype and/or production NU CANFLEX Bruce fuel bundles for pressure drop and endurance tests at representative conditions of temperature, pressure and flow rate. The endurance test duration will be 3000 hours and flow rate 32 kg/s. As part of the endurance test program, flow visualization tests would be needed to investigate the vibration behaviour of the Bruce CANFLEX bundle relative to the standard 37-element bundle and to establish the basis for specifying bundle orientations in the endurance test rig. The bundle characterization measurements combined with pressure tube examinations, done before and after the test, will enable us to measure the extent of material loss due to fretting between the fuel bundles and the pressure tube.

These pressure drop measurements across the fuel channel will provide the basis for the input data for the thermalhydraulic codes, used for licensing purposes.

Twelve NU CANFLEX Bruce bundles would be needed for the out-reactor tests. Endurance/pressure drop tests should start Jan 99.

FUEL HANDLING TESTS

To demonstrate compliance with design requirements [10,11,13], a series of tests with NU CANFLEX Bruce bundles are needed to be performed at the fuelling machine test facilities at GE-C in Peterborough, Ontario. The tests will show that the new bundle design is dimensionally compatible with the fuel channel and fuel handling systems, that the coolant hydraulic drag and fuelling machine ram forces do not adversely damage the fuel bundle, and that the bundles are not adversely affected by the coolant during refuelling. Tests should start Jan 98.

POWER REACTOR DEMONSTRATION TESTS

The purpose of the power reactor demonstration test at Bruce A is to provide the final proof that the new MOX fuel element and bundle designs will successfully meet the specific requirements established by Ontario Hydro.

A small quantity of pre-production bundles, built at LANL, are needed as a final check that the new bundle design can be successfully irradiated in the reactor. For the MOX 43-element bundle design, two demonstration tests have been suggested by OH:

a) a 50 NU bundle irradiation, and

b) a 100 MOX bundle irradiation.

The NU bundles for the first test will be made by one of the Canadian fuel manufacturers. The MOX bundles for the second test will be made under the same conditions as specified in Section 3.3.2. Prior to this test, the results of all preceding tests and the irradiation plan must be assessed and reviewed by the utility and the regulator. An irradiation plan will be prepared and approved by the utility. After these bundles have been discharged from the core and inspected in the fuel bays, a small number of selected bundles and/or elements will transported by irradiated shipping flask to AECL for hot cell examinations.

50 bundle tests should start Jan 00

100 bundle tests should start Jan 02

3.4.3 ELEMENTS AND BUNDLES REQUIRED FOR CANFLEX QUALIFICATION PROGRAM

Considering the bundle requirements for each test site, as shown in Table 3.4.3-1, the total numbers of fuel elements and bundles are summarized below:

Fuel Element Fabrication:

4844 MOX fuel elements (to be fabricated in the USA);

1170 Dy doped/ depleted uranium fuel elements (to be fabricated in Canada) ; and

3268 Natural Uranium elements (to be fabricated in Canada).

Fuel Bundle Assembly (at CRL):

0 Demountable bundles;

6 42-element MOX bundles;

136 43-element MOX bundles; and

76 43-element NU bundles.

Table 3.4.3-1
Number of bundle required for 43-MOX Fuel Qualification Program

	No. of BUNDLES	FUEL COMPOSITION IN EACH RING: OUTER	INTER	INNER	CENTRE	Fuel Element requirements from: ZPI or GEC	LANL or CRL
IN-REACTOR TESTS							
PHASE 1	covered by 37 PHASE 1					none	none
PHASE 2	2 BDL	NU or 15% Dy/DU		~4% Pu	none	70 = 2x35 (outer/inter)	14 = 2x7 (inners)
PHASE 3	4 BDL	~1.5 % Pu	~0.8 % Pu	Dy/DU	none	14 = 2x7 (inners)	70 = 2x35 (outer/inter)
ZED-2 TESTS							
43-ELEMENT	36 MOX	2.6% Pu	4.6% Pu	15% Dy/DU		288 = 36x8 (inners/centre)	1260 = 36x35 (outer/inter)
OUT-REACTOR TESTS							
43-El at SPEL	14 NU	NU	NU	NU	NU	602 = 14x43	none
43-El at GEC FH	12 NU	NU	NU	NU	NU	516 = 12x43	none
BRUCE POWER REACTOR DEMONSTRATION							
43-ELEMENT	50 NU	NU	NU	NU	NU	2150 = 50x43	none
	100 MOX	2.6% Pu	4.6% Pu	15% Dy/DU		800 = 100x8	3500 = 100x35

3.4.4 DESIGN EVALUATIONS

The purpose of the design evaluations, listed below, is to show analytically, that specific design requirements are met. In each case, tests are not needed.

1. Pressure Pulsations: An assessment is needed to ensure that the bundle design will not resonate in acoustically active channels. See Requirement [2]
2. Hydrostatic Pressures: An assessment is needed to show that the element design will successfully withstand hydrostatic testing of the HTS. See Requirement [4]
3. F3SP/Endplate Interaction: An assessment is needed to ensure that the bundle design is dimensionally compatible with the ring support of the downstream shield plug. See Requirement [8]

4. Bundle Strength: The strength of the bundle design needs to be evaluated to ensure that it can successfully withstand the fuelling machine ram loads. See Requirement [12]

5. Reactor Aging: An assessment is needed to show that the element-to-coolant heat transfer for the bundle design is acceptable for a radially crept pressure tube. See Requirement [5]

6. Surface Contamination: No assessment needed if done for 37-element bundles.

7. Sliding Wear: An assessment is needed to show that the sliding wear damage on the pressure tube due to refuelling is acceptable based on the results of previous qualification tests on NU 37-element fuel. See Requirement [7]

8. Refuelling Impacts: An assessment is needed to show that the refuelling impacts associated with the MOX bundles are no worse than previously experienced at other reactors or during previous qualification programs. The acceptability of refuelling impacts for MOX 43-element bundles will be based on the results of the impact tests done in AECL/KAERI program for CANFLEX CANDU 6 bundles. See Requirement [12]

9. Bent Tube Gauge: The currently used bent tube gauge used by the fuel manufacturers to ensure that new MOX bundles can pass through the fuel channel. The tube design must be reevaluated because of possible changes in fuel element swelling rates due to MOX. See Requirement [9]

10. Stress Corrosion Cracking Performance: The fuel failure defect thresholds should be evaluated and modified (if needed) for use with MOX fuel. The PIE results from all MOX fuel elements input will provide some indication if this is needed. See Requirement [14]

11. ASSERT Evaluations: The results of the CHF freon tests will be used to validate ASSERT code predictions of CHF performance of MOX fuel bundles. See Requirement [19]

12. Crevice Corrosion: A brief assessment is needed to show that the effect that crevice corrosion which occurs between the pressure tube and the fuel bearing pads is not a concern for MOX CANFLEX bundles. See Requirement [7]

13. Seismic Qualification: A brief assessment is needed to confirm that the tests previously done for 37-element fuel are sufficiently conservative to address the seismic qualification requirements for MOX CANFLEX fuel bundles. See Requirement [20].

3.4.5 DESIGN REVIEWS

Design reviews are needed as described in Section 3.3.5. Conceptual and final design reviews Jan 99 and Mid 03, respectively.

3.4.6 FUEL DESIGN MANUAL

A Fuel Design Manual is needed as described in Section 3.3.6. (issue by mid 04).

3.4.7 SCHEDULE

Table 3.4.7-1 summarizes the schedule for all deliverables to be produced as a part of MOX CANFLEX Fuel qualification.

Table 3.4.7-1
43-MOX Fuel Qualification Schedule

TEST/ ACTIVITY	START DATE	FINISH DATE
DESIGN ENGINEERING	MID-98	MID-01
NRU -PHASE 1 NRU -PHASE 2 NRU -PHASE 3	- JAN-02 MID-02	- MID-04 JAN-05 (end of irradiations)
ZED-2	JAN-01	MID-02
CHF FREON	MID-99	JAN-01
CHF WATER	JAN-00	MID-02
SPEL - ENDURANCE,ETC	MID-98	JAN-00
GEC - FH	JAN-98	JAN-00
BRUCE DEMO NU MOX	 JAN-00 JAN-02	 MID-01 MID-03 (first bundle out)
DESIGN REVIEWS CONCEPTUAL FINAL	 JAN-99 JAN-03	 JAN-00 MID-03
DESIGN MANUAL	MID-03	MID-04

3.5 PRODUCTION BUNDLE VERIFICATION

To verify that the new MOX fuel production facility can build bundles to perform as designed, a 500-1000 bundle verification program will be performed for each MOX fuel bundle design.

3.6 FUEL QUALIFICATION MATRIX

Table 3.6-1 is a matrix of the fuel qualification program requirements for 37-element and CANFLEX MOX bundles.

Table 3.6-1
Bruce A Mox Fuel Bundle Qualification Program Matrix

TASK		CORRESPONDING DESIGN REQUIREMENTS	TASK NEEDED? FOR OPTIONS	
			MOX 37	MOX 43
Test 1	ZED-2	15,17,19,21	YES	YES
Test 2	FREON CHF	19	YES	YES
Test 3	WATER CHF	19	YES	YES
Test 4	NRU ELEMENTS (PARALLEX)	15,16,18	YES	YES (NO if done for 37)
Test 5	NRU BUNDLE	9,14,15,16,18	YES	YES
Test 6	PRESSURE DROP	1,19	NO	YES
Test 7	ENDURANCE	7,8,12	NO	YES
Test 8	CHARACTERIZATION	8,11	NO	YES
Test 9	SPACER INTERLOCK.	10	NO	YES
Test 10	FUEL HANDLING	10,11,13	NO	YES
Test 11	P. REACTOR DEMO	3,4,9,12,13,15,16,17,18,22	YES	YES
Assmt 1	PRESS PULSATIONS	2	NO	YES
Assmt 2	HYDROSTATIC PRESS	4	NO	YES
Assmt 3	F3SP/EP INTERACTION	8	NO	YES
Assmt 4	BUN STRENGTH	12	NO	YES
Assmt 5	REACTOR AGING	5	YES	YES
Assmt 6	SURFACE CONT.	6	YES	YES
Assmt 7	SLIDING WEAR	7	NO	YES
Assmt 8	REF. IMPACTS	12	YES	YES
Assmt 9	BENT TUBE GAUGE	9	NO	YES
Assmt 10	SCC FUEL PERFORMANCE	14	YES	YES
Assmt 11	ASSERT EVALUATIONS	19	YES	YES
Assmt 12	CREVICE CORROSION	7	NO	YES
Assmt 13	SEISMIC	20	NO	YES

4. SAFETY AND LICENSING

The 1994 study for the US Department of Energy of the feasibility of dispositioning excess weapons plutonium as MOX fuel in the Bruce A reactors addressed issues pertaining to the safety and licensing of these reactors. As was demonstrated in that study, it is possible to operate the Bruce A reactors essentially within the existing operating envelope for natural uranium fuel. Necessary information and experimental data to support licensing submissions would be generated in the course of executing the fuel qualification program. Minor modifications to existing shutdown system process trip parameters were identified as the only modifications to plant systems needed to meet existing licensing requirements.

The present study addresses key aspects of safety and licensing issues that have been affected as a consequence of the optimization of the MOX fuel design. These include the effect of the fuel design on thermalhydraulic performance limits of the fuel, the impact of altered reactor kinetics behaviour on accident analysis and the impact of changes in the reactivity worth of control and shutdown system reactivity devices on accident analysis.

4.1 REACTOR THERMALHYDRAULIC PARAMETERS

Limits exist in CANDU reactors on the thermal power generated in the fuel channels. These limits, referred to as critical channel power, are associated with degradation of boiling heat transfer once critical heat flux (CHF) is exceeded. The 1994 study investigated the effect of the MOX fuel design, in particular the enhanced radial flux depression across a bundle, on CHF and critical channel power through analysis using the Canadian subchannel thermalhydraulic computer code, ASSERT. The results of this analysis indicated that there was no significant impact on critical heat flux performance of MOX fuel, relative to the existing 37-element natural fuel.

The current study shows that the peak bundle and channel powers are slightly reduced in magnitude relative to the 1994 MOX reference 37-element fuel design (see section 2.2.3) and that the radial power distribution across the rings of elements in a fuel bundle is not modified significantly, albeit that the central dysprosium oxide loaded elements produce slightly less power than in the previous design. Similarly, for the advanced CANFLEX fuel design, the axial power distribution along a channel and the radial power distribution between the rings of elements is not significantly different from the 1994 design. Based upon these considerations, the somewhat lower bundle powers at downstream positions in a fuel channel, and the analytical results from the previous study, it is concluded that there will be no detrimental impact of MOX fuel on the critical heat flux performance. Consequently, the thermal power limits at which fuel channels may be operated with MOX fuel is expected to be at least the same, and most probably higher than for the existing natural uranium fuel.

4.2 SHUTDOWN SYSTEM EVALUATION

The presence of plutonium in the fuel, the reduced isotopic purity of the moderator and coolant and the resultant changes to the flux distribution in the core affect the reactivity worth of devices in the control and shutdown systems. These changes in reactivity worth of devices, together with the reduction in the delayed neutron fraction and prompt generation time, can potentially influence the effectiveness of the shutdown function provided by the independent shutdown systems. The impact of the optimized MOX fuel design on shutdown system effectiveness is assessed below.

4.2.1 Small LOCA

In the event of a small break LOCA occurring in heat transport system piping outside the reactor core, the reactivity worth available in both of the shutdown systems, SDS1 and SDS2, is adequate to ensure an effective shutdown. These events are not limiting with respect to either shutdown system speed or depth.

A failure of a fuel channel resulting in discharge of coolant into the moderator, referred to as an in-core break, does, for certain conditions, define the limit on depth of the shutdown systems. In particular, for SDS1 an in-core LOCA can potentially disable some of the SDS1 shutoff rod devices as a consequence of in-core mechanical damage associated with the channel failure. Furthermore, should this failure occur shortly after reactor startup following a long outage when there may be higher levels of neutron poison in the moderator, then it is possible that additional positive reactivity can be introduced by displacing poisoned moderator fluid with unpoisoned coolant. This limiting event has been assessed using the reactivity device worths obtained from the nuclear analysis documented in section 2.2, similar assumptions to those used in existing Bruce A safety report analysis regarding shutoff rod damage, and appropriate operating limits on moderator poison concentration. These assumptions are described below and are summarized in Table 4.2.1-1.

Analysis Assumptions

- The limiting case analyzed to evaluate the reactivity depth of Shutdown System No. 1 (SDS1) is an in-core LOCA with 225 kg/s initial break discharge flowrate and assumed failure of the moderator level based ECIS conditioning logic, as described in the Bruce A Safety Report. A reactor trip initiating shutdown by SDS1 is credited at 247 seconds following the channel failure.
- It is assumed that, prior to the failure event, the reactor is immediately brought to full power following a long shutdown, such that there are high levels of poison in the moderator to compensate for the decay of fission product poisons. This is an extremely conservative assumption, because, in reality, poison would be removed from the moderator as reactor power is increased, resulting in significantly lower poison concentrations once full power is attained. Nevertheless, consistency with existing safety analysis assumptions is maintained.
- A total of 30 mk of moderator poison reactivity holdup is assumed. This includes 20 mk of poison to compensate for Xenon decay, 6.7 mk to compensate for decay of other fission product poisons, and 3.3 mk which is a fuel-ahead allowance. The poison reactivity for fuelling ahead is based upon providing the ability to operate the reactor without fuelling for the same time period as assumed for the natural uranium core (8 mk is assumed in existing safety analysis) and adjusting for the lower fuelling rate and reactivity change per channel refuelled in the MOX core. Note that assuming the additional poison in the moderator for fuelling ahead is conservative because this level of poison could be provided following startup of the reactor by earlier termination of poison removal.
- The group of shutoff rods assumed to be damaged as a consequence of the pressure tube/calandria tube failure are the same as the damage assumed in the existing safety analysis. However, it should be noted that for the current design

with lower moderator isotopic purity, there is a smaller relative change in reactivity worth with the number of rods assumed to be unavailable. The SDS1 depth credited in this analysis is -8.6 mk.

- The reactor regulating system is assumed to have zone controller average level at 70% initially and zone controller reactivity between 70% and 95% level is credited (0.7 mk), plus 5.5 mk for the four mechanical control absorber rods. This gives a total of -6.2 mk available from the reactor regulating system.
- Moderator temperature feedback reactivity is assumed to be the same as for natural uranium fuel, while only half of the moderator downgrading reactivity effect is assumed in this analysis. The moderator downgrading is based upon assuming that the reactor will be operated with smaller differences between the coolant and moderator isotopic purities. This is a conservative assumption since the nuclear analysis has demonstrated that coolant purity has a small effect on reactivities relative to the moderator purity, and therefore limits on purity difference similar to those existing could be employed.

The results for the limiting case of a 225 kg/s in-core break, with the assumptions regarding reactivity worths of credited devices and operating conditions at the time of the failure, identified in Table 4.2.1-1, are summarized in Table 4.2.1-2, together with corresponding results for natural uranium fuel. The results of similar analysis for CANFLEX MOX fuel are summarized in Table 4.2.1-3. The results indicate that sufficient subcriticality margin exists during the first 15 minutes of this event to assure that the reactor remain subcritical and allow operator action to further increase subcriticality. In fact, despite the reduction in the reactivity worth of control and shutdown system reactivity devices in the MOX core, the shutdown depth in the MOX core is higher than in the natural uranium core. Note that once the ECIS system is initiated, or alternatively poison addition to the moderator occurs, a very large increase in subcriticality occurs - either due to light water downgrading of the moderator heavy water, or because of the high poison concentrations that develop in the moderator.

Table 4.2.1-1
Assumptions employed in assessment of SDS1 Reactivity Depth following an In-Core LOCA - Limiting safety report case

CURRENT MOX 37-ELEMENT DESIGN	NATURAL URANIUM	COMMENTS
Break discharge= 225kg/s Reactor Power= 103% FP Failure of ECIS Conditioning Logic	Break discharge= 225kg/s Reactor Power= 103% FP Failure of ECIS Conditioning Logic	Limiting case from Bruce A Safety Report
Moderator poison=30 mk 20 mk Xenon 6.7 mk Fission products 3.3 mk fuel-ahead	Moderator poison= 50.4 mk 28 mk Xenon 14.4 mk Fission products 8 mk fuel-ahead	Existing analysis overestimates fission product reactivity. Fuelling ahead reactivity conservatively assumed in analysis
Full core coolant void = -5 mk Fuel temperature = -2.9 μk/°C	Full core coolant void = 11.4 mk Fuel temperature = -6.0 μk/°C	
Reactor regulating system=-6.2mk Zone controllers = -0.7 mk 4 MCA's inserted = -5.5 mk	Reactor regulating system=-9.1mk Zone controllers = -1.2 mk 4 MCA's inserted = -7.9 mk	Zones assumed to operating at high level to minimize negative reactivity available
SDS1 rods outside damage zone= -8.6mk	SDS1 rods outside damage zone= -12.5mk	Relative reduction in reactivity of shutoff rods not as large in MOX core.

Table 4.2.1-2
Assumptions for Assessment of SDS1 Depth following an Incore LOCA

DEPTH OF SDS1 FOLLOWING AN IN-CORE LOCA WITH FAILURE OF ECIS CONDITIONING SIGNAL
[225 kg/s Initial Break Discharge, Reactor trip @ 247 s, High Moderator poison]

[redacted]

Moderator poison	50.4 mk
Coolant Void	11.4 mk
Fuel Temp	6 mk
Xenon	-28 mk
RRS	-9.1 mk
SDS - Safety Report damage zone	-12.5 mk

TIME (s)	0	40	80	100	140	180	200	220	247	300	400	500	600	700	800	900	925
Moderator poison displacement	0.0	1.3	2.6	3.3	4.5	5.6	6.2	6.9	7.6	8.6	10.3	12.0	13.5	15.0	16.1	16.9	17.1
Coolant void	0.0	0.0	0.0	0.1	0.3	0.6	0.7	0.9	1.0	0.9	1.8	2.3	3.1	3.8	4.6	4.9	5.0
Fuel Temp	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	3.6	3.8	3.8	3.8	3.8	3.8	3.8	3.8
Moderator Temperature	0.0	0.9	1.8	1.9	3.0	3.2	3.4	3.5	3.6	3.3	2.2	1.3	0.6	0.0	-0.7	-1.5	-1.7
Degrading Moderator	0.0	-0.5	-1.0	-1.3	-1.8	-2.2	-2.5	-2.7	-3.0	-3.4	-4.1	-4.7	-5.4	-5.9	-6.4	-6.7	-6.8
Xenon	0.0	0.0	-0.1	-0.1	-0.1	-0.2	-0.2	-0.2	-0.3	-0.3	-0.3	-0.3	-0.3	-0.3	-0.3	-0.3	-0.3
Excess	0.0	1.7	3.3	3.9	5.9	7.0	7.6	8.5	9.0	12.7	13.7	14.4	15.3	16.4	17.1	17.1	17.1
RRS	0.0	-1.7	-3.3	-3.9	-5.9	-7.0	-7.6	-8.5	-9.1	-9.1	-9.1	-9.1	-9.1	-9.1	-9.1	-9.1	-9.1
Total	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-0.1	3.6	4.6	5.3	6.2	7.3	8.0	8.0	8.0
SDS	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-12.5	-12.5	-12.5	-12.5	-12.5	-12.5	-12.5	-12.5	-12.5
Net	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-12.6	-8.9	-7.9	-7.2	-6.3	-5.2	-4.5	-4.5	-4.5

[redacted]

Moderator poison	29.73 mk	
Coolant Void	-5 mk	
Fuel Temp	2.9 mk	
Xenon	-20 mk	
RRS	-6.2 mk	
SDS - Safety Report damage zone	-8.6 mk	
Moderator Temperature Factor	1	Relative to Natural
Moderator-Coolant Isotopic Difference	0.5	Uranium Core

TIME (s)	0	40	80	100	140	180	200	220	247	300	400	500	600	700	800	900	925
Moderator poison displacement	0.0	0.8	1.5	1.9	2.7	3.3	3.7	4.1	4.5	5.1	6.1	7.1	8.0	8.8	9.5	10.0	10.1
Coolant void	0.0	0.0	0.0	0.0	-0.1	-0.3	-0.3	-0.4	-0.4	-0.4	-0.8	-1.0	-1.4	-1.7	-2.0	-2.1	-2.2
Fuel Temp	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.7	1.8	1.8	1.8	1.8	1.8	1.8	1.8
Moderator Temperature	0.0	0.9	1.8	1.9	3.0	3.2	3.4	3.5	3.6	3.3	2.2	1.3	0.6	0.0	-0.7	-1.5	-1.7
Degrading Moderator	0.0	-0.3	-0.5	-0.7	-0.9	-1.1	-1.3	-1.4	-1.5	-1.7	-2.1	-2.4	-2.7	-3.0	-3.2	-3.4	-3.4
Xenon	0.0	0.0	-0.1	-0.1	-0.1	-0.1	-0.1	-0.1	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2
Excess	0.0	1.4	2.8	3.1	4.6	5.0	5.4	5.7	6.0	7.8	7.1	6.6	6.1	5.9	5.2	4.6	4.4
RRS	0.0	-1.4	-2.8	-3.1	-4.6	-5.0	-5.4	-5.7	-6.2	-6.2	-6.2	-6.2	-6.2	-6.2	-6.2	-6.2	-6.2
Total	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-0.2	1.6	0.9	0.4	-0.1	-0.3	-1.0	-1.6	-1.8
SDS	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-8.6	-8.6	-8.6	-8.6	-8.6	-8.6	-8.6	-8.6	-8.6
Net	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-8.8	-7.0	-7.7	-8.2	-8.7	-8.9	-9.6	-10.2	-10.4

DEPTH OF SDS1 FOLLOWING AN IN-CORE LOCA WITH FAILURE OF ECIS CONDITIONING SIGNAL
[225 kg/s Initial Break Discharge, Reactor trip @ 247 s, High Moderator poison]

NATURAL U CORE

Moderator poison	50.4 mk
Coolant Void	11.4 mk
Fuel Temp	6 uk/C
Xenon	-28 mk
RRS	-9.1 mk
SDS - Safety Report damage zone	-12.5 mk

TIME (s)	0	40	80	100	140	180	200	220	247	300	400	500	600	700	800	900	925
Moderator poison displacement	0.0	1.3	2.6	3.3	4.5	5.6	6.2	6.9	7.6	8.6	10.3	12.0	13.5	15.0	16.1	16.9	17.1
Coolant void	0.0	0.0	0.0	0.1	0.3	0.6	0.7	0.9	1.0	0.9	1.8	2.3	3.1	3.8	4.6	4.9	5.0
Fuel Temp	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	3.6	3.8	3.8	3.8	3.8	3.8	3.8	3.8
Moderator Temperature	0.0	0.9	1.8	1.9	3.0	3.2	3.4	3.5	3.6	3.3	2.2	1.3	0.6	0.0	-0.7	-1.5	-1.7
Degrading Moderator	0.0	-0.5	-1.0	-1.3	-1.8	-2.2	-2.5	-2.7	-3.0	-3.4	-4.1	-4.7	-5.4	-5.9	-6.4	-6.7	-6.8
Xenon	0.0	0.0	-0.1	-0.1	-0.1	-0.2	-0.2	-0.2	-0.3	-0.3	-0.3	-0.3	-0.3	-0.3	-0.3	-0.3	-0.3
Excess	0.0	1.7	3.3	3.9	5.9	7.0	7.6	8.5	9.0	12.7	13.7	14.4	15.3	16.4	17.1	17.1	17.1
RRS	0.0	-1.7	-3.3	-3.9	-5.9	-7.0	-7.6	-8.5	-9.1	-9.1	-9.1	-9.1	-9.1	-9.1	-9.1	-9.1	-9.1
Total	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-0.1	3.6	4.6	5.3	6.2	7.3	8.0	8.0	8.0
SDS	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-12.5	-12.5	-12.5	-12.5	-12.5	-12.5	-12.5	-12.5	-12.5
Net	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	-12.6	-8.9	-7.9	-7.2	-6.3	-5.2	-4.5	-4.5	-4.5

MOX CORE - CANFLEX FUEL

Moderator poison	28.41 mk	
Coolant Void	-4.5 mk	
Fuel Temp	2 uk/C	
Xenon	-19 mk	
RRS	-3.48 mk	
SDS - Safety Report damage zone	-7.75 mk	
Moderator Temperature Factor	1	Relative to Natural
Moderator-Coolant Isotopic Difference	0.5	Uranium Core

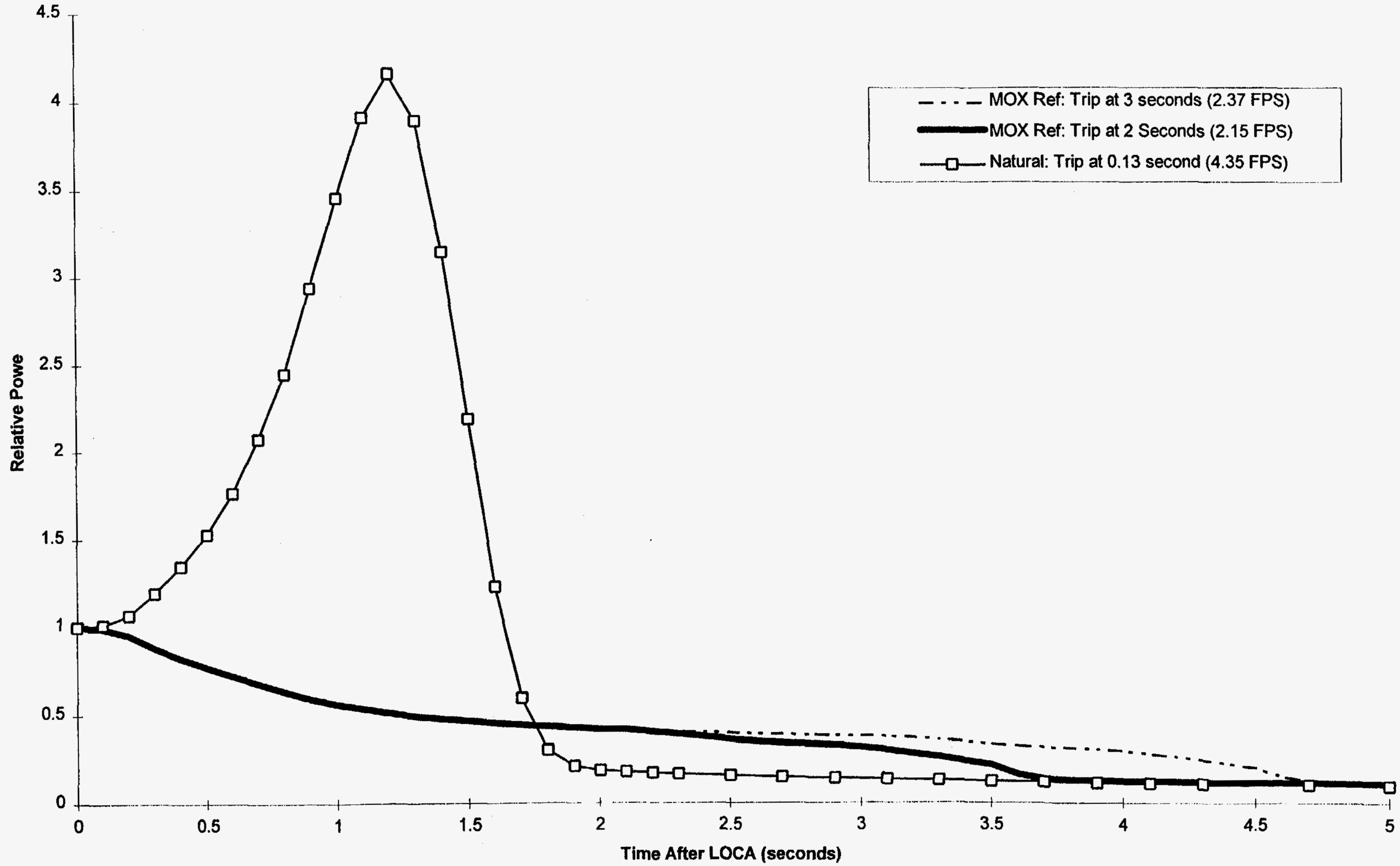
TIME (s)	0	40	80	100	140	180	200	220	247	300	400	500	600	700	800	900	925
Moderator poison displacement	0.0	0.7	1.5	1.9	2.5	3.2	3.5	3.9	4.3	4.8	5.8	6.8	7.6	8.5	9.1	9.5	9.6
Coolant void	0.0	0.0	0.0	0.0	-0.1	-0.2	-0.3	-0.4	-0.4	-0.4	-0.7	-0.9	-1.2	-1.5	-1.8	-1.9	-2.0
Fuel Temp	0.0	0.0	0.0	0.0	1.2	1.3	1.3	1.3	1.3	1.3	1.3	1.3	1.3	1.3	1.3	1.3	1.3
Moderator Temperature	0.0	0.9	1.8	1.9	2.0	1.5	1.0	0.5	0.4	0.0	-0.9	-1.3	-2.2	-2.7	-3.2	-3.6	-4.0
Degrading Moderator	0.0	-0.3	-0.5	-0.7	-0.9	-1.1	-1.3	-1.4	-1.5	-1.7	-2.1	-2.4	-2.7	-3.0	-3.2	-3.4	-3.4
Xenon	0.0	0.0	-0.1	-0.1	-0.1	-0.1	-0.1	-0.1	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2	-0.2
Excess	0.0	1.4	2.7	3.0	4.7	4.5	4.1	3.8	3.9	3.9	3.2	3.3	2.5	2.4	1.9	1.7	1.3
RRS	0.0	-1.4	-2.7	-3.0	-3.5	-3.5	-3.5	-3.5	-3.5	-3.5	-3.5	-3.5	-3.5	-3.5	-3.5	-3.5	-3.5
Total	0.0	0.0	0.0	0.0	1.2	1.0	0.7	0.4	0.4	0.4	-0.3	-0.2	-0.9	-1.1	-1.6	-1.8	-2.2
SDS	0.0	0.0	0.0	0.0	-7.7	-7.7	-7.7	-7.7	-7.7	-7.7	-7.7	-7.7	-7.7	-7.7	-7.7	-7.7	-7.7
Net	0.0	0.0	0.0	0.0	-6.6	-6.7	-7.1	-7.4	-7.3	-7.3	-8.0	-8.0	-8.7	-8.9	-9.3	-9.5	-9.9

4.2.2 Large LOCA

As for the MOX fuel design of the 1994 study, the current 37-element optimized design has negative coolant void reactivity feedback. The full core void reactivity for the current fuel design is -5 mk. Consequently, following a large break LOCA no power excursion will occur, as is the case for operation with natural uranium fuel. The reactor power transient following a large LOCA is shown in Figure 4.2.2-1, together with the transient for the same event occurring in a natural uranium fuelled reactor. In the case of the MOX fuelled reactor, shutdown system action is initiated by process trip parameters that occur later in time than the neutronic trip parameters that are credited in accident analysis for the natural uranium fuelled reactor.

The large LOCA power transient for the optimized MOX fuel design is essentially identical to the results obtained in the 1994 study.

Figure 4.2.2-1
Large LOCA Power transients in MOX and natural-U cores



4.2.3 Loss of Regulation and Loss of Reactivity Control

The 1994 study demonstrated that, although there were changes in the neutron kinetics parameters in a MOX core relative to a natural uranium core, the impact of these changes did not significantly influence either the transient reactor behaviour or the shutdown system effectiveness during a postulated loss of reactivity control accident in which positive reactivity is inserted. A similar conclusion has been obtained in the current study. This is shown in Figures 4.2.3-1 and 4.2.3-2 where reactor neutronic power transients are plotted for a loss of reactivity control events involving: a) simultaneous draining of all zone controllers and b) a parametric reactivity insertion rate of 0.115 mK/S. These results demonstrate that these are no significant differences in the transient response of the reactor nor in the shutdown system effectiveness for MOX fuelled cores as compared with natural uranium fuelled cores.

Figure 4.2.3-1
Loss of reactivity control transients (Natural U and MOX Cores)

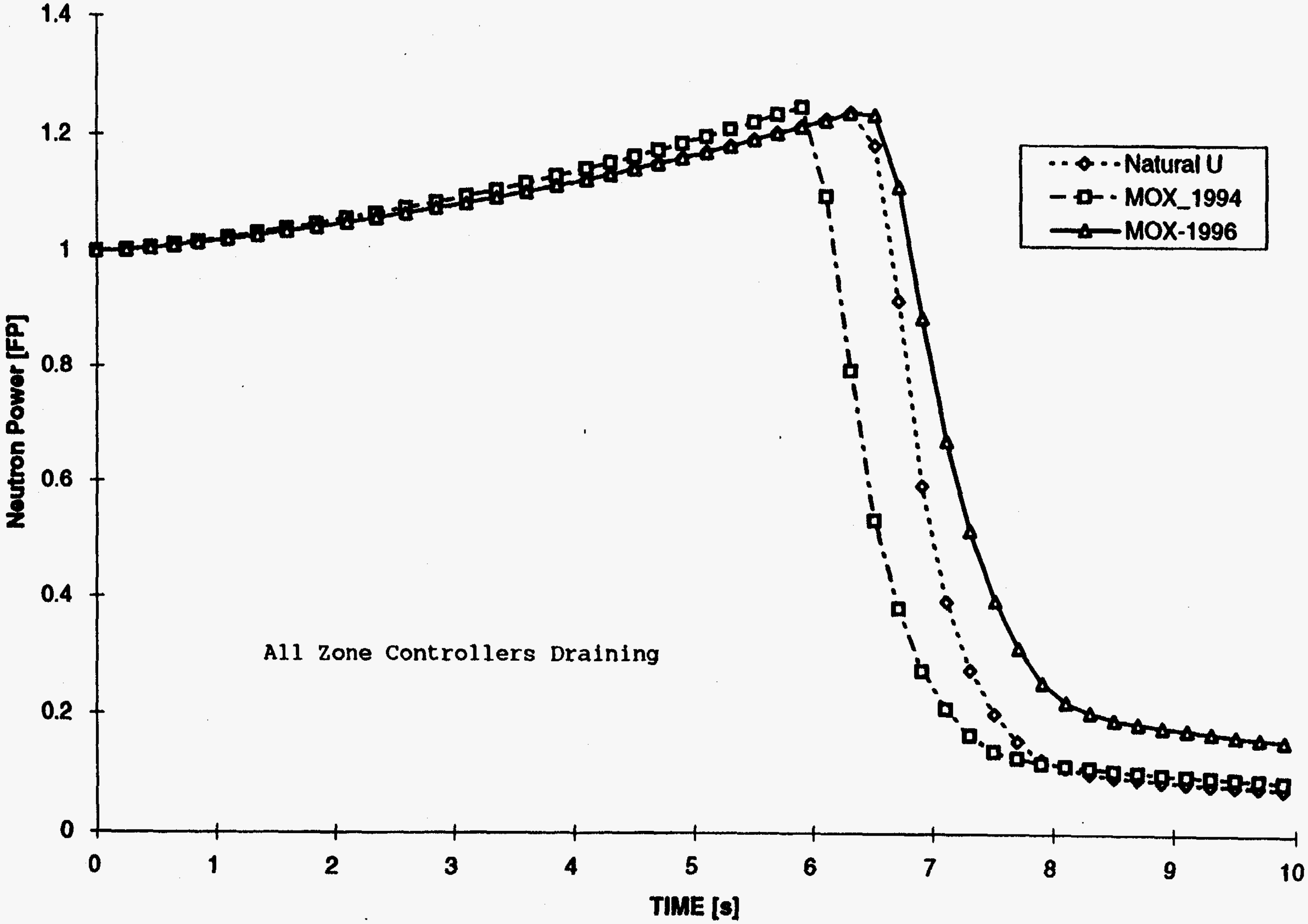
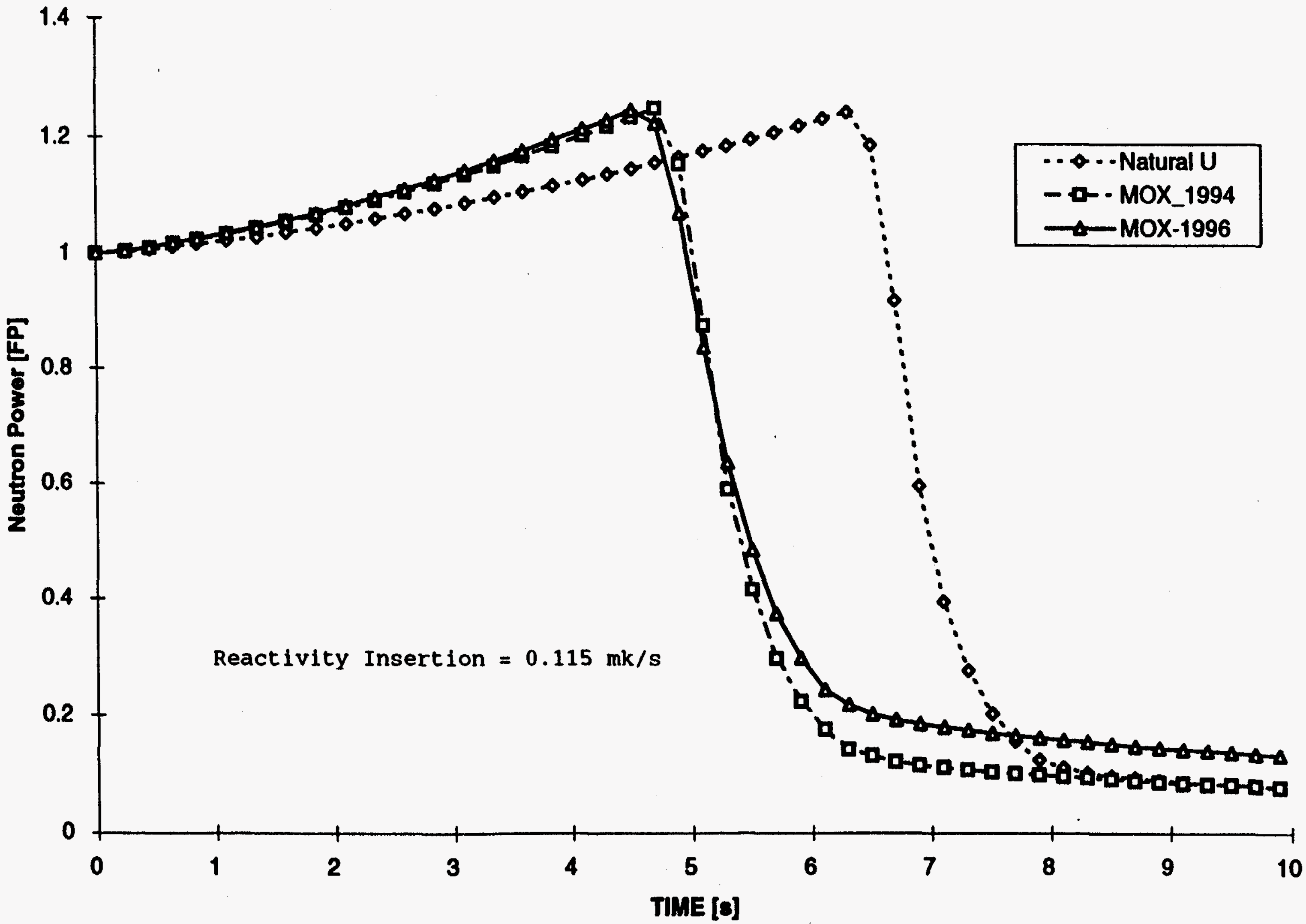


Figure 4.2.3-2
Loss of reactivity control transients (Natural U and MOX Cores)



4.2.4 Overall Shutdown System Assessment

The safety analysis and assessment performed for the optimized 37-element MOX fuel design has demonstrated the feasibility of operating Bruce A reactors within existing safety and licensing limits. Although the safety analysis has not been as extensive as would be required for obtaining approvals to operate with MOX fuel, it has nevertheless focused on identifying limiting events for which potential major modifications to reactor special safety systems may be required. No such limiting events were identified and no major modifications to safety systems are necessary.

4.3 CRITICALITY SAFETY

The content of plutonium in the 37 element and CANFLEX fuel designs has been increased relative to the 1994 design. However, in part because of limited scope and in part because of the results obtained in the previous 1994 study, no new criticality analysis has been performed. Since the amount of dysprosium in the centre elements has been increased from 5% to 15% to offset the increased plutonium fissile content, the optimized fuel is not significantly more reactive than in the previous study. Therefore, similar results to those obtained in criticality analyses performed for the 1994 study are expected to apply to the optimized design. This, again, would indicate that no criticality concerns exist for the optimized fuel design.

5. IMPACT ON OPERATIONS

The impact on operations at Bruce A has been assessed in this study. The aspects of operations that have been examined include:

- operation with reduced moderator and coolant isotopic purity and impact on heavy water management at the station,
- the requirements on receipt and storage of new fuel,
- the transition of the reactor cores from natural uranium fuel to MOX fuel, and
- the effects of MOX spent fuel on the waste fuel repository.

The findings of this study are reported below.

5.1 MODIFICATIONS TO OPERATING CONDITIONS

One consequence of increased plutonium loading in the two outer rings of the CANDU- MOX fuel bundle is a reduction in the isotopic purity of the moderator and coolant to 97%. This purity level is lower than currently employed in the natural uranium fuelled Bruce A reactors. However, operating experience with varying levels of moderator and coolant isotopic purity, within specified upper and lower limits, does exist. Additionally, heavy water upgrader units are operational at Bruce A, as at the other Ontario Hydro nuclear generating stations.

An assessment of operation with lowered moderator and coolant isotopic purity has been conducted by technical staff at Bruce A and no major impediment to such operation has been identified. Depending upon the number of units operating with MOX fuel and the number operating with natural uranium fuel, it may be advantageous, for reduced complexity of operation reasons, to install a new heavy water storage tank to provide physical isolation of heavy water inventories of different isotopic purity. However, this potential modification is considered to be a relatively simple and low cost modification.

5.2 RECEIPT AND STORAGE OF NEW FUEL

The 1994 study proposed that a new fuel receiving area and a hardened, secure storage facility be located adjacent to the central services area of the station. In addition, the need to harden existing new fuel handling areas was identified. The design and assumptions associated with these modifications have been assessed by Bruce A technical staff and Ontario Hydro Nuclear - Architectural staff.

A one month's supply of new fuel for two reactor units stored at site was assumed in the 1994 study. Subsequently Ontario Hydro has established that, on the basis of security of supply considerations, a three month's supply of fuel for two units was desireable and, therefore, this requirement served as a reference assumption for the current assessment. In addition the following pre-requisites were considered essential:

1. Bruce A must prove that it has the physical capability to store the requisite amount of MOX fuel in existing or newly constructed facilities, and
2. Bruce A must prove that it can modify physical security requirements to continue meeting the criteria established by the International Atomic Energy Agency (IAEA),

and which will also be acceptable to the Atomic Energy Control Board (AECB), Canada's nuclear regulatory body.

The assessment of increased new fuel storage requirements involved evaluating the feasibility of a) "in-plant" storage, b) constructing a larger building adjacent to the central services area, and c) constructing a storage facility detached from the station powerhouse. In performing these assessments Ontario Hydro Nuclear - Architectural staff utilized their experience with other high level physical security design issues at Bruce A.

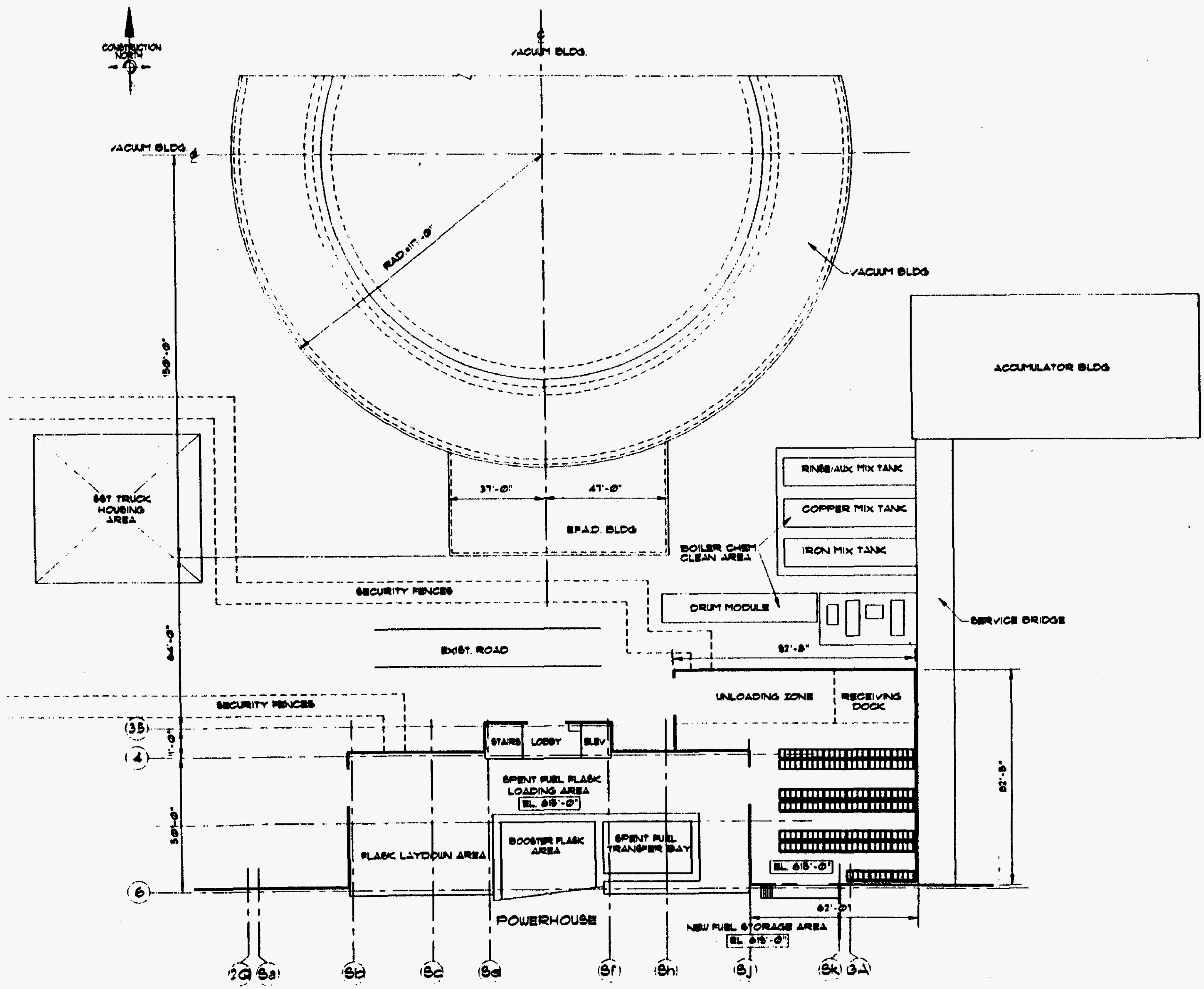
These assessments indicated that;

- "in-plant" storage of the required quantity of MOX fuel is not possible due to the unavailability of the required floor area.
- a storage facility adjacent to the central services area, at the location identified in the 1994 study, can be expanded to meet the three month MOX fuel storage requirement.
- a detached storage facility will not be pursued as an option at the present time.
- future design requirements for detection and delay systems for hardening existing and new facilities will build upon, and be consistent with established existing Bruce A developmental procedures.

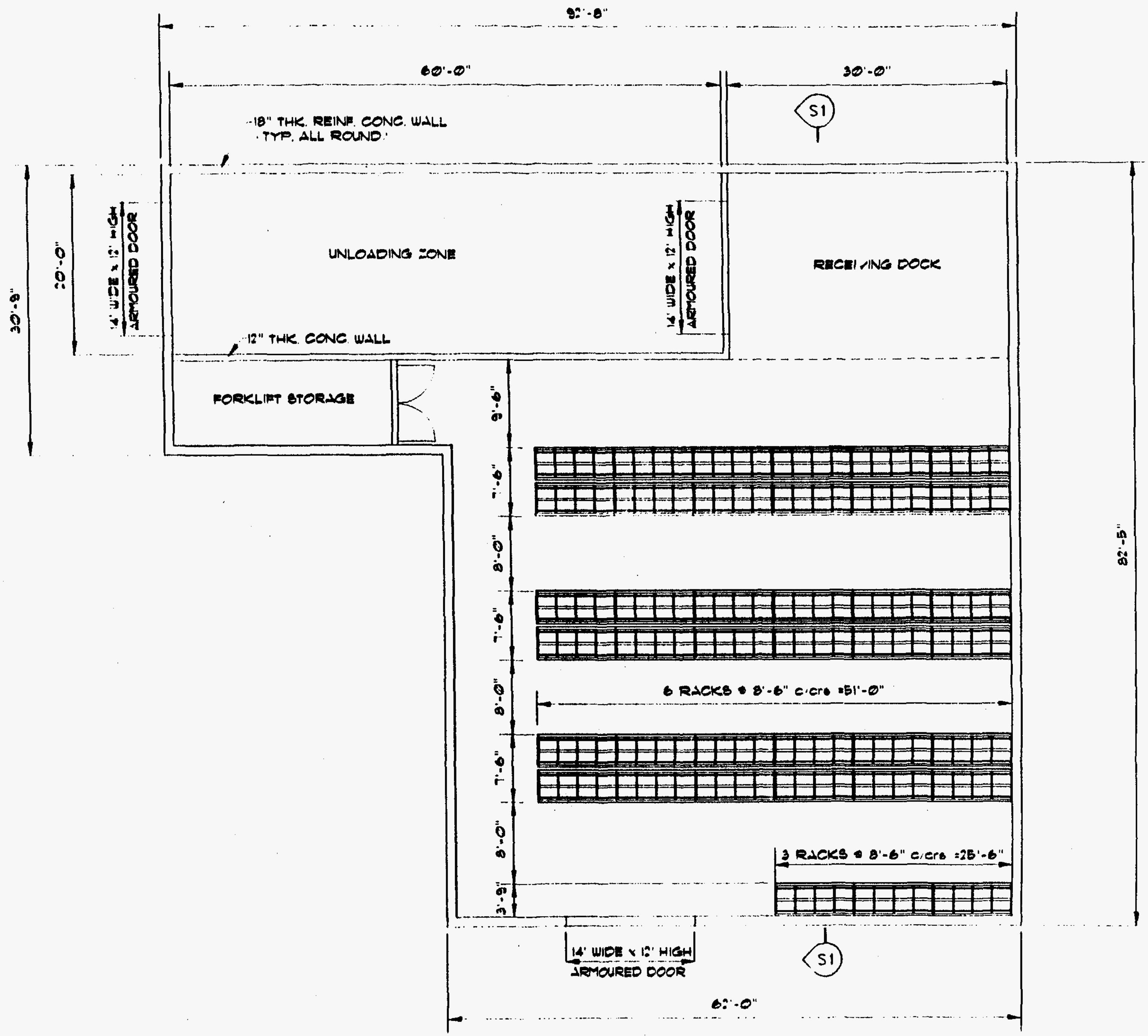
The conceptual layout of the storage facility is shown in Figures 5.2-1, 5.2-2. and 5.2-3. Figure 5.2-1 shows the site plan for the new MOX fuel storage building at the proposed location between the central services area of the powerhouse and the vacuum building. A floor plan and a section view of the storage building are shown in figures 5.2-2 and 5.2-3, respectively.

In addition to confirming the fundamental assumptions made in the 1994 study regarding the new MOX fuel storage building concept, the Ontario Hydro study also confirmed the reasonableness of the cost estimate for this facility. As a result, no changes to the cost estimates provided in the 1994 report are required.

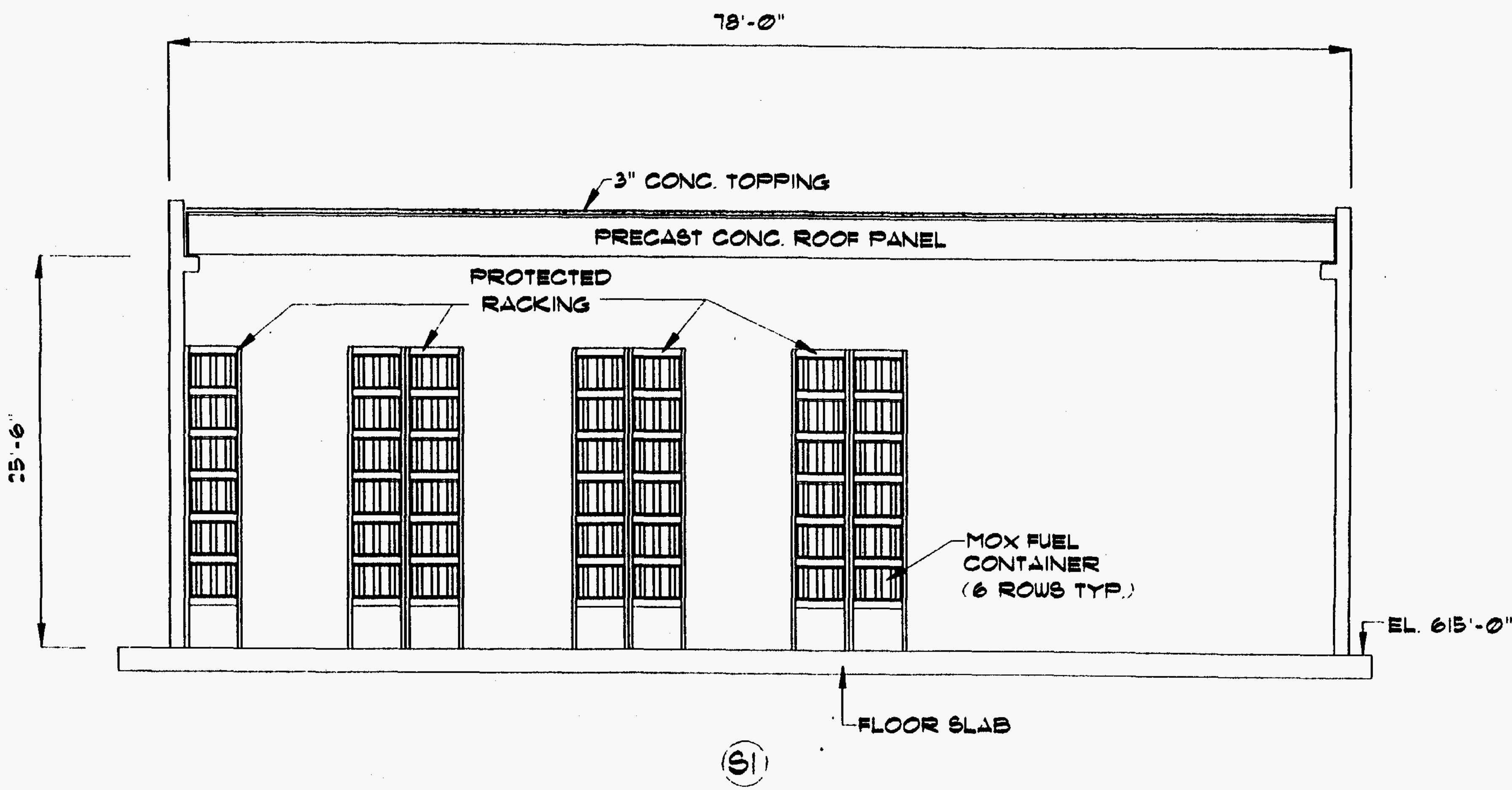
Proposed New MOX Fuel Storage Building - Site Plan



Proposed New MOX Fuel Storage Building - Floor Plan



Proposed New MOX Fuel Storage Building - Section



5.3 TRANSITION FROM NATURAL TO MOX CORES

The approach to transitioning from a natural uranium fuelled core to a MOX fuelled core has been considered. The intent is to treat the first reactor to be transitioned to MOX fuel in an analogous manner to a new reactor core. This would result in the first reactor being fully defuelled of natural uranium fuel bundles and a full core of MOX bundles, with depleted uranium bundles loaded in selected locations in a significant number of fuel channels. This is essentially identical to the manner in which a new reactor is started up with fresh natural uranium fuel. The depleted uranium bundles are used to both reduce the reactivity of the fresh fuel and to provide power shaping for the fresh core. During startup of the new MOX core, a series of reactor physics commissioning tests would be performed to verify the calculated worth of individual and grouped reactivity devices and to confirm the dynamic reactivity characteristics of the shutdown systems. A detailed reactor physics study will be required to establish the configuration of the initial MOX core. This is beyond the scope of the current study.

In subsequent reactors, the transition from natural uranium fuel to MOX fuel will be performed through the process of on-line refuelling. This will result in the gradual replacement of irradiated natural uranium fuel bundles with MOX fuel bundles and will occur over a period of approximately 18 to 24 months. Associated with this gradual transition will be the controlled reduction of moderator and coolant isotopic purity from current purity levels to the target 97% endpoint. It is anticipated that the steps in isotopic purity reduction will be established by detailed analysis, yet to be performed.

5.4 EFFECT OF MOX SPENT FUEL ON WASTE REPOSITORY

In AECLT's 1994 report, it was noted that the spent fuel generated from MOX fuel would be reduced by about 10% as compared with the spent fuel generated from natural uranium. However, the 1994 study did not address the possible effects of this spent MOX fuel, as compared to spent natural uranium fuel, on the design and operation of the Canadian Waste Repository. In the current study, AECL and Ontario Hydro examined the potential implications of this spent CANDU MOX fuel on the planned CANDU fuel waste repository. Results are summarized below:

Although fission product inventories for the MOX and natural uranium CANDU fuels would be similar, actinide concentrations at discharge are significantly higher in MOX fuels, due to the high initial Pu enrichment. As a result of these high actinide inventories, a preliminary analysis of the disposability of used MOX fuel was carried out using the disposal concept proposed for natural CANDU fuel. This analysis included the potential for criticality, implications of a higher decay heat within a MOX fuel container, and the potential for enhanced dissolution of the fuel as a result of alpha-radiolysis of water.

Assuming the standard 37-element MOX fuel was used for the dispositioning of 50 Metric tonnes of Pu, about 150,000 fuel bundles would be generated, requiring about 2000 disposal containers, each containing 72 fuel bundles. It is projected that a disposal vault would contain about 140,000 CANDU fuel containers, so the vault volume required for the MOX bundles and containers would represent about 1% of the total vault volume, if no account is taken of the added heat generation from the MOX fuel.

5.4.1 Potential for Criticality in Repository

A number of potential criticality scenarios were considered which included both flooding of a MOX fuel bundle container with groundwater and complete dissolution of the Pu inventory in the fuel and the redistribution within the container volume. There were no cases where the neutron multiplication factor, k_{eff}, exceeded 0.95 for a flooded container with intact fuel bundles. Also, since the k_{eff}'s were sufficiently below 0.95 there was no necessity to include a neutron poison into the glass beads filling the void space in the container. Calculations have not been performed for 43-element CANFLEX MOX fuel bundles with a higher Pu content.

Various scenarios were also examined involving preferential dissolution and diffusion of U and Zr out of the container leaving Pu distributed within the container, or preferential redistribution of Pu into geometries favourable to achieving criticality. Most scenarios studied were intended to examine the conditions under which redistribution of Pu would produce a K_{eff} greater than 0.95. The scenarios where this occured were found to be highly improbable. Most scenarios indicated that criticality would not occur if a neutron poison such as boron were incorporated into the glass beads filling the void space in the container. Alternatively, if only a few MOX fuel bundles were co-disposed with natural CANDU fuel bundles in a container, the total ^{239}Pu content per container would be reduced, substantially lowering the potential for criticality.

Only a very preliminary examination of the criticality safety of the storage of MOX fuel bundles has been possible in this report. The most obvious future requirement is to critically evaluate the potential scenarios which have been examined to date. A more thorough examination of the physico-chemical processes within a failed container would be required. Also, only standard Bruce bundles have been studied. The 43-element CANFLEX bundle geometry should also be analyzed, as should a range of possible MOX fuel bundle compositions. Survey calculations of the criticality impacts of variations in glass porosities, boron loading of the glass, the possible use of other neutron poisons in the glass other than boron and the oxidation of the container liner from carbon steel into Fe_3O_4 should all be undertaken before final conclusions about the criticality potential are reached.

5.4.2 Effects of Decay Heat on Repository Design and Operation

The decay heat from the reference MOX fuel was shown to be about three times greater than that from natural CANDU fuel. This level of heat output would be sustained for a period of about 10,000 years before it dropped below the maximum heat output from natural uranium CANDU fuel at 100 years. This would result in a prolonged period of high temperature at the container surface and in the clay buffer material. This may require a modification of the vault loading arrangements to limit the surface temperature of the MOX container. This could be accomplished in several ways. The heat output per container with MOX fuel could be decreased by co-disposing of 2 to 3 MOX fuel bundles in a natural uranium CANDU fuel bundle container. This would result in a 5 to 10% increase in the heat output per CANDU fuel container but the additional heat load to the vault could be accommodated by a small increase in the spacing of the containers. This small increase in heat would be unlikely to have a significant impact on container lifetimes. Alternatively, the MOX fuel containers could be evenly dispersed throughout the CANDU fuel disposal vault, again taking into consideration the spacing required to maintain thermal constraints in any vault location. A quantitative estimate of the additional vault volume required for these scenarios would require thermal analysis of vault temperatures as a function of container spacing in the vault, similar to that carried out for the natural uranium CANDU fuel disposal concept. It seems likely that such an analysis would indicate that some increase in vault volume would be required, resulting in some increase in repository cost. It is uncertain as to whether this cost increase would exceed the cost reduction that would accrue from the lower quantities of MOX spent fuel bundles, as compared to natural uranium spent fuel bundles resulting from the plutonium disposition program. Further analyses are required to determine the net effect.

The impact of decay heat from MOX fuel on the vault environment has only been examined on the basis of calculated decay heat per bundle compared to natural CANDU fuel. Similar comparisons should be made using calculated inventories from the 43-element CANFLEX bundle. In addition, a complete thermal analysis of the temperature of the vault should be considered using the scenarios of MOX fuel disposal in a dedicated sector as well as scenarios involving co-disposal with natural CANDU fuel. These cases would give a better indication of the thermal impact on the vault and the potential increase in vault size that would be required to accommodate the additional heat load and to maintain the current vault design constraints on temperature.

5.4.3 Effect on Fuel Dissolution and Cladding Corrosion

Due to the high actinide loading of the MOX fuel compared to natural uranium CANDU fuels, it was necessary to consider the potential for alpha-radiolysis of water to lead to enhanced dissolution of the UO_2 matrix and subsequent release of Pu. It is likely that anoxic conditions will exist in a waste vault deep in plutonic rock and under these conditions the dissolution rate of the UO_2 matrix would be limited by the transport regime and the solubility of UO_2 and not the kinetics of the dissolution process. Alpha-radiolysis, however, could generate oxidizing species in solution that would lead to enhanced matrix dissolution. Electrochemical experiments on UO_2 dissolution in the presence of high α-radiation doses have shown that at the α-dose rates expected from spent natural uranium CANDU fuel, the effect of α-radiolysis on U dissolution is negligible. The dose rate from MOX fuel would be about a factor of 3 greater than natural uranium CANDU fuel and would result in an increase in the predicted dissolution rate by about a factor of 10. The effect of this increase on U release would still be insignificant, however, since the dissolution rate would still be controlled by the solubility of UO_2 and not the kinetics of the oxidative dissolution process.

Although α-radiolysis of water from MOX fuel is likely to have little impact on the dissolution of the fuel matrix. the microstructural changes that occur in the irradiated MOX fuel could lead to enhanced matrix dissolution and fission product release. The irradiated MOX fuel could contain localized inhomogeneities (plutonium rich regions) within the UO_2 matrix, and as a result of the high localized burnup in these regions, there could be substantial migration and precipitation of fission products at the PuO_2/UO_2 interface. This could result in weakening and preferential dissolution along grain boundaries, which could impact on a safety analysis for the release of radionuclides to the vault.

In addition, the manufacture of MOX fuels from weapons plutonium could potentially introduce impurities that are different from those present in natural uranium CANDU fuels. The most important potential impurities are nitrogen and chlorine, which will give rise to ^{14}C and ^{36}Cl, which have been found to be among the most significant contributors to dose in a safety assessment for disposal of used CANDU fuel (Goodwin et al. 1994). In addition, due to the high Dy (and presumably other rare-earths) content of the central elements, there will be significant activation to long-lived radionuclides (e.g. ^{166}Ho) that would require an assessment of their impact on release from the vault. The effect of grain boundary dissolution and impurity concentrations could only be addressed by post-irradiation examinations of irradiated fuels. It is currently planned as part of the Parallex post-irradiation examination of the CANDU MOX fuels, to examine the potential for enhanced radionuclide dissolution and to determine impurity levels of key elements that could be activated to long-lived radionuclides of concern for waste disposal.

In summary, the potential affect on repository performance of substituting CANDU MOX fuel for CANDU natural uranium fuel has been examined for the case of the standard, 37-element design. Higher decay heat levels of the MOX fuel are likely to have a small effect on vault design which will require further study. Also, further examination of the criticality scenarios, and the effect of impurities on cladding corrosion will be required to confirm that present safety analyses and repository performance predictions will remain valid. Also, if the higher burnup CANFLEX fuel is used, for either natural uranium, or MOX missions, additional analyses of the higher burnup on the repository performance will be required.

6. COST AND SCHEDULE

This section summarizes the cost and schedule estimates for the current CANDU MOX disposition case, for disposition of 50 tonnes of surplus weapons plutonium in the Bruce A reactors. Key assumptions used in this estimate were those specified by DOE and ORNL, as adapted to the particular requirements of the CANDU MOX fuel and Bruce operating regime. These key assumptions are as follows:

6.1 COST ASSUMPTIONS

1. All costs are given in 1996 US dollars. Escalation from 94 and 95 dollars assumes 3% annual inflation.
2. Financing costs for facility construction and operation are not included.
3. Land acquisition costs for MOX facilities are not included.
4. Conversion of an existing partially licensed facility in the US Southeast for production of CANDU MOX fuel is assumed, as requested by ORNL. AECL used the previous facility conversion and operating estimates for the Barnwell Nuclear Fuel Plant, reported by AECL report to DOE dated June 30, 1995 as a basis for these new estimates. The slight reduction in facility throughput requirements for the enhanced core design was neglected.
5. The avoided costs for supply of natural uranium fuel, during the period of MOX fuel consumption has been included as in the 1994 study.
6. The cost of pit conversion and transportation of PuO_2 to the MOX site has not been included.
7. The cost of certifying the CANDU MOX fuel bundle transportation package and shipping system, and for the annual transportation costs, remain the same as estimated in the 1994 study, adjusted to 1996 dollars. AECL has remained in close communication with the USDOE Transportation Safeguards Division, and assumptions with regard to packaging and transport remain valid.

6.2 SCHEDULE ASSUMPTIONS

1. Except for Fuel Qualification efforts, which are assumed to begin in 1997, all program elements for both MOX supply and Bruce Preparations are assumed to begin in January 1998.
2. MOX Facility conversion and operation were assumed to be conducted via a privatized, commercial initiative, under NRC license. Parallel licensing and facility modifications were assumed as described in the June 30, 1995 report and as reviewed at that time with the NRC.
3. Following MOX facility startup, an additional 12 months has been included in the schedule to allow for initial Production Fuel Verification of the first 1000 bundles in an operating Bruce A reactor, and the accumulation of sufficient MOX bundles for a fresh core startup and physics testing in a different Bruce A reactor.

4. Subsequent Bruce A reactors used for the plutonium disposition mission are assumed to be converted to MOX fuel operation on-line.

5. Because of the more extensive nature of the CANFLEX Fuel Qualification program at Bruce (both natural U and MOX) it is assumed that the CANFLEX MOX design will not be ready for deployment until approximately 3 years after startup on the reference MOX fuel. Using the throughput figures derived in Section 2, this leads to consumption of about 9 MT plutonium using two Bruce Reactors on reference fuel during the first three years, and 41 MT plutonium using four Bruce reactors on CANFLEX fuel during the remaining 8.5 years, for a total of 11.5 years.

6.3 UP-FRONT COST ESTIMATES

As in the previous studies, three main categories of up-front costs have been evaluated:

Costs for Fuel Qualification, including both Reference fuel and CANFLEX; This totals $77.7 Million; In addition the Fuel Verification program to qualify the initial production fuel via operation of about 1000 MOX bundles in Bruce is expected to cost about $10 million. See section 6.3.1 for details.

Costs for Bruce Site Preparations including new storage building, licensing of Bruce plants on MOX fuel, Provincial Environmental Assessment, and design certification of transportation packages totals $39.3 Million. See section 6.3.2 for details.

Costs for design, construction and startup of a CANDU MOX fuel fabrication facility in the United States totals $182.8 million and is covered in 6.3.3.

The total investment costs for all three categories totals $310 million.

6.3.1 Fuel Qualification Program Costs

The cost estimate for the fuel qualification program is based on the specific program elements outlined in section 3. Two separate programs are estimated; the 37-element bundle qualification program, and the 43-element (CANFLEX) bundle qualification program. Some irradiation tests in NRU are applicable to both programs and costs are included under the 37-element program.

Cost estimates are based on detailed budgetary estimates received in writing from the various test laboratories and test performers that would be responsible to conduct the tests. These include:

AECL CHALK RIVER	NRU irradiation tests, ZED-2 physics tests, critical heat flux (Freon), and PIE of Bruce demonstration tests.
AECL SPEL (Sheridan Park)	CANFLEX Flow visualization and mechanical endurance tests
GE Canada	CANFLEX fuel handling tests
Stern Labs	Critical Heat Flux (water)
Ontario Hydro	Bruce Demonstration tests and Engineering support.

The total costs for 37-element bundle qualification is estimated at $42.3 Million and for CANFLEX at $35.4 Million. The largest single item of cost is the fabrication of the MOX/dysprosia bundles for the irradiation and physics tests, estimated at about $25.5 million. This estimate is for a total of 290 MOX fuel bundles, 90 for Chalk River tests and 200 for Bruce tests. The estimate assumes fabrication at the RFFL at Chalk River, with zircaloy components being supplied by Zircatec Precision Industries. We have not estimated the cost of fabricating these bundles at the Los Alamos TA-55/CMR facilities, although this is another option for future consideration.

A cost breakdown for the CANDU fuel qualification program is shown in the following table 6.3.1-1.

Table 6.3.1-1
Costs for CANDU MOX Fuel Qualification

Test Item	**37-element program**	**CANFLEX program**
1. NRU Irradiation	$21,956	$12,331
2. NRU ZED-2 Physics test	1,370	615
3. Thermal (CHF) Freon	958	991
4. Thermal (CHF) Water	1,838	2,333
5. Fuel Handling		193
6. Hydraulic and endurance		298
7. 90 MOX/DY bundle fabrication (~$87.4 K/bundle)	3,935	3,935
8. Bruce demonstration	833	1,688
9. Bundles for Bruce demonstration 100 MOX for 37 element 100 MOX + 50 NU for CANFLEX	8,740	8,890
10. Engineering Support	1,950	2,632
11. Unspecified Tests	750	1,500
GRAND TOTAL	**$42,330**	**$35,406**

As noted in Section 3, following fuel qualification and prior to startup on a full MOX core, a CANDU MOX Fuel Verification program would be conducted to qualify the new MOX production facility. Costs for production, irradiation, and inspection of this fuel (about 1000 fuel bundles) have not been included in the above estimate. Based on the estimated costs of operating a CANDU MOX facility (see section 6.4.2 below) the costs for production, shipping, insertion, and examination of these fuel verification bundles would be about $10 million.

6.3.2 BRUCE SITE PREPARATION COSTS

As noted in Section 5, Ontario Hydro has reexamined the conceptual design of the proposed new MOX fuel storage building to be erected on the Bruce site, and has made some changes to the earlier arrangement and sizing of this structure. The new arrangement is illustrated in figures 5.2-1 through 5.2-3. The modifications at the Bruce A site include a new MOX fuel storage structure, new fuel handling and reactor units specific modifications. Ontario Hydro has also determined that the budgetary estimates for this structure made and presented in the 1994 study are still about the same. Costs for the remaining Bruce site preparation costs, including licensing, environmental assessment, design and certification of a transportation package for the CANDU MOX fuel, are summarized in Table 9.3 of the 1994 report. These costs adjusted for inflation to CY 1996 are as follows:

Item	Cost (millions of 96$)
AECB Licensing	12.1
Provincial EIS	0.7
Bruce A Modifications	18.4
Design/Certify MOX Transport Package	5.3
Contingencies	2.8
TOTAL	**$39.3M**

It should be noted that the extra cost of revising operating procedures and initial upgrade training of staff is included in the above Bruce A modifications costs, ongoing training costs are included in the annual incremental operating cost estimates in Section 6.4.1 below.

6.3.3 CANDU MOX FUEL FABRICATION FACILITY IN THE U.S.

Several different options exist for siting, erecting and operating a CANDU MOX fuel fabrication facility in the United States. These include use of existing buildings at Hanford, Idaho, Nevada, and Savannah River, as well as the use of private facilities such as those at Barnwell, South Carolina.

AECL Technologies has performed several engineering studies for the DOE under Contract DE-AC03-94SF20218 to examine the feasibility and cost of modifying existing facilities for CANDU MOX fabrication. The most recent completed study focused on the Barnwell Nuclear Fuel Plant owned by Allied Signal in South Carolina. A later study of the Nevada Test Site is underway, but results are not yet available. Therefore, this report summarizes the cost estimates for the Barnwell study, as presented in AECL's report to DOE dated June 30, 1995, with adjustments for inflation to 96 dollars. Note that Decontamination and Decommissioning (D&D) of the MOX facility on completion of the mission is included as an operational cost (section 6.4.2 below), since the plan is to accumulate a D&D fund during facility operation sufficient to D&D upon completion of the mission.

	cost in thousands of 96$
1. Preliminary design/layout	$3,657
2. Final design, building modifications, process equipment refurbishment	76,087
3. Licensing and permitting	12,030
4. Process equipment design, procure install, startup	79,156
5. Program Management	11,886
TOTAL	**$182,815**

6.4 INCREMENTAL COSTS OF OPERATING BRUCE ON MOX FUEL

There are three main components of the annual incremental costs for operating BRUCE on MOX fuel:

Incremental fuel MOX procurement costs vs. natural uranium

Incremental BRUCE operational costs (including security)

Incremental fuel transportation costs

There is a fourth component, not included in this report, but which will nevertheless be a real cost to DOE in connection with the mission; namely the utility charges for incremental financial risk to operate on MOX fuel fabricated from weapons plutonium.

Except for the fueling costs, the estimates for the annual operational costs are the same as the 1994 study, adjusted for inflation. The annual fueling costs have been revised to reflect the 1995 study of Barnwell. Also, because of the higher plutonium loading in the enhanced core design, the mission duration is significantly shorter, leading to much lower cumulative costs for both fuel procurement and operation, as compared to the 1994 report. The estimated net costs for consumption of 50 tonnes of weapons plutonium via a 12 year "hybrid fueling cycle at Bruce A is about $658 million exclusive of front end costs and exclusive of the utility charges for incremental financial risk.

6.4.1 Incremental BRUCE Site Operational Costs

Ontario Hydro estimates that it will cost about $5 million more per year to operate the Bruce A station on MOX fuel as compared to natural uranium fuel. This estimate is the same as in the 1994 report with adjustment for inflation:

	cost in Millions of 96 $
MOX fuel packaging & transportation	$2.2
Fuel receiving and Storage	1.9
Fuel loading, reactor ops, training	1.0
TOTAL	$5.1 Million/yr

Since the number of fuel bundles to be shipped and loaded will stay about the same after shifting from 37-element fuel on 2 reactors, to CANFLEX fuel on 4 reactors, this cost is expected to apply for both operating regimes.

6.4.2 Incremental Fueling Costs

There are three main components of the incremental fueling costs:

a) *The cost of operating the MOX Facility*
b) The costs of transition to and from MOX fuel and from the 37 element design to the CANFLEX design
c) The offsetting costs from avoided natural uranium fuel

a) Facility Operation- The MOX fuel facility costs, as stated above, were derived for the Barnwell study and are shown below, adjusted for inflation:

	(thousands of 96 dollars)	
	37-element	CANFLEX
Materials and Supplies		
UO_2 conversion	4,609	4,982
Cladding components	5,413	7,314
Dysprosia pins	2,710	4,134
Consumables	2,120	2,650
Nat. uranium pellets	435	345
Labor Fees and Management		
Production, direct & indirect	12,879	15,455
Management, G&A, fees	4,240	5,088
Waste Disposal	3,774	4,528
NRC License (I&E) fees	3,498	4,198
Technology transfer fees	5,300	6,360
Facility Support Services		
Landlord and utility costs	7,420	7,420
Security	10,600	10,600
Annual Decontamination fund	3,229	4,384
Capital Replacement	1,590	1,696
Insurance	636	636
TOTAL ANNUAL OPERATING COSTS	**$68,453**	**$79,789**

b) TRANSITION COSTS - In addition to these annual charges, there will be significant one-time costs during the transition to MOX fuel, during the transition from the 37-element MOX to CANFLEX, and again during the transition from CANFLEX back to natural uranium fuel. Bruce site related costs have not been estimated in this study. MOX facility costs for changeover of equipment have been estimated at about $24 million, (see 6/30/95 Barnwell report).

c) NATURAL URANIUM FUEL OFFSETS - Estimates developed for the 94 report have been adjusted for inflation and are shown below:

NU fuel for 2 Bruce reactors -	$17.3 M per year
NU Fuel supply penalty	(1.9 M per year)
Net offset for 2 reactors	$15.4M/yr
Net offset for 4 reactors	$30.8M/YR

6.4.3 Calculation of net fueling costs - Hybrid option

The total mission cost for disposition of 50 tonnes of weapons plutonium using the CANDU Hybrid option is the sum of the up-front costs, the costs for three years operation on 2 Bruce reactors using 37-element MOX fuel, and the cost for 8.5 years operation on 4 Bruce reactors using CANFLEX fuel. This totals $968 million and is calculated as follows.

TIME PERIOD - 3 YEARS OPERATION ON 37-ELEMENT FUEL

MOX Fuel Production	$68.4 M/yr
NU offset costs	(15.4)
BRUCE site costs	5.1
Net Annual costs	$58.1 M/yr
Net Costs for 3 years 58.1 x 3 =	**$174.3 M**

TIME PERIOD - 8.5 YEARS OPERATION ON CANFLEX FUEL

MOX Fuel Production	$79.8 M/yr
NU Offset costs	(30.8)
BRUCE site costs	5.1
Net Annual costs	$54.1 M/yr
Net Costs for 8.5 years 54.1 x 8.5 =	**$459.9 M**

TRANSITION COSTS	**$24 M**
TOTAL MISSION COSTS -(Without FRONT END)	**$658 M**
FRONT END COSTS	**$310 M**
TOTAL MISSION COSTS (U.S. Pu)	**$968 M**

6.4.4 Cost reductions with parallel program to dispose of excess weapons plutonium

A joint Canadian - Russian study is underway to examine the feasibility and cost for disposing of a similar quantity of Russian excess weapons plutonium, also using the Bruce reactors. The design of the MOX fuel would be the same, and the facilities to store and protect the fuel at the Bruce site would be common to both programs. Should parallel missions be authorized and implemented in both countries, there would be cost savings resulting from use of a common fuel design and common facilities at the Bruce site. Assuming that these cost savings are shared equally between the two programs, this would result in a savings to each program of about $54.5 million, derived as follows:

- Fuel Qualification program -	$77.7 M
- Bruce A modifications	$18.4 M
- Bruce licensing/EIS	$12.8 M
- Total of Common Costs	$108.9 M
- 1/2 of Common Costs	$54.5 million

This would reduce the total US mission cost to $914 million.

6.5 ORNL FORMAT

The above facility cost summary data is also presented in a format prescribed by ORNL for the DOE Plutonium Disposition Program. Table 6.5.-1 shows the reactor facility and operations cost summary. The total cost in this case is $186 million. Table 6.5.-2 shows the MOX fuel fabrication facility and operations cost summary. The total cost in this case is $782 million. Note that this total cost contains $308 million savings due to the offset of natural uranium fuel replaced by the plutonium MOX fuel. The grand total cost of the CANDU plutonium disposition mission is $968 million.

Table 6.5-1
Reactor Facility Cost Summary

CAT	End to end alternative	Cost	Basis
	Facility: 2 to 4 reactors Site: Bruce-A in Ontario, Canada		CANDU hybrid case
	Hybrid case: 2 to 4 CANDU reactors first with reference		
	Reactor facility: 2 to 4 Bruce-A reactors with fuel from privatized MOX facility		
	"Preoperational" or "OPC"	Lump sum 1996 $M	
	Front End Costs:		
1	R&D	87.74	MOX fuel qualification
2	NEPA, Licensing, Permitting	12.8	AECB licensing and provincial EIS
3	Conceptual Design	5.3	Design/certify MOX transport pack.
4	Q/A, Site Qualification, S&S	0	Covered in CAT 2 above
5	Post construction start up	0	Covered in CAT 2 above
6	Risk contingency	1.4	Not include contigency in CAT 1
	SUBTOTAL	107.24	Total OPC in 1996 $M
	Captical or TPC Front End Costs (TEC)		
7	Title I,II,III engineering, design, & insp	0	Covered in CAT 8 below
8	Direct & indirect construction/modification	18.4	MOX fuel storage structure
9	Construction management	0	Covered in CAT 8 above
10	Initial spares (Technology dependent)	0	Not considered in this case
11	Allowance for indeterminates	0	Not considered in this case
12	Risk contigency	1.4	
	SUBTOTAL	19.8	Total TEC in 1996 $M
	Total Up Front Cost	127.04	Total TEC in 1996 $M
	Other Life Cycle Cost:11.5 yr Pu Disp. Campaign		
13	Operations & maintenance Staffing (incremental)		
	MOX fuel packaging/transportation: $2.2M/yr		Covers labor and material
	Fuel receiving/storage: $1.9M/yr		Covers labor and material
	Fuel loading/operation/training: $1.0M/yr		Covers labor and material
	Total $5.1M/yr; 11.5 years at $5.1 M/yr	58.65	
14	Consumables including utilities	0	Covered in CAT 13 above
15	Major captical replacement/upgrade	0	No incremental costs expected
16	Waste handling and disposal	0	No incremental costs expected
17	Oversight ($/yr)	0	Not considered in this case
18	M&O contractor fees (% if different than 2%)	0	Not considered in this case
19	Payment in-lieu of taxes to local communities	0	Not considered in this case
20	D&D (% of capital or $ estimate)	0	No incremental costs expected
21	Revenues	0	Not considered in this case
22	Gov't subsidies or fees	0	Not considered in this case
23	Transportation of incoming Pu & outgoing wastes	0	Covered in CAT 13 above
24	Storage of Pu at existing 94-1 site	0	Not applicable
	Total Other LCC	58.65	in 1996 $M
	GRAND TOTAL ALL LCC	185.69	in 1996 $M

Table 6.5-2
MOX Fabrication Facility Cost Summary

EXISTING CANDU ENHANCED CASE: PRIVATELY OWNED MOX FAB PLANT			
CAT	END TO END ALTERNATIVE	COST	BASIS
	Enhanced CANDU Hybrid Case:		170 MT-MOX/YR MOX Capacity
			(Not including non-Pu fuel pins)
	Private MOX plant in SE USA	Lump	3 to 5 MTPu/yr disposition rate
		1996 $M	AECL-T 1995 BNFP MOX supply
	Up-Front Costs:		study results with 3% escalation
1	R&D	0	Included in CAT 3
2	NEPA, Licensing, Permitting	12.03	(see page 11-3 of BNFP report)
3	Conceptual Design	3.66	(see page 11-3 of BNFP report)
4	Q/A, Site Qualification, S&S	0	Included in CAT 3
5	Post construction start up	0	Included in CAT 8 below
6	Risk contingency	0	At 15% included in CAT 2 & 3
	SUBTOTAL	15.69	In 1996 $M
	Captical Up-Front Costs		
7	Title I,II,III engineering, design, & insp	0	Included in CAT 8
8	Direct & indirect construction/modification	155.24	(see page 11-3 of BNFP report)
			(i.e., $76.087M & $79.156M)
9	Construction management	11.89	(see page 11-3 of BNFP report)
10	Initial spares	0	At 20% included in CAT 8
11	Allowance for indeterminates	0	Included in CAT 8
12	Risk contigency	0	Included in CAT 8(25%) & 9(30%)
	SUBTOTAL	167.13	In 1996 $M
	TOTAL UP-FRONT COSTS	182.82	(see page 11-3 of BNFP report)
	Other Life Cycle Cost:11.5 yr Pu Disp. Campaign		
13	Operation & maintenance: 2.0 ave. shifts		
	Staff size = 135 workers; Total 11.3 years		(see page 9-2 of BNFP report)
	3 years at $29.69 M/yr	89.07	(see page 11-5 of BNFP report)
	8.5 years at $35.63 M/yr	302.85	(see page 11-5 of BNFP report)
14	Consumables including utilities; total 11.3 yrs		Assume free PuO2,DUO2 feedstock
	3 years at $33.3 M/yr	99.9	(see page 11-5 of BNFP report)
	8.5 years at $37.44 M/yr	318.23	(see page 11-5 of BNFP report)
15	Major captical replacement/upgrade 11.5 years at $1.7	19.55	(see page 11-5 of BNFP report)
	Transition to CANFLEX in 4th year at an one		
	time cost of $23M	24	(see page 9-5 of BNFP report)
16	Waste handling and disposal	0	Included in CAT 13 (see page 9-4)
17	Oversight	0	Included in CAT 13
18	M&O contractor fees	0	Not considered in this case
19	Payment in-lieu of taxes to local communities	0	Not considered in this case
19A	Nuclear liability & damage insurance		This is an added new CAT
	11.5 years at $0.636M/yr	7.31	(see page 11-5 of BNFP report)
20	D&D; 3 years at $3.229M/yr	9.67	(see page 9-3 of BNFP report)
	8.5 years at $4.384M/yr	37.26	(see same page shown above)
21	Revenues; 3 years at $15.4M/ye	-46.2	Natural uranium fuel offset saving
	8.5 years at $30.8M/yr	-261.8	Natural uranium fuel offset saving
22	Gov't subsidies or fees	0	Not considered in this case
23	Transportation of incoming Pu & outgoing wastes	0	Included in CAT 13
24	Storage of Pu at existing 94-1 site	0	Not considered in this case
	TOTAL OTHER COSTS	599.84	in 1996 $M
	GRAND TOTAL ALL COSTS	782.66	in 1996 $M

6.6 SCHEDULE

The two major components of the schedule are the front end preparations for the irradiation program at Bruce, and the actual irradiation program.

The schedule for conducting the fuel qualification program, preparing the Bruce Site, and preparing and licensing a MOX fuel facility in the U.S. is shown below. The critical path is the preparation of the MOX facilty in the U.S. which has a duration of about 4.5 years. This schedule was developed as part of the Barnwell Nuclear Fuel Plant CANDU MOX Facility Study submitted to DOE on June 30, 1995.

The fuel qualification program which is described in section 3 is projected to take about 6 years for qualification of the 37-element MOX fuel design. However, the MOX facility preparation and licensing schedule is more critical because, upon completion, it will be necessary to accumulate sufficient fuel for a full core loading of MOX fuel, which adds another year to the startup schedule for the first full MOX core Bruce reactor. During this year, the fuel verification program will also be conducted in one of the operating Bruce reactors in order to confirm the quality of the fuel from the new MOX plant.

The major schedule elements are shown in Table 6.6-1 and in the Master Schedule Printout labelled Figure 6.6-1.

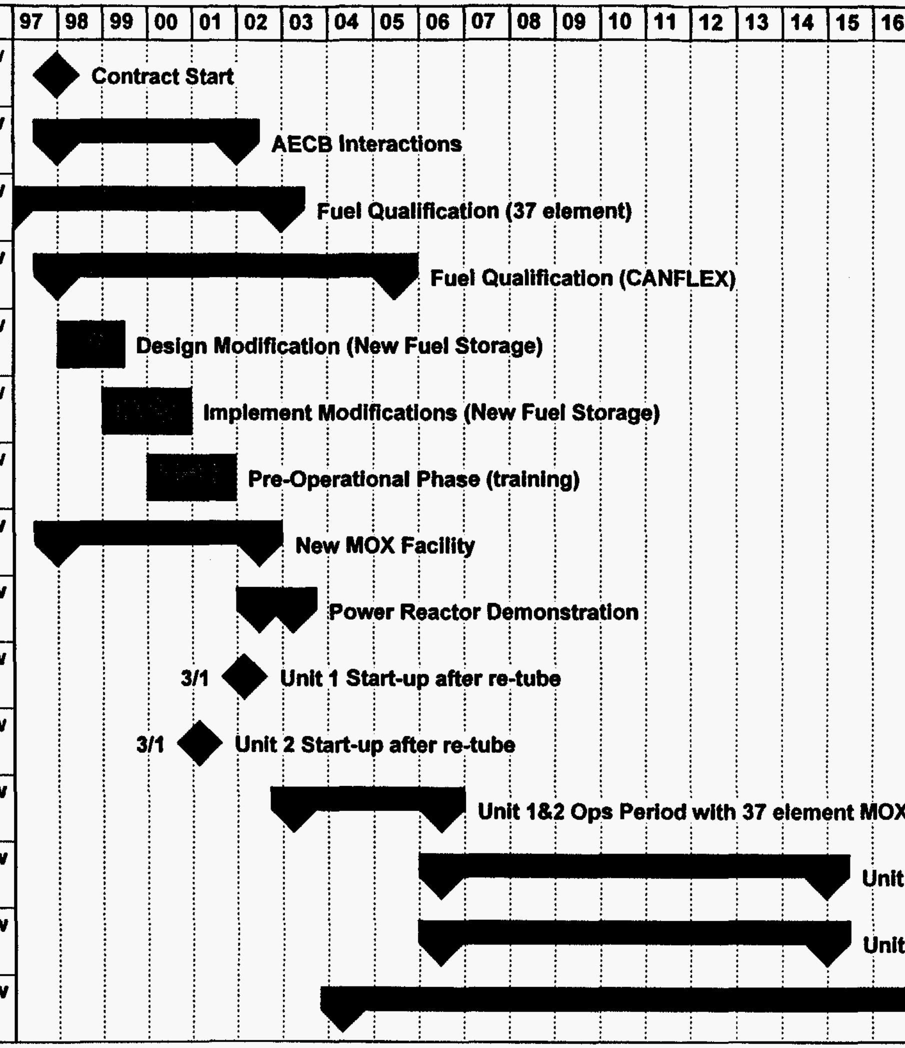
Table 6.6-1
Schedule for CANDU MOX Program at Bruce A

PROGRAM COMPONENT	start	finish	duration
1. Fuel Qualification Program 37-element	1/97	1/03	6 years
2. Fuel Qualification program CANFLEX	1/98	7/05	8 years
3. MOX facility - site, design, equip, checkout and startup	1/98	7/02	4.5 years
4. Accumulate fuel for inital core loading (verify fuel in parallel) **0.4 MT Pu dispositioned in Bruce**	7/02	4/03	9 months
5. Prepare Bruce site - licensing, new storage facility	1/98	1/02	4 years
6. Load first Bruce Reactor with MOX Conduct Physics tests, Full Power on full MOX core	4/03	7/03	3 months
7. Operate 2 Bruce Reactors on 37-element fuel	7/03	7/06	3 years
8. Operate 4 Bruce Reactors on CANFLEX fuel; complete diposition of 50 tonnes of Pu	7/06	1/15	8.5 years

Figure 6.6-1 Master Schedule

Optimization and Implementation Study of Plutonium Disposition using Existing CANDU Reactors

ID	Name	Early Start	Early Fin	Predeces	Duration
1	Contract Start	1/1/98	1/1/98		0w
2	**AECB Interactions**	1/1/98	1/3/02		209.2w
7	**Fuel Qualification (37 element)**	1/1/97	12/31/02		313w
38	**Fuel Qualification (CANFLEX)**	1/1/98	6/29/05		391w
79	Design Modification (New Fuel Storage)	1/1/98	6/30/99		78w
80	Implement Modifications (New Fuel Storage)	12/31/98	12/27/00	79SS+52w	104w
81	Pre-Operational Phase (training)	12/30/99	12/26/01	80SS+52w	104w
82	**New MOX Facility**	1/1/98	7/3/02		235w
90	**Power Reactor Demonstration**	7/4/02	4/2/03	89	39w
93	Unit 1 Start-up after re-tube	3/1/02	3/1/02		0w
94	Unit 2 Start-up after re-tube	3/1/01	3/1/01		0w
95	**Unit 1&2 Ops Period with 37 element MOX**	4/3/03	7/5/06		170w
103	**Unit 1&2 Ops Period with CANFLEX MOX**	7/6/06	12/31/14	95,38	443w
109	**Unit 3&4 Ops Period with CANFLEX MOX**	6/29/06	1/2/15	38FS+52w	444.4w
122	**Spent Fuel Pool**	5/6/04	12/30/22		973.4w



Project:
Date: 9/9/96

Task | Summary | Rolled Up Progress
Progress | Rolled Up Task
Milestone | Rolled Up Milestone