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★ AECL Technologies Inc.

Plutonium Consumption Program
CANDU Reactor Project

Final Report

July 31, 1994

Work performed for the U.S. Department of Energy

Contract DE-AC03-94SF20218

by

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Executive Summary

The United States Department of Energy (DOE) is investigating methods for long term dispositioning of weapons grade plutonium. One such method, identified by the National Academy of Sciences¹ as a "technically attractive possibility for this mission" would be to utilize the plutonium in Mixed OXide (MOX) fuel assemblies in existing CANDU reactors. CANDU (CANAdian Deuterium Uranium) reactors are designed, licensed, built, and supported by Atomic Energy of Canada Limited (AECL), and currently use natural uranium oxide as fuel. The MOX spent fuel assemblies removed from the reactor would be very similar to the spent fuel currently produced using natural uranium fuel, thus rendering the plutonium as unattractive as that in the stockpiles of commercial spent fuel. This report presents the results of a study sponsored by the DOE for dispositioning the plutonium using CANDU technology.

To perform this study, AECL Technologies Inc., the U.S. corporation of AECL, assembled a team to analyze significant technical and cost parameters within the scope specified by the DOE. The team consisted of AECL's key reactor design and support staff, Ontario Hydro, which owns and operates twenty CANDU reactors, the U.S. DOE Hanford site contractor which manages an existing facility that could be converted to fabricate MOX CANDU fuel, Zircatec Precision Industries which produces about one half of the natural uranium fuel assemblies used in Canada, MOX fuel fabrication experts from Babcock and Wilcox, and technical experts from Gamma Engineering Corporation. In addition input was sought from, and meetings were held with personnel from the International Atomic Energy Agency (IAEA), the Atomic Energy Control Board (AECB) which regulates the Canadian nuclear industry, and DOE's Los Alamos, Sandia, and Lawrence Livermore National Laboratories.

Major conclusions follow:

Reference Reactor: Ontario Hydro's 4 x 769 MW(e) Bruce A generating station was chosen as a reference for the study. This four reactor generating station is located, along with its four unit sister station, Bruce B, on Lake Huron, about 300km north east of Detroit. These, as all CANDU reactors, offer many options for fuel cycles. Bruce A is particularly suited for this mission because of the details of the core design, its base load operating mode, its proximity to the US border, and its existing safeguards and security infrastructure.

Nuclear and Fuel Design Reference Option: The study defined the optimum nuclear and MOX fuel design parameters, within the overall study requirements to disposition 50 tonnes of plutonium in 25 years, and the alternate study case of 100 tonnes in 25 years. An optimum nuclear and fuel design utilizes a standard CANDU fuel bundle (figure 1) which contains 37 pencils or elements. The outer 30 elements contain MOX fuel composed of plutonium and depleted uranium oxide. The inner 7 elements contain a mixture of depleted uranium oxide, and a burnable poison, dysprosium oxide, which compensates for the extra reactivity introduced by the plutonium. This fuel would operate within the same burnup and power

¹See the National Academy of Sciences Report Management and Disposition of Excess Weapons Plutonium, p. 14, National Academy Press, Washington D.C. 1994.

rating envelope as standard CANDU fuel, and would have nuclear parameters that allow the reactor to operate within its existing licensing envelope. The study team evaluating fuel supply concluded that the required quantities could be fabricated at the existing Fuel and Materials Examination Facility (FMEF) at the Hanford reservation. Two reactors would be used for the reference case, and four for the alternate case of dispositioning of 100 tonnes in 25 years.

Figure 1

Nuclear and Fuel Design CANFLEX Option: An alternate fuel design utilizing the 43 element CANFLEX fuel bundle was also studied. This fuel bundle, which is currently being qualified for commercial use, has smaller diameter elements which operate at a lower linear power rating, thus permitting higher plutonium concentrations and higher burnups. Use of this design would reduce the quantity of required MOX fuel bundles by almost half, a significant economic advantage. An attractive alternative may be to launch a 2 phase program beginning with the standard CANDU fuel bundle which would allow for earliest implementation, and switch to the CANFLEX bundle once it has finished commercial qualification.

Safeguards and Security: The goal was to maximize security in fabrication, transport, and receipt at Bruce, and to keep the number of sites used to produce the MOX fuel and to disposition the plutonium to a minimum. Methods of transportation had to be proven and reliable. For the reference case, one site will be used to fabricate the fuel, and one site will be used to dispose it. Both road and rail transportation are already routinely used to ship special nuclear materials in the U.S. For the purposes of this study road shipment in existing shipping containers and vehicles was used to evaluate the number of trips and the cost. CANDU reactors are fuelled on power and would require about one shipment of fuel per month for the reference case.

Environment, Safety, and Health (ES&H): There had to be no adverse, and preferably a positive impact on any ES&H concerns. The study compared the current fuel cycle using natural uranium to a MOX cycle which disposes 2 tonnes per year of plutonium (see figure 2). Instead of mining and refining over 6000 tonnes uranium ore per year, the MOX fuel cycle consumes 2 tonnes of plutonium and close to 250 tonnes of depleted uranium waste. The

Figure 2: Comparison of MOX and Natural Uranium Fuel Cycles

resulting quantity of spent fuel is actually about 10% less, due to the slightly higher average burnup which can be achieved with the full MOX core. Fuel handling and packaging methods have been developed to ensure that radiation exposure to power plant workers is equal to or less than the current levels. The fuel fabrication facility design and operation will be licensed by the US Nuclear Regulatory Commission.

Economics, Schedule, and Risk: The objective was to develop a low cost, low risk method for plutonium dispositioning which, in response to the "clear and present danger"² indicated by the National Academy of Sciences, could be implemented as quickly as possible. This has been achieved by designing the fuel and nuclear performance to operate within current experience including safety and licensing envelopes, by limiting changes to the existing reactors to provision for storage and handling of the new fuel, and by utilizing present infrastructures for fuel fabrication and transportation. Costs were broken out into:

²See the National Academy of Sciences Report Management and Dispositioning of Excess Weapons Plutonium, p. 1, National Academy Press, Washington D.C., 1994.

Qualification of the fuel, and confirmation of the nuclear parameters (\$25M)³, modifications to the existing reactor for handling and storage of MOX fuel, including licensing, and design and qualification of packaging and transportation (\$37M), modifications and licensing of FMEF to produce CANDU MOX fuel to disposition 50 tonnes of plutonium in 25 years (\$118M), and the incremental operation and maintenance cost for producing MOX fuel, transportation, and dispositioning at the reactor site of \$54M per year for the duration of the program.

Plutonium Annihilation: Options for a more complete destruction or annihilation of the plutonium were also briefly studied. The CANDU reactor has a superior ability to annihilate plutonium and the minor transuranic actinides. This ability of CANDU is related to the basic design features of using heavy water as coolant and a moderator leading to low neutron absorption rates in the CANDU lattice. For plutonium annihilation, the fuel consists of a mixture of the plutonium isotopes in a non-fertile matrix. The absence of uranium 238 eliminates the source for further creation of plutonium and over 40% of the neutron absorption in the lattice, resulting in a remarkable improvement in neutron economy. The absence of the neutron absorption reduces the fissile requirement of the plutonium annihilator relative to the natural uranium fuelled core. A lower fissile inventory requires a correspondingly higher operating neutron flux level to produce the rated power which is the basis for the CANDU superiority in the annihilation process. This superiority is evident from the high (> 80%) annihilation rate that is achievable per pass through the reactor. By using the on-power fuelling machines to "shuffle" the fuel through one or two additional passes, essentially all of the plutonium may be annihilated. While this option would require substantially more development than the MOX option, it resolves the question of ultimate disposal without requiring highly advanced technology or reprocessing.

Deployment Options: Alternative deployment options were explored and the potential impact on plutonium dispositioning in the former Soviet Union was studied. For example one such alternative would be to set up a parallel CANDU MOX fuel fabrication facility in Russia. The fuel would either be shipped to Canada or utilized in existing or new CANDUs in Europe and Asia. There is potential for many flexible and attractive arrangements. Another alternative deployment option is to incorporate much of the scrap plutonium currently stored as waste at many DOE sites (> 20 tonnes) into the process stream at the MOX fuel fabrication facility. The proposed MOX process has incorporated an oversized scrap recycle system to permit this to occur at a throughput rate in addition to the disposition of the weapons plutonium.

In summary, this study performed for the US DOE has identified practical and safe options for the consumption of 50 tonnes and an alternate of 100 tonnes of plutonium in 25 years in existing CANDU reactors. By designing the fuel and nuclear performance to operate within existing experience and safety and licensing envelopes, and by utilizing existing fuel fabrication and transportation facilities and methods, a low cost, low risk method for long term plutonium dispositioning has been developed. The integrated system can be ready to begin plutonium consumption in 4 years. No changes to the existing reactor system, other than for provision of safe and secure storage of new fuel are required. Alternative deployment

³All costs in this report are in 1993 US dollars.

strategies for the reactors, fuel fabrication, and transportation have been investigated. An annihilation option which uses the unique features of the CANDU system to achieve high levels of utilization of the plutonium offers an attractive option for ultimate disposition without requiring reprocessing of the spent fuel or advanced technology.

1.0 Introduction

The Department of Energy (DOE) of the United States of America is investigating methods for long term of dispositioning of weapons grade plutonium. One such method, identified by the National Academy of Sciences¹ as a "technically attractive possibility for this mission" would be to utilize the plutonium in Mixed OXide (MOX) fuel assemblies in existing CANDU reactors. CANDU reactors are designed, licensed, built, and supported by Atomic Energy of Canada Limited. The MOX spent fuel assemblies removed from the reactor would be very similar to the spent fuel currently produced using natural uranium fuel, thus rendering the plutonium as unattractive as that in the stockpiles of commercial spent fuel. The unique features of the CANDU reactor design, including its heavy water moderated and cooled reactor core, and its on-power fuelling capability allows for a wide range of fuel cycles, including a complete core of plutonium/uranium MOX fuel.

The purpose of this study was therefore to define in detail the composition of the MOX fuel, and to study aspects requested by the DOE relating to economics, safety and licensing, safeguards and security, changes to reactor design, fabrication and transportation of the new MOX fuel, and eventual disposal of the spent fuel. The DOE also asked AECL to briefly study alternate dispositioning options, and methods for annihilation of plutonium using it as a fuel in a non fertile matrix.

With the objective to fully support U.S. DOE in this important initiative, AECL Technologies, Inc., the U.S. corporation of AECL, assembled a study team to analyze all significant technical and cost related parameters within the intended scope of contract specified by the DOE. The team consisted of AECL's key reactor design and support staff, Ontario Hydro, which owns and operates twenty CANDU reactors, the U.S. DOE Hanford site contractor which manages an existing facility that could be converted to fabricate MOX CANDU fuel, Zircatec Precision Industries which produces about one half of the natural uranium fuel assemblies used in Canada, MOX fuel fabrication experts from Babcock and Wilcox, and technical experts from Gamma Engineering Corporation. In addition input was sought from, and meetings were held with personnel from the International Atomic Energy Agency (IAEA), the Atomic Energy Control Board (AECB) which regulates the Canadian nuclear industry, and DOE's Los Alamos, Sandia, and Lawrence Livermore National Laboratories.

Ontario Hydro's 4 x 769 (net) MW(e) Bruce A generating station was chosen as a reference for the study. This four reactor generating station is located, along with its four unit sister station, Bruce B, on Lake Huron, about 300km north east of Detroit.

This report is divided into 11 chapters; chapter one contains the introduction, chapter 2 reviews the CANDU reactor and fuel system in depth and presents the MOX fuel design and the result that no change to the reactor is required. The only changes to the plant are in the facilities to store and handle the MOX fuel. Chapter three discusses the process for licensing,

¹See the National Academy of Sciences Report Management and Disposition of Excess Weapons Plutonium, p. 14, National Academy Press, Washington D.C. 1994.

public, and environmental review of the reactor option. Chapter 4 contains a detailed evaluation of the supply of CANDU MOX fuel, and concludes that the Fuel and Materials Examination Facility at Hanford can be modified to produce the required quantity of MOX fuel at a cost of about \$120M. Production operation could begin in 4 years. Chapters 5 to 10 present results in the areas of Packaging and Transportation of the new fuel, Spent Fuel Storage and Disposal, Safeguards and Security, Environment Safety and Health, Cost and Schedule, and Alternate Deployment Options. Chapter 11 concludes and summarizes the report.

1.1 Technical Synopsis of the Study

As noted above, CANDU reactors offer a high degree of flexibility with respect to fuel cycles. MOX fuel cycles with varying compositions of plutonium/uranium, thorium/uranium, and plutonium/thorium have been tested in several CANDU research and power reactors. The first task of the study group was therefore to define the optimum fuel and nuclear parameters to be used in the study within the requirements set by the DOE for the reference case of dispositioning 50 tonnes of plutonium in 25 years, and the alternative case of 100 tonnes in 25 years. This was done in an iterative fashion using point, and then 3 dimensional reactor physics models.

The resulting nuclear and fuel design utilizes a standard CANDU fuel bundle (see Figure 1.1) which contains 37 pencils or elements positioned in three rings around a central pin. The outer elements contain MOX fuel composed of plutonium and depleted uranium oxide. The plutonium concentration is graded, with the outermost ring having a composition of about 1.2%, and the second ring about 2%. The innermost ring, and central element contain a mixture of depleted uranium oxide, and 5% dysprosium oxide, a burnable poison, which compensates for the extra reactivity introduced by the plutonium. Two reactors would be used for the reference case, and four for the alternate case of dispositioning of 100 tonnes in 25 years.

Figure 1.1

An alternate fuel design utilizing the 43 element CANFLEX fuel bundle was also studied. The CANFLEX fuel bundle is a newer design of bundle which is currently being qualified for commercial reactor use, with post irradiation examination of the first bundles due this year. The CANFLEX fuel bundle has smaller diameter elements in the outer rings which operate at a lower linear power rating, thus permitting higher plutonium concentrations and a higher burnup. Use of this design would reduce the quantity of MOX fuel bundles required by almost half, a significant economic advantage.

Important results of the reference fuel composition and nuclear design are as follows:

The fuel operates within existing power and burnup experience. Present CANDU fuel is extremely reliable; by minimizing changes maintaining this reliability is assured.

The rate of fuelling of the reactor core is slightly lower than at present so no modifications are required to the on-power fuelling machines.

The fuel fabrication rate is below the estimated capability of a facility which could be made by modifying the existing Fuel and Materials Examination Facility (FMEF) at the Hanford reservation. This allows for more economic fuel supply, and a much shorter schedule than could be achieved by constructing a new facility. Operation within existing safety and licensing envelopes is maintained by compensating for the shorter neutron lifetime, and the decrease in the delayed neutron fraction by the addition of dysprosium, a burnable poison, and by other characteristics of the MOX fuel such as a flatter flux profile.

Decay heat generation rates and radioactivity of the spent fuel is similar to the natural uranium fuel currently used so that no changes are required to the current spent fuel storage and handling methods.

Following definition of the reference nuclear and fuel design the study team proceeded in parallel to complete the study as follows:

Nuclear Analysis: A 100 day simulation of reactor fuelling operations was made to refine the basic nuclear parameters including detailed power/burnup histories for use in simulation of fuel performance, and reactivity mechanism worth for use in safety analysis.

Fuel Design: Fuel behaviour, including element stresses and strains, and gaseous fission product formation was simulated using computer models to verify that the reference fuel design met all existing requirements. Evaluations of the fuel's thermal hydraulic behaviour, including determination of critical heat flux were performed.

Fuel Supply: A comprehensive evaluation of the capability of the FMEF to produce CANDU MOX fuel was made which included participation of a CANDU fuel manufacturer, Ontario Hydro, MOX fuel fabrication experts from Babcock and Wilcox, AECL fuel design and fabrication experts, and Westinghouse Hanford personnel.

Safety and Licensing: A systematic review of each of the design basis accidents was performed to confirm that the reactor will be licensable with the full MOX core.

Safeguards and Security: Safeguards and security for the MOX fuel fabrication plant were examined by the fuel supply task team. Safeguards and security for transportation of the new fuel and at the reactor site were examined, including the issue of jurisdiction transfer at the Canadian border. To perform this task meetings

were held with DOE personnel at Sandia. DOE personnel also visited the Bruce site to review the facility. Input was obtained from the International Atomic Energy Agency (IAEA), and the Atomic Energy Control Board (AECB) which is the Canadian regulatory authority.

Environment, Safety, and Health: ES&H issues at the fuel fabrication facility including worker exposure, and waste arisings were evaluated by the fuel task team. Fuel transportation will be performed using proven rail or road methods with a packaging system designed to reduce radiation exposure of personnel to equal to or less than is current experience with current natural uranium fuel. The same packaging system will also limit exposure of personnel at the reactor site to less than current experience.

Changes to Existing Station: The only changes required to the station are associated with storage and handling of the new and spent fuel. A new secure building will be required inside the existing PIDAS (Perimeter Intrusion Detection and Assessment System) fence. The route to the new fuel loading room will require hardening.

Annihilation: A brief study of utilizing the plutonium in a non-fertile fuel matrix was performed. The objective of this physics and fuel study was to achieve high annihilation levels utilizing the unique neutronic and on-power fuelling features of the CANDU system without reprocessing the spent fuel.

Alternative Deployment Methods: While reference cases were adopted for fuel fabrication, the existing reactor site, transportation, and dispositioning rates, alternative deployment options were explored and the potential impact on plutonium dispositioning in the former Soviet Union was studied. For example one such alternative would be to set up a parallel fuel fabrication facility in Russia, which would ship fuel to Canada or utilize existing or new CANDUs in Europe and Asia.

Dispositioning of Spent Fuel: Storage in on site pools, followed by dry storage and eventual burial in either a Canadian or U.S. repository was evaluated.

Economic Evaluation: The cost differential between operating with natural uranium fuel and MOX fuel was evaluated for the entire mission using the economic model provided by the DOE.

1.2 Definition of Commercial Power System Basis

Ontario Hydro operates twenty CANDU units at five Nuclear Generating Stations (NGS) in the Province of Ontario. Eight units with a total net maximum continuous rating of 4124 MW(e) are located at the Pickering site; eight units with a total net maximum continuous rating of 6832 MW(e) are located at the Bruce Nuclear Power Development (BNPD); and four units with a total net maximum continuous rating of 3524 MW(e) are located at the Darlington site, as summarized in Tables 1.1 and 1.2. The nuclear generating stations are operated by Ontario Hydro Nuclear, the nuclear business unit of Ontario Hydro.

Bruce NGS A has been selected as the reference station for this study. Bruce NGS A is a four unit station located on the northernmost point of the 2300 acre BNPD site. BNPD is located on the eastern shore of Lake Huron, approximately 300 km north-east of Detroit (Figure 1.2). In addition to Bruce NGS A, the BNPD site (Figure 1.3) also has the four unit Bruce B station, the Bruce Heavy Water Plant, a steam transformer plant associated with Bruce NGS A providing process steam heat to the site as well as to the Bruce Energy Centre, an industrial development site adjoining the BNPD. The decommissioned Douglas Point prototype CANDU power reactor is located at BNPD, together with a fully operational, safeguarded dry fuel storage facility containing all the spent fuel from operation of Douglas Point.

Although all CANDU reactors can utilize fissile Plutonium in the form of Mixed Oxide (MOX) fuel, Bruce NGS A has been selected as the reference plant because of a number of favourable technical and site-related reasons. These are:

- 1) The reactor is designed as an un-flattened core; that is, it does not have normally inserted thermal neutron absorbing control rods that are used to "flatten" the power distribution in the central, high powered region of the core. The use of enriched fuel in CANDU, as opposed to natural uranium fuel, in conjunction with associated fuel management schemes, tends to flatten the power distribution, which is beneficial with regard to thermal power margins.
- 2) Bruce NGS A is operated as a base-load station, which is considered desirable for the Plutonium disposition mission since this minimizes the total dispositioning time and does not subject fuel to load manoeuvring power changes. Consequently, there is no required change in operating mode for the station, nor is there an impact on Ontario Hydro's grid system control strategies.
- 3) Bruce NGS A is located at a site with a significant, well-developed infrastructure, remote from major population areas, with reasonably close access to the United States of America via the Port Huron/Sarnia border crossing.
- 4) Bruce NGS A has existing IAEA approved safeguards and PIDAS security systems, as do all Ontario Hydro nuclear stations.

As discussed in section 2.1, CANDU reactors utilize a number of pressure tubes (480 in the case of a Bruce reactor) instead of a single large pressure vessel. When the pressure tubes reach the end of their life, the utility makes a decision as to whether to replace them ("retube"), depending upon such factors as demand for electricity and marginal cost of options for incremental generating capacity. Replacing the pressure tubes in a unit takes about two years and has already been proven economically successful as Ontario Hydro retubed all four units at Pickering NGS A, the last of which was retubed in 19 months. Current Ontario Hydro Nuclear business planning projects that the pressure tubes in the four Bruce NGS A units will reach their end of life over the period 1995 to 2008; with Unit 2 starting in 1995 and Units 1, 3 and 4 in the years 2000, 2008 and 2006, respectively.

Figure 1.2 Location of the Bruce Nuclear Power Development Site

Figure 1.3 BNPD Site Layout

Table 1.1

ONTARIO HYDRO NUCLEAR GENERATING STATIONS

STATION	UNIT	FIRST IN-SERVICE DATE/ RETUBED IN-SERVICE DATE
Pickering A	P1	July 1971 / October 1987
	P2	December 1971 / September 1988
	P3	June 1972 / June 1989
	P4	June 1975 / September 1989
Pickering B	P5	May 1983
	P6	February 1984
	P7	January 1985
	P8	February 1986
Bruce A	B1	September 1977
	B2	September 1977
	B3	February 1978
	B4	January 1979
Bruce B	B5	March 1985
	B6	September 1984
	B7	April 1986
	B8	May 1987
Darlington	D1	October 1990
	D2	November 1992
	D3	February 1993
	D4	July 1993

Table 1.2

TECHNICAL SUMMARY OF ONTARIO HYDRO NUCLEAR STATIONS

STATION	NUMBER OF FUEL CHANNELS	CORE THERMAL POWER PER UNIT [MW(th)]	NET STATION ELECTRICAL OUTPUT [MW(e)]	NUMBER OF ELEMENTS PER BUNDLE	NUMBER OF BUNDLES/ CHANNEL
Pickering A	390	1744	2060	28	12
Pickering B	380	1744	2064	28	12
Bruce A	480	2832	3076	37	13
Bruce B	480	2832	3440	37	13
Darlington	480	2860	3524	37	13

It is assumed for the purposes of this study that Bruce NGS A units will be retubed because there is a demand for electricity. Also, for the purposes of this study, it is assumed that during the 25 year mission time the average capacity of the units is 80%, which is generally consistent with Ontario Hydro's experience and, in the case of Bruce NGS A, takes into account factors that have limited capacity in the past, such as transmission line capacity limitations from the BNPD site during part of the 1980's and Ontario Hydro's recent voluntary derating of the eight Bruce units to maintain safety margins while safety-related, corrective design modifications were being implemented.

1.3 Licensing Status of Commercial System Basis

Reactors in Canada are licensed for operation by the Atomic Energy Control Board of Canada (AECB), the nuclear regulatory agency of the Canadian Federal Government. It is the responsibility of the utility operating a nuclear generating station to ensure that the station is sited, designed, constructed and operated to meet safety requirements established by the AECB. Operating licenses are granted for a limited period of time, typically between one and three years and, prior to the expiration of the current license, an application for renewal must be submitted.

Ontario Hydro has successfully obtained and maintained operating licenses for all its nuclear units since the inception of its nuclear power program in the early 1970's, when start-up of the first Pickering NGS A unit occurred. The Bruce NGS A units are licensed in accordance with the AECB Siting Guide requirements which define reference individual and total population dose limits for single failure (process system failure) and dual failure (process system failure plus failure of one of the special safety systems) events, as well as specifying maximum frequencies of occurrence of these failure events. These limits are shown in Table 1.3.

Table 1.3

REGULATORY SITING GUIDE LIMITS APPLICABLE TO BRUCE NGS A

Situation	Maximum Frequency	Individual Dose Limit (1)	Total Population Dose Limit
Single Failure	1 per 3 years	0.5 rem whole body 3 rem thyroid	10 ⁴ man-rem whole body 10 ⁴ thyroid rem
Dual Failure	1 per 3000 years	25 rem whole body 250 rem thyroid	10 ⁶ man-rem whole body 10 ⁶ thyroid rem

NOTE: Maximum individual dose is based upon assuming an average member of the most radiologically sensitive age group remaining at the 914 m (3000 ft) site exclusion boundary throughout the periods of radioactive release.

1.4 Reference Case and Alternative.

The reference case of dispositioning fifty tonnes of plutonium in twenty five years utilizes the standard CANDU fuel bundle containing an average of about 1.5% plutonium. The fuel cycle for this case is shown in Figure 1.4 and compares it to the current fuel cycle using natural uranium. Instead of mining and refining over 6000 tonnes uranium ore per year, the MOX fuel cycle consumes 2 tonnes of plutonium and close to 250 tonnes of depleted uranium waste. The resulting quantity of spent fuel is actually about 10% less, due to the slightly higher average burnup which can be achieved with the full MOX core. The alternate case of consuming 100 tonnes in 25 years could be achieved by doubling the number of reactors (in some ways it may be preferable to operate all four units at the Bruce A station as MOX cores as it eliminates any problems associated with handling two different types of fuel), and by doubling the MOX fuel fabrication rate.

An option for both the reference case and the alternate case is to use the 43 element CANFLEX bundle. The CANFLEX fuel bundle is a newer design of bundle which is currently being qualified for in reactor use, with post irradiation examination of the first bundles due in the fall of 1994. The CANFLEX fuel bundle has smaller diameter elements in the outer rings which operate at a lower linear power rating, thus permitting higher plutonium concentrations and higher burnups. Use of this design would reduce the quantity of MOX fuel bundles required by almost half, a significant economic advantage. This would be particularly attractive in the alternate case (see Figure 1.5) where, instead of mining almost 13000 tonnes of ore and manufacturing and disposing of 430 tonnes of fuel bundles per year, 4 tonnes of plutonium and over 270 tonnes of uranium hexafluoride waste would be consumed, and generate less than 220 tonnes of waste in the form of spent fuel bundles per year.

Figure 1.6 shows the schedule for implementation of either option. In accordance with the instruction given by the DOE, the schedule start date is September 1994. Five main activities were identified:

A start up program where confirmatory tests, engineering, and licensing activities are conducted. This portion of the program would be performed in the first four years of the program. The first two years of this program would be inexpensive and so could be performed in advance of a decision on the exact option.

The modification of the MOX fuel fabrication facility. Again licensing, site selection, and engineering activities could begin immediately, with a hold point prior to any procurement activities.

The detailed design, licensing and procurement of the packaging and transportation route.

Figure 1.4 MOX Fuel Cycle 37 Element Case

Modification of the reactor site which is limited to the design and construction of the new fuel storage building and modification to the transfer route to the new fuel loading room. Station personnel would be trained to handle the new fuel during this period.

A production phase beginning approximately four years after start up, which consumes either two or four tonnes of plutonium per year, using either two or four of the Bruce A reactor units.

As discussed above and in section 4 the MOX fuel could be of the standard 37 element or CANFLEX design, or a phased program. This fuel could be produced in a modified FMEF facility, or, as outlined in section 10 on alternate deployment strategies, in a new or modified facility in the United States, or in the former Soviet Union for dispositioning of Russian plutonium.

As discussed in section 8 the radiation exposure of personnel during the manufacturing, transportation, handling, and disposal of spent fuel during this period would follow ALARA principles, and using the techniques for fuel bundle manufacturing and packaging would be limited to equal to or less than the current levels with natural uranium fuel.

Costs were broken out into: Qualification of the fuel, and confirmation of the nuclear parameters (\$25M)², modifications to the existing reactor for handling and storage of MOX fuel, including licensing, and design and qualification of packaging and transportation (\$37M), modifications and licensing of FMEF to produce CANDU MOX fuel to disposition 50 tonnes of plutonium in 25 years (\$118M), and the incremental operation and maintenance cost for producing MOX fuel, transportation, and dispositioning at the reactor site of \$54M per year for the duration of the program.

²All costs in this report are in 1993 US dollars.

Figure 1.5 MOX Fuel Cycle CANFLEX Case

Figure 1.6 Project Master Schedule

1.5 Cross Reference to DOE Scope and Report Outline

The following matrix shows the cross reference between the sections of this report and the sections of the report outline originally provided by DOE after contract award.

<u>DOE Final Report Outline Section Number</u>	<u>AECLT Final Report Section Numbers</u>
Executive Summary	Executive Summary
1.0 Introduction	1.0 Introduction
1.1 Technical synopsis	1.1 Technical synopsis
1.2 Definition of comm power	1.2 Definition of comm power
1.3 Licensing status	1.3 Licensing status
1.4 Reference case and alternate	1.4 Reference case and alternate
2.0 Technical descriptions	2.0 Technical description; 4.0 Fuel supply; 5.0 Packaging/transportation & 6.0 Spent fuel storage/disposal
2.1 Reactor/reactor cooling	2.1;2.2;2.3;2.4
2.1.1 Mechanical/neutronic des	2.2;2.3;2.8
2.1.2 Cooling/normal op, accid	2.2-2.5; 2.8
2.2 BOP/infrastructure	2.1;2.4;2.8
2.3 Reactor safety systems	2.5;2.8
2.4 Fuel fabrication facility	4.4-4.6
2.5 At-reactor storage	2.4;5.1;6.2
2.6 Fuel management	2.2;2.8
2.6.1 Reactor startup pu inv	2.2;2.7;2.8;2.9
2.6.2 Fuel cycle characteristics	2.2;2.3;2.8
2.6.3 As-charged fuel actinide	2.2;2.3
2.6.4 Discharged fuel actinide	2.2;2.3
2.7 Nuclear material transport	5.2;6.3
2.7.1 Fresh MOX fuel from fab	5.1;5.2
2.7.2 Waste from MOX fab plant	5.1;5.2
2.7.3 Waste from reactor	2.7;5.1;5.2
2.7.4 Spent MOX fuel fr reactor	5.1;5.2;6.3
3.0 Technology needs	2.9 Preparation activities & 6.0 Spent fuel storage/disposal
3.1 Reactor	2.9.2-2.9.4
3.2 MOX fuel	2.9.2
3.2.1 Performance	2.9.2;2.9.5
3.2.2 Fabrication	2.9.2

<u>DOE Final Report</u> <u>Outline Section Number</u>	<u>AECLT Final Report</u> <u>Section Numbers</u>
3.2.3 Compatibility in geolog rep	6.1
3.2.4 Development issues	2.9.1
3.3 New development/test facility	2.9.1
4.0 Operations	2.0 Technical description; 4.0 Fuel supply & 9.0 Cost/schedule overview
4.1 Government/utility/private	2.7;4.7;9.5
4.2 Checkout/startup tests	2.9
4.3 Startup/initial operation	2.9
4.4 Related operating experience	2.1;2.7
5.0 Reactor complex safety/licensing	2.0 Technical description; 3.0 Environmental & 4.0 Fuel supply
5.1 Reactor transient/accident	2.5;2.8;2.9
5.2 Licensing reactors	2.5;2.8;2.9.4
5.3 Licensing fuel fab facility	4.4;4.6
5.4 Environmental impact review	3.0;4.4;4.6
6.0 Estimated cost and schedule	9.0 Cost /schedule overview
6.1 Summary life cycle cost	9.6
6.2 Capital cost detail	9.6
6.3 Operations/maintenance cost	9.5
6.4 Other life cycle cost detail	9.4;9.7
6.5 Anticipated cost risks	9.1;9.6
6.6 Basic schedule	9.1
6.7 Anticipated schedule risks	9.1;9.6
7.0 Safeguards and security	2.0 Technical description; 4.0 Fuel supply; 5.0 Packaging/transportation & 7.0 Safeguards/security
7.1 Fuel fabrication facility	4.4;4.6
7.2 Reactor	2.6
7.3 Spent fuel handling/storage pool	2.6;5.3;7.0
8.0 Environ, safety & health	2.0 Technical description; 3.0 Environmental; 4.0 Fuel supply 5.0 Packaging/transportation; 6.0 Spent fuel & 8.0 ES&H overview
8.1 Transportation safety	5.1;5.2
8.2 Fuel activities	2.5;3.0;4.6.6;8.0
8.3 Waste streams	2.5;2.8;3.0;4.6.6;8.0

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8.3.1 Waste minimization measure	2.7;4.6.6;6.3;8.2.6
8.3.2 Plutonium carryover	2.5;4.6;8.0
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9.3 Other application of technology	2.1;2.3
10.0 Alternative 1	2.0 Technical description; 4.0 Fuel supply & 9.0 Cost/schedule overview
11.0 Public and political acceptance	3.0 Environmental review
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11.2 Likelihood of national pol-a	3.2;3.3
11.3 Likelihood of local/reg accept	3.2;3.3
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2.0 Technical Description - Reactor System

2.1 Current Bruce Station Description

Bruce NGS A comprises four nuclear reactors, four turbine generators and associated equipment, services and facilities arranged as shown in Figure 2.1.1. The main station buildings and structures are: four reactor buildings, four reactor auxiliary bays, a common powerhouse including turbine hall and turbine auxiliary bay running the entire length of the station, a central services area, a vacuum building, an ancillary services building, four pumphouses, a water treatment building, a steam transformer plant, emergency filtered air discharge system building, emergency coolant injection system structures, including an accumulator building, recovery pump room and storage tank, and four standby generator enclosures.

The reactor building (Figure 2.1.2) is a rectangular, reinforced concrete structure which serves as a support and containment enclosure for the reactor and some of its associated equipment. The reactor vault is a portion of the reactor building forming part of the containment system and is connected to the vacuum building via the fuelling duct. The fuelling duct runs the length of the station under the reactor vaults and serves as connection to the fuel handling and fuel storage areas located in the central services area. The reactor vaults are inaccessible during reactor operation above a few percent of full power and services to the reactor vaults can be isolated outside the containment envelope. Provision is made to permit isolation of a non-operating reactor vault from the rest of the station containment system.

2.1.1 Reactor and Auxiliary Systems

The reactor consists of a horizontal, cylindrical tank (the calandria tank) with integral end shields; 480 fuel channel assemblies with integral end fittings; and reactivity control units (Figure 2.1.3). The whole assembly is enclosed by another tank (the shield tank) which is filled with light water. The calandria contains heavy water which acts as neutron moderator and reflector. The calandria, the two end shields and shield tank form an integral multi-compartment structure providing building operational shielding and full shutdown shielding between the calandria and the reactor vault.

Each fuel channel assembly (Figure 2.1.4) consists of a Zirconium-2.5%Niobium alloy pressure tube contained within a Zircaloy-2 calandria tube that provides a gas-filled thermally insulating annulus separating the high pressure and temperature heavy water coolant in the pressure tube from the low pressure and temperature heavy water moderator in the calandria. At each end of a fuel channel the pressure tube is rolled into the hub of an alloy steel end fitting. Each fuel channel, with its contained fuel and heavy water coolant, is supported by the end shield lattice tubes through sliding bearings and partially by the pressure tube/calandria tube annular spacers. The end fitting assemblies are designed with annulus bellows to allow relative axial movement between the fuel channel assemblies and the lattice tubes to accommodate thermal expansion and irradiation induced pressure tube creep.

Figure 2.1.1 Bruce A Station Layout

Figure 2.1.2 Reactor Building

Figure 2.1.3 Reactor Structure

1 Calandria	14 Moderator Inlets
2 Calandria Shell	15 Moderator Outlets
3 Calandria Side Tube Sheet	16 Reactivity Control Rod Nozzles
4 Baffle Plate	17 Booster Rod
5 Fueling Machine Side Tube Sheet	18 Shut-off Rod
6 Lattice Tube	19 Zone Control Rod
7 End Fittings	20 Flux Monitor
8 Feeders	21 Flux Monitor and Poison Injection
9 Calandria Tubes	22 End Shield Cooling Piping
10 Shield Tank Solid Shielding	23 Shield Tank
11 Steel Ball Shielding (End Shield)	24 Shield Tank Extension
12 Manhole	25 Rupture Disc Assembly
13 Pressure Relief Pipes	26 Moderator Inlet Header
	27 Moderator Overflow

Figure 2.1.4 Fuel Channel

1 Channel Closure	10 Outboard End Fitting Sleeve
Bearing and Journal
2 Feeder Coupling	11 Spacer Sleeve
3 Liner Tube	12 Split (Inboard) End Fitting Sleeve
Bearing and Journal
4 End Fitting Body	
5 Channel Annulus Bellows	
6 Fueling Machine Side Tube Sheet	
7 End Shield Shielding Balls	
8 Shielding Sleeve	
9 Shield Plug	

13 Tube Spacer	18 End Shield Lattice Tube
14 Pressure Tube	19 Latch Assembly
15 Fuel Bundle	20 Shield Plug
16 Calandria Tube	21 Liner Tube
17 Calandria Side Tube Sheet	22 Shielding Sleeve
	23 Fixed End Attachment

A removable breech type closure plug is located at outboard ends of end fittings to allow on-line fuelling. A latched steel shield plug is located within a liner tube inside the end fitting to attenuate neutron and gamma flux from the reactor. This shield plug is removed during a fuelling operation and subsequently re-inserted at the end of a fuelling operation.

2.1.2 Heat Transport System

The heat transport system (Figure 2.1.5) circulates pressurized heavy water coolant through the fuel channels to transport heat from the fuel to the steam generators where the heat is transferred to the light water in the secondary side circuit. The heat transport system includes four circulating pumps, six large diameter headers, a system of small diameter feeder pipes connecting the fuel channel end fittings to the headers, and the primary side of eight steam generators and four separate preheaters. These components are connected in a "figure-of-eight" piping circuit arrangement such that the flow in fuel channels is in opposite directions in an interlaced, checker-board pattern. The full power operating values of some key heat transport system parameters are given in Table 2.1.1.

Pressure control is provided by a pressurizer connected to one of the two reactor outlet headers. At high power the pressure is controlled by steam bleed valves connected to the pressurizer and immersion heaters within the pressurizer vessel.

Table 2.1.1

HEAT TRANSPORT SYSTEM OPERATING PARAMETERS

PARAMETER	FULL POWER VALUE
Core Thermal Power, MW(th)	2832
Licensed maximum channel power, MW(th)	7.25 (inner flow zone) 6.95 (outer flow zone)
Licensed maximum bundle power, kW(th)	1035
Reactor Inlet Header Temperature, °C (°F)	251 (483) (inner flow zone) 265 (509) (outer flow zone)
Reactor Outlet Header Temperature, °C (°F)	304 (579)
Reactor Outlet Header Pressure, MPa (psia)	9.18 (1332)
Quality in reactor Outlet Header	0%, saturated
Main System Volume, m ³	266
Number of Heat Transport Pumps	4
Pump Operating MW (hp)	6.41 (8 600)
Pump Rated MW (hp)	8.2 (11 000)

Figure 2.1.5 Simplified Heat Transport System Flowsheet

*The flows indicated are based on the reference design conditions.

2.1.3 Fuel Handling and Storage

On-power fuelling is a unique feature of the CANDU reactor. The Bruce NGS A reactors are fuelled by a system consisting of three fuelling machine systems and associated new and spent fuel handling systems. Each fuelling machine system consists of two fuelling machine heads, head suspensions and elevating tables, and heavy water and air auxiliary systems, all mounted on a transport trolley, shown in Figure 2.1.6. The fuelling machines are controlled remotely from consoles located in the main control room in the central services area. The transport trolleys travel on two sets of rails in the fuelling duct, between the fuel handling and service facilities at the central and east service areas and the bridge and carriage assemblies at each reactor.

New fuel bundles, stored at a new fuel receiving area in the central services area and at new fuel storage racks, are loaded into the fuelling machines at the new fuel loading area. Bundles are transferred to the loading area in crates, bundles uncrated, inspected and transferred to the new fuel transfer mechanisms. The new fuel transfer mechanisms transfer fuel through a transfer port in the containment wall to fuelling machine heads in the fuelling machine room located within containment. Once the fuelling machine head is charged with the required number of new bundles the head and suspension assemblies are lowered onto a trolley by the service area bridges. The trolley is then driven along the fuelling duct to the selected reactor unit to be fuelled.

When the transport trolley is correctly located under the selected reactor the heads and suspensions are picked up and raised by carriages on the reactor bridge to the correct vertical elevation of the channel to be fuelled. The carriages then traverse across the reactor face to the selected end fitting to which the heads are then locked on, as shown in Figure 2.1.7. The closure plugs on the end fittings at both ends of the channel are then removed allowing removal of the shield plugs. Once the channel closures and shield plugs are removed, one fuelling machine inserts new fuel bundles, two at a time, into the channel. This shifts the string of 13 bundles down the channel, pushing two irradiated bundles into an empty fuel carrier of the other fuelling machine head. When the required number of new bundles have been inserted into the channel the shield plugs and channel closures are replaced and the heads unlocked from the end fittings. This procedure is repeated until the required number of channels have been fuelled. The heads and suspensions are then lowered by the bridge onto the transport trolley elevating table, which in turn is lowered onto the trolley. The trolley then returns to the central services area where irradiated fuel is discharged into the primary irradiated fuel storage bay, a reinforced concrete tank with epoxy lined walls and a steel lined floor. This bay is used to store discharged fuel for a minimum of six months after removal from the reactor, following which the fuel is transferred to the secondary irradiated fuel storage bay. The primary irradiated fuel storage bay is sized to store four reactor years of fuel at 80 percent capacity factor.

A small transfer bay and water-filled duct connects the primary and secondary irradiated fuel storage bays. Fuel stacked on trays is moved from the primary bay to the transfer bay by a conveyor, where the tray is transferred by a hoist and tray handling tool to a cable driven cart. This cart, which can carry two trays, is driven through the underwater tunnel to a small receiving bay adjoining the secondary irradiated fuel storage bay. The trays are transferred and stacked by a bridge crane over the secondary bay. The storage capacity of the secondary

Figure 2.1.6 Fuel Handling System

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Figure 2.1.7 Fuelling Machine System at Reactor

irradiated fuel storage bay is approximately 16 station years (64 reactor years) of irradiated fuel.

2.1.4 Moderator System

The heavy water moderator fluid is circulated through the calandria vessel and cooled by heat rejection to heat exchangers. Helium is used as a cover gas over the heavy water. Chemistry control of the moderator fluid is maintained by the moderator purification circuit which contains ion-exchange columns. Values for key moderator system parameters are given in Table 2.1.2, below.

Table 2.1.2

MODERATOR SYSTEM OPERATING PARAMETERS

PARAMETER	VALUE
Number of main moderator pumps	2 at 100%
Main System Volume, m ³	296
Moderator Inlet Temperature, °C (°F)	30
Moderator Outlet Temperature, °C (°F)	65

2.1.5 Reactivity and Power Control

Normal short term control of bulk and spatial reactor power is achieved by light water zone control units. These consist of 14 compartments, distributed within the reactor to define 14 zone regions of the core, containing controllable amounts of light water. The volume of water in the compartments is adjusted in unison to maintain the reactivity balance for bulk power control, and differentially to regulate the spatial power distribution.

In-core self-powered neutron flux detectors, distributed in each of the 14 zone regions, and 22 fully instrumented channel thermal power measurements distributed in 7 axial pairs of zones, provide power measurements utilized by the reactor control system. At low power levels these measurements are augmented by out of core ion chambers.

Additional negative reactivity control to augment the zone controllers is provided by four mechanical control absorber rods, normally positioned outside the core and capable of being driven in the core at variable speed. Long-term reactivity control is maintained by the combination of on-power fuelling and the adjustment of the amount of soluble neutron-absorbing material (boron or gadolinium compounds) in the moderator.

Access to these reactivity control devices and in-core detectors is from the reactivity mechanisms deck directly above the reactor outside containment on the outer surface of the reactor vault ceiling.

2.1.6 Reactor Auxiliary Systems

Some of the more significant of the auxiliary systems associated with the nuclear portion of the plant are:

- Moderator liquid poison system, moderator cover gas system, moderator purification system, heat transport purification system, moderator and heat transport resin handling systems.
- Shutdown cooling system, maintenance cooling system, shield cooling system.
- Liquid zone control system, annulus gas system.
- Irradiated fuel storage bay circulation and purification system.
- Heavy water supply system, heavy water transfer and collection system.

2.1.7 Reactor Special Safety Systems

Four special safety systems are provided to mitigate against the consequences of accidents. They are the two independent shutdown systems, the emergency coolant injection system and the negative pressure containment system, described below.

Shutdown System No. 1

This shutdown system consists of 30 spring-assisted gravity drop shut-off rods (SOR's) of cadmium sandwiched in stainless steel construction. The rods are located vertically above the core in drive mechanisms mounted on the reactivity mechanisms deck. They are maintained out of core by normally energized electromagnetic clutches. An independent triplicated logic system senses the requirement for a reactor trip, based upon two-out-of-three local coincidence voting logic, and de-energizes the direct current clutches to release the shut-off rods.

Shutdown System No. 2

The second independent shutdown system utilizes the injection of gadolinium nitrate solution into the moderator fluid through seven horizontally oriented nozzles. The gadolinium nitrate solution is driven at high pressure from the poison tanks by pressurized helium gas admitted to the tanks when fast-acting solenoid valves are opened on a reactor trip initiation signal.

An independent triplicated logic system, physically separated from the SDS1 logic, is employed to sense the requirement for reactor trip, based upon two-out-of-three general coincidence voting logic.

Emergency Coolant Injection System (ECIS)

The emergency coolant injection system (Figure 2.1.8) is designed to refill the heat transport system and keep it refilled following a Loss of Coolant Accident (LOCA). The system also provides one of the long term heat sinks for emergency core cooling.

The ECIS is common to all four units with a 30 inch common supply header running the length of the station. Injection lines to the individual units contain a parallel pair of normally closed

Figure 2.1.8 Emergency Coolant Injection System Simplified Flow Sheet

light water injection valves located outside containment. An inverted U-bend provides an air gap forming an interface separating the light water of the ECIS from the heavy water of the heat transport system. Four branch lines then penetrate containment, with each line containing a check valve just outside containment in series with a parallel pair of motorized valves inside containment which provide further isolation from the heat transport system. Two of the branch lines connect to the two heat transport system reactor outlet headers, and the other two connect to the reactor inlet headers through the preheater bypass line.

A high pressure accumulator system connects to the common supply header and provides the means of initial high pressure injection of light water into any one of the four units. The accumulator system consists of six large steel tanks located in a building separate from the main station structure. Two of the tanks, containing pressurized nitrogen gas, are normally isolated from the other four tanks, containing light water, by fast acting valves.

Another source of water for the emergency coolant injection system is the grade level tank which connects to the suction side of four 50 percent ECIS pumps. The four pumps also connect to a recovery sump inside containment by a suction line containing a normally closed motorized valve. The pumps discharge to the common header through two of three heat exchangers and a heat exchanger bypass line. The heat exchangers cool the injected water when the system is operating in recovery mode.

The ECIS is initiated upon detection of a LOCA in any unit by triplicated, independent heat transport low pressure signals conditioned by either high reactor vault pressure or temperature signals which act as LOCA conditioning parameters. During the short term phase of operation water from the accumulator tanks is injected at pressure into the unit which initiated the LOCA signal by opening the gas accumulator isolation valves and the isolation valves on the lines to the affected unit. At this time cooldown of the secondary side is also initiated by opening the safety relief valves on the steam generators. The ECIS recovery pumps and cooling water flow to the ECIS heat exchangers are also started.

When the water level in the accumulator tanks becomes low the tanks are isolated to prevent gas entering the heat transport system. Injection of water from the grade level tank by the recovery pumps proceeds until this tank is almost empty, at which time long term recovery mode operation is initiated by opening the sump isolation valves and closing the valves in the heat exchanger bypass line.

Negative Pressure Containment System

The containment system provides a subatmospheric envelope around the nuclear components of the reactor coolant system such that, in the event of failure of these components, any significant amount of released radioactivity will be contained in the envelope.

The containment envelope (Figure 2.1.9) comprises of the following interconnected volumes: reactor vaults, the fuelling duct, fuelling duct extension, Emergency Coolant Injection System (ECIS) recovery sump, central fuelling area, pressure relief duct, pressure relief manifold and vacuum building. The containment volumes are maintained sub-atmospheric, typically 6.9 to 10.3 kPa absolute in the vacuum building and -2.5 kPa to -3.5 kPa gauge in the rest of the containment envelope. Controlled access into the containment for personnel and equipment

Figure 2.1.9 Containment Envelope

is provided by means of 44 airlocks and 10 transfer chambers. Operation of the containment pressure suppression function is automatic and is based upon utilizing the energy released into containment by the LOCA. The pressure relief valves are actuated by a rise in pressure in the pressure relief duct, and the vacuum building dousing spray system is actuated by a rise in the vacuum building pressure.

There are twelve main pressure relief valves, four instrumented main pressure relief valves, four auxiliary pressure relief valves and two reverse flow valves. All of these valves are located in the pressure relief valve manifold. The auxiliary pressure relief valves are pneumatically operated butterfly valves designed to open automatically at a valve manifold pressure of 1.5 kPa(g) and reclose at -5.1 kPa(g). The main and instrumented pressure relief valves open at a valve manifold pressure of 6.9 kPa(g), and reclose at a valve manifold pressure of 2.4 kPa(g) for the main pressure relief valves and -2 kPa(g) for the instrumented pressure relief valves.

The vacuum building dousing system is designed to condense steam discharged into the building and cool the steam/air mixture to limit any further pressure rise. The system consists of an emergency water tank located at the top of the vacuum building connected to a system of spray headers through a vacuum chamber. If the vacuum building pressure rises 26.9 kPa above the pressure in the vacuum chamber, then water is discharged from the emergency water storage tank, over a weir in the vacuum chamber and into the spray headers, from which it sprays into the main chamber of the vacuum building. Vacuum is maintained in the vacuum building and the vacuum chamber by a system of vacuum pumps located in the basement of the building, an area outside of the containment envelope.

Heat removal from containment is provided by a number of air coolers. Each reactor vault has four axial fans and ten main air-to-water heat exchangers, as well as six wall mounted coolers. Four separate air coolers are located in the central fuelling area to remove heat rejected by fuelling machines when parked in this area.

A hydrogen ignition system, consisting of twelve igniters per unit located in the reactor vault and fuelling machine ducts, is provided for controlled removal of hydrogen gas which may be generated during a lower probability design basis accident involving a LOCA, coincident with Loss of Emergency Coolant Injection (LOECI).

An emergency filtered air discharge system (EFADS) is provided to maintain containment subatmospheric in the long term following a LOCA, and to allow controlled and monitored release of radioactivity from containment. The system consists of two 100 percent filters and blowers plus duct work and isolation dampers. Each filter contains a demister, heater, prefilter, upstream HEPA filter, charcoal filter and downstream HEPA filter. Exhaust flow is drawn from the vacuum building and monitored by a post-accident radiation monitoring system prior to being released to atmosphere via the system exhaust stack. The post-accident radiation monitoring system provides on-line radioisotopic analysis for noble gases, gross gamma detection and off-line radioisotopic analyses for particulates, iodine and tritium.

2.1.8 Safety Related Systems

A number of other systems and functions are designed solely to provide safety support functions. They are:

- An emergency boiler cooling system to supply water to the steam generators to establish an adequate heat sink for decay heat removal when normal feedwater supply is unavailable.
- A qualified emergency power supply system to provide power for equipment and instrumentation necessary to maintain and monitor the reactors in a safe shutdown state following a main steam piping failure.
- An emergency venting system to limit internal pressure in the powerhouse in the event of a main steam piping failure.

2.1.9 Secondary Side Systems

The secondary side consists of a conventional, light water, closed steam cycle driving the turbine generator. Steam from the eight steam generators is delivered to two steam drums (Figure 2.1.10) which are each connected in common to a group of four steam generators. The dry steam is delivered by large diameter steam lines to a tandem compound, single shaft turbine directly connected to the generator. The turbine comprises one double-flow high pressure cylinder, followed by live steam reheat and three double-flow low pressure cylinders. The generator is cooled by water and hydrogen and is provided with a rectified AC excitation system.

Connected to the turbine is a surface condenser supplied with cooling water from Lake Huron, suitable for full power operation and capable of accepting approximately 75 per cent of main steam flow as direct rejection. Condensate is returned to the steam generator through five stages of feed heating, consisting of three low pressure heaters, one deaerating heater and one high pressure heater.

Table 2.1.3

SECONDARY SIDE OPERATING PARAMETERS

PARAMETER	VALUE
Steam Drum Pressure, MPa (psia)	4.55 (660)
Steam Flow to Turbine, Mg/s (10 ⁶ lb/h)	1.31 (10.4)
Net Electrical Output, MW(e) Station Efficiency (Net electrical output/total fission thermal power)	769 29.9 %

Figure 2.1.10 Steam Generators and Steam Drum

1. Steam Drum
2. Steam Generator (Boiler)
3. Heavy Water Inlet
4. Heavy Water Outlet
5. Downcomer Annulus
6. Cyclone Separators
7. Steam Scrubber
8. Blow-down Piping
9. 18" Manway
10. Feedwater Inlet

11. Steam Outlet
12. Safety Valve Nozzles
13. Preheater
14. Steam Generator (Boiler) Support
15. Support Hangers
16. Steam Generator (Boiler) Vault Seal
17. Drum Seismic Restraints
18. Steam Generator (Boiler) Seismic Restraints

2.2 Nuclear Analysis and Fuel Management

2.2.1 Reference MOX CANDU Fuel

2.2.1.1 Introduction

CANDU reactors have excellent neutron economy because of the use of heavy water as moderator and coolant. On-power refuelling enables CANDU reactors to operate with minimum excess reactivity in the core. These features allow CANDU reactors to operate with a wide variety of fuel cycles, including natural uranium, slightly enriched uranium, and mixed uranium and plutonium oxide (MOX) fuel. Present CANDU reactors are designed to use natural uranium fuel, which contains only 0.71 wt% U-235. The requirement of disposing 50 tonnes of weapons-grade plutonium over a period of 25 years can be met by using 2 reactors. Each reactor will utilize one tonne of weapons grade plutonium metal per year. This plutonium disposition rate requires the MOX fuel to contain about 1.5% fissile plutonium with the existing 37 element fuel design operating within current burnup limits.

The use of MOX fuel in an existing CANDU reactor, such as Bruce A, introduces changes to the neutronic characteristics relative to the natural uranium core, i.e.

- an increase in the reactivity of the fuel lattice,
- a decrease in the reactivity depth of the control and safety systems,
- a decrease in the delayed neutron fraction,
- a decrease in the prompt neutron lifetime, and
- a decrease in the negative fuel temperature feedback.

These changes nominally results in faster response to reactivity perturbations in the MOX core compared to that in the natural uranium core. The flexibility of the CANDU reactor design allows modifications to be made in the MOX fuel design such that the resulting MOX reactor operates within the safety and licensing parameters for the existing natural uranium CANDU reactor.

The most significant innovation in the MOX fuel design is the strategic placement of burnable poison in the fuel. The purposes of the burnable poison are to

- suppress the excess lattice reactivity and,
- reduce the coolant void reactivity.

Dysprosium is the ideal burnable poison for CANDU reactors because its burnout rate matches the depletion rate of the fissile materials such as U-235 and Pu-239. Figure 2.2.1 shows the arrangement of fuel elements in the standard 37-element design. In the Reference MOX Fuel Design, dysprosium is mixed with 0.2% depleted uranium in the central two rings and plutonium is mixed with 0.2% depleted uranium in the outer two rings. The amount of dysprosium and plutonium in the MOX fuel bundle depend on the requirements for the plutonium disposition rate, MOX fuel production rate and maximum allowable element burnup.

102.5 mm Ref OD Max

495.3 ± 0.8 mm BUNDLE LENGTH

SECTION 1 - 1

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END VIEW

- 1 - ZIRCALOY BEARING PADS
- 2 - ZIRCALOY FUEL SHEATH
- 3 - ZIRCALOY ENDPLATE
- 4 - URANIUM DIOXIDE PELLETS
- 5 - INTER-ELEMENT SPACER PADS
- 6 - PRESSURE TUBE

Figure 2.2.1 37 Element Fuel Arrangement

Coolant void reactivity also depends on the amount of dysprosium in the fuel bundle. In order to offset the faster neutronic response of the MOX fuel, an appropriate amount of dysprosium is added to the fuel to produce a slightly negative void reactivity. It will be seen later that this negative void reactivity adds stability to the MOX core and eliminates the need for enhancements of the existing control and safety systems.

2.2.1.2 Specifications of the Reference MOX Fuel Design

The reference MOX fuel design uses the standard Bruce 37-element fuel bundle geometry. The WIMS lattice code was used to carry out scoping calculations for several combinations of the plutonium and dysprosium contents in the MOX fuel bundle which could satisfy the following design targets:

- the maximum burnup in the discharged MOX fuel elements should not exceed the maximum element burnup for natural fuel, which is about 16,000 MWd/te
- the minimum plutonium metal disposition rate should be one tonne per year per reactor,
- the average MOX fuel burnup should be sufficiently high such that the estimated MOX fuel fabrication rate of 170 te/year can support 2 reactors,
- coolant void reactivity should be between 0 and -5 mk ,
- existing safety and operating limits on maximum channel power, maximum bundle power, and maximum element rating should not be exceeded.

The three-dimensional fuel management code, RFSP, was used to determine the power distribution, fuel burnup and void reactivity in a Bruce A reactor using the burnup dependent MOX lattice cross-sections generated by WIMS. These RFSP calculations were used to select the MOX fuel design that would satisfy all the design targets. Several iterations between WIMS and RFSP were required to establish the specifications for the Reference MOX Fuel Design.

Table 2.2.1 compares the nuclear characteristics of the Reference MOX fuel design with those for the standard natural fuel design. In CANDU reactors, fission is caused by thermal neutrons entering the fuel channel from the moderator. Fuel elements in the third ring are shielded from these thermal neutrons by the elements in the outermost ring, i.e., ring 4. This reduced neutron flux level results in a much lower power output in ring 3 as compared to that in ring 4. The power output of the entire fuel bundle is often limited by the maximum allowable power rating for an individual fuel element. Therefore, it is desirable to design a fuel bundle such that all fuel elements operate at comparable power ratings.

In the Reference MOX Fuel Design, the Pu content of the elements in the third ring is 2%, which is higher than the 1.2% Pu content in the fuel elements of the fourth ring. The higher enrichment level in ring 3 is designed to compensate for the reduced neutron flux level. This

Table 2.2.1
Comparison of Natural Fuel and MOX Fuel Characteristics

	Existing Bruce-A Station		MOX-Fuelled 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) 1.2% Pu Ring 3 (12 pins) 2% Pu Ring 2 (6 pins) 5% Dy Ring 2 (1 pin) 5% Dy	
Bundle Material Composition (fuel material only)	U ²³⁵ 0.13 kg U ²³⁸ 18.67 kg O ₂ 2.53 kg TOTAL: 21.33 kg		Pu 0.23 kg U ²³⁵ 0.04 kg U ²³⁸ 18.36 kg Dy 0.17 kg O ₂ 2.53 kg TOTAL: 21.33 kg	
Average Burnup	8,300	MWd/te	9,700	MWd/te
Maximum Burnup	15,000	MWD/te	15,500	MWd/te
Bundle/FPD	18		15.5	
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	-4.7	mk

enrichment grading scheme enables the elements in ring 3 to operate at a power level comparable to that in ring 4 in spite of the reduced neutron flux level. This allows the fuel bundle to operate at a high power level without exceeding the maximum allowable power rating for individual fuel elements.

The neutron flux level in the central seven elements, which contains 0.2% depleted uranium, 5% dysprosium and no plutonium, is so low that these elements do not produce any significant amount of power as compared to the MOX elements in the two outer rings. Hence, all the power in the MOX bundle is effectively produced by 30 fuel elements instead of 37. This requires the MOX fuel bundles to operate at a lower maximum bundle power limit than that for natural fuel bundles. The 2-bundle shift fuelling scheme chosen for the MOX core flattens the radial and axial power distributions such that the maximum channel and bundle powers in the MOX core, 7000 kW and 780 kW respectively, are lower than those in the natural uranium core, 7200 kW and 950 kW respectively.

The maximum fuel element burnup of the MOX fuel is calculated to be 15,500 MWd/te, which is essentially the same as the maximum fuel burnup attained by natural fuel elements in Bruce A. The MOX fuelling rate is 15.5 bundles per full power day, which is lower than the present fuelling rate of 18 bundles per full power day with natural uranium fuel.

The major difference between the natural uranium fuelled reactor and the MOX fuelled reactor is the reactivity effect during a hypothetical Loss of Coolant Accident. Full core LOCA reactivity in Bruce A is estimated to be +11 mk for the natural uranium reactor and -4.7 mk for the MOX reactor. It will be seen in later sections that this negative void reactivity in the MOX core enhances the performances of the control and safety systems in the existing natural uranium CANDU reactor. The transition from a natural uranium CANDU reactor to a MOX reactor can be achieved without hardware modifications to the existing control and safety systems.

Table 2.2.2 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 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 36.8 g of U-235 and 218.3 g of fissile plutonium. At discharge, each MOX bundle contains 17.4 g of U-235 and 92.0 g of fissile plutonium.

Although the nuclear characteristics of the MOX fuel are different from those of natural uranium fuel, a fuel management scheme has been devised for the MOX fuel to ensure that the performance of the MOX core is comparable to that of the natural uranium core in the areas of fuel power rating, control and safety.

2.2.1.3 Reactor Power Distribution

Figure 2.2.2 shows the arrangement of the fuel channels in a Bruce A reactor. There are 13 fuel bundles in each of the 480 fuel channels. The active length of the core is 12 bundles therefore the end bundles, 1 and 13, are only halfway inside the core region. Figure 2.2.3 shows

Table 2.2.2
Actinide Inventory for NU and Reference MOX Fuel Bundle (g/bundle)

	Fresh		Exit-Burnup	
	NU	MOX	NU 8000 MWd/teHE	MOX 9700 MWd/teHE
²³⁵ U	133	36.8	38.7	17.4
²³⁸ U	18,670	18,3600	18,400	17,800
²³⁹ Pu		218	46.8	78.8
²⁴⁰ Pu		13.5	20.6	57.6
²⁴¹ Pu		0.317	4.32	13.2
²⁴² Pu		0.051	1.28	3.75

Figure 2.2.2 Bruce Channel Arrangement

Figure 2.2.3 Arrangement of Bundles in Channel

the arrangement of fuel bundles in a channel. New fuel bundles are inserted into the inlet end of the fuel channel and irradiated fuel bundles are discharged from the outlet end. A mixed 2-, 4-, and 8-bundle shift fuelling scheme is presently used in Bruce A with natural uranium fuel. The 2-bundle shift fuelling scheme is used in the high power inner region. The 8-bundle shift fuelling scheme is used in the low power outer channels. The 4-bundle shift fuelling scheme is used in the intermediate channels. This mixed fuelling scheme is devised to minimize the power ripple due to refuelling and to optimize fuelling machine usage. The slightly higher fuel burnup of the MOX fuel reduces the fuelling machine usage and therefore allows the use of 2-bundle shift fuelling scheme over the whole core. This simple fuelling scheme significantly reduces the refuelling power ripple in the MOX core.

Figure 2.2.4 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 refuelling intervals in each channel. The power increase upon refuelling, i.e., power ripple, depends on the location of the fuel channel and the number of fresh bundles inserted into the channel. Typical power ripple for a natural uranium CANDU is about 10 percent, i.e., a freshly refuelled channel has a channel power 10% higher than the time-average value. In comparison, typical power ripple for the Reference MOX core is about 7 percent using uniform 2-bundle shift fuelling scheme. The power of the refuelled 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 refuelling again.

Figure 2.2.5 shows the time-average channel power distribution in the Bruce A core using the Reference MOX fuel. Figure 2.2.6 compares the radial power distribution along one row of channels of the MOX core with the radial power distribution of the same row of the natural core. The MOX channel power shape is designed to be slightly flatter than the natural uranium channel power shape. Figure 2.2.7 compares the axial power distributions between a MOX channel and a natural uranium channel. It can be seen that the MOX channel has a flatter bundle power profile than that of the natural fuel channel. Also, the MOX axial power shape is skewed towards the channel inlet end. This skewed power shape increases the value of the Critical Channel Power (CCP), which is the channel power at which dry out occurs on the surface of a fuel element. A higher CCP increases the reactor operating margin by increasing the maximum allowable channel power.

Table 2.2.3 gives the detailed element burnup and element power rating in each fuel ring for each bundle in a high power 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.

Fuel performance data, in the form of maximum element power rating and maximum element power boost due to refuelling, are extracted from a detailed time-dependent RFSP refuelling simulation for 100 full power days (FPDs). Detailed results of the analysis are given in section 2.2.3, Fuel Design Basis.

Table 2.2.3
Burnup and Power Distribution for Reference MOX Fuel (37-Element)
Core-Average Burnup = 9,700 MWd/te

Bundle Position	Burnup (MWd/kg)					Bundle Power (kW)	Element Rating (kW/m)			
	Avg	Ring 1	Ring 2	Ring 3	Ring 4		Ring 1	Ring 2	Ring 3	Ring 4
1	0.5	0.03	0.04	0.53	0.65	180	1.2	1.5	21.9	26.5
2	1.6	0.10	0.12	1.71	2.08	536	2.2	2.7	32.4	36.3
3	2.7	0.17	0.20	2.03	3.51	718	3.4	4.3	47.1	50.1
4	3.9	0.27	0.34	4.29	4.98	754	3.6	4.5	49.4	52.6
5	4.9	0.34	0.42	5.38	6.25	722	4.1	5.3	48.8	49.0
6	5.9	0.46	0.58	6.61	7.41	681	4.8	6.2	47.0	45.2
7	6.8	0.53	0.67	7.62	8.54	642	5.1	6.5	44.3	42.6
8	7.8	0.69	0.87	8.89	9.66	614	5.3	6.8	42.8	40.0
9	8.5	0.75	0.95	9.69	10.53	591	5.6	7.3	41.2	38.2
10	9.5	0.95	1.21	10.96	11.62	565	5.9	7.6	39.3	36.3
11	10.1	1.15	1.47	11.82	12.28	514	5.9	7.6	35.4	32.8
12	10.6	1.20	1.54	12.35	12.83	364	4.6	5.8	25.1	23.2
13	10.6	1.20	1.54	12.35	12.83	119	3.0	3.8	16.4	15.2

Figure 2.2.4 Time Average Bundle Power Natural Uranium

2.27

	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 2.2.5 Time Averaged Bundle Power Reference MOX Fuel

	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								3623	3901	4189	4406	4489	4510	4395	4217	3899	3654								
B						3770	4042	4428	4756	5045	5203	5314	5293	5234	5035	4792	4427	4078	3773						
C				4011	4493	4808	5165	5498	5707	5882	5927	5957	5864	5748	5491	5210	4809	4529	4012						
D				4033	4284	4821	5208	5518	5740	5949	6023	6091	6062	6064	5933	5791	5513	5258	4821	4325	4035				
E			3951	4260	4802	5210	5572	5767	5995	6096	6182	6133	6170	6159	6147	5984	5824	5572	5266	4804	4305	3955			
F		3624	4425	4770	5159	5509	5622	5855	5982	6113	6092	6067	6034	6139	6094	6041	5849	5686	5513	5218	4774	4469	3631		
G		3985	4764	5101	5438	5522	5739	5891	6072	6118	6156	6045	6087	6129	6175	6059	5956	5738	5590	5446	5164	4774	4031		
H		4481	5116	5331	5501	5613	5727	5944	6060	6185	6157	6143	6107	6210	6165	6124	5938	5797	5617	5572	5340	5179	4492		
J	3876	4740	5463	5529	5683	5655	5805	5956	6157	6217	6308	6265	6311	6278	6279	6143	6027	5805	5729	5693	5604	5480	4800	3886	
K	4090	5065	5700	5768	5801	5797	5848	6069	6190	6347	6363	6429	6390	6419	6325	6258	6062	5923	5803	5880	5782	5776	5084	4145	
L	4347	5261	5939	5902	5985	5885	5995	6118	6318	6377	6502	6482	6531	6471	6442	6304	6193	5996	5965	5998	5987	5961	5333	4365	
M	4455	5427	6040	6070	6081	6071	6087	6262	6346	6475	6489	6536	6497	6546	6453	6418	6257	6168	6079	6167	6086	6124	5451	4519	
N	4509	5436	6130	6101	6233	6191	6293	6318	6452	6457	6525	6431	6479	6495	6522	6438	6396	6295	6277	6248	6191	6157	5514	4531	
O	4400	5404	6059	6143	6227	6311	6327	6420	6436	6514	6466	6432	6394	6523	6493	6508	6415	6410	6321	6316	6162	6146	5431	4466	
P	4229	5205	5964	6020	6232	6254	6377	6376	6478	6465	6491	6377	6424	6461	6529	6465	6454	6381	6340	6248	6110	5992	5281	4253	
Q	3991	4965	5708	5913	6080	6218	6266	6381	6403	6493	6463	6491	6453	6518	6473	6472	6376	6346	6228	6165	5933	5791	4993	4054	
R		4660	5437	5673	5972	6029	6186	6249	6397	6417	6494	6460	6505	6465	6478	6385	6321	6190	6109	5989	5757	5465	4732		
S		4177	5011	5556	5958	5898	5975	6202	6297	6428	6430	6486	6451	6481	6409	6361	6199	6047	5908	6035	5681	5084	4202		
T		3611	4521	5213	5706	5907	5882	6040	6238	6295	6379	6343	6384	6354	6350	6228	6105	5885	5977	5727	5286	4548	3667		
U			3969	4658	5222	5647	5898	5974	6099	6246	6242	6244	6214	6287	6230	6156	5973	5960	5663	5289	4682	4030			
V				4075	4630	5156	5590	5657	5894	5993	6072	5994	6029	6050	6040	5886	5713	5599	5215	4650	4134				
W					3983	4515	4984	5404	5668	5891	5947	5958	5935	5981	5881	5712	5405	5033	4529	4035					
X						3716	4195	4575	4942	5164	5326	5349	5373	5312	5197	4938	4612	4200	3756						
Y							3712	3950	4256	4413	4512	4496	4436	4248	3977	3709									

MAXIMUM = 6546 AT M14

Figure 2.2.6 Radial Power Distribution Reference MOX Fuel vs. Natural Uranium

Figure 2.2.7 Axial Power Distribution Reference MOX Fuel vs. Natural Uranium

2.2.1.4 Control

Table 2.2.4 gives the reactivity worths of the control and safety reactivity devices in Bruce A with natural fuel and with the Reference MOX fuel. Equilibrium xenon load, average reactivity due to refuelling a single channel, and full core void reactivity are also given for both cases. It can be seen that the control and safety reactivity device worths in the MOX core are typically 25% lower than the corresponding values in the natural MOX core. This reduction in device worth is due to the 30% increase in the cell average absorption in the MOX lattice over that in the natural fuel lattice. However, the reactivity perturbations in the MOX core, i.e., refuelling and xenon effects, are also proportionately lower than those in the natural core. Hence, control of the MOX core is not expected to be significantly different from that in the natural core. Further discussion of the control of the MOX core are given in Section 2.2.5.

2.2.1.5 Kinetics

Table 2.2.5 gives the kinetics data for natural uranium Bruce A and MOX Bruce A at equilibrium conditions. Most perturbations in a CANDU reactor operating with natural uranium fuel, with the exception of large LOCA, are small and slow and can be easily handled by the control system. The positive void reactivity of natural uranium fuel generates an increase in power prior to operation of the shutdown systems during a LOCA. All modern CANDU reactors are equipped with two independent fast acting shutdown systems. Each system can quickly terminate any increase in power during the most severe LOCA. The void reactivity of the MOX fuel lattice is negative which is a beneficial effect of the dysprosium in the inner region of the fuel bundle. When coolant is voided, the neutron scattering effect due to the coolant in the fuel channel is curtailed. This allows more neutrons to reach the centre of the fuel bundle, i.e., the poisoned region. The negative reactivity generated by the higher neutron absorption in the poisoned inner region more than compensate for the increase in reactivity in the MOX region. The overall coolant void reactivity effect is therefore negative. The implications of the negative void reactivity on behaviour during a LOCA in the MOX core are discussed in detail in section 2.5.

Table 2.2.4
Reactivity Effects in Bruce A Natural and MOX Cores

	Natural Core (mk)	Reference MOX Core (mk)
14 Zone Controllers inserted	-6.0	-4.5
30 SORs inserted	-40.2	-27.0
28 SORs inserted (2 best rods missing)	-31.2	-22.0
4 MCAs inserted	-7.2	-5.5
One channel refuelled	+0.2	+0.1
Full Core Coolant Void	+11.0	-4.7
Equilibrium Xenon Load	-28.0	-22.0

Table 2.2.5

DELAYED NEUTRON DATA FOR EQUILIBRIUM REFERENCE PU-MOX FUEL
AND NATURAL FUEL

GROUP	Reference Pu-MOX Fuel		Natural Fuel	
	DELAYED NEUTRON FRACTION (β)	TIME CONSTANT (s ⁻¹)	DELAYED NEUTRON FRACTION (β)	TIME CONSTANT (s ⁻¹)
1	0.21189E-03	0.47646E-03	0.295E-03	0.612E-03
2	0.81171E-03	0.31239E-01	1.165E-03	0.3155E-01
3	0.68740E-03	0.12065E+00	1.033E-03	0.1218E+00
4	1.46750E-03	0.32595E+00	2.350E-03	0.3175E+00
5	0.50351E-03	0.13651E+01	0.780E-03	0.1389E+01
6	0.14689E-03	0.35989E+01	0.197E-03	0.3784E+01
TOTAL	3.82890E-03		5.819E-03	

Prompt neutron lifetime	= 0.00050 sec (MOX)	= 0.00090 sec (Natural Fuel)
Fuel temperature coefficient	= -3.0 μ k/°C (MOX)	= -6.0 μ k/°C (Natural Fuel)
Full core void reactivity	= -4.7 mk (MOX)	= +11 mk (Natural Fuel)

Table 2.2.6
Reactivity insertion due to 100% Pump Discharge LOCA in Bruce A

Time (second)	Coolant Density (g/cc)	Natural Fuel Void Reactivity (mk) Inserted	Ref. MOX Fuel Void Reactivity (mk) Inserted
0.0	0.800	0.0	-0.0
0.125	0.775	0.34	-0.15
0.250	0.675	1.72	-0.73
0.375	0.625	2.41	-1.02
0.500	0.575	3.09	-1.31
0.625	0.510	3.99	-1.69
0.750	0.450	4.81	-2.04
0.875	0.375	5.84	-2.48
1.000	0.340	6.33	-2.69
1.125	0.290	7.01	-2.97
1.250	0.250	7.56	-3.20
1.500	0.200	8.25	-3.50
1.750	0.175	8.59	-3.64
2.000	0.155	8.87	-3.76
2.250	0.150	8.94	-3.79
2.500	0.145	9.01'	-3.82

2.2.2 Advanced MOX CANDU Fuel - CANFLEX Bundle

2.2.2.1 Introduction

The major difference between the Reference MOX design and the Advanced MOX design is the MOX fuel fabrication requirement. For the same Pu disposition rate of 1 tonne per year per reactor, the Advanced Fuel Design requires only half the MOX fuel production rate of the Reference Fuel Design. This reduction in MOX fuel fabrication requirement is due to the increase in the maximum allowable fuel element burnup. The larger number of fuel elements in the CANFLEX fuel design reduces the element power rating, thus allowing the fuel elements to reach higher burnup without undue degradation of fuel performance.

The amount of Pu in the Advanced Fuel Design is higher than that used in the Reference Fuel Design in order to increase the fuel discharge burnup and reduce the MOX fuel fabrication requirement. The maximum allowable fuel element burnup increases from 15,000 MWd/te to about 28,000 MWd/te. The CANFLEX fuel bundle design is specifically designed to enable the fuel elements to achieve burnups of over 30,000 MWd/te. Hence, the CANFLEX fuel bundle geometry, as shown in Figure 2.2.8, is chosen for the Advanced MOX bundle design. Detailed descriptions of the features of CANFLEX fuel bundle design are given in Section 2.3, Fuel Design.

2.2.2.2 Specifications of the Advanced MOX Fuel Design

The specifications of the Advanced MOX Fuel Design are obtained from a series of iterations between the lattice code WIMS and the 3-D fuel management code RFSP. The procedures are similar to those used to establish the specifications of the Reference MOX Fuel Design.

Table 2.2.7 compares the nuclear characteristics of the Advanced MOX Fuel Design with those for the Reference MOX Fuel Design. In both designs, plutonium is mixed with 0.2% U-235 depleted uranium in the outer two rings, whereas dysprosium is mixed with the 0.2% U-235 depleted uranium in the inner two rings. The plutonium content is increased from 230 g. in the Reference MOX bundle to 375 g. in the Advanced MOX bundle. The amount of dysprosium is increased from 170 g. to 253 g., correspondingly. This allows the discharge burnup of the Advanced MOX fuel bundle to be increased to 17,100 MWd/te. The rate of refuelling is 9 bundles per full power day (FPD) for the Advanced MOX Fuel Design as compared to 15.5 bundles for the Reference MOX Fuel Design. The higher Pu content of the Advanced Fuel Design increases the maximum rippled bundle power from 770 kW to 800 kW. However, the maximum element rating in the Advanced Fuel Design is still about 10% lower than that in the Reference Fuel Design because the bundle power is spread over 43 elements rather than 37 elements. The Advanced MOX core has a slightly higher power form factor, i.e., higher power flattening, than the Reference MOX core. This enables both the Reference MOX core and the Advanced MOX core to operate with the same maximum rippled channel power of 7.0 MW(th).

Table 2.2.8 compares the uranium and plutonium contents in the Advanced MOX fuel with those in the Reference MOX fuel. Each Advanced MOX fuel bundle contains 352.3 g. of fissile plutonium when it is fresh and 161.9 g. of fissile plutonium at discharge burnup. The corresponding values in the Reference MOX fuel bundle are 218.3 g. and 92.0 g. respectively.

Table 2.2.7
Comparison of Advanced and Reference MOX Fuel Characteristics

	Advanced MOX-Fuelled Bruce A Station		Reference MOX-Fuelled 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 (21 pins)	2.1% Pu	Ring 4 (18 pins)	1.2% Pu
	Ring 3 (14 pins)	3.6% Pu	Ring 3 (12 pins)	2% Pu
	Ring 2 (7 pins)	6% Dy	Ring 2 (6 pins)	5% Dy
	Ring 2 (1 pin)	6% Dy	Ring 2 (1 pin)	5% Dy
Bundle Material Composition (fuel material only)	Pu	0.37 kg	Pu	0.23 kg
	U ²³⁵	0.04 kg	U ²³⁵	0.04 kg
	U ²³⁸	17.70 kg	U ²³⁸	18.36 kg
	Dy	0.25 kg	Dy	0.17 kg
	O ₂	2.97 kg	O ₂	2.53 kg
	TOTAL:	21.33 kg	TOTAL:	21.33 kg
Average Burnup	17,100	MWd/te	9,700	MWd/te
Maximum Burnup	28,000	MWd/te	15,500	MWd/te
Bundle/FPD	9		15.5	
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	-1.7	mk	-4.7	mk

Table 2.2.8
Actinide Inventory for Advanced and Reference MOX Fuel Bundle (g/bundle)

	New		Exit-Burnup	
	Advanced MOX	Reference MOX	Advanced MOX 17,100 MWd/teHE	Reference MOX 9700 MWd/teHE
²³⁵ U	36.8	36.8	20.4	17.4
²³⁸ U	17,700	17,900	17,600	17,800
²³⁹ Pu	351	218	146	78.8
²⁴⁰ Pu	21.7	13.5	87.4	57.6
²⁴¹ Pu	1.30	0.317	15.9	13.2
²⁴² Pu	0.08	0.051	3.94	3.75

2.2.2.3 Reactor Power Distribution

Figure 2.2.9 shows the time-average channel power distribution in the Advanced MOX Bruce-A core. Figure 2.2.10 compares the radial power distribution between the Advanced MOX core and the Reference MOX core across the central row of fuel channels. Figure 2.3.11 shows the axial bundle power distribution along a high power channel in the Advanced MOX core and in the Reference MOX core. The fuel management scheme in the Advanced MOX core has been designed to give channel and bundle power shapes comparable to those in the Reference MOX core.

Table 2.2.9 gives the detailed element burnup and element power rating in each ring for each bundle in a high power channel. The bundle power shape is obtained from a time-average calculation. The bundle powers have been normalized to give a maximum channel power of 7.0 MW(th), which is the highest expected in the Advanced MOX core. The axial bundle and element power shapes are used for CCP evaluations.

Fuel performance data are extracted from a rippled core calculation using a random fuel burnup distribution designed to simulate the Advanced MOX core at equilibrium. Time-dependent refuelling simulations have not been carried out for the Advanced MOX core, however, based on the results obtained from the 100 FPDs refuelling simulation in the Reference MOX core, the power boosting effect due to refuelling in the Advanced MOX core is not expected to be a critical factor. Detailed results of the fuel performance evaluation are given in Section 2.3.2, Fuel Design Basis.

2.2.2.4 Control

Table 2.2.9 gives the reactivity worths of the control and safety reactivity devices in Bruce A for the Advanced MOX fuel core and the Reference MOX fuel core. Equilibrium xenon load, average reactivity due to refuelling a single channel, and full core void reactivity are also given for both cases. The control and safety reactivity device worths in the Advanced MOX core are typical 10% lower than those in the Reference MOX core. This reduction in reactivity device worth is due to the increase in the absorption cross-section of the Advanced MOX fuel lattice. Based on the stability analyses carried for the Reference MOX core, this small decrease in controller reactivity worth is not expected to significantly affect the control of the Advanced MOX core.

2.2.2.5 Kinetics

Table 2.2.10 gives the kinetics data for the Advanced MOX core and the Reference MOX core at equilibrium conditions. Because of the higher Pu content in the Advanced MOX fuel core, the delayed neutron fraction and the prompt neutron lifetime are slightly lower than those in the Reference MOX core. This indicates that the power transients in the Advanced MOX core will be slightly faster than those in the Reference MOX core. A detailed discussion of a LOCA transient in the Advanced MOX core is given in section 2.5.

Figure 2.2.8 CANFLEX Bundle Geometry

Figure 2.2.9 Time Average Power Distribution Advance Fuel Design

	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								3604	3866	4128	4352	4407	4450	4309	4166	3829	3640							
B						3801	4125	4512	4824	5121	5253	5383	5331	5302	5069	4867	4465	4156	3749					
C						4038	4532	4908	5267	5602	5789	5979	5992	6051	5918	5843	5541	5310	4847	4550	3974			
D					4062	4389	4925	5318	5592	5790	6003	6048	6142	6075	6113	5934	5847	5523	5359	4855	4409	3992		
E				3967	4364	4904	5325	5660	5803	6018	6090	6195	6118	6187	6122	6155	5944	5859	5587	5364	4831	4383	3898	
F			3634	4453	4859	5268	5588	5634	5836	5930	6066	6019	6032	5958	6090	5990	5995	5760	5689	5514	5306	4785	4468	3574
G			4063	4838	5201	5512	5534	5714	5830	6005	6022	6086	5953	6028	6009	6094	5927	5894	5637	5587	5437	5240	4768	4083
H			4554	5197	5373	5521	5599	5678	5892	5977	6121	6068	6095	6017	6144	6042	6049	5814	5740	5523	5573	5297	5232	4487
J	3875	4801	5530	5551	5683	5614	5755	5879	6094	6128	6266	6192	6271	6185	6205	6014	5949	5679	5676	5608	5606	5459	4834	3816
K	4066	5120	5749	5784	5772	5757	5770	6009	6101	6305	6292	6395	6312	6374	6223	6178	5931	5839	5683	5836	5709	5800	5058	4100
L	4326	5294	5996	5888	5966	5811	5935	6028	6276	6305	6462	6403	6487	6379	6388	6195	6105	5859	5881	5893	5954	5928	5343	4275
M	4408	5480	6064	6080	6023	6019	5985	6204	6269	6443	6413	6501	6417	6497	6361	6351	6125	6062	5946	6095	6008	6129	5424	4455
N	4492	5456	6195	6070	6212	6087	6218	6213	6414	6385	6495	6368	6452	6411	6470	6334	6295	6143	6165	6142	6146	6135	5515	4447
O	4350	5473	6082	6172	6163	6258	6205	6362	6351	6486	6394	6414	6332	6478	6405	6435	6286	6286	6188	6242	6107	6154	5426	4404
P	4233	5231	6059	6005	6245	6161	6321	6271	6429	6385	6466	6320	6403	6385	6469	6352	6355	6251	6243	6182	6087	6010	5298	4201
Q	3960	5063	5755	5998	6061	6222	6175	6347	6303	6444	6380	6454	6373	6463	6367	6387	6276	6259	6159	6144	5943	5832	5032	4022
R		4708	5575	5723	6084	6019	6206	6180	6363	6313	6432	6349	6430	6355	6396	6292	6263	6142	6102	6030	5807	5540	4779	
S		4274	5075	5898	6129	6044	5986	6222	6210	6371	6306	6399	6323	6386	6299	6292	6157	6067	5989	6209	5861	5150	4255	
T		3518	4518	5373	6002	6120	6036	6041	6249	6212	6329	6240	6316	6257	6291	6183	6122	5979	6198	5967	5454	4500	3581	
U			3888	4863	5428	5963	6102	6098	6095	6264	6192	6237	6167	6265	6199	6172	6041	6177	5927	5507	4848	3962		
V				4029	4841	5347	5858	5747	6005	6021	6137	6018	6087	6073	6093	5948	5821	5819	5421	4826	4105			
W					3929	4550	5130	5628	5829	6091	6095	6148	6090	6156	6038	5892	5585	5193	4528	3999				
X						3676	4350	4707	5120	5294	5490	5477	5528	5441	5346	5077	4756	4321	3731					
Y							3733	3911	4237	4353	4476	4433	4397	4197	3950	3696								

MAXIMUM = 6501 AT M12

Figure 2.2.10 Radial Power Distribution Advanced Fuel Design vs. Reference MOX Fuel

Figure 2.2.11 Axial Power Distribution Advanced Fuel Design vs. Reference MOX Fuel

Table 2.2.9
Burnup and Power Distribution for Advanced MOX Fuel (43-Element)
Core-Average Burnup = 17,100 MWd/te

Bundle Position	Burnup (MWd/kg)					Bundle Power (kW)	Element Rating (kW/m)			
	Avg	Ring 1	Ring 2	Ring 3	Ring 4		Ring 1	Ring 2	Ring 3	Ring 4
1	0.6	0.02	0.03	0.64	0.86	228	1.3	1.5	22.9	29.7
2	1.6	0.07	0.08	1.71	2.29	623	1.9	2.2	31.7	39.8
3	4.7	0.22	0.27	5.18	6.62	763	2.8	3.5	42.6	46.0
4	9.4	0.55	0.69	11.16	12.64	762	4.2	5.4	47.5	41.9
5	11.5	0.72	0.91	13.94	15.23	721	4.8	6.3	46.4	38.3
6	12.4	0.85	1.07	15.32	16.19	696	5.0	6.6	45.2	36.4
7	12.8	0.88	1.11	15.82	16.71	648	4.9	6.4	42.3	33.7
8	13.3	0.91	1.15	16.44	17.36	583	4.6	6.0	38.2	30.2
9	15.6	1.26	1.62	19.89	19.83	530	5.1	6.8	34.9	26.8
10	19.3	1.86	2.41	25.13	23.98	500	6.8	8.9	32.1	24.9
11	20.6	2.18	2.83	27.09	25.44	466	7.1	9.2	29.4	23.2
12	20.8	2.20	2.85	27.27	25.60	357	5.5	7.2	22.5	17.8
13	20.9	2.21	2.86	27.40	25.73	122	3.5	4.6	15.5	12.1

Table 2.2.10
Reactivity Effects in Bruce A Advanced and Reference MOX Cores

	Advanced MOX Core (mk)	Reference MOX Core (mk)
14 Zone Controllers inserted	-3.9	-4.5
30 SORs inserted	-24.7	-27.0
28 SORs inserted (2 best rods missing)	-20.2	-22.0
4 MCAs inserted	-5.0	-5.5
One channel refuelled	+0.2	+0.1
Full Core Voided	-1.7	-4.7
Equilibrium Xenon Load	-20.0	-22.0

Table 2.2.11

DELAYED NEUTRON DATA FOR EQUILIBRIUM
ADVANCED AND REFERENCE MOX FUEL

GROUP	Advanced Pu-MOX Fuel		Reference Pu-MOX Fuel	
	DELAYED NEUTRON FRACTION (β)	TIME CONSTANT (s^{-1})	DELAYED NEUTRON FRACTION (β)	TIME CONSTANT (s^{-1})
1	0.2021E-03	0.4591E-03	0.21189E-03	0.47646E-03
2	0.7909E-03	0.3112E-01	0.81171E-03	0.31239E-01
3	0.6606E-03	0.1318E+00	0.68740E-03	0.12065E+00
4	1.4039E-03	0.3281E+00	1.46750E-03	0.32595E+00
5	0.4887E-03	0.1374E+01	0.50351E-03	0.13651E+01
6	0.1391E-03	0.3568E+01	0.14689E-03	0.35989E+01
TOTAL	3.6852E-03		3.82890E-03	

Prompt neutron lifetime	= 0.00046 sec (Advanced)	= 0.0005 sec (Reference)
Fuel temperature coefficient	= -2.0 $\mu\text{k}/^\circ\text{C}$ (Advanced)	= -3.0 $\mu\text{k}/^\circ\text{C}$ (Reference)
Full core void reactivity	= -1.7 mk (Advanced)	= -4.7 mk (Reference)

2.3 Fuel Design

2.3.1 Reference MOX Fuel Design

2.3.1.1 Overview

Section 2.3.1 assesses the fuel performance of the reference MOX fuel design. This assessment is based on the performance of UO_2 fuel under similar operating conditions, on AECL's and world experience with MOX fuel, and on MOX fuel modelling. This section discusses the design and evolution of the 37-element natural uranium CANDU fuel bundle, which is the basis for the reference MOX fuel design; the current natural uranium operating power/burnup envelope in Bruce A; AECL's experimental MOX fuel performance data; MOX physical properties that impact on fuel performance; and the basis for the reference MOX fuel design.

Power/burnup histories for MOX fuel resulting from the full core reactor physics simulations are compared with those for natural uranium, and the MOX power histories are shown to fall within the natural uranium envelope. A power boost assessment is performed for the reference MOX fuel, showing that no fuel defects are anticipated from power boosting resulting from refuelling. Further confirmation of fuel performance for the reference MOX fuel is provided by modelling the performance of the fuel for the high-power envelope, using the code ELESTRES. This is a severe, or bounding, test of fuel performance. The predicted gas release, internal gas pressure, and sheath strains for the reference MOX high-power envelope are shown to be well below that for natural uranium.

The thermalhydraulic performance of the reference MOX fuel design is then assessed. It is shown that reactor overpower trip setpoint margins are the same as for natural uranium fuel.

2.3.1.2 37-Element Fuel Design

The CANDU 37-element fuel bundle is fabricated from seven basic components (see Figure 2.2.1), using two main materials: zircaloy and UO_2 . When loaded with fuel pellets the bundle weighs approximately 24 kg, of which 90% is UO_2 .

The fuel element contains the UO_2 pellets in a Zircaloy-4 sheath. A thin graphite layer (known as CANLUB) on the inside surface of the sheath reduces the effects of pellet-clad interaction (PCI) and protects against stress-corrosion cracking (SCC). Endcaps are resistance welded to the ends of the sheaths to seal each fuel element. Endplates are also resistance welded to the endcaps to hold the 37 elements in the bundle assembly. The elements are arranged in rings of 6, 12 (intermediate ring) and 18 (outer ring), around the central element. Spacers are brazed to the elements at their mid-points, to provide the desired inter-element spacings. The bundle is spaced from the reactor pressure tube by bearing pads brazed near the ends and at the mid-point of each outer element. Beryllium metal is alloyed with the Zircaloy to make the braze joints. A more complete description of the bundle and its components is given below.

UO₂ Pellets. The UO₂ pellets are designed to

- maximize the amount of fissile material in each fuel element,
- minimize dimensional changes during irradiation,
- control fission-gas releases,
- minimize circumferential ridging of the sheath, and
- be economical to produce.

The material properties of the UO₂ that strongly affect the thermal behaviour and hence fission-gas release are maintained within limits; for example, density (typically 97% of the theoretical density) and oxygen-to-uranium ratio. The high-density pellets ensure dimensional stability. Other features of the UO₂ pellets, such as pellet dishing and length/diameter (L/D) ratio, are designed to minimize sheath strains and to accommodate fission gases.

To help in maintaining low center-temperatures, and hence low levels of fission-gas release within the elements, a fine surface finish is specified for the pellet surfaces, and for the inside of the clad. This maximizes the heat-transfer coefficient between the fuel and the contacting sheath. Such finishes are produced with typical fabrication techniques such as wet grinding.

Fuel Sheath. The Zircaloy-4 sheaths are designed to

- contain the UO₂ under normal operating conditions,
- minimize neutron absorption,
- minimize corrosion and hydrogen/deuterium pickup,
- minimize strain effects,
- minimize resistance to heat transfer,
- minimize hydraulic head loss,
- be strong enough to withstand normal operating loads (including re-fuelling and reactor power manoeuvring).

Zircaloy-4 has low neutron absorption, good corrosion resistance and low hydrogen pickup. Material properties and heat treatments are specified to optimize sheath ductility at high irradiation levels. An important feature of the CANDU fuel element is the use of thin-wall clad. The minimum wall thickness is approximately 0.38 mm, and has been used continuously since the first charge for the NPD reactor in 1958. Because the clad is designed to collapse into direct contact with the fuel pellets, pellet clad interaction (PCI) was recognized as a fact from the start of the program. PCI can cause unacceptable sheath strains, for example, due to in-reactor densification of the UO₂. Thus the need was established for the high-density, irradiation-stable fuel pellets, which avoid this problem.

CANLUB Coating. A thin layer of graphite is applied on the inner surface of the sheath, to reduce the effects of PCI, and to protect against SCC. Improvements to the coating are continuously being investigated by AECL, and it is expected that changes may be implemented in current production, particularly for high-burnup fuels. Any pertinent changes that are proven in service would also be adopted for MOX fuel.

Endcaps. The endcaps are designed with the correct profile and strength to take the loads

of the latches in the fuel channel, due to the coolant hydraulic drag. They are thin, to minimize neutron absorption. The welds are amenable to automatic fabrication techniques, and can be 100% examined non-destructively to provide a very high quality weld.

Filling Gas. The fuel elements are filled with unpressurized helium, to allow all elements to be helium leak tested during fabrication. The helium also enhances the pellet/sheath heat conductance at start-of-life.

Endplates. In addition to holding the elements in the correct configuration, the endplates have to be flexible, to allow differential axial expansion among the elements. Also, they have to distribute axial loads (for example, during refuelling) to all elements in the bundle. The endplates are made as thin as possible, to minimize the amount of neutron-absorbing material in the assembly, and to minimize the gap between the UO_2 fuel in adjacent bundles.

Inter-element Spacers. Small spacer pads, approximately 2.5 mm x 8 mm, are brazed at the mid-length of all elements. They are mounted at a small angle to the axis of the element, so that any two mating spacers are angled in opposite directions. This skew increases the width of possible contact between spacers, and prevents interlocking. Avoidance of a grid spacer minimizes the parasitic material present in the bundle. The spacers contribute to the good heat-transfer performance and hence high critical heat flux of the bundle, by introducing turbulence to the coolant at each bundle mid-plane.

Bearing Pads. Three pads, approximately 2.5 mm x 25 mm, are brazed near the ends and at the center of each outer element. The pads support the bundle in the fuel channel and fuel handling system, and prevent mechanical damage to the sheath. The surface wear and fretting to the pressure tube caused by the bearing pads is within the design allowance.

The CANDU fuel bundle has a number of important differences from the PWR assembly. The simplicity of the bundle is the most obvious, and results in very low fuelling costs, and facilitates automated fabrication. The collapsible fuel cladding in CANDU ensures good heat transfer from the fuel to the coolant, and lower fuel temperatures than PWR fuel at the same powers. The pellets are of higher density than for PWR fuel, so in-reactor sintering (and resultant volumetric changes) is minimized. Neither pore-formers nor gas-plenums are required for CANDU fuel. A grid-spacer is not required to maintain inter-element spacing. CANDU fuel pellets have high surface finish. Some of these design features will be referred to in Section 4.

The simplicity of the CANDU fuel design has proven itself suitable for mass production to very high quality levels. This has been established with consistent operation of extremely high reliability, and over 1 000 000 fuel bundles of the 37-element design (half each made by the two Canadian fabricators) have been irradiated to date.

2.3.1.3 Evolution of the CANDU Fuel Design

The 37-element CANDU fuel design has evolved from the designs of the 22 MWe NPD reactor, which used 7- and 19-element bundles, the 220 MWe Douglas Point, KANUPP and RAPP reactors' 19-element bundle designs, and the Pickering 28-element design. The 37-

element design is used in more than 16 large power reactors, including the Bruce units. This evolution in CANDU fuel design parallels the trend to greater subdivision in LWR fuel, and continues with the development of the CANFLEX bundle, which is the reference for the advanced MOX fuel design described in Section 2.3.2. The increased subdivision has allowed an increase in nominal maximum power ratings from the 220 kW of the NPD bundles to about 1 000 kW of the Bruce 37-element design.

To originally qualify the 37-element fuel designs, prototype irradiations of two bundles were performed in the NRU reactor during the 1970s. The fuel elements in one test were initially irradiated at high power of 72 kW/m, which slowly declined to about 48 kW/m at a final burnup of about 12 500 MWd/te. The power level was just above the linear powers corresponding to the outer elements of a bundle operating along a 1035 kW bundle power envelope. This was the original license limit for bundle power at Bruce. The power history for the second test was less severe; i.e., slightly lower powers and final burnups than the first test.

The 37-element design is also the reference fuel for the Bruce B, CANDU 6 and Darlington reactors.

2.3.1.4 CANDU Fuelling

CANDU reactors are fuelled on-line using an automated fuel handling system, to enable close control of core reactivity and efficient utilization of its natural uranium fuel. The full power refuelling requirements for one Bruce A unit is 18 bundles per day. As noted in Section 2.2.1.3, Bruce A uses a mixed 2-, 4- and 8-bundle shift fuelling scheme with natural uranium fuel. The reference MOX fuel will require only a 2-bundle shift fuelling scheme throughout the core. The typical fuelling sequence for a 2-bundle shift is as follows:

- New fuel bundles are loaded in pairs into the fuelling machine.
- The fuelling machine is coupled onto the upstream end of the channel to be refuelled, and a second fuelling machine is coupled onto the downstream end of that channel.
- Closure plugs and shield plugs are removed from the channel ends, and the upstream fuelling machine inserts a pair of bundles into the channel.
- The downstream fuelling machine receives two bundles into a fuelling magazine; shield plugs and end closures are replaced, and the refuelling machines disengage from the channel.

With a 2-bundle shift fuelling scheme, several channels can be refuelled before the refuelling machine needs to discharge the spent fuel through the spent-fuel port to the spent-fuel bay.

On-power refuelling provides great flexibility in fuel management, and facilitates the use of different fuel cycles or bundle designs. Currently, that flexibility is used with natural uranium fuel in several ways: different regions of the core are refuelled at different rates to achieve

a particular distribution of channel power; different regions of the core are refuelled with different fuelling schemes (adding fewer bundles reduces the refuelling ripple, but increases the fuelling machine usage); fuel can be shuffled from one channel to another (this is done in the initial fresh core to increase the burnup); different channels are sometimes fuelled with different fuels (in the initial core, depleted uranium is loaded into the center of the central channels to flatten the axial and radial powers). Defects can be discharged with the reactor at power, thereby reducing fission product contamination of the coolant.

This flexibility will facilitate the introduction of MOX fuel in several ways. First, a power reactor demonstration irradiation will be facilitated by the ability to load a few MOX bundles into the most appropriate channels, before loading a larger fraction of the core with MOX fuel. The ability to choose the number of bundles added to the channel during refuelling enables the size of the refuelling ripple with MOX fuel to be controlled. The rate of introduction of the MOX fuel into the core can also be controlled with on-power refuelling. Note that in the reference MOX core, the fuelling rates were chosen to give a similar power profile as natural uranium; different refuelling rates could be used in different regions of the core to further flatten the radial channel power profile across the core, thus lowering the peak channel power relative to natural uranium.

2.3.1.5 CANDU Fuel Performance and Defect Mechanisms

CANDU fuel designs are based on the very extensive experimental loop irradiations carried out since the late 1950s in AECL's NRX and NRU research reactors. Performance data from early tests spanned a large range of power ratings on single elements, as well as on full size bundles. Later tests subjected fuels to very large power ramps, and included tests to burnups of approximately three times the design. Test results were used to optimize the later designs now in use.

About 1 000 000 fuel bundles have been irradiated to date in Canada. Fuel performance has been excellent, with the successful irradiation without defects of over 99.9% of the bundles. The element defect rate is lower than 0.1%. This defect rate is not an equilibrium or steady-state "rate". Often, the annual defect rates have been zero at some Canadian CANDU stations. The majority of the defects reported have occurred either in a small number of batches of fuel with manufacturing flaws or as a result of non-standard operation. In those instances, the defect rate is high and is referred to as a "defect excursion". The number of such defect excursions in Canada has been very small (Table 2.3.1), and in every case the industry has reacted quickly to identify and implement solutions. The defect excursions have had no impact on station capacity factors. Fuel defects can be dealt with at full power primarily because most CANDUs, including Bruce, have on-power failed fuel detection and location capability, and all have on-power fuelling capability, so that defected fuel can be removed promptly.

The defects that have occurred in CANDU fuel elements and bundles fall primarily into three categories: manufacturing and "unassigned" (50%), SCC (35%), and fretting (15%). Manufacturing defects are a comparatively rare event and represent a small statistical probability in a very reliable mass-production product. The manufacturing defects found in CANDU power reactor fuel have been primarily caused by incomplete endcap welds, porous

Table 2.3.1 Defect Excursions in CANDU Fuel - Canadian Experience

PERIOD	REACTOR	EXTENT	LOCATION	DESIGN	REASON
Early 70's	Pickering	110 bundles	Ridges	Non-CANLUB	SCC
Early 70's	Douglas Point	80 bundles	Ridges	Non-CANLUB	SCC
1980	Douglas Point	50 bundles	Sheaths	CANLUB	Incomplete Welds
1983/84	Bruce A	40 bundles	End-caps	CANLUB	SCC
1985	Pickering	N/A *	Sheaths	CANLUB	Fretting by debris
1988	Pickering	36 Bundles 290 Elements	Ridges	CANLUB	SCC
1991/92	Point Lepreau	20 Bundles	Sheaths	CANLUB	Primary Hydriding

*N/A Not available

endcap barstock, and excess hydrogen within the element due to improper curing of CANLUB. The quality program that would be applied to the MOX fuel fabrication would assure the same low probability of manufacturing flaws with MOX fuel as for natural uranium.

SCC results from high sheath stresses associated with a sudden increase in power (power-ramp), combined with a high concentration of corrosive fission products within the pellet-to-sheath gap. CANLUB was introduced to provide protection against SCC defects, and is believed to immobilize the corrosive fission products (such as iodine). Based on previous fuel performance at the power and experimental reactors, fuel defect thresholds for SCC have been established (Figure 2.3.1). These thresholds provide burnup-dependent operational limits for ramped power (final power after boosting), and power increase. To use these diagrams, the final power of the fuel, after a power increase at a given burnup, is compared to the threshold line for ramped power. If the point falls above the threshold line, it lies in a region of finite defect probability. Similarly, the power increase as a result of the ramp is compared with the threshold curve for power increase. Only if both thresholds are violated is the fuel indicated to fail. The threshold curves in Figure 2.3.1 are used in this study to assess power-ramp performance of the reference MOX fuel in Section 2.3.1.11.

Figure 2.3-1(a) Stress-Corrosion Cracking Threshold for Ramped Power

Figure 2.3-1(b) Stress-Corrosion Cracking Threshold for Power Increase

Fretting defects are normally associated with the initial operation of a new CANDU plant, due to debris left behind in the primary heat transport system during reactor construction. Fretting by debris will hence not be an issue with MOX fuel in Bruce A.

2.3.1.6 Natural Uranium Power/Burnup Histories

This section summarizes the power envelope for Bruce A, for fuel within the normal burnup range. This information will subsequently be compared to that for the reference MOX fuel. Section 2.3.2.3 examines the extended burnup fuel performance data from Bruce A, applicable to the high-burnup advanced MOX fuel design.

The bundle average discharge burnup for the 37-element natural uranium fuel in Bruce A is 8 300 MWd/te, and the nominal peak outer element discharge burnup is about 15 000 MWd/te (although some elements have reached burnups exceeding 25 000 MWd/te). Figure 2.3.2 shows the distribution of outer-element discharge burnup over a nine-month refuelling period in Bruce A. The two distinct peaks correspond to two burnup regions in the core, used to flatten the radial channel power distribution, thereby lowering the peak power.

Figure 2.3.3 shows a typical snapshot of outer-element power and burnup at a particular point in time (1987 December 1). Each point in the plot corresponds to the outer-element rating and corresponding burnup for one bundle in the core. This gives a visual indication of the element power/burnup distribution in the core. Also shown is the "power envelope", a curve that envelops the power/burnup snapshots over a long period of time. The power envelope shows the highest element power as a function of element burnup, and is used in fuel performance modelling as a severe, or bounding, test of fuel performance. Figure 2.3.4 quantifies the information in Figure 2.3.3 in the form of a 3-D histogram, which shows the number of elements in the core for various power/burnup intervals. Data is shown for both the intermediate and outer rings (and the cumulative data for both rings), even though the peak rating always occurs in the outer ring of fuel with natural uranium. This facilitates the later comparison with the reference MOX fuel (where the peak rating can occur in either the intermediate or outer ring). Note that the total number of elements in the two outer rings is 187 200 (480 [channels] x 13 [bundles] x 30 [elements]).

These curves will be compared to those for the reference MOX fuel in Section 2.3.1.11.

Figure 2.3.2 Outer Element Discharge Burnups Bruce A Unit 1: 1987 May to 1988 Feb
(4044 Bundles)

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Figure 2.3.3 Snapshot of Outer-Element Powers and Burnups for Natural Uranium Fuel in Bruce A

Figure 2.3-4 (a) Distribution of Outer-Element Powers and Burnups for Natural Uranium Fuel in Bruce A

Power (kW/m)	Burnup (MWd/te)									Total
	0-2000	2000-4000	4000-6000	6000-8000	8000-10000	10000-12000	12000-14000	14000-16000	16000-18000	
50-60	288	2556	4266	3114	666	0	0	0	0	10890
40-50	2610	3908	4590	4374	3798	1044	0	0	0	20322
30-40	5238	5364	3330	2484	2736	2088	198	0	0	21438
20-30	5904	4158	2916	1620	2376	3240	882	18	0	21114
10-20	8244	972	1620	1386	1890	3276	1044	18	0	18450
0-10	9828	54	1026	3222	1764	3294	900	0	18	20106
Total	32112	17010	17748	16200	13230	12942	3024	36	18	112320

Figure 2.3-4 (b) Distribution of Intermediate-Element Powers and Burnups for Natural Uranium Fuel in Bruce A

Power (kW/m)	Burnup (MWd/te)									Total
	0-2000	2000-4000	4000-6000	6000-8000	8000-10000	10000-12000	12000-14000	14000-16000	16000-18000	
50-60	0	0	0	0	0	0	0	0	0	0
40-50	660	2904	3552	1320	0	0	0	0	0	8436
30-40	3984	4272	3864	3288	1176	0	0	0	0	16584
20-30	6024	4536	2700	2112	1800	132	0	0	0	17304
10-20	5844	1788	1692	1560	3492	456	12	0	0	14844
0-10	7572	108	3048	2244	3816	900	24	0	0	17712
Total	24084	13608	14856	10524	10284	1488	36	0	0	74880

Figure 2.3-4 (c) Distribution of Outer/Intermediate Element Powers and Burnups for Natural Uranium Fuel in Bruce A

Power (kW/m)	Burnup (MWd/te)									Total
	0-2000	2000-4000	4000-6000	6000-8000	8000-10000	10000-12000	12000-14000	14000-16000	16000-18000	
50-60	288	2556	4266	3114	666	0	0	0	0	10890
40-50	3270	6810	8142	5694	3798	1044	0	0	0	28758
30-40	9222	9636	7194	5772	3912	2088	198	0	0	38022
20-30	11928	8694	5616	3732	4176	3372	882	18	0	38418
10-20	14088	2760	3312	2946	5382	3732	1056	18	0	33294
0-10	17400	162	4074	5466	5580	4194	924	0	18	37818
Total	56196	30618	32604	26724	23514	14430	3060	36	18	187200

2.3.1.7 AECL Experimental MOX Irradiation Experience

AECL commenced irradiation testing of (U,Pu)O₂ MOX fuel more than twenty-five years ago. Irradiation testing in the NRU reactor loops, and post-irradiation examination (PIE), is still being conducted as part of AECL's advanced fuel cycle program. The objectives of these experiments have varied from early interest in the ceramic properties of MOX fuel to the present multi-bundle demonstrations of the irradiation performance of Canadian-built fuel. The MOX fuel irradiated was fabricated both off-shore and by AECL in the Recycle Fuel Fabrication Facility at the Chalk River Laboratories (RFFL, described in Section 4).

The program has involved the irradiation of more than 350 fuel elements. Linear power ratings as high as 75 kW/m and burnup levels approaching 50 000 MWd/te have been attained. Plutonium concentrations in the fuel ranged from 0.5 to 3.0 wt% Pu in total heavy elements. Some defects have occurred from deliberate and severe power ramping of fuel bundles, in order to determine defect thresholds. Irradiation behaviour has generally been comparable to that of UO₂ under similar conditions.

Early MOX fuel irradiations in 1966 and 1968 tested fifteen fuel elements in the NRX reactor. The enrichment of the plutonium was between 0.5 and approximately 3 wt% Pu in (U,Pu)O₂. These tests were designed to investigate the effect of PuO₂ heterogeneity on fuel performance; plutonium migration and homogenization; and the effect of oxygen/metal ratio on the irradiation behaviour of MOX fuel.

In the mid-1970s, six 19-element fuel bundles with approximately 3 wt% Pu in (U,Pu)O₂ were irradiated in the NPD reactor at peak linear element ratings of 40-45 kW/m, declining to 15 kW/m at end-of-life when the maximum burnups were 18 000 MWd/te. One bundle remained in NPD for 14 years, before being discharged with an outer-element burnup of 49 000 MWd/te. Dimensional changes to the fuel were minimal, and the test was concluded without incident. This test demonstrated the excellent fuel performance of MOX fuel to high burnup, at power ratings up to 45 kW/m. In order to determine power-ramp thresholds for MOX fuel, one bundle, with high-density annular pellets, was power ramped from less than 20 kW/m to 55 kW/m; this severe power ramp produced some fuel defects. Two other bundles, with low-density pellets survived severe power ramps to 60 kW/m. When the power rating of these bundles was increased to 70 kW/m, one defect occurred.

In the early 1980s, six 37-element bundles containing an approximately 0.5 wt% fissile addition of plutonium were loaded into the NRU loops at Chalk River, for testing to extended burnup. These bundles were irradiated to burnups in excess of 21 000 MWd/te, with maximum linear element ratings of 68 kW/m at the start of irradiation. These bundles are currently being examined at Chalk River. A further six bundles are currently undergoing intermittent irradiation in the NRU loops. Initial examination has revealed no problems with this fuel.

To summarize, AECL irradiations have indicated that MOX fuel behaviour is generally comparable to that of UO₂ under similar conditions. PIE of MOX fuel with declining power histories to extended burnup has revealed no life-limiting parameters not previously recognized. The principal concern in extended-burnup fuel performance (whether MOX or

UO₂) is fission-gas release and resultant high internal gas pressure; this, of course, is more of an issue with the high-burnup advanced MOX design (see Section 2.3.2). Provided internal gas pressure can be kept low enough, there should be no fuel performance problems. The element internal design for reduced gas pressure has been addressed in both the reference and advanced MOX fuel designs.

Two areas of fuel performance requiring additional investigation are clear from the earlier examinations of MOX fuel irradiated at AECL: stoichiometry effects (i.e., oxygen/metal ratio) in irradiated MOX fuel, and the influence of the fissile particle or agglomerate size in dry-blended MOX. Further PIE is scheduled for irradiated MOX fuel during 1994/95 and this will add to our understanding of the irradiation effects of dry-blended material and of stoichiometry. While these issues need to be addressed so that suitable specifications can be written, it is expected that established pellet fabrication practices for commercial MOX fuel fabrication will assure adequate starting microstructure.

AECL's high-burnup irradiations in research reactors of natural uranium UO₂ (some to burnups over 30 000 MWd/te, where the plutonium content is about 0.5%; see Section 2.3.2.4) also constitute a form of MOX fuel experience, since most of the power generated in burnups over about 7 000 MWd/te comes from plutonium. The plutonium in this case is in solid solution with the uranium matrix, and is uniformly distributed (apart from the macroscopic variation across the pellet). Also, the power ratings in high burnup natural uranium fuel is relatively low. Nonetheless, this experience further confirms our expectation that the performance of MOX is comparable to that of UO₂.

World experience in the utilization of MOX fuel in LWRs is favourable. Issues being addressed in MOX utilization for LWRs are in the areas of reactor physics and core management, rather than fuel performance. The U.S.A., France, Germany, Belgium, Japan and other countries have operated extensive R&D programs on MOX fuel, while France, Germany and Belgium are routinely using partial LWR re-loads of MOX fuel. In addition, the FUGEN reactor in Japan has used MOX fuel with no reports of fuel performance problems. This world experience gives additional confidence in the use of MOX fuel in CANDU.

2.3.1.8 MOX Fuel Characteristics

Plutonium homogeneity. Uniformity of microstructure is an issue that can effect MOX fuel performance because of the possibility of "hot spots" in plutonium-rich regions. To ensure a uniform microstructure, most fabrication processes start with fine ceramic-grade powders (micron-sized particles). The UO₂ and PuO₂ powders are mixed together, often in a ball mill or other type of mill to break up agglomerates and ensure intimate mixing on the scale of the size of the particles themselves. To facilitate forming a uniform distribution, it is common to use a pre-mix stage (master blend) that which is high in PuO₂ content. This is milled, then blended with UO₂ to yield the correct final composition, and milled again to ensure uniformity.

A suitable fabrication route ensures that PuO₂ forms a solid solution with UO₂ at the concentrations found in MOX fuel. In spite of this, since the starting form of the PuO₂ is particulate, the regions in the vicinity of the original PuO₂ particles are richer in plutonium as the plutonium diffuses into the surrounding UO₂ matrix. This forms what are often termed Pu-

rich "particles". These "particles" do not have the size or composition of the feed PuO_2 particles. They are much larger regions of solid solution into which the Pu has diffused, and thus have a much lower Pu content than the original Pu particles. The diffusion process continues in-reactor to further homogenize the fuel. Alpha autoradiography measures these Pu-rich areas, not actual PuO_2 particles.

There has been much fabrication and in-reactor experience with MOX fuel. The pellet fabrication process for the CANDU MOX fuel would follow established practices that would ensure the required degree of homogeneity (see Section 4.4). Hence, ensuring an acceptable degree of plutonium homogenization in the MOX fuel fabrication route is not seen to be a problem.

Thermal conductivity. Another important parameter affecting fuel performance is thermal conductivity. The thermal conductivity of UO_2 has been thoroughly investigated. While several open literature papers discuss the effects of plutonium content on the thermal conductivity of MOX fuel, the plutonium content is usually 20% or greater, which is outside the range of interest for CANDU's. The fact that there are few measurements for a few per cent plutonium is largely because the data is not much different from UO_2 , and the effect of a few percent plutonium is often masked by other effects (such as porosity or stoichiometry). For the plutonium enrichments of CANDU MOX fuel, the UO_2 thermal conductivity can be used.

Fission yields. Many aspects of fuel behaviour depend on the presence and mobility of fission products generated within the fuel pellets. Table 2.3.2 shows the fission yields for several fission product isotopes corresponding to U-235 and Pu-239. The yields for the iodines and xenons are within a narrow 20% band between the two fissile sources, whereas the fission yields for the kryptons are about 60% lower for Pu-239 than for U-235. These findings are consistent with measurements, where the Xe/Kr ratios are much higher for MOX compared to UO_2 . To predict fission-product inventories, fractional releases to the gaps, etc., most fuel codes, including ELESTRES, rely on fixed values for the cumulative fission yields. In reality, the yields are functions of the amounts of fissile sources (U-235, Pu-239 and Pu-241), which vary with fuel burnup and with starting enrichments. To model MOX fuel, Pu-239 fission yields were used in ELESTRES.

Table 2.3.2 Comparison of Fission Yields for U-235 and Pu-239

Isotope	Cumulative Fission Yields* for Fissile Source		Ratio Pu-239/U-235	Yields Currently Used in ELESTRES
	U-235**	Pu-239***		
Xe-133	6.57	6.72	1.023	6.19
Xe-133m	0.193	0.234	1.212	0.19
Xe-135	6.59	7.45	1.131	7.2
Xe-138	7.65	4.84	0.633	6.17
Kr-85m	1.31	0.565	0.431	1.27
Kr-88	3.59	1.32	0.368	3.49
Kr-87	2.54	0.92	0.362	2.47
I-131	2.85	3.75	1.316	2.8
I-134	7.63	7.2	0.944	7.52

* Yields as predicted by ORIGEN and WIMS codes.

** Yields for U-235 are representative of natural uranium fuel at zero burnup.

*** Yields for Pu-239 are representative of natural uranium fuel at high burnup and MOX fuel at all burnups.

2.3.1.10 Reference MOX Fuel Design

The well-proven 37-element geometry was chosen for the design. Fuel composition was determined largely on the basis of ensuring that the power/burnup envelope was well within the current experience in Bruce A, as well as by meeting a disposition rate of around one tonne per year in one Bruce A reactor (hence 50 tonnes in 25 years in two Bruce A reactors). This set limits on the maximum plutonium composition, both in terms of the resultant burnup and peak linear element ratings. The relative plutonium enrichment in the intermediate (2%) and outer (1.2%) rings was chosen on the basis of balancing several performance objectives: minimizing the peak ratings for both fresh fuel and discharge fuel, and optimizing the thermalhydraulic performance of the bundle.

The use of dysprosium as a neutron absorber in the center elements has two benefits: more plutonium is required to achieve a given burnup (thus maximizing the plutonium disposition

rate), and the void reactivity is reduced.

Minor changes were made in the internal element design for the reference MOX fuel. These changes are within the current experience base for CANDU fuel. Some changes were made to enhance fuel performance with MOX fuel, by accommodating fission-gas release and reducing clad strains; eg. a smaller ratio of pellet length/diameter (L/D) for MOX pellets, a different shape of the chamfer on the pellet ends, a different dish depth and land width, and a slightly lower pellet density. Other changes were made to facilitate automated fabrication; eg. a slightly larger diametral clearance to facilitate automated loading of the pellets. In addition, different L/D ratios were used for each enrichment, to allow a visual means of confirming the enrichment of a pellet. The MOX fuel elements also contained a natural uranium pellet at each end of the stack, to reduce flux peaking at the end of the element. (The thermal flux increases between adjacent bundles, because the D₂O coolant and Zircaloy endcaps and endplates are less absorbing than the fuel.) The optimization of the pellet design was based on extensive experience in both irradiation testing, and fuel modelling.

2.3.1.11 Reference MOX Power/Burnup Histories

MOX fuel performance was assessed in several ways. Qualitative assessments are made in this section by showing that the MOX power histories resulting from the full-core reactor physics simulations fall within the natural uranium envelope. Power-ramp performance for MOX fuel is assessed by comparing the power- and power-boost envelopes for the MOX fuel derived from a fuel management simulation to the SCC threshold curves described in Section 2.3.1.5. Further confirmation of fuel performance for the reference MOX fuel is provided in the next section, by modelling the performance of the reference MOX fuel for the high-power envelope using the code ELESTRES.

The MOX power/burnup data was derived from full-core reactor physics calculations. "Time-average" calculations yield the average powers for every bundle position in the core. Another type of calculation performed is called "instantaneous", which provides a snapshot of the powers at a point in time. Finally, a 100-day "time-dependent refuelling simulation" was performed, in which the refuelling of individual channels was modelled to arrive at a better estimate of actual peak powers and power boosts. The starting point for this simulation was an instantaneous calculation.

The bundle average discharge burnup for the MOX reference fuel based on the time-average calculation is 9 700 MWd/te, compared to 8 300 MWd/te for Bruce A natural uranium fuel.

Figure 2.3.5 shows the distribution of peak element discharge burnup for the reference MOX fuel, resulting from the 100-day time-dependent refuelling simulation. This is similar to the distribution for natural uranium fuel (Figure 2.3.2). Both distributions have a double peak, again a result of the two radial burnup zones in the core that were used to flatten the channel power distribution. The median of the burnup peaks for the MOX fuel are higher than for natural uranium, because of the higher bundle average burnup and the larger spread in burnup across the MOX bundle (because of the dysprosium in the central seven elements). However, whereas the natural uranium core has a few bundles with peak element burnups greater than 14 000 MWd/te, none of the MOX fuel had burnups above that value.

Figure 2.3.5

Distribution of Peak Element Burnup in Discharged MOX Bundles

Figure 2.3.6

Snapshot of Intermediate- and Outer-Element Powers and Burnups for Reference
MOX Fuel in Bruce A

Figure 2.3.6 shows a typical snapshot of peak linear element rating and corresponding element burnup for the reference MOX fuel for an arbitrary point in time in the refuelling simulation. Each point in the plot corresponds to the peak element rating and corresponding element burnup for a bundle in the core. (The peak rating for the MOX fuel initially occurs in the outer ring of fuel; for element burnups greater than about 8 000 MWd/te, the peak rating shifts to the intermediate ring; see Section 2.2.) Superimposed on this snapshot is the natural uranium envelope from Figure 2.3.3. The power envelope for the reference MOX fuel is considerably lower than for natural uranium.

Figure 2.3.7 quantifies the information in Figure 2.3.6 in the form of a table that shows the distribution of peak element powers and corresponding element burnups. This can be compared to Figure 2.3.4 for natural uranium fuel. In the natural uranium core, only 9% of the intermediate and outer elements have burnups greater than 10 000 MWd/te, whereas for the MOX reference fuel the fraction is 20%. In the natural uranium core, 6% of the intermediate and outer elements have peak linear element ratings between 50 and 60 kW/m; in the MOX core, this number is less than 0.5%. Hence, the MOX core has a larger fraction of elements operating at higher burnups, and fewer elements operating at high power, compared to the natural uranium core.

The most important reactor physics determinant in fuel performance is the power history; i.e., the bundle (element) power as a function of burnup as it moves along the channel as a result of refuelling. With a two-bundle shift fuelling scheme, fuel that is initially loaded into bundle position 1 is moved to axial bundle position 3 upon the next visit of the fuelling machine to the channel. It then moves to positions 5, 7, 9, 11, and 13 with subsequent refuelling, before being discharged from the core. Likewise, fuel that is initially loaded into position 2 is subsequently shifted to positions 4, 6, 8, 10 and 12 before being discharged. Given a very long refuelling simulation, one can follow individual bundles along the channel. The 100-day refuelling simulation was, of course, not long enough for this, so representative power histories were obtained from the time-average axial power distributions. Table 2.2.3 lists the average element powers and burnups for a typical channel in the MOX fuelled core, normalized to the maximum channel power of 7 000 kW. The power histories for the intermediate and outer elements can be derived from that table by assuming that the power of a bundle in any axial location remains constant during the time spent in that position. For example, the power of the bundle in axial position 1 is constant until its burnup at that location is reached; upon refuelling, the bundle power increases to the average power at bundle position 3; once the burnup at position 3 is achieved, the channel is refuelled and the bundle assumes the power level at axial position 5, and so on. This procedure is repeated for the even-bundle positions. This is an extremely conservative measure of the maximum element power history, since in reality, any channel will be at the maximum power for only a short period of time. Generally, the power of a channel peaks during refuelling, and then decreases until it is refuelled again.

Figures 2.3.8 and 2.3.9 show the resultant maximum linear element power/burnup histories for the MOX reference case based on the time-average axial bundle power distributions, for

Figure 2.3.7 Distribution of Outer/Intermediate Element Powers and Burnups for Reference MOX Fuel in Bruce A

Power (kW/m)	Burnup (MWd/te)									Total
	0-2000	2000-4000	4000-6000	6000-8000	8000-10000	10000-12000	12000-14000	14000-16000	16000-18000	
50-60	135	585	30	0	0	0	0	0	0	750
40-50	3060	7605	7515	2190	0	0	0	0	0	20370
30-40	9840	12435	12705	17235	17610	8655	15	0	0	78495
20-30	10830	3540	3795	5955	8865	9615	2490	0	0	45090
10-20	6180	0	0	360	4620	4560	1320	0	0	17040
0-10	10875	0	0	30	2715	6330	5505	0	0	25455
Total	40920	24165	24045	25770	33810	29160	9330	0	0	187200

2-65

intermediate and outer elements, respectively. One general observation is that only relatively fresh fuel (burnup less than 2 000 MWd/te) experiences power boosting during refuelling. This is good from the perspective of SCC, since low burnup fuel is resilient to power boosts (since there are relatively few fission products, and the cladding is still ductile.) Figures 2.3.8 and 2.3.9 define a high-power envelope for the intermediate and outer elements. Again, this is a very conservative measure for fuel performance assessments, since very few elements would ever follow the high-power envelope. These power envelopes were used in the ELESTRES modelling, described in the next section.

Figure 2.3.10 compares the high-power envelopes for the intermediate and outer MOX elements derived in this fashion, with the power envelope resulting from the time-dependent refuelling simulation. The curve from the time-dependent refuelling simulation shows the peak element power, for either intermediate or outer rings. There is good agreement between the time-average results (normalized to a peak channel power of 7 000 kW) and the results from the time-dependent simulation. The peak ratings from the time-dependent refuelling simulation are greater than the time-average results up to a burnup of 6 000 MWd/te, and are generally lower at higher burnups. This is due to the power ripple in a channel upon refuelling. Also shown is the high-power envelope for natural uranium fuel for Bruce A. Again, the MOX envelope falls well below that of natural uranium.

Finally, an assessment was made of the power-boost performance of the reference MOX fuel, by comparing the power and power-boost envelopes derived from the time-dependent refuelling simulation for the reference MOX fuel to the threshold curves for SCC (Figure 2.3.1). Figure 2.3.11 shows the results. In the power-boost envelope for MOX fuel, the power boosts at low burnup are a result of power increases during refuelling as the bundles are shifted towards the center of the core. The power boosts at higher burnup are the result of the "refuelling ripple" from refuelling neighbouring channels. The MOX "ramped-power" and "power-boost" curves are both below the SCC threshold curves, and hence there is little likelihood of SCC failure. Both ramped-power and power-boost curves for the reference MOX fuel are below the curves for natural uranium for Bruce A.

These results give us confidence in the performance of the reference MOX fuel. The next section assesses the high-power MOX fuel performance using the ELESTRES fuel modelling code.

2.3.1.12 Reference MOX High-Power Performance Assessment

The ELESTRES code models the behaviour of CANDU fuel elements under normal operating conditions. The focus is on calculations of fission-gas release, internal and interfacial pressures, and pellet ridge strains. A single element is modelled by accounting for the radial and axial variations in stresses and displacements. The constituent equations are physically (rather than empirically) based, and include such phenomena as pellet-to-sheath heat transfer, temperature and porosity dependence of pellet thermal conductivity, burnup-dependent neutron flux depression, burnup- and microstructure-dependent fission-product gas release, and stress-, dose- and temperature-dependent constitutive equations for the sheath. The interactions and feedbacks among these parameters are considered in a dynamic manner throughout the irradiation history.

Figure 2.3.8

Reference MOX Intermediate-Element High-Power Envelope for a 2-Bundle Shift

Figure 2.3.9

Reference MOX Outer-Element High-Power Envelope for a 2-Bundle Shift

Figure 2.3.10

Comparison of Reference MOX High-Power Envelopes with Natural Uranium

Figure 2.3-11a: SCC Ramped-Power Assessment for Reference MOX Fuel

Figure 2.3-11b: SCC Power-Boost Assessment for Reference MOX Fuel

ELESTRES was used to compare the continuous high-power operation of the reference MOX fuel to that of natural uranium. Power histories corresponding to Figure 2.3.11 were used, with the appropriate fuel properties and pellet dimensions. As noted in Section 2.3.1.10, small changes were made to the internal design of the reference MOX bundle, to optimize its performance. A conservative (flatter) radial heat generation rate was used for the MOX pellets than would be expected, which will result in a small over-prediction of centerline temperatures and gas release.

Table 2.3.3 summarizes ELESTRES predictions for several important fuel performance parameters, for natural uranium and the reference MOX fuel in Bruce A. Predictions of total fission gas released to the "free inventory", maximum internal pressure and sheath strains for the reference MOX design are well below the values corresponding to the natural uranium high-power envelope. This is due to the lower peak element ratings for the MOX reference design, which reduces fission-gas release, and the optimized internal design of the MOX fuel elements, which reduces internal pressure and sheath strains.

It is emphasized that the high-power envelope does not represent the typical power history for either the natural uranium or the MOX fuel, but is rather a severe test of fuel performance. The design intent is to show that if an element can successfully operate along the high-power envelope, it can successfully operate at any power history below the envelope. The ELESTRES predictions provide good confidence in the performance of the reference MOX fuel.

2.3.1.13 Performance of Dysprosium Fuel

The above fuel performance assessments have focused on the MOX elements in the reference MOX design. The central seven elements contain 5% dysprosium mixed with depleted uranium. As shown in Section 2.2, these elements operate at a very low power level (less than 10 kW/m), and to a very low burnup (less than 3 000 MWd/te). There is extensive world experience with burnable poison fuel, and while much of this is with gadolinium, the chemical behaviour of gadolinium and dysprosium is similar. For linear element ratings < 25 kW/m there is no evidence of in-reactor gadolinium diffusion or of enhanced grain growth. Dysprosium is a fission product, and so is already present in irradiated CANDU fuel.

AECL is currently irradiating dysprosium fuel elements in NRU at Chalk River Laboratories. These irradiations cover a range of dysprosium concentrations, power levels, burnups, and fabrication routes. The first PIE of these elements will take place in the fall of 1994. No performance difficulties are anticipated, and the current program at AECL should provide the confirmation of acceptable fuel performance for the dysprosium fuel for both the reference and advanced MOX fuel designs.

2.3.1.14 Reference MOX Thermalhydraulic Performance

The objectives of the thermalhydraulic assessment are to determine the critical heat flux (CHF) for the reference MOX fuel design, and its impact on critical channel power (CCP), the channel power at which CHF first occurs on any fuel element in the channel. The neutron OverPower (NOP) system also referred to as the Regional Overpower Protection (ROP) system in CANDU reactors protects the fuel from experiencing sustained heat transfer degradation, resulting from overpowers that could arise from perturbations in the power distribution, or

Table 2.3.3 ELESTRES Predictions for Natural Uranium and
Reference MOX Fuels for High-Power Envelopes

	Natural U. Outer	MOX Ref. Intermediate	MOX Ref. Outer
Enrichment (%)	0.7	2	1.2
Pellet Design	Current	Optimized	Optimized
Coolant Pressure (MPa)	9.9	9.9	9.9
Peak Element Power (kW/m)	60	49	53
Final Burnup (MWd/te)	12 600	12 300	12 500
Gas release (mm ³) *	39 000	14 000	17 000
Gas release (%) *	19	7	9
Gas on grain boundaries (%) *	16	14	13
Gas within grains (%) *	65	79	78
Internal gas pressure (MPa) **	11	3	4
Interfacial pressure (MPa) **	32	20	21
Total ridge strain (%) **	1.5	0.4	0.4

* End-of-life values.

** Peak values during the irradiation.

from a slow loss-of-reactivity-control power transient. The NOP system prevents fuel sheath "dryout" following a gradual increase in either local or bulk neutron power, by limiting channel powers to below the CCP. The CCP values are used to establish the neutron flux detector trip settings for the NOP system of the reactor.

Dryout is a term used to denote a process of progressive degradation of heat transfer from the fuel sheath. This degradation is associated with the change in heat transfer mode from forced convection nucleate boiling, through transition boiling, to stable film boiling. In CANDU reactors dryout typically occurs under two-phase coolant boiling conditions and is characterized by the progressive spread of a film-boiling drypatch on a fuel element, with a resultant increase in the temperature of the sheath region subjected to film boiling. The broadly analogous terms used for thermal limits in light water reactors are "boiling crises", "departure from nucleate boiling (DNB)" and "burnout", however, burnout behaviour is not typically exhibited in CANDU dryout heat transfer.

In CANDU analyses, the CCP of a channel under overpower conditions is calculated assuming nominal (i.e. 100% full power) constant header-to-header pressure drop, inlet header temperature and outlet header pressure corresponding to steady state operation at a specified power level and heat transport flow operating configuration. A review of all the channels for the current Bruce A design indicates that channel O16, located in the inner zone of the primary heat transport system (PHTS), is the channel with the maximum power, lowest flow and maximum exit quality for nominal operating conditions at 100% full power. Channel O16 is thus the limiting channel with the lowest CCP, and was used for the CCP assessment.

The steady-state thermalhydraulics code NUCIRC is used in calculating CCP for use in NOP design and performance analysis, and in licensing submissions. The NUCIRC code calculates individual channel flows, cross-sectional averaged physical properties of the D₂O coolant at various locations in the PHTS, and dryout power for any number of channels. NUCIRC uses a flux-corrected, cross-sectional averaged CHF correlation that was derived from empirical data obtained on electrically-heated, Bruce-type, 37-element fuel bundle simulators operating at reactor conditions of interest. This CHF correlation is used to calculate CCP's. However, this CHF correlation cannot account for the different radial and axial power profiles for the MOX fuel relative to natural uranium.

These effects were accounted for in the subchannel code, ASSERT, which calculates conditions for individual flow zones or subchannels in a fuel bundle. ASSERT accounts for cross mixing between adjacent subchannels, cross flows generated by pressure differences between the subchannels, and the local heat flux in the fuel. ASSERT uses tube-CHF look-up tables to predict CHF in the subchannels, based on these local conditions. These tables present empirically based CHF for discrete values of pressure, mass flux, and quality.

ASSERT was used to calculate the dryout powers and the corresponding pressure drops, for several constant dryout flow rates for the 37-element natural uranium and reference MOX bundles. Correction factors were then applied to the two-phase friction multiplier and CHF correlations in NUCIRC to reproduce the ASSERT values of dryout power and pressure drop at the same constant flow conditions as used in ASSERT. Since in-reactor the head-to-header pressure drop remains constant, and not the channel flows, NUCIRC was then run with

constant header-to-header pressure drop, inlet header temperature and outlet header pressure to calculate CCP.

ASSERT Calculation of CHF

ASSERT simulations for the natural uranium and reference MOX bundles provided predictions of CHF and pressure drop for a range of constant dryout flows. Representative axial fluxes along the channel and radial heat fluxes through the bundle were used for natural uranium and MOX fuel. For a dryout flow of 16.9 kg/s, the dryout power (or CHF) for the reference MOX fuel (9625 kW) was 1.1 % higher than for natural uranium (9522 kW), while the pressure drop was 3.7% higher (593 kPa compared to 572 kPa).

This improvement in CHF is the net effect of the different axial power distribution along the channel, and the radial power distribution through the bundle for MOX fuel compared to natural uranium. Table 2.3.4 shows these separate effects. The steeper power distribution through the reference MOX bundle reduces CHF by 8% compared to natural uranium. The axial power distribution for the reference MOX fuel with the 2-bundle shift peaks at the inlet end of the channel, and results in a lower relative power at the dryout location at the downstream end of the channel; this increases CHF by 6% compared to the natural uranium axial power shape. However, the inlet-end skewed axial power distribution also leads to a higher level of boiling, and hence a higher pressure drop.

Table 2.3.4 ASSERT Calculation of CHF (dryout at constant flow)
for Natural Uranium and Reference MOX Fuel

	Axial power: NU Radial power: NU	Axial power: NU Radial power: MOX	Axial power: MOX Radial power: NU	Axial power: MOX Radial power: MOX
Dryout power (kW) at constant flow	9 522	8 812	10 050	9 625
Total Pressure Drop (kPa)	572	526	621	593

The initial location of dryout predicted by ASSERT is bundle position 10 or 11, counting in the direction of flow, for both the natural uranium and MOX fuel. With natural uranium fuel, dryout first occurs predominantly on either the inner or intermediate ring of rods, while in the reference MOX fuel, ASSERT predicts dryout first occurring always on the intermediate ring of rods.

To calculate the CCP, the effect of the increase in both CHF and pressure drop on the dryout power at constant header-to-header pressure drop (rather than constant flow), must be determined. NUCIRC simulations of the same ASSERT cases were first performed to derive

the correction factors to the two-phase friction multiplier and CHF correlations used in NUCIRC, thus permitting evaluation of CCP for each case.

NUCIRC Calculation of CCP

A comparison of the flow-power characteristics predicted by ASSERT and NUCIRC for 37-element fuel strings with natural uranium and MOX fuels over a range of critical flows typical of Bruce A channels enabled correction factors to be calculated for the cross-sectional averaged two phase pressure drop multiplier and CHF correlations. These corrections resulted in consistent calculations of dryout powers for the natural uranium and MOX fuels for ASSERT and NUCIRC, at constant flow. NUCIRC was then used to calculate the CCP at constant header-to-header pressure drop for channel O16, for natural uranium and the reference MOX fuel. The nominal conditions at 100% full power and at dryout predicted by NUCIRC for channel O16 are presented in Table 2.3.5.

Table 2.3.5 NUCIRC Calculation of CCP for Natural Uranium
and Reference MOX Fuel

	Natural Uranium	Reference MOX
Nominal Flow (kg/s)	25.32	24.95
Nominal Power (kW)	6535	6536
Exit quality (%)	0.07	0.38
Dryout Flow (kg/s)	17.45	16.99
Dryout Power, CCP (kW)	9187	9166

The axial flux shape for the MOX fuel is skewed toward the inlet end of the channel, leading to slightly longer boiling length in the fuel channel, a higher pressure drop, and consequently, lower flow for a constant header-to-header pressure drop. In principle, a slight reduction in core flow and slightly higher levels of bulk boiling are expected for the PHTS with MOX fuel. NUCIRC predicts the same dryout power for the reference MOX and natural uranium cases, within the uncertainty of the calculation.

Conclusions

ASSERT predicts that both the CHF and pressure drop are higher for the reference MOX fuel than for natural uranium. However, within the accuracy of the calculation, the NUCIRC prediction of CCP is essentially the same in both cases. Therefore, based on the results of this scoping study, no change to the NOP trip setpoints will be required for the reference MOX fuel.

2.3.2 Advanced MOX Fuel Design

2.3.2.1 Overview

Section 2.3.2 assesses the fuel performance of the advanced MOX fuel design. This bundle has an average burnup (17 100 MWd/te) that is nearly double that of the reference MOX design. The higher burnup reduces the required MOX fuel fabrication requirements for a given plutonium disposition rate by nearly half. This would have significant economic advantages (in terms of MOX fabrication, transportation, and disposal), with the additional benefit, in the alternative case of disposing of 100 tonnes of plutonium in 25 years, of being able to use the same fuel fabrication facility as for the reference case.

The advanced MOX fuel design is based on CANFLEX, a 43-element bundle with two element sizes. This has about 20% lower peak power ratings than the 37-element bundle, which facilitates the achievement of extended burnup. The CANFLEX bundle and development program is described in the following section.

The CANDU extended burnup experience is discussed in Section 2.3.2.3, both in terms of power reactor experience in Bruce A, and in terms of research reactor irradiations. The data base supporting extended burnup operation for CANDU fuel includes over 200 bundles that have achieved burnups greater than 17 000 MWd/te in Bruce A, and 25 UO₂ bundles and 15 elements that have been irradiated to burnups greater than 17 000 MWd/te in Canadian research reactors. While the data generally indicates good fuel performance at extended burnup with the current 37-element fuel design, there appears to be a power/burnup threshold above which defects are observed, associated with increases in diametral strains and fission-gas release. This is addressed in the advanced MOX design by reducing the peak ratings relative to the 37-element bundle, thereby reducing gas release, and by optimizing the internal design to accommodate the fission gas.

The power/burnup histories for the advanced MOX fuel design are discussed, and compared to natural uranium in Bruce A. The design changes indicated for the advanced MOX design are sufficient for extended burnup operation, as confirmed by modelling the fuel performance for the high-power envelope. Fission-gas release is shown to be below that for natural uranium fuel. Hence, we have good confidence that the advanced MOX fuel design will achieve the same excellent fuel performance as the current 37-element bundle with natural uranium fuel.

The section concludes with an assessment of the thermalhydraulic performance of the bundle.

2.3.2.2 CANFLEX Fuel Design

The CANFLEX bundle is a 43-element geometry, with two element diameters (see Figure 2.2.8). The elements are arranged in rings of 7, 14 and 21, around the central element. The central eight elements are larger (nominal clad diameter of 13.5 mm) than the outer 35 elements (nominal clad diameter of 11.5 mm). For comparison, the nominal clad diameter for the 37-element bundle is 13.1 mm. The greater number of fuel elements and the two element sizes reduce peak element ratings by about 20% relative to the 37-element bundle. The improvement in linear element rating applies to a range of fuel types, including natural uranium, slightly enriched uranium, and the MOX fuel design for this application. The lower rating facilitates the achievement of extended burnup in CANDU by lowering the central fuel temperature, and hence fission-gas release. Fission-gas release exhibits an exponential dependence on fuel power. At peak powers of less than 40 kW/m, fission-gas releases are low (< 1%). Between 40 and 50 kW/m, fission-gas releases are generally in the range of 1-10%; at powers in the range of 50-60 kW/m, releases of up to 26% have been observed for CANDU fuel.

The second major feature of the CANFLEX bundle is thermalhydraulic enhancement. CHF-enhancing devices have been devised and are currently being tested. The objective is to achieve an improvement in critical channel power of between 5 and 6% for the CANFLEX bundle.

The CANFLEX bundle has been designed to be compatible with existing fuel handling systems in the CANDU 6 reactor. Minor design modifications (such as the location of the bearing pads, and endcap profile) will be required to make the bundle compatible with Bruce A.

The program to develop and demonstrate the CANFLEX bundle was started by AECL in 1986. In 1991, KAERI (the Korean Atomic Energy Research Institute) joined the program. KAERI is focusing on fabricating bundles for flow and endurance tests. The AECL focus is on the thermalhydraulic optimization of the bundle, and the in-reactor demonstration of the bundles in the NRU reactor. The current program is aimed at demonstrating the bundle to the point that would enable a power reactor demonstration irradiation by 1996. The major program elements over the next three years are as follows:

1994/95

- Complete the irradiation of the first CANFLEX bundle to natural uranium burnup in NRU. Initiate the irradiation of several other CANFLEX bundles in NRU. These will be taken to extended burnup, and four of these bundles will have a composition and burnup similar to the advanced MOX design (with enriched uranium rather than plutonium).
- Complete handling tests and fuelling machine compatibility tests.
- Finalize the location and design of CHF-enhancement appendages on the bundle, and finalize the design drawings.
- Complete fabrication of 50 natural uranium CANFLEX bundles (KAERI) for flow, endurance and handling tests, and for reactor

physics measurements.

1995/96

- Complete flow and endurance tests.
- Initiate water CHF test.
- Complete ZED-2 reactor physics measurement of reactivity coefficients and reaction rates.

1996/97

- Complete water CHF test.
- Issue final reports on CANFLEX irradiation, reactor physics measurements, thermalhydraulic tests, and safety and licensing.

2.3.2.3 Bruce A Extended Burnup Experience

Section 2.3.1.6 dealt with the natural uranium Bruce A operating experience that was applicable to the lower burnup reference MOX fuel design. The advanced MOX fuel design has a bundle average discharge burnup of 17 100 MWd/te, and a maximum element discharge burnup of about 32 000 MWd/te. This section discusses the extended burnup UO₂ experience in Bruce A.

Approximately 230 thirty-seven element bundles have achieved burnups in excess of 17 000 MWd/te in the Bruce reactors. The majority of these bundles have not been inspected in the irradiated fuel bay or examined in hot-cells, because they showed no evidence of having failed in-reactor. (Bruce reactors have on-power failed-fuel monitoring systems). Hence, it is expected that the majority of these bundles performed without failure. The power/burnup histories of about 65 of these bundles have been used to extend the natural uranium power envelope to higher burnup. The resultant high-power envelope extended to high burnup is shown as a heavy line in Figure 2.3.12.

Of those bundles exceeding a bundle-average burnup of 17 000 MWd/kg in Bruce, seven have undergone PIE at Chalk River. Four of those bundles (J24533C, J24546C, J98315C and J98324C), had high gas release (around 24%), and fuel failures in the outer elements. The failures had the characteristics of SCC, although the bundles had not undergone power ramps. One other bundle (J03311W) had high gas release, but no fuel failures. Bundle F04857C experienced lower powers, and had much smaller gas release (3-10%), and no failures. Figure 2.3.12 superimposes the power histories of these bundles on the high-power envelope from Bruce A. Generally, the power histories of those bundles that had defects crossed above the high-power envelope for part of their operating history.

Evidence supports the existence of a power/burnup threshold above which a degradation of fuel properties occurs, leading to high fission-gas release, high sheath strains, and possibly fuel failures. Currently, there is not enough data to define a defect threshold for high-burnup CANDU fuel. Fuel performance is dependent not only on the power/burnup history, but is

very sensitive to details of the element design, such as the endcap weld geometry, the shape and density of the pellets, and the internal void in the element. Nonetheless, the high-power envelope for natural uranium fuel will be useful as a guide in assessing the power envelope for the advanced MOX fuel. Since this envelope applies to the current element design, optimization of the internal element design will enable operation at a higher power/burnup level than indicated by this envelope. Changes to the internal design of the element that reduce fission-gas release at high burnup and enhance fuel performance have been identified, and will be necessary for the advanced MOX design. Essentially the same changes have been incorporated into both the reference and advanced MOX designs.

2.3.2.4 Experimental Extended Burnup Experience

Most of the experimental UO_2 irradiations in AECL research reactors have involved slightly enriched uranium UO_2 , so that power histories similar to those of natural UO_2 fuels in power reactors could be obtained. A large proportion of these have been specific to natural uranium burnups, around 8 000 MWd/te. Table 2.3.6 lists the bundles and elements relevant to extended burnup fuel performance in CANDU, which we have arbitrarily defined as $\geq 17\,000$ MWd/te. This AECL experience constitutes 25 bundles and 15 elements. Many of the bundles are still undergoing examination and documentation at AECL, while some are still undergoing irradiation.

Ten of the bundles (BDL-406) were natural- UO_2 fuelled Bruce production bundles irradiated as "filler" bundles in the NRU U-1 and U-2 loops when insufficient experimental bundles were available. These bundles had peak linear element ratings < 44 kW/m, and have been irradiated to outer element burnups exceeding 37 000 MWd/te, with no indications of any life limiting attributes. The NDP-56 experiment consisted of ten 19-element bundles without CANLUB sheath coatings, irradiated in the NPD reactor to outer element burnups around 33 000 MWd/te at peak ratings < 34 kW/m. Fission-gas release for the elements punctured and sampled was $< 0.3\%$ of that formed. No defects were revealed by PIE. Hence, for low-power irradiations, there is no apparent heat-transfer deterioration leading to enhanced gas release.

Experiment NRU-229 was particularly interesting. It was an early prototype test irradiation of several 37-element bundles. One bundle was taken to a maximum burnup of about

29 000 MWd/te, with a linear power output in the range 60-80 kW/m. PIE indicated maximum diametral expansions at pellet interfaces of 0.9%, and maximum fission gas release by the outer elements of 13% of that formed. Metallographic and ceramographic examinations revealed no unusual features.

Many of the other irradiations investigated specific features of the element design that led to improved power-ramp (SCC) performance. Included were tests of various CANLUB coatings and cladding materials, different fuel densities, and different pellet geometries. For example, one bundle in experiment BDL-416 was power ramped from 40 to 70 kW/m at an outer-element burnup of 6 300 MWd/te, then successfully irradiated to a burnup of 26 200 MWd/te. In experiment BDL-401, different pellet geometries were tested. The bundle was subjected to a power ramp from 28 to 55 kW/m at an outer-element burnup of 7 500

Figure 2.3.12

Bruce Power/Burnup Histories

Table 2.3.6 Experimental Natural And Slightly Enriched
Uranium Fuel Irradiations

Experiment	Bundle or Element	Outer Element Linear Power, kW/m	Outer Element Burnup, MWd/te
BDL-406	AAH	39	17 500
BDL-406	GC	44	18 300
BDL-406	GE	44	17 500
BDL-406	GF	44	37 500
BDL-406	XY	36	20 400
BDL-406	ZR	41	22 900
BDL-406	AAM	41	16 700
BDL-406	ZS	39	29 200
NPD-56	10 Bundles	28-34	29 200 - 33 300
BDL-416	AAW	72	26 300
NRU-229	JC	66	29 200
BDL-400	PY	68	16 300
BDL-401	HR	55	17 900
BDL-413	ZY	69	20 000
BDL-413	ZX	72	21 700
DME-190	4 Elements	55-65	1 > 19 000
		55-65	3 > 21 000
DME-191	8 Elements	55-65	3 > 19 000
		55-65	5 > 21 000
DME-195	3 Elements	55-65	3 > 21 000

MWd/te, and then successfully irradiated to an outer-element burnup of 17 000 MWd/te. Experiment BDL-413 tested two different CANLUB coatings. One bundle was power ramped at moderate burnup (4 000 MWd/te), and ramped again at 17 000 MWd/te without defecting. The bundle continued irradiation at a declining power to a final outer element burnup of 22 000 MWd/te. The demountable element experiments DME-190, -191 and -195 again proved the SCC-tolerance of various CANLUB coatings.

The research reactor irradiations of extended burnup CANDU fuel show that irradiations to high burnup can be achieved if peak ratings are kept below about 40 kW/m. In this case, gas release and sheath strain will be small. At higher ratings, means of accommodating enhanced fission-gas release must be incorporated into the element design for extended burnup operation.

Hence, while the Bruce high-burnup experience suggests a power/burnup threshold for current CANDU fuel above which a deterioration of fuel performance occurs, the experimental irradiation data base also demonstrates design features that can be incorporated to facilitate the achievement of high burnup. These design features are part of the element design of both the reference and advanced MOX fuels.

2.3.2.5 Advanced MOX Fuel Design

The choice for the advanced MOX fuel design was dominated by high-burnup performance considerations. It is recognized that this option will take longer to demonstrate and implement than the less-demanding reference case, but the timescale for implementation would be consistent with a natural transition from the reference to the advanced MOX fuel designs in Bruce A. As in the reference MOX fuel design, the relative plutonium enrichment in the intermediate (3.6%) and outer (2.1%) rings was chosen on the basis of balancing several performance objectives: minimizing the peak ratings for both fresh and discharge fuel, and optimizing the thermalhydraulic performance of the bundle.

The choice of the CANFLEX geometry allowed a 20% reduction in peak linear element ratings over that achievable with the same plutonium composition and burnup with the 37-element geometry. Since fission-gas release at ratings over 40 kW/m increases exponentially with increasing power rating, the lower ratings with CANFLEX are an important feature for the powers and burnups of interest. The resultant peak ratings (for the highest rated channel) are lower than 50 kW/m, and most of the core operates at lower than 40 kW/m. To further enhance the fuel performance of the advanced MOX fuel at extended burnup, optimization of the internal element design was carried out, as in the reference MOX case.

2.3.2.6 Advanced MOX Power/Burnup Histories

The bundle average discharge burnup for the advanced MOX fuel based on the time-average calculation is 17 100 MWd/te, compared to 8 300 MWd/te for Bruce A natural uranium fuel and 9 700 MWd/te for the reference MOX fuel. The estimated peak element discharge burnup is about 32 000 MWd/te.

Figure 2.3.13 represents a snapshot of the peak linear element ratings and corresponding

element burnups for every bundle in the core at an arbitrary point in time. It is not perfectly representative, since in this method no bundle has a burnup greater than the time-average discharge burnup (whereas, in reality, some bundles would have burnups higher than time average). Nonetheless, it is indicative of the power/burnup distribution for the advanced MOX fuel. The corresponding high-power envelope is also shown. This was used in the ELESTRES modelling, described in the next section. Also shown is the natural uranium high-power envelope from Figure 2.3.12. The MOX power/burnup values fall well below the natural uranium high-power envelope.

The peak linear element ratings for the MOX fuel are lower than 50 kW/m, and hence below the power threshold curve for SCC defects shown in Figure 2.3.1. Therefore, power boosting resulting from refuelling a channel is not expected to result in SCC failures for the advanced MOX fuel design. Power boosts at higher burnups, resulting from the refuelling "ripple" of fuelling neighbouring channels, or from the movement of reactivity devices (such as the water level in the liquid zone controllers), will be small, and are not expected to pose a problem.

Figure 2.3.14 shows the distribution of outer- and intermediate-element powers and burnups for the advanced MOX design, corresponding to the snapshot above. This can be compared to the distribution of power/burnups for natural uranium in Figure 2.3.4, and for the reference MOX case in Figure 2.3.7. In the advanced MOX fuel design, 96% of the intermediate and outer elements have powers lower than 40 kW/m, and hence would experience low gas release. Fuel elements with burnup greater than 24 000 MWd/te have ratings below 30 kW/m.

2.3.2.7 Advanced MOX High Power Performance Assessment

ELESTRES was used to compare the continuous high-power operation of the advanced MOX fuel to that of natural uranium. The same high-power history was used for both intermediate and outer elements of the advanced MOX design (from Figure 2.3.13), and the results were almost identical for both. Appropriate fuel properties and pellet dimensions were used for the advanced MOX fuel. Table 2.3.7 compares the ELESTRES results for the advanced MOX fuel to those for natural uranium in Bruce A. Predictions of gas release, internal pressures and sheath strains are all lower for the advanced MOX design than for natural uranium. This is due to the lower peak element rating and optimized internal pellet design for the MOX fuel elements. The internal gas pressure for the advanced MOX design is well below coolant pressure.

Figure 2.3.13

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

Figure 2.3-14: Distribution of Outer/Intermediate Element Powers and Burnups for Advanced MOX Fuel in Bruce A

Power (kW/m)	Burnup (MWd/te)													Total
	0-2000	2000-4000	4000-6000	6000-8000	8000-10000	10000-12000	12000-14000	14000-16000	16000-18000	18000-20000	20000-22000	22000-24000	24000-26000	
50-60	35	140	35	0	0	0	0	0	0	0	0	0	0	210
40-50	420	1855	2870	2170	700	0	0	0	0	0	0	0	0	8015
30-40	3185	5320	6370	6825	9800	10430	9660	6300	1890	1085	35	0	0	60900
20-30	4690	4095	3675	4725	4830	5110	6580	10535	13055	12950	9765	5005	490	85505
10-20	5040	2450	945	490	315	210	315	945	3605	6650	7175	5075	1225	34440
0-10	9730	1645	0	0	0	0	0	0	770	4130	8540	2030	2485	29330
Total	23100	15505	13895	14210	15645	15750	16555	17780	19320	24815	25515	12110	4200	218400

The high-power envelopes are similar up to the nominal discharge burnup of the intermediate elements in the reference MOX fuel (12 500 MWd/kg), and the resultant fuel temperatures and fractional gas releases into the "free inventory" are similar up to that burnup (about 7% of the total gas inventory in both cases). However, the absolute gas release for the advanced MOX fuel at 12 500 MWd/te is lower by the ratio of the volumes of the fuel stacks (1.28; the outer elements in the advanced MOX design are smaller than those of the 37-element bundle). The advanced MOX fuel releases virtually no more fission gas for the rest of its burnup history, since its power ratings are lower than 40 kW/m. Hence, the fractional gas release at end-of-life for the advanced MOX fuel is half of that at mid-burnup (since the gas inventory has doubled, and there has been no further release), and half of that for the reference MOX fuel. Even though the burnup (and total gas inventory) at end-of-life for the advanced MOX is double that of the reference MOX fuel, the absolute gas release is lower. This shows the importance of lowering peak element ratings in extended burnup applications. With ratings below about 40 kW/m, the fission gas is contained largely within the grains.

Again, the high-power envelope used for this assessment is a conservative measure of fuel performance. As mentioned earlier, at any point in time, 96% of the advanced MOX fuel will have ratings lower than 40 kW/m, and hence, very low gas release.

In conclusion, the combination of lower peak ratings and optimized internal element design should ensure good fuel performance for the advanced MOX fuel.

Table 2.3.7 ELESTRES Predictions for Natural Uranium and Advanced MOX Fuels
for High Power Envelopes

	Natural U. Outer	Advanced MOX Intermediate/ Outer
Enrichment (%)	0.7	3.6 / 2.1
Pellet Design	Current	Optimized
Coolant Pressure (MPa)	9.9	9.9
Peak Power (kW/m)	60	51
Final Burnup (MWd/te) *	12 600	25 000
Gas release (mm ³) *	39 000	10 700
Gas release (%) *	19	4
Gas on grain boundaries (%) *	16	11
Gas within grains (%) *	65	85
Internal gas pressure (MPa) **	11	3
Interfacial pressure (MPa) **	32	12
Total ridge strain (%) **	1.5	0.2

* End-of-life values.

** Peak values during the irradiation.

2.3.2.8 Advanced MOX Thermalhydraulic Performance

As in the reference MOX fuel case (Section 2.3.1.14), a two-step process was used to calculate the CCP for the advanced MOX fuel. First, ASSERT was used to calculate the CHF and corresponding pressure drop for a range of constant dryout flows for the advanced MOX fuel (which has the CANFLEX geometry). These results were used to derive correction factors in NUCIRC to account for the geometry, and the axial and radial power profiles for the

advanced MOX fuel.

While CANFLEX will include CHF-enhancement appendages, these calculations were performed without such devices; standard bearing pad and spacer locations were assumed on the advanced MOX bundle.

ASSERT Calculation of CHF

ASSERT simulations for the advanced MOX bundles provided predictions of CHF and pressure drop for a range of constant dryout flows. Nominal axial fluxes along the channel and radial heat fluxes through the bundle were used for the advanced MOX fuel. For a dryout flow of 16.9 kg/s, the dryout power (CHF) predicted for the advanced MOX fuel (9315 kW) was 2% lower than for the natural uranium 37-element bundle (9522 kW), while the pressure drop was 15% lower (496 kPa compared to 572 kPa), due to the higher flow area in the CANFLEX bundle, and slightly lower total resistance. The initial location of dryout predicted by ASSERT for the advanced MOX bundle is bundle position 12, counting in the direction of flow, in the intermediate ring of elements.

To determine the effect on CCP, the effect of the decrease in both CHF and pressure drop on the dryout power at constant header-to-header pressure drop (rather than constant flow), must be determined. NUCIRC simulations of the same ASSERT cases were performed to derive the correction factors to the two-phase friction multiplier and CHF correlations used in NUCIRC, thus permitting evaluation of CCP for each case.

NUCIRC Calculation of CCP

The nominal conditions at 100% full power and at dryout predicted by NUCIRC for advanced MOX fuel in channel O16 are presented in Table 2.3.8. NUCIRC predicts a slightly greater (2%) CCP for the advanced MOX compared to the 37-element natural uranium reference case. Until definitive measurements of pressure drop and CHF are made for the CANFLEX bundle, it is prudent to conclude that the CCP for the advanced MOX bundle is the same as for the reference 37-element bundle, within the accuracy of the calculation.

Table 2.3.8 NUCIRC Calculation of CCP for Advanced MOX Fuel

	Natural Uranium (37-element)	Reference MOX (CANFLEX)
Nominal Flow (kg/s)	25.32	26.53
Nominal Power (kW)	6535	6536
Exit quality (%)	0.07	-0.09
Dryout Flow (kg/s)	17.45	17.78
Dryout Power, CCP (kW)	9187	9380

Conclusions

ASSERT predicts that both the CHF and pressure drop are lower for the advanced MOX fuel than for the natural uranium 37-element bundle under constant flow conditions. Within the accuracy of the calculation, the CCP calculated by NUCIRC for constant pressure drop conditions, is the same in both cases. It should be noted that the CANFLEX bundle is being designed to include CHF-enhancement appendages, and these would also be applied to the advanced MOX bundle. An estimated increase in CHF of 28% for natural uranium fuel, due to the appendages, more than compensates for the slight pressure drop penalty incurred by the addition of buttons. The estimated increase in CCP for the natural uranium CANFLEX bundle with CHF enhancement is in the range of 5-6%, which would add significant extra margin to the NOP trip setpoints. However, CHF enhancement has not been included in this assessment for the advanced MOX fuel. It is anticipated that this would increase the CCP by several percent over the existing natural uranium 37-element bundle without CHF enhancement.

2.4 Plant Changes

Engineering changes in the existing CANDU plants that may be required to receive, store and handle the MOX fuel were assessed, and are discussed below. This review was done with assistance from Ontario Hydro personnel at Bruce A, and with the intent of ensuring that the conversion of Bruce A units to MOX fuel does not result in any negative effects on station operability. From the standpoint of engineering changes to the plant, the largest item is the need for enhanced security for the MOX fuel. For cost information associated with these changes see section 2.9 and chapter 9.

2.4.1 Receipt and Storage of New Fuel

New Fuel Storage Building: A secure facility for receiving, inspecting and storing new fuel is required. Existing facilities and procedures used for natural uranium fuel are inadequate for mixed oxide (MOX) fuel, mostly because of the low level of security required for natural uranium fuel. An approximate size and layout for this new building was established, based on a requirement to maintain a minimum fuel inventory for two units for one month. A location inside the main station building was evaluated, but judged too disruptive to essential station maintenance operations. A possible location for a new building is shown in figure 2.4.1, and the route for fuel transfer to the new fuel loading room is indicated.

The new fuel storage building will be an IAEA Category 1 facility, with reinforced concrete floor, walls and roof, controlled access entrances, alarms and other security features described in section 2.6.4.

New Fuel Loading Room: This room contains the equipment that loads fuel into the fuelling machines, which carry it to the reactors. Because it will contain a minimum inventory of one day's fuel for two units (31 bundles, containing 7.13 kg Pu), it will be hardened to Category 1 requirements, with additional access control and security provisions.

Occupational Radiation Exposure: Radiation fields from new MOX were compared to fields from natural uranium fuel, and measures required to protect personnel inspecting and handling new fuel were assessed. Gamma fields from MOX fuel are about four times higher than for natural uranium fuel (20 mrem/h contact for MOX fuel, 5 mrem/h contact for natural uranium fuel - ref section 4.1.8). In addition to the gamma field, the MOX fuel will have a 5 mrem/h neutron field. This analysis assumed no removal of Americium during fuel fabrication.

MOX fuel bundles will be placed in shielding sleeves (water extended polyester surrounded by stainless steel) at the fabrication plant, and left in these until being loaded into the reactor fuelling machine. This will allow MOX fuel to be handled with radiation doses to personnel equivalent to, or lower than natural uranium fuel. The shipping containers and fuel storage facilities will provide additional shielding. In the new fuel loading machine, a ram will push the bundle out of the sleeve and into the machine's magazine.

Additional local shielding, simple equipment that helps the worker maintain an arms length distance from the fuel, and a bar code reader or video camera to read the bundle serial number will help minimize exposure.

Criticality: A criticality analysis of 7 new MOX fuel bundles (without shielding sleeves) in the 55 gallon drum, filled with water, shows that there is no danger of criticality. New fuel will be stored and handled at the plant in shielded containers to eliminate any danger of criticality.

2.4.2 Reactor Operation

Fuelling Rate: The fuelling rate of MOX fuel was compared to that of natural uranium fuel and the ability of the fuel handling system to meet the required fuelling rate was assessed. The fuelling rate for MOX fuel is 15.5 bundles/day per unit, lower than the rate for natural uranium fuel (18 bundles/day). The fuel handling system is capable of fuelling reactors with MOX fuel.

New Fuel Loading: All new fuel loading machines will be modified to accommodate MOX fuel in shielding sleeves. If natural uranium fuel is in use on one or more units, it will be loaded using shielding sleeves in the same way as MOX fuel.

Fuelling Machines: A fuelling machine carries 16 bundles at a time in a magazine flooded with heavy water. This has been assessed and there is no danger of criticality.

Radiation fields and thermal heat generation associated with high burnup MOX fuel are very close to those of natural uranium fuel taken to a similar burnup. No modification of fuelling machines is necessary.

Reactor Control: Additional requirements for reactivity control systems were evaluated, including:

- requirements for changes to hardware or controls
- profile and worth of liquid zone control units
- speed and depth of SDS1 shut off rods
- type, number, and spatial disposition of flux detectors/reactivity measuring devices

No significant changes have yet been identified.

Moderator and Endshield Cooling Systems: The capability of the moderator and endshield cooling systems to reject additional heat caused by the increased neutron fields produced by MOX fuel was evaluated. The moderator system heat exchangers are oversized for the current heat loads and can accommodate the additional heat without changes. The primary heat load for the endshields does not come from neutron fields, and the 10% increase in neutron fields is judged to not be significant.

In-core Components: The effects on core components (pressure tubes, calandria tubes, reactivity control mechanisms) of increased neutron fields were evaluated. Fast neutron fluences from MOX fuel at pressure tube and calandria tube locations are very close to those from natural uranium fuel. There will be no significant difference in the neutron induced

effects on core components.

2.4.3 Discharge, Handling and Storage of Irradiated Fuel

See Section 6 for a description of spent fuel management at Bruce A and the proposed scheme for adapting this to MOX fuel. The most significant change to the plant is the replacement of the equipment in the primary spent fuel bay with new machinery to load the fuel into storage modules similar to those in use at Ontario Hydro's Darlington and Pickering stations. The design of this equipment will be based on equipment used for the same purpose at Darlington. In addition, new racks will be needed in both the primary and secondary spent fuel bays to store the modules. The conveyor that moves fuel through the transfer duct from the primary to the secondary spent fuel bays will be modified to accept the modules.

2.4.4 Advanced Fuel Bundle

Sections above deal with plant changes required for the reference design fuel bundle, which contains 230 g plutonium (in 37 elements). The advanced bundle, containing 380 g plutonium (in 43 elements) was also evaluated. The major changes to the plant will be unaffected - the security and safeguards provided for the reference bundle will be adequate for the advanced bundle. Additional radiation shielding for new fuel, and a new spent fuel storage module may be needed for irradiated fuel in the spent fuel bays.

2.4.5 Reactor Control System Effectiveness

2.4.5.1 Background

Bruce A reactors employ a number of subsystems including light water zone control absorbers, mechanical control absorbers (MCA's), on power fuelling and moderator poison addition and removal for control of reactivity and regulation of reactor power during normal operation. A digital computer is employed to manage these systems. Additional information on the function of these systems is provided in is provided in section 2.1.5

Reactor physics parameter changes associated with the change to MOX fuel have some impact on the bulk and spatial dynamic characteristics of the Bruce A reactor core and hence, also to have some impact on the behaviour of the closed loop control characteristics of the core. A brief reactor control study was undertaken in order to assess the feasibility of continuing to use the present computer-based bulk and spatial reactor regulating system algorithms with this type of fuel, to propose control system design changes if warranted, and to identify more detailed design-support analysis that might be needed if any significant complications are uncovered.

The physics parameter changes which influence the Bulk power reactor control system behaviour include: the delayed neutron fraction, the prompt neutron lifetime), change from a small positive to a small negative void reactivity feedback coefficient, fuel temperature feedback coefficient, and reactivity control device worth. Detailed values for these, and related parameters are summarized in section 2.2 and Appendix C. To assess the combined effect of these changes, a frequency domain and time domain analysis was undertaken.

The principal constraining factors are the digital control computer sampling frequency, and the zone control fill/drain and control valve characteristics. Control system gain parameters can be adjusted easily if required.

Spatial flux variations can occur in CANDU reactors due to their large size. In addition to automatic bulk reactor power regulation, the CANDU design also use an automatic spatial flux shape control system for control of spatial xenon reactivity feedback. The change to MOX fuel introduces physics parameter changes which influence the spatial control system behaviour. A scoping assessment of spatial reactor regulation, based on examination of the first few reactor harmonic modes, and the associated modal reactivity worths, and a comparison of these with corresponding parameters for a natural uranium fuelled reactor was undertaken.

2.4.5.2 Bulk Reactor Control Loop

To assess the impact of the change to MOX fuel on the Bruce A bulk reactor control loop, a linearized Bode frequency domain control system analysis, supplemented by a time domain step response calculation was carried out. This was done at both very low and normal reactor power levels (1.0×10^{-6} of Full Power, and Full Power), to account for power dependent effects in the flux measurement system dynamics. This analysis included dynamic modelling of the reactor point kinetics, void and temperature feedback effects, and measurement and control system behaviour, including the digital control computer sample and hold behaviour based on the present Bruce A digital reactor control algorithm.

The analysis showed that the various reactor physics parameter changes described above will have only a small net effect on the bulk reactor control system behaviour. The shapes of the frequency response curves for the natural uranium and MOX fuelled cores are similar. With a modest adjustment to the primary control feedback gain, control system gain and phase margins can be made almost identical. With these gain adjustments in place, the closed loop control system step response behaviour (in response to step reactivity disturbance inputs which conservatively model the effect of channel refuelling), is also very similar between the natural uranium and MOX based cores. It is not expected that any hardware changes to the bulk reactor control system will be needed for the MOX based core.

2.4.5.3 Spatial Flux Control

To assess the impact of Reactor Physics parameter changes on the reactor spatial flux control system, important harmonic modes and the associated modal subcriticalities were computed. These important modes are described as side to side, top to bottom and end to end flux tilts. The ability of the spatial control system to stabilize the reactor was evaluated.

Based on this assessment, the light water spatial control system algorithm is expected to function satisfactorily for the MOX core, and to provide suitable spatial xenon stabilization action without changes to existing hardware. However, we do recommend that MCA operation be examined more closely in order to ascertain whether the automatic MCA drive control algorithm or MCA banking arrangement should be modified in order to optimize system behaviour under upset conditions.

2.5 Safety and Licensing Summary

2.5.1 Introduction

All CANDU reactors have been designed to meet Canadian Atomic Energy Control Board (AECB) licensing requirements. A key element is demonstrating through safety analysis that the design meets specified limits on public dose consequences for a broad range of design basis accidents. The regulatory requirements for public dose consequence limits are broadly similar to those established by the US NRC. Appendix C provides a summary of the Canadian regulatory requirements and the approach to safety and licensing analysis, together with a discussion of these in the context of NRC requirements.

CANDU reactors, utilizing natural uranium fuel and heavy water moderator, have some unique safety characteristics. The neutron spectrum contains a relatively large fraction of delayed neutrons and photoneutrons. This leads to a relatively slow response to changes in reactivity. Also, a reduction of coolant density initiated by a loss of coolant leads to an increase in reactor power. In recognition of this two independent, physically separate, fast-acting shutdown systems are provided to rapidly shut the reactor down should any reactor trip parameters deviate beyond pre-set limits. These shutdown systems are also independent and physically separate from the reactor control system.

The introduction of MOX fuel modifies the reactor kinetics of the reactor core and introduces physical factors that are not relevant to natural uranium fuel. Consequently, this necessitated a review of safety and licensing issues in four areas:

- reactor dynamics,
- exposure of plant personnel to radiation fields from new fuel,
- criticality of new or lightly irradiated fuel, and
- changes in fission product content of irradiated fuel.

One specific goal of this study was to achieve a fuel design that resulted in reactor behaviour not significantly different from that of natural uranium fuelled reactors. The intent of this goal is to maintain operation of a MOX fuelled reactor within the existing safety and licensing envelope. In order to achieve this end, burnable poison material is introduced into the central fuel elements to modify the reactivity characteristics. The change in reactor response associated with the plutonium in the MOX fuel is compensated by the dysprosium poison material which leads to a negative reactivity feedback when coolant density is reduced. The magnitude of this feedback was controlled to avoid introducing new safety and control issues.

Natural uranium fuel used in CANDU reactors is only slightly radioactive and is manually handled during the fuel loading process. Addition of plutonium to the fuel increases radioactivity to the extent revised handling procedures and equipment are necessary. Analysis was undertaken to establish data needed to design measures for handling of fresh fuel at the reactor site.

CANDU reactors operating with natural uranium fuel can only attain criticality in the presence of heavy water moderator. Enrichment with plutonium raises the possibility that fuel may

reach criticality in the presence of light water or in specific geometries. Extreme scenarios were evaluated and found not to result in criticality concerns.

Enrichment of the fuel with plutonium allows higher fuel burnup to be attained than is characteristic of CANDU reactors operating with natural uranium and alters the fission product content of spent fuel. Fission product loading and decay heat levels have been computed to assess emergency cooling and containment system capability, and evaluate consequences should an accident occur.

The following sections address these issues. Emphasis is placed on the modified reactor physics characteristics of a MOX fuelled core as they relate to reactor shutdown and control capability. As demonstrated below, the goal to ensure reactor safety equivalent to the natural uranium fuelled CANDU reactor has been achieved.

2.5.2. New and Irradiated Fuel Dose Characteristics

The impact of MOX fuel on current practice for handling new and irradiated fuel has been assessed. MOX fuel generates somewhat higher dose rates than natural uranium, all other factors being equal. The design goal for new fuel handling is to maintain radiological dose to operators at or below current levels in order to demonstrate acceptable plant worker safety. New fuel handling dose assessment is based on contact dose rates from gamma and neutron fields from the new natural and MOX fuel which are presented in Appendix C.

Section 2.4 describes the minor modifications to the new fuel handling system that include shielding sleeves and other measures introduced to ensure that new MOX fuel handling results in doses equal to or less than those associated with handling new natural uranium fuel.

Dose information developed also includes the contact dose from spent fuel. Although this is not directly relevant to safety during remote handling of spent fuel, it is needed to demonstrate that the CANDU spent fuel standard is maintained. Information presented in Appendix C confirms that the dose from irradiated MOX fuel is equivalent to that of irradiated natural uranium fuel up to 250 years after discharge from the reactor.

2.5.3. Criticality Safety

The natural uranium fuel used in CANDU reactors can only become critical in the presence of heavy water in a fuel/moderator core configuration close to the designed lattice arrangement. Existing CANDU's, therefore do not require special precautions to prevent criticality outside the reactor as do reactor designs which depend on fuel enriched in fissile content.

Accident scenarios may be postulated which might lead to a concentration of fissile material from fuel bundles in a moderating material. These range from accidental immersion in light water during transport, through some accidental mode of flooding with heavy water at the plant, to multiple fuel bundle damage in the irradiated fuel bay. A review of some scenarios was undertaken. This indicated the probability of flooding fresh fuel with heavy water outside of the reactor is impossible due to the relative elevations and separation of fresh fuel storage

areas and heavy water sources. Transport is in special containers designed to preclude criticality under all conceivable situations. Attention was thus focused on an assessment of criticality in the irradiated fuel bay. The scenario of light water covered fresh fuel in full close packed handling modules in an infinite array, representative of a full fuel bay was analyzed. The results show that criticality would not be achieved for the scenarios considered. A detailed criticality assessment would be performed as part of the licensing process.

2.5.4 Reactor Thermalhydraulic Parameters

In CANDU reactors thermal power limits in the fuel channels exist which are associated with a change in two-phase boiling heat transfer regime from nucleate boiling, through transition boiling, to film boiling. These limits are referred to as the critical channel power and are defined for each fuel channel by the occurrence of critical heat flux (CHF) at any fuel bundle element in that channel. This condition is referred to as onset of dryout, because as fuel element power is increased beyond the power level at which CHF occurs for the prevailing coolant thermalhydraulic conditions, a process of film boiling drypatch spreading on the element occurs. This drypatch spreading results in progressively degraded cooling that is associated with post-dryout (PDO) heat transfer.

The introduction of MOX fuel causes some changes to the distribution of heat generation, both within fuel bundles and axially along the 13 bundle fuel string within a fuel channel assembly, relative to the natural uranium fuelled core. Within a fuel bundle, variation in power distribution amongst the elements occurs radially because of the enrichment in the outer two rings of 30 elements of the MOX fuel bundle and the presence of burnable poison in the inner 7 elements. As a result the power generated in the inner 7 elements is much less than for the natural uranium bundle. Axial variation in power distribution along the fuel string is reduced through use of a refuelling scheme that results in more uniform power generation which is made possible by the use of enriched fissile material content.

These factors were studied extensively as reported in Sections 2.2 and 2.3. to determine their importance to thermalhydraulic parameters. The studies show that the impact on critical heat flux performance of the 37-element MOX bundle is insignificant relative to the 37-element natural uranium bundle. Calculated critical channel powers are essentially identical for MOX and natural uranium fuelled channels. In order that the calculated thermal power limits be accepted in any future licensing submissions it will be necessary to verify the critical heat flux with full-scale water CHF experimental data, as is done for the existing natural uranium fuel bundle.

2.5.5 Shutdown and Control System Evaluation

2.5.5.1 Introduction

The Bruce reactors are each fitted with two completely independent shutdown systems which ensure rapid shutdown in the event of any malfunction. A separate, independent control system regulates reactivity during normal operation to control reactor power. These systems are described in more detail in section 2.1 of the report. The introduction of MOX fuel somewhat modifies the reactor dynamic response so that a systematic review of the

capability and performance of these systems is required. The following section addresses the capability of the existing shutdown systems. The control system is discussed in detail in section 2.4 on plant changes.

2.5.5.2 Shutdown System Trip Parameter Effectiveness

The full set of design basis accidents, for which trip coverage analysis is normally done, was assessed. Cases where neutronic trips play a role were systematically identified and the required effectiveness of alternative trips were evaluated.

The assessment of trip parameter effectiveness performed for this study, and documented in section 6 of Appendix C, established the trip parameter coverage summarized in Table 2.5.1. This table presents the trip coverage provided by the existing trips for the various categories of design basis accidents in a MOX core and presents a comparison with the coverage provided in a natural uranium core, which is documented in the current Safety Report for the station. For some design basis accidents, typically LOCA and loss of flow events, neutronic trips will not occur due to the negative coolant void reactivity feedback in the MOX core. However, the existing process trip parameters will provide effective protection for these events.

For important accident categories, such as large break LOCA, in-core LOCA and loss of reactivity control, for which either trip parameter coverage is changed, reactivity worths are affected or core kinetics play an important role, specific analyses have been performed as part of the trip parameter assessment to demonstrate that specific safety design objectives are met. The large break LOCA has received particular attention. A detailed discussion follows.

2.5.5.3 Large Break LOCA

In natural uranium fuelled CANDU reactors reactivity increases following loss of coolant from the pressure tubes. A key parameter when evaluating reactor safety is the energy deposited in the fuel during the power increase prior to shutdown system initiation by neutronic indications of increasing neutron flux levels. Current generation CANDU reactors are equipped with two fast-acting independent safety shutdown systems to assure energy deposition is limited to acceptable levels.

Appendix C explains in some detail the reactor physics of the MOX fuel design that ensures the two shutdown systems will meet safety design objectives applicable to large break LOCA accidents. The basic approach entails engineering of slightly negative void reactivity to ensure that timely shutdown of the MOX-fuelled reactor occurs to limit challenges to the integrity of the fuel and fuel channels. A point kinetics analysis of LOCA response was performed to compare the power transient and energy deposition in the natural uranium fuel core with that in the MOX fuel core for the first few seconds following initiation of a large LOCA. Figure 2.5.1 shows the power transient for the first 5 seconds following initiation of LOCA. The analysis for natural uranium is based on reactor shutdown initiated by an early neutronic trip. The analysis for MOX fuel is based on a process trip signal 3 seconds following initiation of LOCA. The detailed process trip assessment in Appendix C shows that timely process trips will occur and that there is not a strong sensitivity to the timing of the reactor trip. The

energy deposited in the fuel 5 seconds after LOCA initiation is less for the MOX fuelled case indicating that fuel heatup will be reduced and that existing safety design objectives are met.

Analyses of additional accident scenarios are described in more detail in section 6 of Appendix C. Highlights of the analysis indicate that:

- a) shutdown system reactivity depth is maintained for the case of an in-core small break LOCA which dilutes the poison concentration in the moderator.
- b) loss of reactivity control events are not significantly different in a MOX core than in a natural uranium core, although there is a change in the neutron kinetics parameters characterizing the two cores. Reactor safety will not be affected negatively by changes in control characteristics. Any changes in neutronic parameters influencing safety will be intercepted by the neutronic trip signals of the two independent shutdown systems resulting in shutdown.
- c) the decision and action time associated with establishing alternate heat sinks for secondary side failures is increased slightly in a MOX core as a consequence of the lower (short term) decay heat for MOX fuel, relative to natural uranium fuel.

TABLE 2.5.1
SUMMARY OF TRIP PARAMETER COVERAGE FOR MOX & NATURAL URANIUM CORES

ACCIDENT CATEGORY	TRIP PARAMETERS PROVIDING PROTECTION							
	MOX CORE				NATURAL URANIUM CORE			
	SDS1		SDS2		SDS1		SDS2	
Large Break LOCA	HTLP	HTLF	HTLP	HTLF	NOP	NHLR	NOP	NHLR
Small Break LOCA (including in-core LOCA and single channel failure events)	HTLP	PLL	HTLP	PLL	HTLP	PLL	HTLP	PLL
Loss of Reactivity Control	NOP	HTHP NHLR	NOP	HTHP NHLR	NOP	HTHP NHLR	NOP	HTHP NHLR
Loss of Reactor Power Regulation	NOP	HTHP	NOP	HTHP	NOP	HTHP	NOP	HTHP
Loss of Flow	HTHP	HTLF	HTHP	HTLF	HTHP	NOP HTLF	HTHP	NOP HTLF
Feedwater Line Failures	SGLL	Manual HTHP	SGLL	Manual HTHP	SGLL	Manual HTHP	SGLL	Manual HTHP
Steam Line Failures	HTLP	PLL SGLL	HTLP	PLL SGLL	HTLP	PLL SGLL	HTLP	PLL SGLL

TRIP PARAMETER LEGEND

NOP = Neutron Overpower
HTHP = Heat Transport High Pressure
HTLF = Heat Transport Low Flow
SGLL = Steam Generator Low Level

NHLR = Neutron High Log Rate
HTLP = Heat Transport Low Pressure
PLL = Pressurizer Low Level

Figure 2.5.1 LOCA (100% Break) Power Transients in Natural, Ref. Mox, and Advanced MOX Cores.

2.5.6 Post LOCA Decay Heat and Radionuclide Content

2.5.6.1 Introduction

Decay heat level and radionuclide content of the MOX fuel, of importance to accident analysis, was calculated for discharge burnup and irradiation representative of equilibrium conditions (mid burnup). This information is needed to assess the performance of safety systems and quantify accident dose consequences. Information on decay heat is also needed for the assessment of irradiated fuel storage options. Section 7 of Appendix C establishes decay heat at discharge burnup and mid burnup for the assessment of spent fuel handling and accident conditions, respectively.

2.5.6.2 Decay Power

The decay power rundown for a MOX-fuel bundle discharged at exit-burnup is presented in Appendix C, which also shows the power rundown curve for a natural uranium fuel bundle discharged at exit-burnup. At early decay times, the decay heat power from MOX-fuel is slightly less than that from natural uranium fuel. This trend continues to about one month when decay power of the two fuels become about equal for the following 30 years. After 30 years the decay power from the MOX fuel becomes as much as 1/3 greater due to the higher actinide quantities which dominate when the high-energy decay gammas from fission products die away.

Information provided in Appendix C allows an estimate of total decay power from a core operating at equilibrium burnup to be estimated. Again, the magnitudes of the decay power from the two bundles are close and behave similarly to those observed for exit-burnup.

2.5.6.3 Radionuclide Content

Analysis presented in Appendix C shows that there are relatively minor variations in the inventory of some fission products relevant to accident consequence in the MOX fuelled core, relative to the inventory in natural uranium fuel. These variations are to be expected because of the shift in the yields for specific fission products generated from fissioning plutonium-239 versus uranium-235 and the different ratios of these fissile isotopes in irradiated MOX and natural uranium fuel.

2.5.7 Releases From Fuel in Accident Conditions

Canadian licensing requires comparison of consequences of a wide range of postulated accidents with dose limits. The assessment, in part, depends on analysis to estimate fractional releases of radionuclides from any fuel overheated beyond the normal operating range following LOCA. Accidents of relevance include the case of large LOCA and those which expose fuel from a single channel to overheating. For example, it is postulated that the 13 bundles in a channel could be discharged from the reactor into containment. As concluded in Appendix C the fraction of fission product content released from MOX fuel during accidents will be approximately the same, or reduced relative to natural uranium fuel. Factors that reduce the fission product source term for LOCA are the reduced heat input to fuel

immediately following LOCA initiation, reduced local maximum operating temperatures of the MOX fuel and reduced decay heat for several days post LOCA.

2.5.8 Accident Consequences and Dose Assessment

Based upon the above considerations regarding radionuclide content and fission product releases from fuel, fractional releases from fuel for postulated accidents which could discharge small quantities of fuel into containment will be similar to that from natural uranium fuel.

The introduction of MOX fuel to the Bruce A reactors will not significantly affect reactor safety and AECL dose limits would continue to be met. Thus, there is no significant difference in overall risk between operating with either a MOX or natural uranium core.

2.5.9 Safety and the Advanced CANDU Fuel Bundle

A more highly enriched MOX fuel bundle has also been evaluated for potential use in CANDU. The purpose of that bundle is primarily to reduce fuel production requirements. The implications on reactor safety are discussed briefly here. The safety related analysis was not performed to the level of detail presented for the reference MOX bundle. The fuel design again makes use of dysprosium poison to compensate excess reactivity and to provide for negative coolant void reactivity. The design includes a more finely subdivided fuel bundle which reduces element power ratings relative to the reference case. This facilitates operation of the reactor to higher fuel burnup approaching twice that of the reference MOX fuel. The bundle contains nearly twice as much plutonium, commensurate with doubled burnup. Projected implications on safety relative to the reference MOX fuel follow:

- a) Doses from fresh fuel will be doubled but fuel throughput will be halved implying similar dose conditions for operators all other factors being equal. The arrangements for operator protection discussed in Section 2.4 will provide adequate protection.
- b) Additional precautions may be needed to preclude criticality as the plutonium content is approximately doubled as compared to the reference MOX fuel. This could be accomplished through thickening of neutron absorbing components of the fuel handling modules.
- c) Analysis presented in Section 2.3 of the main report indicates the thermal hydraulic performance of the advanced bundle will be satisfactory with greater margins to dryout than the reference MOX fuel. This is a result of the more finely subdivided fuel bundle with lowered element power rating.
- d) The advanced bundle has been designed to behave similarly to the reference MOX fuelled case with respect to reactor trips and control.
- e) Decay heat and radionuclide content of the advanced fuel is greater than for the reference fuel. The increased decay heat will have some influence on the performance of the emergency cooling system and moderator heat sink which will tend to increase fission product release fraction from the fuel. We expect that safety margins inherent in these systems and others will accommodate this without dose limits being exceeded.

2.5.10 Summary and Conclusions

Reactor safety issues arising from the use of MOX fuel in CANDU reactors have been reviewed. The MOX fuel has been designed to avoid new safety issues. This is accomplished by the introduction of dysprosium poison. This ensures that bulk reactivity effects are comparable to the natural uranium fuelled reactor. The addition of dysprosium to the inner fuel elements makes void reactivity negative for the MOX fuelled core rather than positive as for the natural uranium fuelled core. A modest benefit ensues for dose consequence analysis of large break LOCA analyses. Based on this study the AECL dose limits will be met for the entire spectrum of accidents which must be considered for Canadian licensing.

The introduction of enriched fuel to CANDU reactors introduces a need to consider the potential for criticality outside the heavy water moderated core. Natural uranium fuel normally used in CANDU reactors cannot achieve criticality outside the core. Analysis of representative scenarios indicates that arrays of the reference fuel in handling modules in the irradiated fuel bay cannot become critical. Although the analysis is not as extensive as will be required for licensing we believe it provides a strong indication that criticality issues can be managed through use of the fuel handling modules adapted from CANDU fuel storage practise.

The conclusion of this study is that reactor safety, in the overall sense of risk to the public and to plant personnel arising from accident situations is unaffected by the introduction of plutonium enriched MOX fuel.

2.6 Safeguards and Security

This section covers the safeguards and security requirements for the Bruce Nuclear Power Development (BNPD) focusing on the specific needs of to carry out the mission of dispositioning excess weapons-grade plutonium. For the reference case, (37 element MOX Fuel Bundle), the safeguards and security requirements are addressed under the following headings:

- Bases of Study
- Existing Site Security Arrangements
- Additional Site Security Requirements
- Additional Site Safeguards Requirements

The Advanced Fuel Design case has also been reviewed to identify any significant changes to the safeguards and security plans developed for the existing 37 element fuel design.

2.6.1 Bases of Study

For the reference case of 37 element CANDU MOX fuel, the bases of study are that the fuel will be delivered to the Bruce site in convoys of the DOE SST's. These deliveries are expected to be consistent with the fuelling requirements of the CANDU reactors assigned to this mission. It is also presumed that the Special Nuclear Materials (SNM) that are being shipped have undergone an IAEA safeguards accountancy and sealing procedure at the reference fuel fabrication site, FMEF Hanford, prior to shipping. It is further presumed that the packaging has received approvals from both the US government authorities of concern, as well as the appropriate Canadian government authorities. (These issues are addressed in the Packaging and Transportation Sections and in the Security and Safeguards Overview Section)

2.6.2 Existing Site Security Arrangements

Section 2.1 provides an overview of the whole of the Bruce Nuclear Power Development (BNPD) site. The BNPD site currently has an unarmed security force of 60 members who are responsible for serving the day-to-day operations of the overall site. These duties include BNPD perimeter access control, access control and emergency response to each of the generating stations, access control and emergency response for the Bruce Heavy Water Plant, personnel badging and ID control, vehicle entry control, and security monitoring activities. The site has a central Security Monitoring Room (SMR) which is manned at all times. Each nuclear station has a secondary SMR which deals with access to the specific station and serves as backup to the central SMR. The secondary SMR's are also manned at all times.

Signals from the existing security systems distributed throughout the site are brought back to the central SMR and the secondary SMR's via fibre-optic cables. The field systems include electronic fence detection systems, video surveillance systems, motion detectors, and alarms. The security systems in place meet existing Canadian and IAEA requirements for the current mission of the facilities.

In the event of a threat to the reactor facilities, direct, supervised lines are used to obtain

assistance from specially trained, armed response forces located off-site, but within the local area. This armed force is provided by the Ontario Provincial Police (OPP) and has an initial response time of 20 minutes. The initial response force can be augmented with armed reinforcements within 90 minutes and special tactical units within 2 hours. These tactical units are specially trained in counter-terrorism, containment, and negotiating strategies.

2.6.3 Additional Site Security Requirements

Current security practice at the reference disposition site Bruce A has no need for particular concern regarding the new fuel for loading into the reactor(s). To accommodate the use of MOX fuel of the reference design, additional security requirements would include the establishment of a new fuel receiving and storage area, a secure transfer route from the storage area to the new fuel loading room, and hardening of the new fuel loading room.

The key cost element to be addressed is that of building the new fuel receiving and storage facility. This facility will have to be a Category 1 (Inner Area) facility and provide a secure docking area which could accommodate a US DOE Safe Secure Transport (SST) for the delivery of new MOX fuel (see discussion in Section 5). Preliminary assessments have indicated that there would be no difficulty in locating such a facility within the PIDAS fenced area around the Bruce "A" site. The existing security fence meets the international requirements for such exclusion barriers and is fully capable of providing intrusion detection and assessment.

The new fuel receiving and storage facility (described in Section 2.4) should be built to provide a delay factor consistent with the response time of the appropriate security force assigned to deal with any potential incidents involving the MOX fuel. This security force has existing duties as a response team for the Bruce Nuclear Power Development site and would respond to any incident involving a threat against the reactor facilities. The initial studies indicate that a 90 minute delay would be sufficient to meet these security needs. Several US DOE and DOD codes are available to assist in defining the specifications but the determining factor will be the compliance with IAEA regulations and with the Canadian regulatory body (Atomic Energy Control Board) requirements. A possible building arrangement is shown in Figures 2.4.1.

Basic security systems required for this facility would include four levels of access control such that an individual requiring access would have to be accompanied by at least one other authorized individual (the 2 person rule), have a valid access control card and Personal Identification Number (PIN), and be further authorized by a biometric security device such as hand geometry or retinal scan. The facility itself would be monitored by external motion detection and video surveillance.

The security requirements, built-in delay factors and fuel bundle storage schemes have been evaluated using guidelines developed for such facilities by the US DOE and the IAEA¹.

¹ ITC 10, IAEA Course on Physical Protection of Nuclear facilities and Materials; Vol. 2, "Designing a Physical Protection System"

Fencing, barriers, motion detection and alarm systems would be installed in strategic areas to ensure coverage and provide early warning of intrusion attempts. The evaluation indicated that a delay time of 90 minutes would be sufficient to allow an initial armed emergency response team to arrive at the site. The tactical response units would be at the site within 2 hours.

Current commercially available security devices and systems have been used for the purposes of estimating the costs associated with the study. It would be prudent, however, to specify the devices and systems to be used closer in time to the actual construction of the facility to take advantage of any technological advances.

The transfer to the new fuel loading room could be effected via a tunnel or an overhead bridge, both of which can be hardened and secured to prevent any unauthorized access. The transfer route will be chosen on the basis of least disruption to Bruce station operations. While the transfer of one container at a time of new MOX fuel could technically be classified as a Category 2 movement, there are advantages to maintaining the Category 1 requirement throughout the transfer process to reduce the risk of diversion of the material. Thus the entire process of receiving, storage, transfer, and eventual loading of the new MOX fuel will be carried out in a Category 1 facility.

The new fuel loading area will require some engineering work to seal off potential unauthorized access points which currently exist. The main entryway to this room would have to be modified to include the four levels of access control described earlier for the receiving and storage facility. This would entail the construction of a vestibule which would serve as a man-trap and ensure the integrity of the Category 1 requirement for the new fuel loading room. A more detailed description of the new fuel loading room is presented in Section 2.4.

Another change required for this mission would be to have the personnel identified as authorized staff for access to the MOX fuel receiving and storage facility as well as the new fuel loading room, cleared to the appropriate security level for such access. The personnel clearances would have to include any employees who would be handling existing natural uranium fuel for the other two reactors as the two types of fuel would share the same room for fuel loading. This is a cost that would be incremental to the existing activities. The approximate time to obtain such a clearance is 10-12 months.

It is anticipated that there would be a need for 5 additional security personnel to support this mission and 5 fuel handling personnel for the handling of the MOX fuel. Costs associated with additional security requirements are identified in section 2.6.6 and chapter 9.

2.6.4 Additional Site Safeguards Requirements

The existing full-scope safeguards regime at the Bruce facility have been put in place by the IAEA. These safeguards consist of an accounting scheme which identifies and tracks the natural uranium fuel from reception through the loading into the reactor, through the on-power cycle, and subsequent discharge to the spent fuel storage bays. There is a device in use which counts the spent fuel bundles (Spent Fuel Bundle Counter, or SFBC) as they are

discharged into the spent fuel storage bays. A schematic representation of the process is provided in Figure 2.6.1. Once in the spent fuel storage bays, the fuel comes under an IAEA Containment/Surveillance (C/S) system which records all movements of spent fuel. Some spent fuel has been put in specially-designed cages which have an IAEA tamper indicating seal as a closure. A schematic view of the seal is shown in Figure 2.6.2.

The safeguards associated with the new mission of dispositioning the weapons-grade Plutonium as MOX fuel will require several significant changes to the existing regime. On delivery of the MOX fuel via the transport system selected for the mission, the IAEA would verify that the seals placed on each container by the IAEA at the shipping site (FMEF, Hanford) are intact. A device currently in use at the PFPP in Japan, and developed by LANL, would then be used to verify that the container did indeed contain the right amount of MOX fuel. Two of these devices, based on passive neutron counting², will be required. The second device will be used in the new fuel loading room prior to insertion into the new fuel loading magazine to ensure no diversion has occurred. These measurements can be integrated with the accountancy program to assure tracking of fuel bundles and assays.

Because Bruce is a multi-unit station, and this mission is dedicated to the disposition of weapons grade material, the IAEA would wish to augment the current arrangement of relying only on SFBCs and will require the installation of Core Discharge Monitors (CDMs) to reconcile the gap between actual core discharge of fuel and possible diversion prior to discharge to the spent fuel bay. These devices have been installed in the latest generation of CANDU multi-unit reactor station. Figure 2.6.3 is a schematic representation showing the existing CDM arrangement in use at the Darlington Nuclear Generating Station. There may be a requirement to have the CDMs made capable of distinguishing between spent natural uranium fuel and spent MOX fuel. This may be an area for further study.

The costs associated with the installation and use of such devices are detailed in section 2.6.6 and chapter 9.

2.6.5 Advanced Fuel Design

No significant changes to the security and safeguards regime are anticipated with the introduction of the advanced fuel. Current evaluations do not identify any parameters that indicate that new approaches be adopted nor that any new assay equipment may be required. It is anticipated that recalibration of the assay equipment will be required. The procedures and protocols put in place for the reference fuel may have to be modified somewhat but the impacts on costs are expected to be negligible.

²H.O. Menlove, et al., "CANDU MOX (CMOX) Counter Design and Operation Manual", LA-12101-M (ISPO-336), Los Alamos National Laboratory, (October 1991)

Figure 2.6.1 Core Discharge Monitors

Schematic for Bundle Counter Installation

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Figure 2.6.2 IAEA Security Seal

Darlington CDM System

2.6.6 Cost and Schedule

The estimates for cost associated with the safeguards and security aspects of the dispositioning of plutonium at the Bruce Nuclear Power Development site are based on having two Bruce units dedicated to the mission, with MOX fuel arriving on a monthly basis in sufficient quantities to meet fuelling requirements and have an additional float supply of one month's fuel. They are also based on the use of US DOE SST's for the transportation of the MOX fuel. All mission costs are inclusive of start-up costs.

The costs associated with the new MOX fuel receiving and storage facility at the Bruce site as presented here are exclusive of construction costs, which are covered elsewhere and but are inclusive of delay systems, access control and sensing equipment, and alarm systems as well as licensing costs associated with a Category 1 facility.

The safeguards needs for this facility are expected to include material accountancy systems, plutonium assay equipment, along with handling tools and inspection activities. The total safeguards and security costs for the new MOX fuel receiving and storage facility are anticipated to be \$3.6M for the initial start-up phase and annual operation and maintenance (O&M) cost of \$1.8M/year. The new fuel loading room estimates are exclusive of construction "hardening" costs, but inclusive of access control systems, surveillance systems and alarm systems.

The safeguards associated with the new fuel loading area include the installation and use of plutonium assay equipment, an accounting system station, containment and surveillance systems and inspection. The total safeguards and security costs for the new fuel loading room are anticipated to be \$1M for the initial start-up phase and an O&M cost of \$0.2M/year.

The safeguards associated with in-reactor operations include the installation of core discharge monitors, refurbishment of spent fuel bundle counters and inspection. The cost of the system has been annualized at \$0.2M/year to reflect the fact that it must be periodically replaced due to the high radiation field.

Bruce Nuclear Power Development site central security services associated with the disposition project would require additional staff, additional communications equipment, and additional security monitoring room equipment for the primary and secondary security control rooms. Other cost items identified are the development and delivery of training programs, operational protocols, response strategies, and liaison with other security forces. These costs are anticipated to be \$1.1M in the start-up phase and O&M costs of \$0.5M/year.

The existing infrastructure will be used for safeguards requirements. Additional costs associated with the dispositioning of plutonium are covered in the areas identified above.

To summarize, the costs for the safeguards and security measures identified in this section are anticipated to total about \$5.7M during start-up and O&M costs \$2.7M for the overall mission. These have been included in the Bruce A modification costs (Table 9.3, and the overall O&M costs (Table 9.5).

While it is recognized that many activities need to be carried out prior to committing the reactors to this mission, the overall schedule for the mission is dependent on the earliest production of MOX fuel to supply reactor operations. As a consequence, all the security and safeguards measures must be in place for the receipt of the MOX fuel. The longest lead-time factor governing the safeguards and security requirements dealt with in this section is the construction, commissioning and licensing of the Category 1 fuel receiving and storage facility. The preliminary design and subsequent detailed design should start as early as possible to ensure that all the appropriate security and safeguards measures are addressed for this facility.

2.7 Operations and Waste Management

2.7.1 Impact on Plant Operations

The introduction of MOX fuel to the Bruce A plant will have virtually no impact on the day-to-day operation of the station. The only significant difference between a natural uranium fuelled plant and a MOX fuelled plant will be in the area of new fuel handling. Training programs will be required for station operating staff to familiarize them with the requirements imposed by the MOX fuel in terms of criticality concerns, increased radioactivity of the new fuel and safeguards and security changes as a result of the Category 1 status of the fuel storage and handling facilities. Criticality concerns will also dictate the need for minor modifications to the handling of the spent fuel and, while these will represent changes with respect to existing procedures at the Bruce A station, the new procedures will more closely resemble procedures already in place for the handling of irradiated natural uranium fuel at other Ontario Hydro sites.

2.7.2 Waste Management

Although the contents of the MOX CANDU fuel bundle differs from the natural uranium CANDU fuel bundle, externally the bundles are identical except for the level of radiation associated with the new fuel. This difference, however, will not result in any increase in the quantity or hazard level of waste produced at the plant. (Handling of the spent MOX fuel will be dealt with separately in chapter 6) The Bruce Nuclear Power Development has facilities on site for the storage of low, medium and high level radioactive waste. There will be no changes required to these facilities.

2.8 Logistics for Achieving Alternate 1 - 100 MT

As noted in other sections, the option presented for achieving the alternative case of dispositioning of 100 tonnes in 25 years is to increase the number on reactor units to 4. At the four unit Bruce A station this may be an advantage over operating two of the units on natural uranium fuel and two on MOX fuel, as it will eliminate the need for physical barriers and procedures to ensure the two types of fuel do not become interchanged. If the reference 37 element fuel design is used, then the fuel supply requirement (in tonnes of MOX fuel per year) would and the number of trips from the supply plant would double, thus making development of the CANFLEX particularly attractive for this alternative.

2.9 Preparation Activities for Plutonium Disposition in CANDUs

2.9.1 Introduction

This section summarizes the activities that will be required to prepare the reactor for full implementation of the plutonium disposition program. Preparation related to safeguards and security at the reactor is discussed in section 2.6. Preparation activities the fuel supply facility and transportation are detailed in chapters 4 and 5 respectively. As requested by DOE, the program is assumed to begin in the third quarter of 1994. The overall plan (figure 1.6) leads to initiation of Bruce A reactor operation on MOX fuel in about 4 years.

There are several groups of preparation activities, with significant overlap and interdependence among them:

- Confirmatory testing of nuclear and fuel performance
- Licensing activities for the Bruce station
- Reactor facility additions (fuel storage and handling)
- Establishing, licensing and implementing a MOX fuel supply program

Following these activities, and after initiation of operation on MOX fuel, a transition is made from natural uranium to MOX fuel at Bruce. Fuel supply and transportation preparation activities are described in chapters 4 and 5 respectively. The remaining phases are described below

2.9.2 Confirmatory Tests

While operation of Bruce reactors on MOX fuel will not require any new technology, it will be necessary to confirm the predicted fuel and nuclear performance in order to establish a firm basis for licensing and for establishing operating parameters. These programs will be a joint effort involving the plant operator (Ontario Hydro), the reactor designer (AECL) and various suppliers, including U.S. companies and National Laboratories (such as Los Alamos) involved in production of trial MOX fuel bundles for use in the fuel and physics confirmatory tests.

2.9.2.1 Physics

Reactor physics experiments will be carried out at the zero energy reactor, ZED-2, at the Chalk River Nuclear Laboratories to measure the following lattice properties of the Reference Pu-MOX fuel and, at a later stage, the Advanced Pu-MOX fuel design :

- reactivity of the Pu-MOX fuel lattice compared to the natural uranium fuel lattice,
- void reactivity of the Pu-MOX fuel lattice,
- flux distributions across the fuel bundle before and after coolant void, and
- coolant and fuel temperature reactivity effect of the Pu-MOX lattice.

The results of these measurements should establish the accuracy of the reactor physics calculations of the Pu-MOX fuel design.

In order to keep the cost of these experiments as low as possible, consistent with obtaining

high quality data, it is proposed to use the substitution technique in which seven channels, each containing five bundles of the Pu fuel, are successively substituted in a critical host lattice. The critical buckling of the substituted Pu fuel lattice can be derived from the change in critical size of the mixed core, as can the change in buckling corresponding to changes in temperature or coolant density in the Pu fuelled rods. Measurements would be made with the substituted rods "cooled" by heavy water and with the coolant voided. These measurements would be made with the substituted rods both at ambient temperature (about 21°C) and at operating temperature, about 300°C.

In addition to the above reactivity measurements, detailed fine structure reaction rate distributions would be measured in a special demountable bundle of the Pu fuel located in the center of the region of substituted fuel. These distributions would be measured at ambient temperature both with and without heavy water coolant. Foils containing U235, Pu239, In, Cu, U238, Li, Mn, Al, and Dy would be irradiated at various positions in the cell, but especially between the pellets in the fuel elements. The various reaction rate ratios would be compared with calculated values to validate the calculation.

For each fuel design the following activities will be carried out:

1) Prerequisites:

35 bundles of the Pu fuel design, one "demountable" bundle for fine structure measurements.

2) Preparations:

Criticality calculations for fuel storage,
Pre predictions of reactor critical heights,
Safety assessment of the experiments,
Acquisition of foil materials, manufacture and calibration of foils,
Acquisition of other hardware,
Set up reference core,

3) Substitution Measurements:

Ambient heavy water cooled substitutions, including flux plots,
Ambient voided substitutions, including flux plots,
Hot heavy water cooled substitutions, including flux plots,
Hot voided substitutions, including flux plots,
Analysis of measurements,
Reporting results,

4). Fine Structure Measurements:

Loading of foils etc. (requires use of the RFFL because of Pu fuel),
Irradiation in ZED-2,
Disassembly and counting of foils,
Analysis of results,
Reporting of results.

The above steps are carried out twice: once with coolant and once without.

The cost estimate for the above is: \$0.75M, including project management, but excluding the production of the 35 fuel bundles. The fuel bundle cost is estimated at \$3.9M (see Section 4).

The above estimates are for one fuel design. To perform the same series of measurements for the other design doubles the cost to \$1.5M; fuel bundle cost is estimated at an additional \$4M.

The plutonium oxide feed material is assumed to be free issue, and the cost of disposal of the fuel, once the qualification tests are complete would be factored into existing spent fuel storage costs.

2.9.2.2 Fuel Design

The fuel verification program for either the reference or advanced MOX fuel entails the research, development, and demonstration necessary to qualify the fuel for large-scale implementation in Bruce A. This comprises two main elements: (1) confirmation of design in experimental facilities, and (2) Prototype bundle irradiation in Bruce A.

Confirmation of design. This involves demonstration that the fuel design meets all the requirements for normal operating conditions. This is necessary to obtain utility and the regulatory approval for a demonstration irradiation of prototype MOX bundles in Bruce A. The following fuel activities would be included:

- irradiation of elements and bundles in the NRU research reactor followed by PIE, to provide information for finalizing the design and fabrication processes, and for confirmation of irradiation behaviour
- measurements of critical heat flux to demonstrate that the design provides acceptable Neutron Overpower Protection (NOP) trip margins
- mechanical integrity tests to demonstrate acceptable in-reactor mechanical performance

For the reference MOX design, the geometry of the bundle is unchanged from the current fuel, so most out-reactor tests (fuel handling, flow visualization, and endurance tests) are not required. (Some confirmation of mechanical integrity of the bundle will be required, because

of the different thermal expansion of the inner and outer elements.)

The total cost of this part of the program is estimated to be \$18M for the reference MOX design, with the majority of the cost for irradiation and subsequent examination.

Prototype bundle irradiation in Bruce A. A prototype irradiation in a power reactor of about 50 bundles will be required before proceeding to a full core implementation of MOX fuel. This irradiation will use fuel fabricated using the same processes as in the full-scale MOX fuel fabrication plant. The cost of this is estimated to be \$3.2M for the reference MOX fuel.

The above are the two major program elements in the fuel confirmatory program. The next stage after this start-up program would be the transition to the full MOX core. The transition strategy is to convert from natural uranium to MOX through normal fuelling, loading about 15 MOX bundles per day or by loading new bundles during a major maintenance outage. The transition would serve as the large-scale power reactor demonstration irradiation.

Finally, there would be an ongoing fuel surveillance program (as is in place for all CANDU fuels) to provide additional information on the irradiation performance of MOX fuel discharged from Bruce. This would take place starting with the transition to the full core of MOX fuel, and would involve visual examination of discharged MOX bundles in the spent fuel bay, and PIE in the hot-cells at Chalk River of about one bundle per year, to further establish the database on CANDU MOX fuel behaviour.

Confirmation of Design: Reference MOX Fuel

MOX Fuel Characteristics

In order to irradiate MOX fuel in CANDU, several activities would be required to ensure characterize the performance of CANDU MOX fuel, both as-fabricated and irradiated. Several fuel fabrication issues would be addressed through experimental and analytical work, including assessments of the degree of plutonium homogeneity required and its effect on subsequent fuel performance; the blending/fabrication techniques required to ensure the necessary degree of homogeneity; the effect of fuel stoichiometry (oxygen to metal ratio) on performance. This information would feed into the MOX fuel fabrication program, the NRU irradiation testing, and the fuel design. This would be a 2-year program, with estimated cost of \$0.7M.

NRU Irradiation Testing

The NRU research reactor at CRL has two vertical loops, U1 and U2, each capable of irradiating full size CANDU fuel bundles at power reactor conditions. These two loops have been used extensively in the development and testing of CANDU fuel. Power output of the bundles under test is determined by the initial enrichment and by the local neutron flux, which can be adjusted during the course of the irradiation by appropriate management of the driver fuel. The U2 loop has two "legs" connected in series, each of which can accommodate six bundles, with a total loop power removal capability of 7 MW. The U1 loop also has two "legs", one of which is dedicated to the Blowdown Test Facility for severe accident research; the other leg can accommodate up to six bundles and has a power capability of 3.5 MW. In

principle, a total of 18 MOX bundles could be irradiated simultaneously in the reactor. It will be assumed that one third of the available loop space will be available for irradiating MOX fuel (i.e. 6 bundle positions). It takes about one year for an irradiation to natural uranium burnup, at CANDU power levels.

To confirm fuel behaviour prior to loading into Bruce for prototype test irradiation, and before the transition to the full core, a number of MOX bundles would be irradiated in NRU. Irradiation testing in NRU would take place continuously over a 2 1/2 year period. The irradiations would envelope the power/burnup history expected in Bruce. These will be a combination of "demountable", and full bundles. The "demountable" bundles use a 37-element geometry, in which the outer 18 elements can be easily removed and replaced. The demountable bundle can provide fuel characteristics as a function of burnup (since elements can be removed at selected burnups, and replaced with fresh elements to continue the irradiation). These irradiations would provide the earliest possible information on:

- the effect of plutonium homogeneity on performance; this would be used in finalizing the plutonium / depleted uranium powder specifications, and in selecting the blending process
- the effect of oxygen-to-metal ratio on performance; this would provide information on the stoichiometry required for the as-fabricated fuel, and the effect of deviations from the specified value
- the effect of impurity levels on performance; again used to finalize the powder specification
- the effect of pellet geometry and element design on performance (required before finalizing the internal element design)
- the effect of power/burnup history on performance (by irradiating otherwise identical elements at different power levels to various burnups); the enrichments would be chosen so that some elements are irradiated at high ratings close to, or above, the threshold for high fission gas release; this would provide information on that threshold, and hence on fuel performance margins.

Two demountable bundle irradiations would be mounted as soon as fuel was available. Elements would be fabricated to test the effect of each of the parameters above on performance. The irradiation would be designed so that some elements could be removed at an interim burnup (about 5,000 MWd/te, after 4-5 months of full power irradiation), and examined to provide feedback before finalizing the design and fabrication process. One of the demountable bundles would be irradiated for one year, the other for two years.

Four MOX bundles would also be fabricated at the same time for the earliest possible irradiation testing in NRU. Two of these bundles would be irradiated at high power levels, beyond the nominal discharge burnup for the reference MOX fuel (about 1 year irradiation). The other two bundles would be irradiated to about double the discharge burnup for the reference MOX fuel (about 2 1/2 year irradiation). The plutonium enrichment of the

demountable bundles, and of the first four full bundles could be higher than that of the reference MOX bundles, in order to achieve the desired high power rating at NRU flux levels.

After the completion of the ZED-2 experiment, three additional bundles would be irradiated in NRU. These would have the same composition as the reference MOX bundles. Two of these bundles would be initially irradiated at low power, and then power ramped. The power/burnup histories of the NRU irradiation would be more severe than for the prototype irradiation in NRU, to help establish performance margins.

All bundles would be examined in detail in the hot cells after irradiation. PIE would include the following:

- Visual inspection
- Disassembly and profilometry
- Puncture tests for measurement of fission gas release and analysis of gases
- Metallography of cladding and fuel of selected elements
- Burnup analysis

The results of the PIE would be documented and form a part of the package to the Atomic Energy Control Board (AECB) for regulatory approval to load prototype bundles into Bruce A.

To cost this activity, it was assumed that there will be on average 6 bundles undergoing irradiation in NRU in the first two years, and 4 bundles undergoing irradiation for half of the third year. A total of 9 bundles would be irradiated over the 2 1/2 year period. The total cost of the NRU irradiation and PIE would be \$9.3 M, comprised of the following:

- fabrication cost of MOX bundles (two of which are demountable) included in the ZED-2 program
- cost of NRU irradiation: \$8.3M
- cost of PIE for 9 MOX bundles: \$700K
- cost of irradiation planning, monitoring, project management, and documentation: \$350K over 2.5 years

Critical Heat Flux Measurement

Because of the steeper radial power profile across the MOX bundle compared to natural uranium, it will be necessary to measure the critical heat flux of a full length string of electrically heated bundles that simulate the radial and axial power distributions along the channel. The tests would be performed in a full-scale, hot, pressurized light-water loop that can duplicate reactor channel power and coolant conditions. In parallel, scoping tests would be performed in a low-pressure, low-temperature Freon loop to determine the effect of variations in the radial power profile on CHF, and the effect of crept/uncrept pressure tubes on CHF. These tests would provide corrections to the water CHF tests for these effects. The total duration of these tests, including design, fabrication of the electrically heated elements, performing, analyzing, and documenting the tests is 20 months, at a total cost of \$2.9M (\$2.2M for the water-CHF test, and \$0.7M for the Freon tests).

Fuel Design

The steeper radial power profile across the MOX bundle results in different stresses on the endplate than in the 37-element bundle, because of differential thermal expansion between the inner and outer elements. Fuel bundle stress analyses and as out-reactor strength tests would be performed to confirm adequate design margins. This activity is estimated to take about 2 years at a cost of \$0.5M including an allowance for some design adjustments.

Assessments will be performed to finalize the design, and to check that the design meets the requirements. The design assessments include: design requirement specifications, preparation of technical specifications of powder and pellets, design calculations including modifications of computer models, review and incorporation of data from NRU irradiation, design review, and preparation of the design manual for submission to the AECB for prototype bundle irradiation. The activities, which will be completed before the loading of the prototype bundles in the Bruce A reactor, are estimated to cost about \$1.8M.

Prototype Bundle Irradiation in Bruce A: Reference MOX Fuel

The main objective of the prototype irradiation in Bruce A is to demonstrate acceptable fuel performance for MOX fuel fabricated using the same processes that would be used in the full-size industrial fuel fabrication facility. The irradiation and subsequent PIE would also verify that the fuel performance is consistent with current understanding and fuel models, and would demonstrate the compatibility of the fuel design with the fuel channel and the fuel handling system.

For the reference fuel, the prototype irradiation would involve the following activities. Up to 50 MOX bundles would be fabricated using prototypical production processes, and irradiated in the Bruce A reactor through their normal burnup cycle. Before loading into the reactor, metrology of some of the bundles will be performed to confirm that the bundle geometry of the MOX fuel is unchanged from current design, and to provide as-fabricated dimensions for comparison after irradiation. Following discharge, these bundles will be visually inspected at the fuel bay to check for unexpected appearances or features. Several selected bundles or elements would be shipped to the hot cell for detailed PIE. Performance parameters such as the amount of fission gas, ridge height, would be measured and compared to expected values. This will provide information for validating fuel modelling and reactor physics codes.

The irradiation and PIE would take place over 3 years. The total cost would be \$4.7M, including:

- fabrication of 50 MOX bundles (\$3.3 M)
- Demonstration support and PIE of 10 bundles (\$0.5 + \$1.4M)

Not included is the cost of transportation of the MOX fuel to the station.

In parallel, safety and licensing submissions to the Atomic Energy Control Board for transition to a mixed oxide core, and for the full unrestricted use of mixed oxide fuel following the transition would be prepared. The results of the prototype irradiation and PIE results would

be documented and form a part of the submission package. The cost is included in Section 2.9.4 on AECB Licensing.

Overview: Advanced MOX Fuel

The plutonium disposition strategy assumes that the reference 37-element bundle will be used first for MOX fuel, since it can be quickly implemented. The advanced MOX design would follow at a later date. The two main differences of the advanced design from the reference that would require specific verification testing are the geometry (43-element CANFLEX rather than 37-element), and the discharge burnup (double that of the reference MOX fuel). The advanced MOX fuel qualification would focus on these aspects; much of the generic MOX development done for the reference MOX fuel would also apply to the advanced MOX design (such as generic irradiation testing, fuel fabrication development, powder/pellet specification, etc.). Moreover, much of the generic development and testing of the CANFLEX bundle will be done outside of the MOX program, on a compatible timescale. The CANFLEX program includes CHF-enhancement technology. Many of the activities required for qualification of the advanced MOX bundle could be done in parallel to the program above for the reference MOX fuel.

The total cost of the verification program for the advanced MOX fuel would be about \$17M over a four year period.

Confirmation of Design: Advanced MOX Fuel

NRU Irradiation Testing

Some of the irradiation testing required for the advanced MOX design could piggy-back that for the reference design. For example, demountable bundle irradiations could test fuel performance at the higher burnups required for the advanced design, and be initiated at the same time as for the reference design. In addition, irradiation testing in NRU of several CANFLEX bundles for a range of power/burnup histories will be required. It is assumed that four advanced MOX bundles would be irradiated at the high power envelope to high burnup (30,000 MWd/te, about 3 years), and that two others would be power ramped after irradiation at lower power to a burnup of 5,000 - 10,000 MWd/te. Fabrication, irradiation and PIE of these CANFLEX bundles over a four year period would cost approximately \$10M.

Critical Heat Flux Measurement

The same Freon- and water-CHF measurements will be required for the advanced MOX design as for the reference design. The total cost will be about \$2.9M.

Out-Reactor Tests

The CANFLEX program includes CANDU-6 specific out-reactor tests, such as fuel handling, flow visualization, vibration, fretting and endurance tests. Tests specific to Bruce A will be required for the advanced MOX fuel design. As with the 37-element bundle, analysis and tests will be required to confirm adequate design margins for the strength of the bundle.

These activities will take about 2 years at a cost of about \$3.6M.

Fuel Design

Assessments will be performed to finalize the design, and to check that the design meets the requirements. The design assessments include: design requirement specifications, preparation of technical specifications of powder and pellets, design calculations, including modifications of computer models, review and incorporation of data from NRU irradiation, design review, and preparation of the design manual for submission to the AECSB for prototype bundle irradiation. Since the work performed for the reference MOX fuel will be used extensively for the advanced MOX fuel, the activities are estimated to take 2 years at a cost of \$0.7M.

Prototype Bundle Irradiation in Bruce A: Advanced MOX Fuel

Most of the fabrication issues would have been addressed in the reference MOX fuel program. The prototype irradiation of the advanced fuel would focus on the different bundle geometry, and higher burnup.

About 20 MOX bundles would be fabricated and irradiated in the Bruce A reactor to their nominal discharge burnup (peak element burnups as high as 30,000 MWd/te). As in the reference case, following discharge, these bundles will be visually inspected at the fuel bay, and several selected bundles or elements will be shipped to the hot cell for detailed PIE.

The irradiation and PIE would take place over 4 years. The total cost would be \$3.9M, including:

- fabrication of 20 MOX bundles (\$3.0M)
- PIE of 10 bundles (\$.9M)

In parallel, safety and licensing submissions to the Atomic Energy Control Board for transition to the advanced MOX fuel will be prepared. The results of the prototype irradiation and PIE results will be documented and form a part of the submission package. The cost is included in Section 2.9.3 on AECSB Licensing.

2.9.3 Plant Changes

Plant changes required to prepare the Bruce Station for receipt and use of MOX fuel are described in Sections 2.4 and 2.6. These include a facility for receipt and storage of new fuel and new storage racks for the spent fuel and associated control and logistic modifications. A program to design, license, construct and test these changes is required, also as described in section 2.4. This program has 4 phases. Schedule and estimated cost, including contingency, follows:

1)	Phase 1 - Startup, Preliminary design
(duration; 12 months) | \$0.7M |
|----|--|--------|

2)	Phase 2 - Detail Design (duration 12 months)	\$0.8M
3)	Phase 3 - Construction & Procurement (duration 18 months)	\$13.6M
4)	Phase 4 - Installation and Commissioning including 7 day plant outage and operators retraining (duration 10 months)	\$2.2M
5)	Contingency (15%)	\$2.6M
	Total Estimated Cost	\$19.9M

2.9.4 AECB Licensing

Separate licensing activities are required for the confirmatory test phase, and for the Bruce plant operational phase. These are described in the following sections

A number of activities involving production of test fuel and testing of fuel leading up to placing fuel in the Bruce reactor will require licensing of research facilities. Licensing of MOX fuel tests to be conducted in reactor and relicensing of the Bruce A reactor for operation with MOX fuel will also be required. Safety analysis involving criticality will be needed.

2.9.4.1 Licensing of Research Facilities and Tests

AECL Research has licensed facilities available to conduct needed physics and irradiation tests. Licensing for the irradiation of MOX fuel bundles in the NRU reactor and physics tests in ZED - 2 for this project requires routine notification of the AECB and subsequent approval. This should be initiated by the second quarter of 1995 to easily meet the third quarter of 1995 schedule for irradiation in NRU and physics tests in ZED - 2.

As noted in Section 4, there are two possibilities for producing trial bundles for use in the confirmatory testing program: Los Alamos TA-55, and the RFFL at Chalk River. Regulatory approval for operating the TA-55 is not addressed in this report. The RFFL manufacturing facility is in the process of being relicensed by AECB for general advanced CANDU bundle development involving use of enriched uranium and plutonium from a number of sources. This will be operational by mid 1995. Licensing for the preparation of MOX fuel bundles for this project requires routine notification of the AECB and subsequent approval. This should be initiated by the end of 1994 to easily meet the mid 1995 initiation of fuel manufacture target.

2.9.4.2 Reactor Licensing

The introduction of MOX fuel into the Bruce reactor will be a 2 stage process with irradiation of a limited number of bundles (up to 50) followed by careful inspection. Should quality standards be met, the test irradiations will be followed by routine on power fuelling with MOX bundles replacing natural uranium bundles as they reach the end of life. Licensing of the test

program and operation with the full scale transition to MOX will be required. Early attention will focus on issues related to handling a limited number of MOX bundles. Fuel handling, criticality, refuelling and physics/thermalhydraulics of isolated bundles will thus require early licensing analysis. This will be followed with full scale safety analysis of the full core and licensing by AECB. Development and approval of licensing criteria for MOX fuelled reactor will be required. Total cost of reactor licensing is estimated at \$11.4 million.

New Fuel Handling and Prototype Bundle Irradiation

The quantity of new fuel stored on site and the total content of plutonium in the fuel will require that it be classified as Category 1 material and appropriate security provisions be established to meet regulatory requirements of both the AECB and the IAEA. All fuel movements into and out of the secure storage facility will be subject to full scope IAEA safeguards and it will be necessary to demonstrate to the AECB appropriate measures for safe handling of new fuel that meet regulatory requirements.

This demonstration will be needed for the prototype irradiation as will analysis of criticality for a limited number of bundles and safety assessment of isolated bundles in a natural uranium core.

Analytic Methods Development

Some review and training in criticality assessment methods will be required. Plutonium generates more fission products. Fuel may respond somewhat differently from natural uranium in accident conditions. This work should start with project initiation with a focus on criticality issue followed by development of refined fission product assessment techniques as information is gathered from irradiation tests. Schedule from beginning 4th quarter 1994 to end 1998 with transition to MOX core. Estimated cost is included in the full scope safety analysis.

Full Scope Safety Analysis

The safety and licensing of Bruce NGS A for operation with MOX fuel will require a full scope re-analysis of the entire design basis to demonstrate that safety design objectives continue to be met and that the regulatory limits imposed by the AECB are met. This re-analysis will involve a complete update of the Safety Report to include any changes in plant design, such as those identified with new fuel storage and handling and spent fuel handling, as well as the design basis accident analysis. The latter will also postulate and address criticality accidents associated with both new and spent fuel. This would cost of about \$11.4 million based on recent experience from similar projects.

Plant Staff Training

The special security requirements for fresh fuel and criticality considerations will require additional training for staff at the plant to ensure requirements are met. This will be scheduled to begin in earnest in mid 1998 for completion by the end of 1998. Limited training will be provided earlier for the prototype irradiations.

2.9.5 Transition to Equilibrium Core

The Reference Pu-MOX fuel bundle has been designed to give a fuel discharge burnup of 9,700 MWd/te, which is only slightly more than the discharge burnup of 8,300 MWd/te for the standard natural fuel in Bruce A. This means that the reactivity effect of the Reference Pu-MOX fuel bundle is only slightly higher than that of the natural bundle. The use of 2-bundle shift fuelling scheme reduces the perturbation effect due to refuelling with the Reference Pu-MOX fuel bundles. Therefore, the transition from natural fuel to the Reference Pu-MOX fuel in Bruce A can be achieved simply by refuelling the reactor with the Reference Pu-MOX fuel bundles on power. The fuelling rate is 15.5 Pu-MOX fuel bundles per full power day and there are 6,240 fuel bundles in the core. Hence, it will take at least 400 FPDs to displace all the natural fuel bundles with the Reference Pu-MOX bundles. However, because channels are refuelled at a different rate in the reactor, some channels will take longer than 400 FPDs to have the natural fuel bundles completely replaced by the Reference Pu-MOX bundles. It is expected to take at least 600 FPDs to reach the equilibrium Pu-MOX core condition. The same procedure can be used to replace the Reference Pu-MOX fuel bundles with the Advanced Pu-MOX fuel bundles at a later stage if it is required. An alternative Strategy for the transition would involve start up after a major scheduled outage with a full MOX core. In this case the 7 day outage time allowed for in section 2.9.3. would not be incurred. The particular transition strategy would be optimized as part of the implementation program.

2.10 Plutonium Annihilation Analysis

2.10.1 Neutronics

The CANDU reactor has a superior ability to annihilate plutonium and the minor transuranic actinides. This superiority is evident from the high (> 80 %) annihilation rate that is achievable per fuel cycle thereby eliminating the need for reprocessing and refabrication cycles.

This ability of CANDU is related to the basic design features of using heavy water as moderator and a fuel handling system that provides on-power refuelling capability. Both design features lead to low neutron absorption rates in the CANDU lattice due to the low neutron absorption cross section of the heavy water and the absence of burnable poisons for reactivity suppression. As a result, CANDU can be operated with a variety of low grade (low fissile content) fuels as these provide sufficient reactivity in a CANDU lattice.

The extent of CANDU's fissile economy is illustrated in Table 2.10.1 (note that CANDU AB stands for CANDU Actinide Burner) where the fissile inventory for several reactor concepts is compared for 1 GW(e) cores. The inventory needed for CANDU is between 2 and 4 times lower. This is the case for a reactor that is fuelled with natural uranium.

Table 2.10.1

FISSILE INVENTORY (TE) OF 1 GW(E) CORES

FBR	PWR	CANDU	CANDU AB
3 to 4	2 to 3	1	0.05

For plutonium annihilation, the fuel consists of a mixture of the plutonium isotopes in a neutronically inert matrix. Several materials for the matrix have been proposed and are discussed in Section 2.10.2. The absence of ^{238}U eliminates the main source of the plutonium isotopes. It also eliminates the main neutron absorber of the lattice resulting in a remarkable improvement in neutron economy. The loss of the neutron absorption in ^{238}U which comprises over 40% of the neutron absorption in the CANDU lattice, reduces the fissile requirement of the CANDU AB (Actinide Burner) relative to the natural uranium fuelled CANDU. The fissile inventory for a 1 GW(e) core drops to 50 kg.

A lower fissile inventory requires a correspondingly higher operating neutron flux level to produce the rated power. The higher operating neutron flux level is the basis for the CANDU superiority in the annihilation process. In particular, the on-power refuelling system is used to move the fuel into regions of higher flux as its fissile content depletes during irradiation. Through proper fuel management of this type a very high percentage of the plutonium can be eliminated before the fuel is discharged.

In order to appreciate the extent of the CANDU flux advantage, the thermal neutron flux levels

in the fuel for the reactor concepts of Table 2.10.1 are compared in Table 2.10.2. Note that the thermal flux level in the CANDU AB is comparable to the fast flux level in the FBRs of current design.

Table 2.10.2

NEUTRON FLUX LEVEL IN FUEL

Reactor Type	Neutron Flux (n/cm ² *s)	
	Thermal (< .625 eV)	Fast (500 keV)
FBR	-	0.5E+16 - 1.0E+16
PWR	8.0E+13	3.0E+14
CANDU	1.4E+14	0.7E+14
CANDU AB	5.0E+15	0.7E+14

To understand the annihilation process, the capture and fission cross sections of the plutonium isotopes in thermal and fast energy spectra must be considered, (Table 2.10.3). In a fast spectrum, all the plutonium isotopes are fissionable, but the fission cross section is relatively small; about 3 barns for 0.8 mev neutrons. In a thermal spectrum, only the isotopes with odd mass numbers are fissionable, but the fission cross sections are large; several hundred barns.

Table 2.10.3

NEUTRON CROSS-SECTIONS
(barns)

ISOTOPE	THERMAL		FAST	
	σ_c	σ_f	σ_c	σ_f
Pu 238	540	18		2.5
Pu 239	269	748	0.02	2.0
Pu 240	290			2.0
Pu 241	358	1011		2.0
Pu 242	19			1.5
Pu 243	87	196		

The annihilation process in CANDU involves first conversion from even to odd and then fission

by thermal neutrons of the odd numbered isotopes. The high thermal neutron flux level provides an appreciable annihilation rate in spite of the conversion step required first. Reaction rates (per nuclide) of the plutonium isotopes in CANDU and FBRs of current design are compared in Table 2.10.4. The reaction rates in CANDU are between one and two orders of magnitude higher.

Table 2.10.4

Comparison of Reaction Rates (Relative Units)

ISOTOPE	FBR		PWR		CANDU AB	
	capture	fission	capture	fission	capture	fission
Pu 238		1.25E+16	4.32E+16	1.44E+15	2.70E+18	9.00E+16
Pu 239	1.00E+14	1.00E+16	2.15E+16	5.98E+16	1.35E+18	3.74E+18
Pu 240		1.00E+16	2.32E+16		1.45E+18	
Pu 241		1.00E+16	2.86E+16	8.09E+16	1.79E+18	5.06E+18
Pu 242		7.50E+15	1.52E+15		9.50E+16	
Pu 243			6.96E+15	1.57E+16	4.35E+17	9.80E+17

The absence of ^{238}U has a major impact on the fuel management strategy that is used in the annihilation option. As formation of ^{239}Pu is eliminated, the reactivity of the lattice drops rapidly with fuel burnup. The refuelling rate required to maintain criticality is significantly higher compared with the reactor that burns natural uranium. The refuelling rate is kept within the capability of the Fuel Handling System by adjustment of the initial fissile content of the bundles. An increase in fissile content, however, requires reactivity suppression which reduces the thermal neutron flux level and subsequently increases the plutonium content of the discharged bundles.

The fuel management strategy used is, therefore, a balance between the fissile content and the refuelling rate to obtain the optimum benefit from the fuel handling capability that is available. The optimization includes the option of shuffling the fuel between bundle positions and/or between channels.

With a bundle plutonium inventory of 73 g, (isotopic mix shown in Table 2.10.5), the ^{239}Pu content at discharge from a once-through (no shuffling) cycle is reduced by 87 %. The isotopic mix of the discharged bundle is shown in Table 2.10.6.

Table 2.10.5
Bundle Plutonium Content

Isotope	g.
Pu 239	68.5
Pu 240	4.23
Pu 241	0.26
Pu 242	0.02

Table 2.10.6
Isotopic Mix of Discharged Bundle

Isotope	g.
Pu 239	9.2
Pu 240	15.2
Pu 241	3.3
Pu 242	2.7

To assess the performance from the plutonium disposition viewpoint, the annihilation rate is expressed in terms of the critical mass of the plutonium isotopes. Table 2.10.7 gives the critical masses of the isotopes used in this assessment. Using the once-through (no shuffling) cycle in a 480 channel (950 MW(e)) CANDU, 190 critical masses are inserted per year of which 158 critical masses are annihilated. The fuelling rate required for this cycle is six fuel channels per FPD. The bundle fissile content is low enough to refuel the channel completely. It is estimated that the Reactor Regulating System is capable of handling this rate of refuelling.

Fuel shuffling will be required to annihilate the plutonium remaining in the bundle at the end of the once-through cycle. One additional pass (or at most two) is sufficient to reduce the plutonium content of the discharged bundle to a level that is comparable to the losses experienced in fuel reprocessing.

Table 2.10.7
Critical Mass of Plutonium

Isotope	kg.
Pu 239	10
Pu 240	38
Pu 241	12
Pu 242	79

The additional pass would entail an increase in the fuelling machine duty. It is expected that some modifications will be required to the design of the Fuel Handling System to achieve this capability. However, the conceptual design of the of the system will remain unchanged. A discussion on Fuel Handling System design and capability is given in Section 2.10.3.

The absence of ^{238}U and the predominance of ^{239}Pu affects the dynamics of the CANDU lattice. The coolant void reactivity coefficient is negative. The fuel temperature reactivity coefficient is zero for fresh fuel but tends to become positive with irradiation. However the power coefficient of reactivity is negative or zero throughout the fuel life. As the power coefficient of reactivity is near zero for the currently operating CANDU reactors, there appear to be no implications on the Reactor Regulating System of using this type of fuel.

There is a reduction in the delayed neutron fraction relative to the natural uranium CANDU. The value for this lattice is between 3.0 and 3.8 mk compared with 6 mk. However, the neutron generation time for the lattice is significantly longer (2 ms) and the discontinuity between the sub-prompt and super-prompt neutron transient is relatively small.

2.10.2 Selection of Fuel Inert Matrix Material or Carrier Material

A number of non-uranium-based candidate materials to hold, or *carry*, plutonium exist. SiC, with its excellent thermal conductivity and very high melting temperature appears to be one of the best. The critical test that must be performed on SiC is to demonstrate that it is fully compatible with water at fuel operating temperatures. Silicon, alloyed with the plutonium (2-3 vol. %), has very high thermal conductivity; however, it has the lowest melting temperature (1410 °C) of the candidates considered. Potential swelling and blockage of the fuel channel as a result of a large temperature excursion must be addressed. If this can be resolved, it represents a very good candidate. Finally, CeO₂ and MgAl₂O₄ are good candidates, with thermal conductivities higher than UO₂. However, if very high thermal conductivities are required for reactor physics reasons, these latter candidates could be eliminated and focus placed on SiC and Si (or other metal, e.g., Zr).

2.10.2.1 Introduction

The objective is to find a suitable inert matrix material or carrier material to hold the Pu. Its function is to dilute the plutonium, since pure plutonium or PuO₂ would be much too highly fissile, without producing any new plutonium during the irradiation (as UO₂ does). The following sections identify the properties required for such a carrier material, list and assess a selection of materials that appear the most promising, and give a development program for selecting, testing and qualifying a candidate material.

2.10.2.2 Properties Required For a Carrier Material

Neutronics: All elements in the carrier material must have low neutron absorption.

Melting Temperature: A high melting temperature is required so that fuel operating temperatures are below approximately 0.5 of the melting temperature (Kelvin). This ensures that thermally activated processes, such as fission gas diffusion, are acceptably slow. A high melting temperature also ensures that there is an acceptable safety margin between fuel operating temperature and melting temperature, to accommodate potential off-normal excursions.

Compatibility with Coolant and Clad: Chemical compatibility with water at coolant temperatures is a requirement. Compatibility with the clad is also required, although the clad need not necessarily be zirconium or a zirconium alloy.

Phase Stability: Phase transformations as the fuel temperature cycles up and down are unacceptable if there is a volume change associated with the transformation and whether the material would lose mechanical integrity. Similarly, a loss of crystallinity, thus forming an amorphous phase, as the fuel is irradiated is unacceptable if it would lead to similar problems. This latter point is one of the aims of the test program outlined below.

Thermal Conductivity: High thermal conductivity is advantageous, since this allows low fuel operating temperature (for a given element power rating). This provides increased safety margins in the carrier material. Thermal conductivity at least as high, and preferably higher,

as UO_2 is required.

Solid Solubility with plutonium: In order to disperse the plutonium on an atomic scale in the carrier material, the carrier should be able to dissolve 1-3% plutonium. MOX fuel with small particles of PuO_2 in a UO_2 matrix have been used successfully in LWR power reactors, but reactor operators must be conservative in the burnup limits of the fuel. Currently, the trend is to fabricate MOX fuel with the plutonium dispersed as much as possible. Fuel assemblies with improved fuel have been used in power reactors, and are currently awaiting post-irradiation examination.

Irradiation Properties: A major irradiation test program would be required, as detailed below.

2.10.2.3 Candidates

A list of candidate materials to hold the plutonium includes ZrO_2 , BeO , MgAl_2O_4 , CeO_2 , SiC and ThO_2 . The last material listed, ThO_2 , is included because no new plutonium is produced, even though its irradiation produces new fissile material, ^{233}U . Thus, use of ThO_2 to hold the plutonium would achieve the goal of effectively destroying plutonium. If, however, a subsidiary goal were to not produce *any* new fissile material, ThO_2 would not be an acceptable candidate.

Since there is a considerable base of irradiation experience on ThO_2 , it would clearly be the simplest of the candidates to qualify for use as a carrier material for plutonium. It will not be further discussed in the following.

Other candidates considered were Al_2O_3 and silicide dispersion fuels. Alumina (Al_2O_3) was rejected because it has long been known that becomes amorphous with a volume change of 30% when used as a fuel. Uranium silicide dispersion fuels are used in CRL's research reactor NRU. These fuels consist of a meat of either U_3Si or U_3Si_2 dispersed in an Al matrix. Currently, uranium of 20% ^{235}U enrichment is used in NRU, but high-enriched uranium, 93% ^{235}U , has been used in the past. Thus, plutonium could be substituted for the U to make a plutonium-silicide dispersion fuel. However, the Al matrix has too low a melting temperature, 600°C, for safety considerations, and must, therefore be ruled out.

Alloys of plutonium and metals were considered, for example Zr alloyed with plutonium. An apparently promising candidate is Si, as it is cheap, abundant, and has good thermal properties. Also, there is irradiation experience with uranium silicides (larger amounts of uranium than required of plutonium for this purpose). This candidate is retained as a representative of metallic alloys and silicides.

2.10.2.4 Properties of the Candidate Materials

Relevant properties, not including irradiation experience, are discussed first; this is followed by a discussion of published material on irradiation experience.

(1) ZrO_2

Melting Temperature ($^{\circ}\text{C}$)	2715
Thermal Conductivity ($\text{W/m}\cdot\text{K}$)	1.9 (100°C)
	2.3 (1000°C)

Zirconia has a high melting temperature and good neutronic properties (absorption cross section of 0.184 barns for thermal neutrons) as long as hafnium, a persistent impurity is removed. Commercial grade Zr contains 1-3 % Hf, but nuclear grade Zr is essentially pure (~ 50 ppm Hf).

Thermal conductivity increases with temperature, in contrast to most materials. However, it is probably too low, being even lower than for UO_2 , which has thermal conductivity values of 9 $\text{W/m}\cdot\text{K}$ at 100°C and 3 $\text{W/m}\cdot\text{K}$ at 1000°C .

It is insoluble in water, and should be compatible with zircaloy sheath.

Phase stability is a potential issue. Zirconia undergoes phase transformations with large associated volume changes at $900\text{-}1000^{\circ}\text{C}$ (monoclinic to tetragonal) and at 2370°C (tetragonal to cubic). These phase transformations (the high temperature one is unimportant) cause cracking and would eventually disintegrate the fuel, after repeated temperature cycling through the phase transition temperature.

However, additions of yttrium, calcium, magnesium or possibly other materials will stabilize the high temperature cubic phase of zirconia. (This cubic phase is the fluorite structure, the same as UO_2 , PuO_2 , and ThO_2 .) Approximately, 8 at.% of the additive⁽²⁾ is required for full stabilization, but smaller amounts provide partial stabilization. The fracture toughness that this stabilization provides makes this material an intensely studied material in the ceramics field. The thermal neutron absorption cross sections of these additives are not large (Y, 1.28 barns; Ca, 0.43 barns; Mg, 0.63 barns), so while providing an extra degree of complexity in fabrication and possibly in irradiation, maintains this material as a good candidate.

Thus, zirconia's poor thermal conductivity is its most serious drawback. The fuel would have to run at lower powers than urania. Other, more acceptable candidates are sought.

(2) BeO

Melting temperature ($^{\circ}\text{C}$)	2530
Thermal Conductivity ($\text{W/m}\cdot\text{K}$)	220 (100°C)
	20 (1000°C)

The very high thermal conductivity, combined with a high melting temperature, are strong

advantages. The conductivity does decrease rapidly with temperature, but is still high at 1000°C and the fuel operating temperature is determined more by its low temperature conductivity (above coolant temperature) than its high temperature value. It has very low neutronic absorption (0.008 barns for thermal neutrons) and is a neutron moderator. It does not undergo phase transitions (hexagonal over its whole temperature).

A potential drawback to BeO is its toxicity. The hazard is highest when using powders, as fabrication would probably require. However, since plutonium is toxic already, the extra toxicity of BeO might be acceptable. Unknown is whether the material would form a solid solution with plutonium. The toxicity of BeO would make laboratory testing more difficult, but Whiteshell Laboratories has operated a Be laboratory in the past. Moreover, U could be substituted for plutonium to quickly and approximately examine solubility.

Its irradiation behaviour is, however, questionable (see next section).

(3) MgAl_2O_4

Melting Temperature (°C)	2135
Thermal Conductivity (W/m·K)	10.7 (100°C)

Spinel (MgAl_2O_4) is a candidate suggested by the Europeans. The French, supported by the Transuranium Institute at Karlsruhe, FRG) are actively considering it for an inert matrix (carrier) material for both LWR and Fast Breeder Reactors. An advantage is its perceived resistance to swelling in-reactor. The argument is that for swelling to nucleate, a number of vacancies must congregate together. Furthermore, there must be Mg, Al, and O vacancies *in stoichiometric proportions*, i.e., in ratios of 1:2:4. Since the probability that this will happen randomly is small, the material is resistant to swelling. It has a thermal conductivity of about 10.7 W/m·K at 100°C, marginally higher than that for UO_2 . Melting point is sufficiently high, 2135°C. Thus, it is a promising candidate.

(4) CeO_2

Melting Temperature (°C)	2600
Thermal Conductivity (W/m·K)	10.9 (100°C)

This material, and the next, are recent additions to the list of candidates. CeO_2 was added³ because its crystal structure is identical to UO_2 . Since UO_2 has good properties, except for thermal conductivity, and, especially, good irradiation performance, it is thought that this material could have similar good properties. Its thermal conductivity is about 10.9 W/m·K at 100°C, slightly higher than that for UO_2 , and its melting temperature is very high, 2600°C. Because of its close similarity with UO_2 , this is a *very* promising candidate.

³ suggested by Hj. Matzke, Transuranium Institute, FRG.

(5) Si

Melting Temperature (°C)	1410°C
Thermal Conductivity (W/m·K)	108 (100°C)

Silicon is abundant, cheap, and has good thermal properties. Also, there is irradiation experience with uranium silicides (with larger amounts of uranium than of plutonium required for this purpose). Silicon is generally relatively inert, although it is attacked by dilute alkali. Most acids do not affect it. Presumably, control of the coolant pH, together with the use of a Zr sheath would neutralize concerns about attack by the coolant. The ease with which silicon becomes amorphous in-reactor needs to be investigated. Possibly the use of amorphous silicon, which can be fabricated easily, is preferable. The volume change and glass transition temperature (temperature at which amorphous silicon crystallizes) may vary with plutonium content. (This could be investigated using uranium as a substitute for plutonium, initially).

The use of silicide fuels (U_3Si and U_3Si_2) for CANDU was investigated in the 1970's by AECL. They were not selected because they softened and swelled above a certain temperature, which depended on the burnup, and there was a reactor control problem, associated with the low heat capacity. These concerns would be re-addressed for the purer silicon-plutonium fuel.

(6) SiC

Melting Temperature (°C)	2700
Thermal Conductivity (W/m·K)	85 (100°C)

SiC is a material with a considerable industrial base of use, and, thus also, a large data base on properties, especially compared to other candidates considered above. It has a very high thermal conductivity and a high melting temperature. Both elements, Si and C, are low neutron absorbers, and C will also act as a neutron moderator. Although compatible with water at low temperatures, it must be tested for compatibility with water at reactor coolant temperatures, 300°C and above. Fabrication methods have been developed for industrial application, although these may require modification to incorporate plutonium. As long as this material tests compatible with water at high temperatures, it appears to be an excellent candidate.

The key thermal properties of the above candidates, together with those for UO_2 are collected in Table 2.10.8.

TABLE 2.10.8

Melting Temperatures and Thermal Conductivities of Candidate Carrier Materials

MATERIAL	Melting Temperature (°C)	Thermal Conductivity (W/m·K)		Heat Capacity ¹ (J/cm ³ ·K) (J/g·K)	
		100°C	1000°C		
ZrO ₂	2715	1.9	2.3	2.55	0.46
BeO	2530	220.	20.	3.07	1.02
MgAl ₂ O ₄	2135	10.7			
CeO ₂	2600	10.9		2.55	0.36
SiC	2700	85.		2.15	0.67
Si	1410	108		1.65	0.71
UO ₂	2878	8.75	3.21	2.58	0.24

Note: The heat capacity per *unit volume* is more relevant, for calculating temperature increases for a given input of heat, than per *unit weight* because the volume of the fuel element will be constant whatever the carrier material.

2.10.2.5 Irradiation Experience

The only program to test carrier materials in-reactor, with a fissile component to introduce fission damage to the carrier material, was a Bettis program [1-3] in the early 1960's to select a material for a full-core loading of the Shippingport test reactor. They tested Al₂O₃, ZrO₂, ZrO₂+CaO, BeO and ZrSiO₄, all containing large amounts of UO₂ (>20 wt. %). Only ZrO₂ behaved satisfactorily for them: Al₂O₃ became amorphous after only a very short time in reactor; BeO exhibited high swelling (from fission gas bubble formation) and the BeO matrix material "suffered considerable fission fragment damage". The authors of the Bettis reports suggest that it is the cubic crystal structure of ZrO₂ which provides its resistance to damage. This was supported, for example, by the results on the candidate material ZrO₂+CaO+UO₂, which existed as a two-phase material, one phase cubic and one non-cubic. The cubic phase showed resistance to damage, whereas the non-cubic phase did not. Al₂O₃, BeO, and ZrSiO₄ are all non-cubic and did not test well. However, CeO₂, MgAl₂O₄, and SiC, candidates listed above but not tested by Bettis, all form cubic crystal structures. Therefore, there is reason to expect that one or more of these candidates will be acceptable as a carrier material for plutonium in CANDU.

The Bettis results show that ZrO₂ might make a good carrier material for our purposes. However, it is the least desirable of the suggested candidates from a thermal conductivity consideration. Therefore, a program of developing a carrier material for plutonium in CANDU

would begin with the candidates CeO_2 , MgAl_2O_4 , and SiC .

2.10.2.6 Program to Select and Qualify a Carrier Material

There is strong interest in developing and qualifying a carrier material, especially by the French, for both LWR's and FBR's. Their interest is to burn actinide waste products generated in the normal LWR fuel cycle. To select a carrier material from the above candidates, accelerators can be used for initial screening, as described below. This must be followed by a substantial in-reactor irradiation program (*with* dissolved fissile material, although not necessarily plutonium initially). Simultaneously, an out-reactor properties-determination program must be undertaken to establish that there are no other deleterious properties of the candidate material, such as reaction with water or the selected clad material.

Much of the material presented at this working group meeting was on the chemistry of separation of the actinides from the rest of the irradiated LWR fuel, and on refabrication of new fuel elements containing these actinides for burning in FBR's or LWR's. The fabrication methods centered on sol-gel fabrication to produce a homogeneous mix of the actinides in the new fuel. One concept was to produce *low*-density pebbles by sol-gel methods for subsequent pressing and sintering into regular pellets.

(1) Accelerator Testing

Since there are too many candidates to test directly as fuels in-reactor, the TANDEM accelerator would be used to provide an initial screening. All of the promising candidates would be subjected to bombardment by 70 MeV (fission-fragment energy) iodine at typical fuel operating temperatures. The bombarded areas would be analyzed to check for unwanted phase changes. The best two or three candidates, based on the accelerator results and the earlier analysis of material properties, would be selected for reactor testing.

Estimated time: 1-2 years.

(2) Reactor Testing

After laboratory fabrication tests, with uranium substituting for plutonium, fuel rods will be made for testing in NRU using a demountable bundle (allowing individual rods to be removed during the irradiation for post-irradiation examination). Test powers would range for the elements of the bundle by fabricating with different concentrations of uranium. This will provide an excellent first in-reactor test-program, since the fuel matrix material will be expected to perform similarly with regard to fission gas release, swelling, and other performance indicators, whether the fissile component is plutonium or ^{235}U . Following a successful test program with uranium, fuel rods with plutonium will be made in CRL's Recycle Fuel Fabrication Facility.

The test program would span the expected operating range of fuel power densities and temperatures in CANDU reactors. Multi-bundle tests in a CANDU reactor would be required, finally, before partial-core or full-core loading.

It is estimated that this program would require about 10 years.

2.10.3 Fuel Handling Considerations

CANDU reactors feature on-power refuelling. While the reactor is operating, a pair of fuelling machines latches onto opposite ends of one of the fuel channels. The latching operation seals the fuelling machines to the fuel channel, and makes them into temporary extensions of the fuel channel pressure boundary. The fuelling machines remove and store the channel closures and shield plugs, and then one machine pushes fuel bundles into the fuel channel. The other machine receives the "overflow". The fuelling machine magazine cylinders each hold two fuel bundles, so the bundles are inserted into the fuel channel and removed from it two at a time.

The fuel bundles received by the second fuelling machine can either be returned to the same fuel channel (to the same or a different position), put into another channel, or discharged to the spent fuel bay. It is possible, by juggling fuel back and forth between the two machines, to change any pair of bundles in the channel. During typical normal fuelling, 4 new fuel bundles are inserted at one end of the channel and 4 highly irradiated bundles are discharged at the other end. During subsequent refuelling operations, after a 4 month burn period, these 4 bundles will then be displaced a further 4 positions, entering into and residing in a higher flux region of the core. Repeated fuelling will eventually see these bundles moved through the core, with different residence positions, and ultimately discharged with high burnup. A typical residence time is about one year in reactor. The core is divided into 2, 4 and 8 bundle push regions.

With natural uranium fuel, the Bruce A fuelling machines visit 4 fuel channels per full power day (FPD). The fuelling rate for the plutonium annihilation option is estimated to be 6 channels per FPD, about 1.5 times the current rate.

Fuel handling experts at Bruce A believe that the existing three trolley (2 fuelling machines are carried by each trolley) system could deliver a 50% increase in performance. This would require a series of performance enhancing modifications, additional operating personnel, and an increase in maintenance staff to reduce downtime. A fourth trolley, providing two more fuelling machines, would increase the fuelling rate an additional 33%.

Some work has been done by the fuelling machine designer (GE Canada) on an advanced fuelling machine design, offering increased performance over the existing design. This would provide an alternative way to increase the fuelling rate. Six new machines on three trolleys might be less expensive (even including design and development costs) than adding a fourth trolley, because the service area required for the fourth trolley would not be required.

In summary, it appears that the increase in the fuelling rate required for plutonium annihilation at Bruce A is achievable, and that the work required to accomplish this is a practical modification to the current design fuel handling system. downtime.

2.10.4 Conclusions

- 1). It is feasible to annihilate over 80 % of the initial ^{239}Pu in currently operating CANDU reactors and using refuelling rates that are available with current fuel handling technology. Higher burnouts would be achieved with fuel shuffling.
- 2). Some reactivity suppression, using poison in the fuel is desirable, to reduce refuelling rates and consequently reduce bundle fabrication costs.
- 3). The absence of ^{238}U and the predominance of ^{239}Pu has an influence on the lattice neutronics. However, the lattice stability and kinetics are acceptable throughout fuel life.
- 4). The power coefficient of reactivity is negative or near zero for the equilibrium fuel situation.

3.0 Environmental Review of Reactor Systems

3.1 Environmental Review Process

Any major proposal, plan or program undertaken by Ontario Hydro is subject to review and approval under the provincial Environmental Assessment (EA) Act, unless it is within the scope of an existing EA approval or exemption. Furthermore, if the undertaking is nuclear and requires approval at the federal level from the AECB, review under the federal EA process is also required. In cases where both jurisdictions are involved, there tends to be coordination of review processes.

3.1.1 Provincial EA Process

The general requirements of the provincial EA process are specified in Section 5(3) of the EA Act. In this Act, "environment" is defined as including social, cultural and economic aspects as well as the physical and biological environment. It is noteworthy that the Act requires evaluation of not only the proposed undertaking, but also of alternative methods of implementing the undertaking and alternatives to the undertaking. It should also be noted that the Act does not prohibit a feasibility study, including research, or any other activities necessary to comply with the Act (Section 5(2)).

Because construction of Bruce GS A was well under way before the EA Act came into force in 1976, the program of "planning, designing, constructing, operating and maintaining" Bruce GS A was formally exempted from the application of the Act (Exemption Order OHN-14, October 1976). Additional rationale for the exemption was the fact that parts of the project were subject to review and approval under provincial environmental protection legislation. Apart from this explicit rationale, of course, Bruce GS A was approved and licensed by the AECB.

Ontario Hydro's interpretation of this EA exemption is that ongoing operation and maintenance of Bruce A, including basic rehabilitation to ensure continued safe and reliable operation, does not require further provincial EA approval. On the other hand, major modifications (whereby the purpose, use, capacity or location of the facility would change substantially) cannot be presumed to be covered by the exemption. While there appear to be good reasons for positioning a proposal to operate with MOX fuel within the scope of the existing exemption, it must be recognized that the Minister of Environment has the power to suspend or revoke an exemption or alter its terms and conditions. Consultation by Ontario Hydro with the Ministry of Environment and Energy would be necessary to ascertain the applicability of the EA Act and existing exemption to a proposal to utilize MOX fuel.

3.1.2 Federal EA Process

EA reviews at the federal level are presently governed by the Environmental Assessment and Review Process (EARP) Guidelines Order (1984). However, the Guidelines Order will eventually be superseded by a new Canadian Environmental Assessment Act (CEAA). This Act was passed in June 1992, but has not yet been proclaimed in force. Four of the fifteen or so regulations which will define the scope and application of the Act must be finalized before the Act is formally proclaimed and implemented. These four regulations are (1) The Law List, (2)

Exclusion List, (3) Inclusion List and (4) Comprehensive Study List. Public review of draft regulations ended in December 1993, but it is uncertain at this time when the regulations might be finalized and the new Act proclaimed. For purposes of this study, it is assumed that the existing EARP Guidelines Order will continue to apply.

The Guidelines Order applies to any proposal that involves a federal proponent, federal lands, federal funding or requires a federal approval decision (e.g. by a regulatory agency). Thus, the EAR process would be invoked in this case to the extent that approval at the federal level from AECEB is required.

In general, the Guidelines Order requires a proponent to "screen" or assess a proposal with regard to:

- (a) the potential environmental effects of the proposal and the social effects directly related to those environmental effects, including "any effects that are external to Canadian territory"; and
- (b) the concerns of the public regarding the proposal and its potential environmental effects.

Consistent with the Guidelines Order, the AECEB's process for EARP screening involves the following basic steps:

- Proponent initiates informal discussions prior to submission of proposal
- Proponent initiates environmental evaluation and establishes a program to obtain public views (monitored by AECEB staff)
- Proponent submits proposal and documentation on environmental evaluation
- AECEB staff review submission with reference to:
 - their Exclusion List (types of proposals that would not cause adverse effects)
 - their Automatic Referral List (types of proposals that would likely cause significant adverse effects and, thus, would be automatically referred for more detailed assessment and public review)
- Section 12 of the Guidelines Order - Are the identified effects:
 - insignificant/mitigable?
 - unknown/further investigation and assessment required?
 - adverse/significant/unacceptable?
- Section 13 of the Guidelines Order - Are public views:
 - in favor/indifferent?
 - opposed/further public review desirable?
- AECEB staff complete screening report and recommend to AECEB Board
- AECEB Board makes decision on proposal and informs proponent and Federal Environmental Review Office (FEARO) to either:
 - proceed with any necessary mitigation and follow-up, or
 - modify proposal and rescreen, or
 - refer for public review by a Panel, or
 - postpone or reject.

- postpone or reject.

3.1.3 Approach to Environmental Review of Bruce A Operation with MOX Fuel

The review process envisaged by Ontario Hydro to address a program to disposition plutonium in Bruce reactors would consist of the following activities:

- Consultation with Government ministries/departments/agencies, early in the definition phase of the project, would be necessary to confirm applicability (or otherwise) of Ontario/Federal EA processes.
- Description of key features of the project, such as:
 - Modifications to Bruce A facilities,
 - Modifications to Bruce A operation,
 - Reference case (2 units, 50 Mg Pu/24 yrs),
 - Operation within existing "licensing envelope",
 - Alternative case (4 units, 100 Mg/25 yrs),
 - System for transportation of MOX fuel from Hanford, Washington,
 - Facilities and operational system,
 - Security,
 - Modifications to used fuel storage and disposal.
- Scoping of key issues and concerns (coordinated with evaluation of public views/concerns and potential socio-economic impacts.)
- Develop positions that address the following major factors:
 - Public safety (radiological focus)
 - Natural environment effects
 - Public concerns & socio-economic impacts.
- Screening/assessment/documentation and submission.

3.2 Environmental Review Schedule

The duration and complexity of the review process will depend on whether the proposal:

- (a) needs only to be screened under the federal EAR process (likely the shortest duration);
- (b) needs to be reviewed under both federal and Ontario EA processes, i.e. if the proposal is deemed beyond the scope of the existing Bruce A exemption (more time could be required); or
- (c) is likely to be referred for full public review, conducted by a Panel, presumably under a joint federal-provincial process (likely the longest duration).

Consultation with appropriate Government bodies, as suggested under 3.1.3, would be necessary to develop a realistic schedule. However, based on recent Ontario Hydro project experience, the minimum duration (for a federal EAR screening process) is estimated to be of the order of 12 months, all of which could be managed within the normal AECB licensing process, at an estimated cost of \$0.7M. If a more comprehensive environmental review becomes necessary, including public hearings, the project schedule could be extended considerably (by several months at least, and perhaps up to a year).

3.3 Public Input

An important component of the Definition Phase of a proposal to utilize MOX fuel in Bruce NGS A involves public affairs initiatives because it is crucial that Ontario Hydro undertake such a project in the interest of its stakeholders. The concept must be assessed for its full socio-economic and public acceptability implications.

From Ontario Hydro's perspective the public affairs objective is to support internal decision making regarding the use of MOX fuel in Bruce reactors. The goals therefore are to:

- work closely with key stakeholders during the Definition Phase of a MOX project to develop an understanding of the project and its implications, and to assess its feasibility and suitability;
- provide opportunities for secondary stakeholders to be involved during the study; and,
- provide responsive media relations initiatives.

3.3.1 Nature of the Proposed Project

The use of mixed oxide fuel in Bruce reactors has many characteristics which must be considered in the design of an appropriate public affairs approach.

It would involve existing stations in host communities with a long-established relationship with Ontario Hydro and familiarity with its nuclear operations. The project involves the existing operations, and does not require additional, or new nuclear operations, or facilities.

At the same time, the project would involve communities who have no immediate experience with the nuclear operations of Ontario Hydro, such as those communities along a potential transportation corridor outside the Bruce Nuclear Power Development area.

The nature of the project poses a potentially wide range of risks and benefits, both real and perceived, which may influence how stakeholders respond to this proposal. These could include security, safety, political and international issues, as well as potential social, economic and environmental effects. Ontario Hydro has the responsibility to assess and respond to these issues as part of the Definition Phase of a project.

3.3.2 Stakeholder Attitudes: Existing Information

Primary Information

Ontario Hydro conducted a workshop with key stakeholders of the ten host communities surrounding Bruce Nuclear Power Development (BNPD) in June 1994 in order to assess their preliminary response regarding possible future detailed study of the use of plutonium fuel and to seek their input into the design of an appropriate stakeholder involvement approach. The host community generally supported the study and reacted positively to the proposal.

Workshop participants were asked to list the issues that should be considered in a detailed study of the potential use of plutonium in the local reactor. The following issues were raised:

- public and employee health and safety concerns;
- security (any increase in the possibility of terrorism activity);
- potential environmental impacts such as impacts on water, the possibility of contamination;
- potential emissions;
- risk:
 - any increase in the risk of nuclear accident,
 - any potential radiation risk to the employees and the public;
 - any risk to the environment;
 - any risk associated with the spent fuel;
 - any risks associated with transportation;
- used fuel management, storage and disposal;
- potential socio-economic impacts on the community:
 - economic benefits;
 - negative impacts and offsets;
 - potential increase in the international profile of Bruce;
 - any potential benefits on local employment;
 - positive aspects on tourism;
 - any impacts on the viability of community;
- technical aspects:
 - alternatives to using Bruce A for plutonium dispositioning;
 - the rate of use Mixed Oxide Fuel;
 - alternatives to using Bruce A for the mission.
- impacts on Ontario Hydro:
 - fuel cost and rates;
 - maintenance costs (e.g. refuelling);
 - the anticipated longevity of the Bruce A operation;
 - implications on the viability of the retubing of Bruce A;
 - the cost efficiency of the proposal;
 - potential implications on the financial stability of Bruce A;
- implications on the fuel suppliers, and related employment; and,
- timing, how long would plutonium be used as a fuel.

Participants were asked what conditions must be satisfied so that the key stakeholders and the community could support a detailed study on the use of plutonium in the local reactor. The following conditions were suggested:

- the issues and concerns listed above should be addressed;
- that safety must be ensured;
- that the community must be involved in planning and decision-making;
- that an open and honest process for stakeholder involvement be in place; and,
- that an equity/humanitarian offset benefit (i.e. for allowing the destruction of nuclear weapons through this means) should be considered.

The participants of the workshop also provided input on their preferences regarding their involvement in any future studies. They expressed a desire for active participation during the Definition Phase. The extent of that involvement ranged from co-planning, possibly using the

existing Impact Advisory Committee in the community, to a consultation/feedback approach for providing input throughout the process.

Secondary Research Information

Existing public attitude research findings on general attitudes of Ontarians toward issues associated with nuclear power is used here to identify potential issues and concerns of stakeholders¹.

Historically in Ontario, slightly more than 50 percent of residential customers have been in favor of using nuclear power to generate electricity, and slightly less than 50 percent have been opposed, while nuclear power has historically received greater support from the business community²;

About 80 percent of the Ontario public respond that rate is the most important issue for Ontario Hydro, while only 2 percent mention nuclear power³;

Most Ontarians believe that existing plants should be kept but that no new plants should be built. Only 28% are strongly opposed to nuclear power. Support for the use of nuclear power tends to be somewhat higher among men, university graduates, and those with higher incomes⁴;

When asked what the most important advantages of nuclear energy are, almost 60% refer to environmental advantages. Major concerns include health and safety considerations associated with nuclear energy, disposal of nuclear waste, effects of radiation exposure, and safety of aging plants.⁵;

A summary of public attitude research findings based on five studies conducted between 1991-1993 on nuclear waste concluded that the public expects Ontario Hydro to⁶:

¹ Issue Management Analysis: Proposal To Use Plutonium Fuel At Bruce A, Corporate and Public Affairs, Ontario Hydro, 1994.

² Ontario Hydro Customer Satisfaction Index: The Summary Report of All Customer Classes 1993 Results. Book 3 of 3: Hydro Issue Analysis, Goldfarb Consultants, June 2, 1993.

³ The Goldfarb 1993: Energy and the Environment, Sector Book 7. Goldfarb Consultants, 1993.

⁴ The Energy Monitor, 1992. International Environmental Monitor Ltd. (Biannual). [Joint venture of Environics Research Group Ltd. and Synergistics Consulting Ltd]. October 1992.

⁵ The Energy Monitor, 1992-2. International Environmental Monitor Ltd. (Biannual). [Joint venture of Environics Research Group Ltd. and Synergistics Consulting Ltd]. October 1992.

⁶ FEARO Public Affairs Plan, Ontario Hydro, November 1993.

- disclose all information on worker and public health and safety associated with its nuclear operations;
- maintain close working relationships with host communities, involving them in decision making; and
- support effective third party review and regulation;
- operate at standards consistently exceeding government regulation;
- make decisions on nuclear waste management on the basis of safety, not cost;
- have effective plans to control nuclear fuel at every point in its life-cycle; and,
- operate according to "no danger/no defect" standards of worker and public safety and to demonstrate care, precaution, and prudence in its actions.

Case studies prepared in 1992 focusing on experiences in Alberta and Manitoba relating to the transportation of hazardous waste identified a key factor in the success of a route selection process is the overall approach to public consultation⁷. Emphasis on the integrity of the process, being open, willing to listen and address public concerns was critical to the success of route selection. In two case studies⁸, this approach resulted in the avoidance of extensive route selection processes; planning focused on optimizing health and environmental protection and on minimizing risk. In addition, a public consultation program that shares information and uses a joint problem-solving/partnership approach appears to help the public accept technical constraints related to transportation.

In summary, the available research information regarding stakeholder attitudes is limited, as is secondary research findings because public attitude research does not specifically address the issue of utilization of MOX fuel. In addition, the primary information is focused only on the host communities, and does not include the potential corridor communities. Clearly, more public attitude research and primary information would be required.

3.3.3 Stakeholder Selection

There are potentially many interested parties who could be included in an involvement program. A wide range of stakeholders can be expected to be interested in a proposal to utilize MOX fuel to disposition weapons grade plutonium. In order to focus the participation consistent with the program purpose, selection criteria would be used. The following selection criteria are suggested. Key, or primary stakeholders are those who may be either directly affected by the proposal; may be partners in the undertaking; may have a direct interest in the proposal; or may have a significant influence on the decision on whether or not to proceed with the proposal. Secondary stakeholders are those who may have an interest in or have an informed opinion regarding the proposal; who may be Hydro "watchers"; or who may be indirectly affected by the proposal. Tertiary stakeholders are those who have little direct stake or interest in the proposal and generally, do not directly influence planning and decision making.

⁷ Canadian Case Study Analysis of the Transportation Component of Site Selection Processes - A Discussion Paper, Prepared for Atomic Energy of Canada Limited, Hardy Stevenson & Associates, November 1992.

⁸ Ibid, page 47.

Based on the above criteria, a **preliminary** list of potential stakeholders would include:

Key Local Stakeholders

Host community (Bruce Communities)
Transportation corridor communities
Power Workers Union (Ontario Hydro)
The Society of Professional & Management Staff (Ontario Hydro)
Local Interest Groups
Local Aboriginal Communities

Key Institutional Stakeholders

Ontario Hydro
Atomic Energy of Canada Ltd.
Canadian Nuclear Association
Atomic Energy Control Board
Local MPPs
Minister & Deputy Minister Environment & Energy
Municipal Governments
Ministry Of Transportation
Royal Canadian Mounted Police/Ontario Provincial Police
Department of National Defence
Department of Natural Resources
Department of Foreign Affairs and International Trade

Secondary Stakeholders

Peace/Disarmament Groups
Energy Interest Groups
Fuel Suppliers/Existing Contractors

Tertiary Stakeholders

General Public

3.3.4 Stakeholder Issues

Based on the available research information, the following are potential issues/concerns of key and secondary stakeholders that are relevant to the design of a stakeholder involvement program.

Transportation and On-site Security

Until the weapon-grade plutonium is irradiated in a CANDU reactor, there may be concerns that the fuel may be stolen or otherwise diverted to countries or organizations for non-peaceful purposes.

Public and Worker Health and Safety

Plutonium is viewed by some as one of the most dangerous substances on earth⁹. Thus, stakeholders and employees would likely be concerned about the possibility of plutonium release and its radiation risk in both normal and abnormal circumstances.

Reactor Safety

Since Ontario Hydro's CANDU reactors have never used mixed oxide fuel, some stakeholders may be concerned about whether the use of this fuel may increase the possibility of reactor accidents.

Fuels Storage and Spent Fuel Disposal

The storage of fuels, and the disposal of spent fuels may also be a concern.

Concerns About the Potential Negative Impact on Suppliers

Stakeholders may be concerned about the potential loss of Canadian employment with suppliers resulting from the reduced consumption of natural uranium and fuel bundle fabrication in the United States.

Environmental Impacts

The potential impacts of implementation of the MOX proposal on the environment, and the potential for recycling should be addressed. Potential impacts on socio-economic aspects of the community, both positive and negative, would be important considerations.

Public Consultation/Involvement

Communities affected by proposals desire a role in decision making.

3.3.5 Overall Strategic Approach

There are a broad range of stakeholders with varying issues and concerns potentially interested in participating in a Definition Phase study. The needs and expectations of these various stakeholders will vary and the approach to stakeholder involvement must be flexible to respond to differing issues and needs.

Specific public attitude research should be conducted to assist in scoping issues and study design during the Definition Phase. A range of research could be undertaken, such as:

- qualitative research (focus groups) with residents in the host community and along the transportation corridor to determine their views towards the proposal. Focus groups

⁹Issue Management Analysis, Ontario Hydro, 1994.

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would provide a foundation on which to develop further testing of the concept. They enable researchers to establish consumer reactions in order to provide actionable strategic answers.

- quantitative research (telephone interviews or mail surveys) with various interest groups to determine their opinions on the proposal. Such research would provide in-depth clarification of potential problem areas, as perceived by residents, that may be encountered with the use of plutonium in nuclear reactors.

Since the Bruce host community expressed concerns with the timing in undertaking such research, it would have to be conducted in consultation with the host community.

The stakeholder involvement program would focus on the key stakeholders in order to meet the goal of improving mutual understanding of project implications. Different tactics are appropriate to meet the varying needs and expectations for the degree of involvement, and to ensure flexibility to respond to the varying issues of concern.

Key Local Stakeholders

A joint planning approach is recommended to provide a dynamic process in which opportunities are provided to resolve issues, to develop a consensus, and to develop stakeholder acceptability for the study and the outcome. A Steering Committee/Working Group would make decisions binding on Ontario Hydro with respect to the decisions to proceed with the Demonstration Phase and with seeking approvals to proceed with the proposal.

Key Institutional Stakeholders

Consultation is recommended throughout the decision-making process to provide advice and input at key decision points. Support from these stakeholders is essential in order to proceed with the demonstration phase and with the approvals.

Secondary Stakeholders

An information-feedback approach is recommended for the secondary stakeholders. At key milestones opportunities would be provided to solicit their feedback with respect to the study.

Tertiary Stakeholders (General Public)

The approach recommended is to provide responsive, focused media relations initiatives. Information would be provided to the public through the media.

3.3.6 Proposed Program Elements

Stakeholder Involvement Program

The purpose of this program is:

- to facilitate an open and accessible process as well as informed decision-making;
- to contribute to promoting the public acceptability of the project;
- to support the project by improve information input from stakeholders to the decision process;
- to ensure that an early and meaningful involvement of stakeholders would be in place consistent with the Sustainable Energy Development principles that are part of Ontario Hydro's corporate mission.

The program will be developed according to the following principles:

- to be acceptable to Ontario Hydro, as well as to the affected stakeholders;
- to be flexible in order to accommodate changing circumstances and project requirements;
- to be open, responsive and fair;
- to strengthen partnerships by establishing and enhancing mutual trust, respect and understanding between Ontario Hydro and stakeholders; and,
- to be initiated at the early stage of the project.

Key Local Stakeholders

In order to facilitate a joint planning process, Ontario Hydro would establish a Steering Committee/Working Group, comprised of representatives from the key local stakeholders and Ontario Hydro representatives. This group would scope issues, help design the study method, review information, develop evaluation criteria, evaluate alternatives, and make recommendations on the most effective course of action for Ontario Hydro.

Sub-Committees could be struck on each of the key issues for the detailed study, such as, the natural environment, socio-economic, security, health and safety, spent fuel management, technical and financial aspects. The Steering Committee/Working Group would coordinate the Sub-committees and steer the process.

In identifying the appropriate representatives of the key local stakeholder groups such as the transportation corridor communities, input would be sought from key decision makers of those communities before setting up the Steering Committee/Working Group and the Subcommittees.

The essential aspect of this joint-planning approach is that this joint Steering Committee would make recommendations on proceeding to the demonstration phase, and seeking approvals for the undertaking.

To supplement the Steering Committee/Working Group, information centers and a forum would be undertaken. Information centers would be undertaken at strategic locations at the beginning and key project milestones to inform and to seek input from key local stakeholders. A forum would be conducted at the conclusion of the study to provide an opportunity for key local stakeholders to be informed of the results of the Definition phase and to discuss the results. Key local stakeholders would be invited to both the information center and the forum using targeted public notification.

Key Institutional Stakeholders

Opportunities for regular briefings and presentations would be offered. Publications/printed material and regular progress reports would also be provided to these institutional stakeholders to ensure their issues are addressed and that their input is solicited during the study.

Secondary Stakeholders

An "Introductory Backgrounder" and subsequent "Newsletters/Status Reports" summarizing study progress and identifying opportunities for more direct involvement would be provided to secondary stakeholders at key stages of the study process determined by key stakeholders. This could be undertaken through mail-outs with invitations to provide comments, or response papers for consideration by the Steering Committee.

Tertiary Stakeholders (General Public)

A reactive media strategy would be undertaken to ensure responsible focused and timely information.

Socio-economic Impact Assessment Program

The purpose of the socio-economic impact assessment component is to:

- provide the socio-economic assessment information necessary to recommend the appropriate course of action for Ontario Hydro Nuclear with regard to this project.
- contribute to scoping of issues and concerns;
- contribute to improving public acceptability through:
 - assessing externalities (i.e. social) and recommending the range of measures in avoiding and mitigating potential effects;
 - facilitating future enhancement of the potential positive impacts and avoiding and reducing the negative impacts of the project;
- meet the legislative requirements as outlined in Section 3.1;
- contribute to refining the appropriate scope, strategy and approach to undertaking the project; and,
- contribute the necessary documentation in order to proceed with the next phase.

A socio-economic impact assessment will augment the approvals process and the stakeholder involvement process through activities designed to:

- identify, in consultation with communities/stakeholders, the range of concerns and effects associated with, and to conduct a preliminary determination on the degree of significance of potential effects on the host community, and potential corridor communities, and other potentially affected parties;
- provide input to the project team in order to facilitate the avoidance of such potential impacts wherever possible and to develop a range of possible mitigation measures for potential effects;

- enhance potential benefits of the project components wherever feasible (enhancement); and,
- identify, assess and manage cumulative effects, at a preliminary level, where applicable.

A socio-economic impact assessment would include the following features.

A) Project Orientation

This program phase includes gaining an understanding of project components and activities, a literature review, and an initial meeting with key stakeholders.

B) Identification of the Study Area

Based on the information obtained during project orientation, the study area would be identified along spatial boundaries. Also other stakeholders that could be potentially affected by the proposal would also be identified.

C) Issue Scoping

Issue scoping workshops in potential host and corridor communities would gain information and insight on the scope of issues and concerns related to the project.

D) Community Profiling

Baseline socio-economic data and future trends for the study area would be developed through primary and secondary data collection, field research, key informant interviews, and surveys. Data collected would include demographic information, the economic base, the character of the built environment, the character of community services, local planning information and administration resources, and social and cultural characteristics.

E) Analysis of the Range of Potential Effects and The Determination of Their Possible Significance

Data collected in the above would be analyzed to determine the effects on key stakeholders and potential host and corridor communities of the project under both the normal and potential accident conditions. Some examples of these effects are, quality of life changes, (i.e. potential health and safety effects, associated stress, the quality of life) and economic base change. A key element in the assessment process is the understanding of real and perceived risk. The Stakeholder Involvement Program will be integrated with this socio-economic impact assessment in order to validate information and conclusions and to determine the significance or impact of change consistent with local values.

F) Development of Possible Impact Management Measures

After socio-economic impacts are determined, measures to manage these impacts would be formulated in consultation with key stakeholders and affected communities. Impact management comprises those measures designed to mitigate, compensate, enhance or offset

potential impacts and measures to ensure continuing monitoring, contingency planning, and community liaison.

G) Communication and Documentation of the Results

The findings of the assessment would be provided to the Project Team and incorporated into the study throughout the planning process. The socio-economic assessment would be documented and become a component of the documentation for the approvals process.

4.0 Fuel Supply

The CANDU fuel supply study covered all facets of technical and programmatic parameters for determining the feasibility of using existing U.S. DOE facilities, as well as new facilities, to manufacture CANDU MOX fuel. AECL Technologies assembled a team of individuals with experience in MOX fuel fabrication, CANDU fuel fabrication, CANDU fuel design, utility fuel supply needs, and the existing MOX fuel facility at Hanford, Washington, known as FMEF (Fuel Materials and Examination Facility). Organizations comprising this team are presented in section 4.11.

Several members of the Fuel Supply Team toured the Zircotec Precision Industries CANDU fuel fabrication facility, in Port Hope Ontario. This trip familiarized team members with the CANDU design and current CANDU manufacturing practices and served as a planning and coordination activity for the team. The Fuel Supply Team also visited the FMEF at Hanford, to become familiar with the equipment and facilities for MOX fuel fabrication located there.

Based on the information it had gathered, the team agreed on an overall strategy for the production of CANDU MOX fuel assemblies. This strategy formed the basis for the balance of the Fuel Supply Team's efforts which were focused on the preparation of an initial design concept for a CANDU MOX fuel fabrication facility, and the development of initial cost estimates for the construction, startup, operation and the eventual decontamination of such a facility.

This report presents the results of the Fuel Supply Team's efforts. The chapter is organized in eleven sections as follows:

Section 4.1 summarizes the main conclusions of our study, including technical bases, cost, and schedule. Section 4.2 summarizes the main features of the CANDU fuel design that influence the fabrication and processing approach and FMEF layout. Section 4.3 summarizes commercial experience in Canada in producing the natural uranium fuel for Canada's 22 CANDU power reactors, and also summarizes experience at AECL's Chalk River Laboratory in fabricating CANDU MOX fuel and other special fuels such as CANFLEX and Dysprosium loaded fuel.

Section 4.4 describes the basic strategy that was selected for the MOX fabrication process and the layout at the FMEF. It provides the basis for the selection of design throughput of the facility, and the reasons for selection of separate pin lines for each of the two outer rings. Section 4.5 describes the assumptions and requirements for the "supplied" materials, including that to be provided by the U.S. Government (plutonium oxide and depleted UF₆) and that to be supplied to FMEF from commercial sources (conversion of UF₆ to UO₂, zircaloy components, dysprosium elements, and natural uranium end pellets).

Section 4.6 presents the results for the reference case which has a "design" rating of 170 tonnes per year of MOX fuel, and which utilizes the reference bundle (37 pin design). This section describes the facility layout, the various processing stations within the facility, the provisions for internal recycle, as well as utilization of DOE stockpile plutonium oxide, the anticipated waste arisings, the expected occupational radiation exposure, the recommendations for licensing and operation of the facility, and provisions for safeguards and security including IAEA inspections. Section 4.7 presents the results of our study for alternative strategies, designed to consume up to 100 tonnes of weapons plutonium in 25 years.. Section 4.8

presents a brief evaluation of decontamination and decommissioning of FMEF following the MOX fuel campaign. Section 4.9 describes the ownership and operational arrangements recommended by the study team for the planning, design, installation and operation of the MOX CANDU fabrication program. Section 4.10 presents our preliminary estimates of the costs including both modification costs and operational costs. It also includes the basis for our 4 year best effort schedule estimate for installing, licensing and checkout of the facility.

4.1 Conclusions

The Fuel Supply Team has concluded that it is practical, economical, and efficient to utilize the existing FMEF facility at Hanford to produce 157 tonnes/yr of CANDU MOX fuel at FMEF utilizing 2.1 tonnes/yr of plutonium metal. The facility will yield 9,050 standard (37 pin) MOX bundles per year, each containing .23 kg of plutonium metal, and sufficient to fuel two BRUCE A reactors at 80% capacity factor. This same facility could also be used to produce 164 tonnes/yr of the advanced CANFLEX fuel (43 pin) utilizing 4 tonnes/yr of plutonium metal/yr, sufficient to fuel 4 BRUCE A reactors at 80% capacity factor. Figures 4.1 and 4.2 illustrate the material flow streams for these two CANDU MOX options as compared to the traditional natural uranium fuel cycle in use at Bruce. Figure 4.3 illustrates the key characteristics of a CANDU MOX bundle (reference 37 pin design) as compared to a CANDU natural uranium fuel bundle.

The fuel supply team also concluded that the FMEF can be prepared for this mission, including modifications, process equipment procurement, installation and testing, with all necessary licensing and permitting, within a period of 4 years and for a cost of about \$115 million. (This cost estimate is overnight costs and does not include financing which obviously depends on the ownership arrangements) This cost estimate is based on a very detailed review of every process function required, a detailed layout of these process functions within the existing FMEF facility, the current status of the facility, and the licensing, permitting, safeguards and security requirements. It is based on the extensive experience of task team members in both MOX (FFTF) fuel fabrication, and CANDU fuel fabrication.

With regard to operating costs, we have concluded that the overall MOX fuel supply costs to Ontario Hydro for two Bruce Reactors (standard 37 pin design) would be about \$64 million per year, including site security and landlord functions, and ultimate decontamination of the facility, but not including financing costs. This compares with about \$16.3 million per year for the current natural uranium fuel supply. Also not included are the transportation costs, the Bruce Facility modification and licensing costs, and an initial fuel demonstration program.

The estimate assumes that the FMEF facility will be leased at no cost to the operator. The estimate includes \$20 million/year for DOE to provide the necessary physical security and landlord functions at the Hanford site.

Some of the key assumptions, results, and conclusions of the study team are as follows:

- (1) FMEF SUITABILITY, NATURAL PHENOMENA, PHYSICAL SECURITY, AND PROCESS SYSTEMS. We found that the existing FMEF facility is quite suitable for this fabrication program. The structure has been designed to withstand external events such as tornado and earthquake as required by current DOE orders and NRC

Figure 4.1 Annual Material Flow Balance - 2 Bruce-A Reactors

Figure 4.2 Annual Material Flow Balance - 4 Bruce-A Reactors

Figure 4.3 Natural Uranium versus Reference MOX Fuel Bundles

	Existing Bruce-A Station	MOX fueled Reference Bundle
Fuel Bundle Dimension (mm)	102 (4.02") Dia 495 (19.5") Long	Same as existing fuel
Pellet Material Composition (% based on weight of heavy metal in fuel)	Natural UO ₂ in all rings (37 pins)	Ring 4 (18 pins) 1.2% Pu Ring 3 (12 pins) 2 % Pu Ring 2 (6 pins) 5 % Dy Center (1 pin) 5 % Dy
Bundle Material Composition (kg) (fuel material only)	U ²³⁵ 0.13 U ²³⁸ 18.67 O ₂ 2.53 Total: 21.33	Pu 0.23 U ²³⁵ 0.04 U ²³⁸ 18.36 Dy 0.17 O ₂ 2.53 Total: 21.33
Average Burnup (MWd/ton)	8,300	9,700
Maximum Burnup (MWd/ton)	15,000	15,500
Bundle/FPD	18	15.5
Fuel Management Scheme (bundle shift)	2,4,8	2
Maximum Channel Power (KW)	7,200	7,000
Maximum Bundle Power (KW)	975	800
LOCA Void reactivity (mk)	>10	-4.7

regulations; essential utilities are in place, including HEPA filtered ventilation systems and reliable and redundant power supplies. Security features at FMEF include well designed vaults and handling equipment for Special Nuclear Materials. Many of the SNM storage vaults, finished assembly loading stations (truck bays) and process systems and equipment originally installed for FFTF fuel fabrication are usable for the CANDU MOX line, including such items as the two large sintering furnaces. Overall, we estimate 20 to 30% of the cost of new process equipment can be avoided by using existing process equipment originally designed for FFTF fuel fabrication.

The FMEF, with its two large shielded cells and other infrastructure, might also be suitable for use in storage of plutonium pits and conversion of pits to oxides. Such usage was outside the scope of this study.

- (2) **PREREQUISITES TO FULL MOX PRODUCTION** - Our study confirmed that no new technology is required to achieve the MOX fuel supply for CANDU bundles. However, practical engineering judgment requires several stages of commercial fuel demonstration as a prerequisite to full operation on a new fuel system such as MOX fuel. The first stage would be the production of about 40 fuel bundles to detailed AECL/utility approved MOX fuel design specifications, and subsequent irradiation in existing test reactors such as NRU. (Some of these bundles would also be used first in a critical facility test described elsewhere in this report.) The second stage would require the production of about 50 bundles for operation in a commercial reactor such as Bruce A. And the third stage would involve production of up to 1000 bundles and operation in Bruce during the transition from Natural U to MOX. This latter would be the initial production run at the modified FMEF, and would not begin until the utility had received AECB approval to operate on a full MOX core. Costs for the first two stages are assumed as part of the MOX design confirmation phase and are considered under the nuclear and fuel design segments of this report. Costs for the 1000 bundle production run are included as part of this fuel supply task.

It is our recommendation that DOE arrange for this initial CANDU MOX fuel production (about 90 bundles containing about 25 KG of plutonium) at the earliest possible time, even before the policy decision on plutonium disposition is made, and before a particular reactor system is chosen. The existing TA-55 facility at Los Alamos could be used for this fabrication/operational demonstration. Some of the process equipment approaches suggested in this report could be demonstrated as part of this TA-55 program, thus further accelerating the eventual disposition program once a policy decision is made by the U.S. Government. An additional benefit of such an early program phase would be the participation of Russian fuel fabrication specialists, as described in item 9 below.

- (3) **SCHEDULE** - Figure 4.4 is a rough schedule for implementing this program. Because of the "clear and present danger" warning of the National Academy, and in accordance with DOE requested schedule assumptions, it assumes that critical path work begins immediately, and this includes the trial bundle fabrication and testing program mentioned above. We estimate that the work in the first two years of this program will cost about \$25 million. We suggest that DOE consider funding these efforts (including licensing preparations) in parallel with policy development activities, as this would allow

Figure 4.4 Program Plan and Schedule for MOX Fuel Supply

the plutonium disposition program to begin 2 years earlier than otherwise. Such early activities would lend substantial impetus to a comparable Russian program, which is critical from an overall non-proliferation point of view.

- (4) **LICENSING AND REGULATION:** In order to proceed with a materials conversion program such as this, neighbors from the surrounding community and the general public must be convinced that the facility will be operated with public health and safety, and environmental protection, uppermost in priority. In addition to a full defense in depth approach during design, installation and test of the fabrication facility, we propose that the operation of this facility be licensed by the independent regulatory agency, the NRC. NRC has extensive experience in licensing similar facilities, and can be counted on by the public to do a thorough job. We believe the existing facility, and our proposed processing operations will be fully consistent with all NRC safety and environmental regulations, and have incorporated into our program schedule and cost estimates sufficient time and money to allow such a thorough review and licensing activity.
- (5) **SAFEGUARDS AND ACCOUNTABILITY:** Our study is based on the assumption that the entire MOX fabrication facility will be under the umbrella of international (IAEA) safeguards agreements. These will require extensive measurements and accounting by the facility operator, as well as provisions for permanent and frequent IAEA inspections and measurements. For example, we have assumed a near real time accountancy system. Of the four MOX fuel fabrication facilities currently under IAEA safeguards throughout the world, only the PFPF in Japan has such a system in operation. (There are recent reports of some troubles with this system) We have conservatively assumed that such a system would be used at FMEF and have allowed for the costs (\$10 million) in our estimates. However, there may be opportunity for innovative solutions to the measurement and accounting systems to be installed that provide the operator and IAEA the necessary assurances at significantly less cost than such a system. We suggest such innovative systems be considered in future phases of this program.
- (6) **REDUCTION IN "WASTE ARISING" USING MOX VS. NATURAL URANIUM CYCLE:** As shown on Figure 4-1, the use of MOX fuel instead of natural uranium fuel in two CANDU reactors allows for major reductions in waste generation, and in fact leads to reduction of materials currently stored as waste in the U.S. (2 tonnes/yr of plutonium metal and 170 tonnes/yr of depleted uranium in the form of UF_6), and at the same time eliminates the mining of some 6,300 tonnes of uranium ore, with associated mill tailings. We estimate that the MOX plant will generate about 260 to 390 55 gallon drums of TRU waste per year, consisting of compacted solids and miscellaneous equipment (for purposes of this report, the availability of WIPP for TRU waste disposal is assumed). The LSA waste is estimated at less than 50 drums per year, with hazardous (non-radioactive) waste at 30 to 50 drums per year. No mixed wastes are anticipated.
- (7) **OCCUPATIONAL RADIATION EXPOSURES:** Unshielded, we estimate (using calculations and some measurements) that the average contact dose on a standard (37 pin) CANDU MOX fuel bundle will be 15 to 25 m^r/hr, including about 5 m^r/hr neutron dose and the balance gamma dose. In the bundle assembly area, a shielded work

station consisting of a 4 inch panel of water extended polyester (WEP) sandwiched between 1/8 inch sheets of stainless steel would reduce these dose levels to below 1 mr/hr. A thinner (about 1 inch of WEP) cylindrical shield would be used to handle the bundles after assembly and in storage. Contact dose rates would be under 5 mr/hr. Storage would be in existing concrete "tile" holes (300 exist in FMEF for storage of FFTF fuel) and in the standard SST shipping containers, which should further reduce general area radiation levels. Based on the proposed processing layout, and recent experience from LANL TA-55 and Belgonucleaire, we are confident the yearly operator dose can be kept under 500 mR, consistent with new NRC standards.

- (8) **REDUCTION IN FUELING COSTS USING CANFLEX:** The advanced CANFLEX design is a more efficient fuel bundle because it can utilize higher plutonium concentrations, and longer burnup, without exceeding core thermal limits. However, several years of additional testing are required before the utility will have sufficient confidence to use this new design in commercial reactors. Therefore, one scenario might be to start the fabrication and commercial operation programs using the standard bundle design, and then switch to CANFLEX after the new fuel system is proven, perhaps in five or six years. The process lines proposed herein can accommodate this switch. The cost of producing 162 tonnes of CANFLEX fuel per year would only be marginally higher than 157 tonnes/yr of standard bundles (perhaps 20%). Such an arrangement could allow the modified FMEF facility to increase its plutonium utilization rate from 2 tonnes/yr to 4 tonnes/yr (using 4 Bruce Reactors) and thereby shorten the entire program from 25 years to about 15 years. This would lead to substantial savings. (Estimated at about \$500 million for the entire program)
- (9) **ESTABLISHING A "TWIN" FACILITY FOR CANDU MOX FABRICATION IN RUSSIA:** As stated in the National Academy study, disposition of excess weapons plutonium in Russia is of great importance; in fact the idea of disposing of 100 tonnes in 25 years is being considered because it may be possible to dispose of Russian and U.S. excess plutonium in one facility. An alternative approach which has many attractive features is to establish "twin" facilities in Russia and in the U.S. to produce MOX fuel for reactors in either or both countries. A joint "fuel supply team" which includes Russian as well as U.S. and Canadian participants could be set up to finalize the design of the MOX facility and then to arrange for installation, licensing, safeguards and operation in the separate countries. The net costs to both countries (U.S. and Russia) would be significantly reduced by this type of joint program, as compared to separate and unique facilities being designed and erected in each country. The CANDU MOX fuel produced in the Russian MOX facility could be utilized in either new CANDU reactors to be erected in Russia, existing CANDU reactors now in startup in Romania, or possibly shipped to the Bruce site on Lake Huron via secure, safeguarded, oceangoing vessel such as the type now being used for shipment of plutonium from European reprocessing plants. Since it may take some time to establish and implement a new CANDU reactor program in Russia, one approach to project initiation would be to utilize Bruce for the early production runs (e.g. 2 to 3 years) and thereby confirm the quality of the Russian produced fuel, and then, when the new Russian CANDU plant is ready, direct the fuel deliveries to Russia instead of Canada. As a first step, we suggest that Russian engineers be invited to participate in the next design phase of the FMEF modifications, and also to participate in the trial bundle fabrication at Los Alamos as suggested above.

Russian engineering participation would be funded as part of the program costs.

- (10) **DISPOSITION OF 100 TONNES OF PU VIA MOX FABRICATION IN THE UNITED STATES:** As requested by DOE, we have also examined the cost and approach that could be used to fabricate and dispose of 100 tonnes of plutonium in the U.S. in 25 years. This would require a facility producing 340 tonnes/yr of CANDU MOX fuel. We examined two approaches, a new greenfield facility, or an expansion of FMEF. We concluded that a new greenfield facility would cost about \$1 billion. A further expansion of FMEF would cost an additional \$80 million (a total of about \$200 million.)
- (11) **ALTERNATIVE U.S. SITES FOR MOX FABRICATION:** While the Hanford FMEF facility is the preferred facility for production of CANDU MOX fuel from the standpoint of existing fabrication infrastructure, we were also asked to consider other alternatives that might reduce the extent of plutonium transportation required. In this regard, two members of the fuel supply team visited and inspected the new Device Assembly Facility (DAF) at the Nevada Test Site. The future mission for this facility is unclear and it is possible that about 2/3rd of the existing facility may become available for new missions. If it were used for MOX fuel fabrication, and the Nevada Test Site were also being used for pit storage and pit conversion, this would reduce the extent of road (or rail) transport significantly. This might offset some of the extra costs associated with outfitting a facility for MOX fabrication that has no existing processing capability. It also could ease the policy concerns of Nevada citizens regarding long term pit storage by adding a processing component to the site, thereby assuring the citizens that plutonium would be shipped off-site after processing. Our brief inspection of the DAF facility indicated it probably has sufficient floor space, the necessary security features, and HVAC and utility supplies for this new mission. Also, favorable layouts (e.g. using the long corridors) could probably be developed to allow a reasonably efficient processing capability. Further study by the entire fuel supply team is required to confirm this conclusion.
- (12) **CONSUMPTION OF PLUTONIUM WASTE PRODUCTS FROM DOE SYSTEM:** The MOX processing equipment outlined in this report includes an oversized MOX recycle station. Normally this recycle process equipment is required for internal recycle of rejected product including powder, green pellets and fired pellets, so that the net yield of the facility is close to 100%, and a minimum amount of plutonium is incorporated in the waste stream. The fuel supply team found that, by increasing the size of this station, it will be possible to utilize much of the plutonium scrap in the DOE system, such as the green FFTF fuel pellets, scrap plutonium oxide and metal at the Rocky Flats, Savannah River, Hanford, Oak Ridge and LLNL facilities. This scrap material can be processed through the recycle unit, along with internal recycle, and blended into the normal process stream at a ratio of up to 10 to 20%. This could significantly reduce the environmental and cost burden to DOE's Environmental Management programs.
- (13) **OWNERSHIP AND OPERATIONAL RESPONSIBILITY:** An essential prerequisite for the proposed plutonium disposition in CANDU option is that the utility (Ontario Hydro for the purposes of this study) must be assured that:
 - (a) switching from their present natural uranium supply situation to this MOX supply

- will not disrupt their overall fuel supply plan (for Bruce A and their other 16 reactor plants) which has been developed and implemented at great effort over many years; and
- (b) the MOX fuel supply for the two Bruce A reactors which use MOX will be of sufficient quality and reliability (guaranteed supply) that they can fully depend on this supply to meet their customers' electricity needs.
 - (c) Arrangements are made for direct, on-site quality assurance surveillance by utility inspectors, including qualification of fabrication processes.

To achieve this level of assurance by the utility, it is essential that the utility and its' current fuel suppliers, be directly involved in the entire MOX fuel fabrication program, from inception (now) through development and licensing, through production and transportation.

One way to achieve this is through the establishment of a private venture to design, license and operate the production line within FMEF. In this regard, the study team has received expressions of interest from private firms interested in participating in a privately financed fuel supply program, subject to suitable assurances from the Government on feed material supply, negotiation of a long term (25 year) fuel supply contract with the utility, and up front guarantees of licensing (by NRC) of the facility. Securing these guarantees, particularly the licensing guarantee, prior to investment in facility modifications, may be difficult to achieve. Therefore, the team considered two other arrangements for ownership and operation, in addition to direct private licensing and ownership:

Government License/Own; Private Operation The first alternative would be Government ownership (including investment in the modifications and responsibility for licensing) with operational responsibility assigned to one or both of the present fuel suppliers, as agreed to in discussions among DOE and the utility. In this case, work on modifying the FMEF could proceed immediately upon the Record of Decision, and NRC licensing would proceed in parallel with equipment procurement and facility modifications. This would allow completion and startup within the 4 year period shown in Figure 4.4. A variation of this alternative would be for the facility to be operated by a government contractor, under license to existing Canadian fuel suppliers, and with training, product and process qualification under the supervision of the existing fuel suppliers and/or the CANDU operating utility. While this arrangement would appear, on paper, to lead to lower capital costs (government vs. private financing) the ultimate costs could be higher because of the absence of the profit incentive associated with private ventures.

Private License/Own/Operate A second alternative would be to delay the schedule for process equipment procurement and modifications to FMEF (the bulk of the investment required) until after the NRC has granted a license for operation. Once that license has been issued, there would be sufficient basis for private investment into the modification and operation of the facility. This option has the advantage of retaining the entire fuel fabrication operation in private hands and under the contractual control of the utility. This would also meet the intent of recent privatization desires expressed by DOE Management. However, its' major disadvantage is a delay in initial production of MOX fuel bundles by about 2 years. Considering the "Clear and Present Danger" concern expressed by the National Academy of Sciences, this delay might prove unacceptable to the U.S. Government.

AECL, and its supporting subcontractors, believe that the choice between these alternatives are beyond the scope of this study, and suggest that DOE continue to evaluate the pros and cons of each, in consultation with the operating utility, in order to arrive at a final decision.

4.2 Introduction to CANDU Fuel and Evolution of the Design

The reference fuel bundle design for this study is the standard design used in the Bruce and Darlington CANDU reactors, described in Chapter 2 and illustrated in Figure 2.2.1. It is an assembly, 19 inches long, of 37 elements of Zircaloy-clad fuel pellets held together by two endplates, to form a fuel bundle weighing about 25 kg. Because it consists of only seven basic parts it is simple, and the design has proven itself suitable for mass production to very high quality levels. This has been established with consistent operation to extremely high reliability. More than 1,000,000 CANDU bundles (half each made by the two Canadian fabricators) have been irradiated to date. The element linear ratings of the MOX fuel have been chosen to be within the envelope of the current Bruce reactors with natural uranium fuel.

The basic simplicity of the CANDU fuel bundle has led to very low fabrication costs. A relatively small process line is required and manpower needs are moderate. Therefore, fabricating CANDU MOX fuel bundles in the existing FMEF facility can build on the excellent CANDU natural uranium fabrication experience, can use most of the mass-production techniques of current CANDU fabrication, and therefore will yield relatively low MOX fabrication costs as compared to other commercial MOX fuels.

Chapter 2 also describes the advanced CANFLEX fuel bundle, a 43-element fuel assembly in which the outer rings of elements are slightly smaller in diameter than those of the 37-element designs. It is designed to interface with existing fuel handling systems, and to be fully compatible with the 37-element fuel bundles. This will allow a reactor to convert from one design to the other without the need for an intermediate shutdown. It has the advantage that the outer element linear ratings are lower than standard element ratings, offering: 1) higher reliability for very high burnups, and 2) safety improvements, while being essentially "transparent" to reactor physics and fuel management. Figure 2.2.8 illustrates the CANFLEX design.

The development of MOX fuels for use in CANDU reactors was addressed as early as 1981. A recycle fuel fabrication laboratory was built at AECL's Chalk River laboratories, and a quantity of over five hundred elements containing 0.5 wt% PuO₂ were built. Some of these were assembled into fuel bundles and successfully irradiated in the NRU reactor. This facility is being recommissioned for a 1995 fabrication campaign, and could be used to fabricate test bundles for this program for irradiation in the NRU reactor.

CANDU FUEL DESIGN FEATURES RELEVANT TO FABRICATION

FUEL ELEMENT DESIGN

The individual fuel elements are approximately 19 inch (490 mm) long and contain three basic component parts: the UO₂ pellets, the sheath (with the Canlub coating on the inside surface) and the endcaps. In addition, each element has spacer pads and/or bearing pads brazed to the outer surface.

(a) The UO_2 pellets are designed to:

- maximize the amount of fissile material in each fuel element,
- minimize volumetric changes during irradiation,
- control fission gas releases,
- minimize circumferential ridging of the sheath, and
- be economical to produce.

The material properties of the UO_2 are maintained within limits, for example density (typically 97% of the theoretical density) and oxygen-to-uranium ratio, which strongly affect the thermal behaviour and hence fission gas release of the UO_2 . Other features of the UO_2 pellets, such as pellet dishing and length/diameter ratio, are designed to optimize fuel performance; several programs related to fundamental UO_2 performance (and advanced fuel technology) are used to assess fuel performance.

The pellets will have a length to diameter (L/D) ratio consistent with the desired burnup and heat rating. The current CANDU natural uranium pellets have an L/D of approximately 1.4, and that for the MOX fuels will be optimized at a lower value, bearing in mind the higher burnup targets of the MOX fuel. The L/D of the outer ring of elements will be lower than that of the other rings and will serve two purposes. First, the value will be optimized to reduce fabrication costs while recognising the performance requirements, and second, the different L/D between the two different plutonium loadings (enrichments) will provide an easy visual means of separating the enrichments in fuel fabrication operations, should the need arise.

Although current CANDU fuels are made with a very high surface finish, there may be a case for reducing the requirements for the MOX fuel. This will help keep down fabrication costs, and would be decided later in the program, even though the reference surface finish for the MOX fuel is the same as for current CANDU fuels.

(b) Fuel Sheath: The Zircaloy-4 sheaths are designed to:

- contain the UO_2 under normal operating conditions,
- minimize neutron absorption,
- minimize corrosion and hydrogen/deuterium pickup,
- minimize strain effects,
- minimize resistance to heat transfer,
- minimize hydraulic head loss,
- be strong enough to withstand normal operating (including re-fuelling and reactor power manoeuvring) loads.

Zircaloy-4 is chosen because of its excellent nuclear characteristics of low neutron absorption, good corrosion resistance and low hydrogen pickup. Material properties and heat treatments are specified to optimize sheath ductility at high irradiation levels.

(c) Canlub Coating

A thin layer of graphite is applied on the inner surface of the sheath, to reduce pellet-clad interaction and gross corrosion cracking. It should be noted that improvements to the coating

are continuously being investigated by AECL, and it is expected that changes may be implemented in current production, particularly for high burnup fuels. Any pertinent changes that are proven in service, would be adopted also for the MOX fuels.

(d) Endcaps

The fuel elements are sealed at each end with an endcap, resistance welded to the sheath. The endcaps are designed with the correct profile and strength to take the loads of the latches in the fuel channel, during re-fuelling. The endcaps are designed to be essentially "butt" welded to the clad, and hence provide a small internal gas volume outboard of the weld; they use a minimum of material. The welds are amenable to automatic fabrication techniques, and can be 100% examined non-destructively to provide a very high quality weld.

(e) Filling Gas

The fuel elements are filled with unpressurized helium to allow all elements to be helium leak tested during fabrication. The helium also enhances the pellet/sheath contact conductance at start of life.

An important feature of the CANDU fuel element is the use of thin-wall clad. The minimum wall thickness is of approximately 0.38 mm (0.015 inch) has been used continuously since the first charge for the NPD reactor in 1958. Because the clad is designed to collapse into intimate contact with the fuel pellets, pellet-clad interaction (PCI) was recognised as a fact from the start of the program. PCI can cause unacceptable sheath strains, for example, due to in-reactor densification of the UO_2 . Thus the need was established for the high density "irradiation-stable" fuel pellets which avoid this problem. High density UO_2 pellets, typically 97% theoretical density, have been used, therefore, since the start of the Canadian program. This feature, together with the introduction of the Canlub coatings, has reduced the potential PCI problem to an insignificant level for CANDU fuels.

To help in maintaining low center-temperatures as much as possible, and hence to maintain low levels of fission gas release within the elements, a fine surface finish is specified for the pellet surfaces, and indeed for the inside of the clad. This is generally higher than for other fuel designs, eg. for free-standing clad fuels, and the better surface finish maximises the heat-transfer coefficient between the fuel and the contacting sheath. Such finishes can be produced with typical fabrication techniques such as wet grinding.

FUEL BUNDLE DESIGN

The fuel bundle, Figure 2.2.1, consists of 37 elements held together as an assembly by two endplates. The elements are arranged in three rings, of 18, 12 and six elements plus one central element. The endplates are simple stampings, and consist of three concentric rings held together by a number of radial webs.

The elements are spaced from each other by the endplates at each end, and by inter-element spacers at the element mid-length. Simple resistance welds are used to attach the elements to the endplates, and beryllium brazing is used to attach the spacers to the clad. Bearing pads, also brazed to the outer elements of the bundle, support the bundle in the fuel channel. The

bundle is approximately 19 inch (500 mm) long, 4 inch (100 mm) diameter and weights approximately 53 lbs (24 kg).

a) Endplates

In addition to holding the elements in the correct bundle configuration, the endplates have to be flexible to allow differential axial expansion among the elements. Also they have to distribute axial loads (for example, during refuelling) to all elements in the bundle. The endplates are made as thin as possible, to minimize the amount of neutron-absorbing material in the assembly, and to minimize the gap between the UO_2 fuel in adjacent bundles.

(b) Inter-element Spacers

Small spacer pads, approximately 0.1 inch x 0.25 inch (2.5 mm x 8 mm) are brazed at the mid-length of all elements. They are mounted at a small angle to the axis of the element so that any two mating spacers are angled in opposite directions. This increases the width of possible contact between these inter-element spacers, to allow for normal distortions in-reactor and to maintain the desired spacing of all elements at the bundle mid-plane. Note that no grid spacer is used, and in this way a minimum of parasitic material is present in the bundle. The spacers contribute to the good heat-transfer performance and hence high critical heat flux of these bundles, by introducing turbulence to the coolant at each bundle mid-plane.

(c) Bearing Pads

Three pads, approximately 0.1 inch x 1.0 inch (2.5 mm x 25 mm) are brazed near the ends and at the centre of each outer element. The pads support the bundle in the fuel channel and fuel handling system, and prevent mechanical damage to the sheath. The pads are designed to minimize surface damage to the pressure tube caused by sliding and fretting.

FUEL FABRICATION

Conventional manufacturing processes are used throughout the fabrication of CANDU-PHWR fuel, with custom-designed welding and brazing equipment for the assembly of the elements, and their final assembly into bundles. The bundle manufacturers are responsible for the manufacturing details and are required to provide test results which demonstrate that their product will meet stated service requirements.

The current manufacture of fuel bundles involves the following major process operations:

- Test the incoming UO_2 powder and Zircaloy materials;
- Make high density UO_2 pellets (typically 97% of theoretical density);
- Fabricate Zircaloy structural components;
- Load the pellets into the sheath, and weld the end caps to the sheath to complete the elements;
- Assembly elements into bundles; and
- Final inspect, and package ready for shipping.

Whereas the first core load (built from 1958 to 1961) for the 22 MWe NPD reactor used inert gas fusion welding for the endcap to clad welds and for the element to endplate welds, all subsequent production has used resistance welding. Brazed spacers and bearing pads were introduced in 1967 and have been used ever since. The only other process or design change that has been introduced since then has been the introduction of Canlub in 1972/3.

Today, manufacturing emphasis is on improved productivity and process control, for example, microprocessors are contributing to improvements in the already high-quality product, and to the very low fabrication costs of the fuel.

TECHNICAL SPECIFICATIONS FOR FUEL

The design requirements and pertinent design features for CANDU fuel are included in comprehensive technical specifications. Although AECL has developed its own specifications these are very similar to the ASTM specifications, and in some cases the latter specifications would be applicable with minor exceptions. However, the AECL specifications have been developed from the extensive R&D work on CANDU fuels and the excellent power-reactor performance, and they reflect the differences between CANDU and for example, LWR fuels. Major components and materials covered by the AECL specifications are:

- Sinterable UO_2 powders, and UO_2 pellets;
- Zirconium alloy sheet and bar;
- Zirconium alloy seamless tubing for reactor fuel sheathing;
- Canlub coating of fuel sheathing;
- Beryllium brazing material and braze joints for element sub-assemblies;
- Bundle specification and design drawing.

In the early years of fuel production the specifications were based largely on fundamental research work. The specifications have been continuously reviewed, and recent improvements and updating have been made using up-to-date manufacturing and reactor performance data. Because of the accent on neutron economy in CANDU fuel, a key feature of the specifications is the strict control of impurities with large, neutron capture cross-sections. They are used today for the procurement in Canada of fuel for CANDU reactors.

QUALITY ASSURANCE

Any complete Quality Assurance program has to include control of the whole cycle of fuel design, fabrication, research and development, commercial reactor performance, an post-irradiation examination and evaluation. The quality and the manner of control of ongoing research and development and the operation of the commercial reactors have been amply demonstrated by the excellent performance of CANDU fuel in commercial reactors.

The Canadian Standards Association (CSA) standards which have been developed and used since 1975 for the procurement of CANDU fuel, are the well known CSA Z299 series of quality program standards. These standards have been received favourably around the world, and have been translated into several languages including Spanish, French, Japanese and German. The CSA Standards are very similar to the ISO 9000 series of standards.

Formal quality program standards were first used in Canada in 1969 for procurement of nuclear fuel. The CSA standards, specifically the CSA Z-299.1 (Quality Assurance Program) and CSA Z-299.2 (Quality Control Program) were applied in 1975. They are now widely accepted, clearly understood, and their implementation is well co-ordinated. All fuel for first core loadings is procured in Canada to the top quality level, CSA Z299.1, and all subsequent fuel orders are specified to comply with the CSA Z299.2 quality level. It is expected the MOX fuel initially will be fabricated to the CSA Z299.1 level, or equivalent, and that subsequent fabrication will be to the second level, Z299.2.

OVERVIEW OF CANDU MOX VS STANDARD CANDU DESIGN

The CANDU MOX fuel bundle design contains only minor differences from the standard CANDU design. Except for the substitution of MOX and Dy/Du Oxide pellets for the natural uranium oxide pellets, the impact of these design differences on manufacture of the fuel bundle are only minor and will be easily accommodated.

REDUCTION IN FUEL PELLET LENGTH-TO-DIAMETER RATIO The length-to-diameter ratio of the pellets used in the CANDU MOX design will be reduced to about 70% of the current CANDU design value. The predominant impact of this change on manufacturing will be to increase the number of pellets that must be produced per fuel bundle. With an automated pellet pressing and handling system this is not significant. In fact, shorter pellets press more consistently and hourglass less during sintering which means less material will need to be removed in centerless grinding of the pellet cylindrical surface.

CANLUB COATING THICKNESS AND BAKING CONDITIONS The thickness of CANLUB graphite coating in CANDU MOX fuel will be increased relative to that required for the standard CANDU design. In addition, the baking conditions will be specified differently from standard CANDU fuel. Both of these changes are to address the performance of the fuel at the higher burnups that MOX fuel will experience. CANLUB coating and baking are both processes that lend themselves to considerable process flexibility such that neither change will present a problem for the existing CANDU fuel manufacturers who will provide the fuel tubing with CANLUB coating already applied.

NATURAL URANIUM OXIDE END PELLETS: Normally partial length end pellets are used in manufacture as a means of meeting overall stack length requirements. In the case of MOX fuel, this would normally require a separate pelletizing step for the partial length pellets. To avoid this, and to enhance performance (reduced axial flux peaking) the partial length end pellets will be made from natural uranium oxide and supplied separately to the MOX facility. They will also have a slightly reduced diameter, to facilitate end cap welding, and reduce the likelihood of plutonium contamination of the weld.

4.3 CANDU Fuel Fabrication Experience

There are two Canadian CANDU fuel fabricators that supply Canada's nuclear industry requirements. They are Zircotec Precision Industries (ZPI) and Canadian General Electric (CGE). These commercial fabricators have been producing fuel since the 1950's as a supplier to Canada's nuclear research program, and both have produced their 500,000th bundle in 1993, for a combined total of more than one million bundles (or 22,000 Mg U) to date. The combined

fuel manufacturing capacity of ZPI and CGE is approximately 2,400 Mg U per year.

One of the well-recognized strengths of CANDU is the high quality and excellent performance of CANDU fuel. This is ensured in the manufacturing stage by comprehensive Quality Assurance Programs implemented by both the commercial fabricators and the utilities. Their QA programs satisfy the requirements of the Canadian Standards Association (CSA) CAN3-Z299.2 standard.

The excellent performance of CANDU fuel is demonstrated in its service record. Of the more than 1,000,000 bundles that have been irradiated, less than 0.1% have developed defects. This number, however, is not a steady-state defect rate. In fact, the majority of the defects have occurred in batches, which are referred to as "defect excursions", due to several causes such as non-standard reactor operations and manufacturing defects. The number of defect excursions is small, and in each of these instances, prompt and immediate remediation was implemented to minimize any adverse effects on reactor performance. With the CANDU reactor's on-power refueling capability, defective fuel can be quickly removed from the reactor. Investigation of the cause of a defect excursion has in each case resulted in changes to further improve the performance of CANDU fuel.

The defects that have occurred in CANDU fuel fall into three primary categories. One category is fretting of the fuel elements (or pins) due to debris from the primary heat transport system, which accounts for 15% of the total number of defects. The second category is stress corrosion cracking (SCC) which accounts for 35% of the defects. CANLUB graphite coating on the inside of the fuel sheaths was introduced to mitigate SCC-induced defects. The third category consists of manufacturing defects combined with other unassigned causes (the utilities lump these two together in their reporting). This category represents the largest group, about 50% of the total number of defects. Although the actual number of manufacturing defects is not known, the defect rate that could be attributed to manufacturing flaws is less than 0.05% of the total number of bundles.

EXPERIENCE WITH PRODUCTION OF NATURAL URANIUM CANDU FUEL

Manufacture of CANDU fuel is accomplished with a blend of automated and semi-automated equipment designed to provide a viable, cost effective production operation. Machine automation is utilized to provide consistency of performance in areas where human input could result in excessive variability. However operator input is maintained in those areas where needed to provide added dexterity or to provide a quick feed-back of problems with machine operation or with process control. For the MOX design, these operations will be re-engineered, as necessary, to provide equivalent dexterity and control while eliminating the hands on features which would cause unacceptable personnel radiation exposures.

In many instances, process and performance inspection is built into the machine, the output being used to control the machine such that non-conforming product is recognized and physically removed. This resulting blend has been shown to yield high quality product in a cost effective manner. The individual operations used in the manufacture of CANDU fuel are described in the following sections.

The essential steps in fuel pellet manufacturing process include:

- production of a free flowing powder to ensure continuity of production and uniform die filling,
- incorporation of a press die lubricant to the powder,
- pressing of green compacts using hydraulic or mechanical powder compaction presses,
- sintering of green compacts to high density, and
- diameter sizing by centerless grinding.

The individual pins that make up the CANDU fuel bundle are called fuel elements. Manufacture of a fuel element involves the attachment of small zirconium alloy components on the surface of the fuel tube using a brazing process. The fuel pellet stacks are then loaded into this "sub-assembly" and zirconium alloy caps are welded to each end of the sub-assembly to hermetically seal the pellets inside the fuel element.

The CANDU 37 element bundle consists of an assembly of fuel elements in an array of three concentric rings surrounding a central element. Element arrangement in the bundle is maintained by attachment of each end of each element to a zirconium alloy end plate. Completed fuel bundles are subjected to a series of inspections and non-destructive tests against all specification requirements.

CHALK RIVER FUEL MANUFACTURING EXPERIENCE:

In addition to this commercial experience, AECL's Chalk River Laboratory has experience in specialty fuel manufacture including the following:

DYSPROSIUM POISONED FUEL - AECL Research undertook a program to fabricate and qualify dysprosium-poisoned fuel for use in an advanced fuel bundle concept. During the early stages of the program, various routes for fabricating dysprosium-poisoned pellets were investigated and one was chosen that yielded the most uniform microstructure (grain size, grain size distribution, dysprosium distribution). At this stage, sufficient fuel pellets were made for approximately 20 CANDU elements using pellets made by various fabrication routes and these elements were assembled and incorporated in a demountable element (DME) bundle for irradiation testing. The dysprosium content in the various fuel elements was varied from 1% to 15%. The irradiation is ongoing.

Once the fabrication route was chosen (based on microstructure and throughput), a pilot facility for blending dysprosia and urania (depleted in U-235, designated here as DU) powders was established and approximately 200 kg of dysprosium-poisoned fuel powder was produced. This powder was pressed on an automatic press and sintered into fuel pellets using a refractory metal furnace. One problem encountered was that since DU powder is a specialty item, the properties of the powder were unpredictable and not well controlled. For example, the powder was from old stock at the supplier and had to be reduced and passivated prior to blending to suppress cracking in the sintered pellets. This problem will be avoided in the future, as the required DUO₂ powder will be provided in a controlled manner, following standard CANDU fuel manufacturing practice.

Sufficient poisoned fuel pellets were fabricated from this blended powder for approximately 300 CANDU elements. These elements were fabricated and incorporated into 38 prototype bundles

using dedicated assembly facilities. The fuel bundles are to be used in demonstration irradiation and reactor physics tests. Initial reactor physics measurements have been completed and the prototype irradiation is ongoing. The program is now focused on producing a high-burnup version of this bundle concept, based on the CANFLEX carrier bundle.

CANFLEX FUEL

Since the start of the CANFLEX fuel development program in 1986, several bundles have been fabricated for irradiation testing, thermal hydraulic assessments, and other testing. AECL, in collaboration with commercial fuel fabricators, has identified and is developing appropriate fabrication technology for the CANFLEX bundle. To date, a total of 56 bundles (at 43 elements per bundle or 2,048 total elements) have been manufactured. Plans are underway for production of an additional 40 bundles or more for irradiation in a power reactor.

Some of the bundles manufactured to date have been used for thermal hydraulic performance optimization and flow and fuel handling tests. These tests have provided data necessary to maximize the critical power of the CANFLEX bundle, and demonstrate the bundle's flow endurance and compatibility with existing fuel handling systems.

Other bundles have been used for irradiation testing in NRU. One bundle is currently being irradiated (started in February, 1994) and two are awaiting installation when one of the NRU loops becomes available. Other bundles will be inserted into the test sections to replace others that are removed for interim post-irradiation examination. These irradiation tests will demonstrate the performance of CANFLEX bundles under extended burnup conditions.

Reactor physics measurements on 35 bundles are also planned in the zero power ZED-2 facility. Following the physics and irradiation testing, a demonstration irradiation of about 40 bundles is planned in a power reactor for the period 1996 - 1997.

MOX FUEL FABRICATION IN THE RFFL: Plutonium in various forms has been handled in AECL (Chalk River Labs) since 1960. Different research activities were carried out in glove boxes between 1960 and 1970, including development of MOX fuel fabrication technology, measurement of physical properties, production of fuel samples for experimental irradiation, and others. In 1970, a decision was made to re-model the plutonium laboratory and install new facilities to focus on MOX fuel fabrication technology. Installation of the new facilities was complete by 1975. The facilities, collectively referred to as the Recycle Fuel Fabrication Laboratories (RFFL), are designed to produce experimental quantities of MOX fuel for reactor physics tests or demonstration irradiations.

Subject to the special precautions due to the presence of plutonium (e.g., essentially all operations done inside glove boxes), the processes employed in the RFFL follow conventional natural UO_2 practice. The fabrication line was designed for the production of sealed individual fuel elements (pins) starting from UO_2 or ThO_2 powders as the major component and PuO_2 or $^{233}\text{UO}_2$ as the minor component.

The first step in the process is green pellet production. Weighed amounts of the starting oxide powders are mixed either in single-stage or double-stage blending, the latter being used for more dilute mixtures to achieve better homogeneity. After blending, the MOX powder is fed

to the pre-press which converts the mixed powder into compacts, which are, in turn, fed to a granulator. The resulting free-flowing granules are then suitable for final pressing into green pellets using a single-cavity hydraulic press.

Green pellets are loaded to a batch furnace, where sintering is done in a dilute hydrogen cover gas. Sintered pellets are then centerless ground to a specified diameter and surface finish. The pellets are washed, and dried in warm air. Acceptable pellets are loaded into empty sheaths that already have one end cap welded and all appendages brazed in place (these sub-assemblies are supplied by commercial fabricators). Loaded sheaths have their second end cap welded on using a tungsten inert gas welding system. The sealed elements are then helium leak-tested, scanned for surface alpha contamination, and assayed by neutron interrogation. Following assay, the elements are ready for bundle assembly.

FABRICATION CAMPAIGNS IN THE RFFL: Extensive commissioning and test production programs using natural UO_2 were implemented in the RFFL. This enabled several modifications and optimization of process and safety-related equipment prior to the introduction of Pu in the fabrication line in 1979.

From 1979 to 1987, several MOX fuel fabrication campaigns were run in the RFFL producing various quantities of fuel with different compositions, see Table 1. The first campaign consisted of (U, Pu) O_2 fuel with 0.5 wt.% Pu in heavy element (HE). Later, about 1.3 Mg of (Th, Pu) O_2 fuel elements were produced containing up to 2.3 wt.% Pu in H.E. The last campaign was particularly challenging since it involved the fabrication of 1350 elements containing 1.4 wt.% ^{233}U in H.E. (due to the presence of ^{232}U which has a gamma-active daughter).

The fuel elements and bundles produced in the RFFL were used for test irradiations in NRU and for physics tests in the zero power ZED-2 reactor. Overall, there were about 150 bundles, containing over 3 Mg MOX, fabricated in the RFFL before operations in the facility were suspended. In late 1987, the RFFL was placed in a state of active standby, a condition when no fuel fabrication activities are conducted, but the monitoring and ventilation systems in the facility are maintained.

Currently, a project to rehabilitate the RFFL and resume MOX fuel fabrication is underway. The re-commissioning (using inactive material) is scheduled for mid-1995, with the regulatory approval to resume MOX operations anticipated by late 1995. The initial campaign will consist of the production of thirty seven, 37-element (U, Pu) O_2 bundles containing 0.3 wt.% Pu in H.E. destined for physics tests in ZED-2.

TABLE 4.1 FUEL PRODUCED IN THE RFFL

EXPT.	DATE	FUEL TYPE	QUANTITY
DP-12	1977-78	Natural UO ₂	50, 19-element bundles
BDL-419	1979-80	(U, Pu)O ₂ @ 0.5% Pu	15, 36-element bundles
BDL-422	1981-83	(Th, Pu)O ₂ @ 1.75% Pu	6, 36-element bundles
BDL-430	1982	Natural ThO ₂	1, 36-element bundle
WR1-1012	1982	(Th, U)O ₂ @ 1.8% Pu	2, 21-element bundles
	1982	(Th, Pu)O ₂ @ 2.3% Pu	2, 21-element bundles
WR1-1010	1982-85	(Th, Pu)O ₂ @ 2.3% Pu	1332 elements
BDL-432	1986-87	(Th, U)O ₂ @ 1.4% U-233	1350 elements

4.4 Fabrication Strategy

Each MOX fueled CANDU Bruce reactor will consume 15.5 bundles of fuel per full power day. Based on two reactors fueled with MOX fuel, and an assumed 80% capacity factor, the net fabrication requirement is therefore 9050 CANDU MOX bundles per year. Based on a nominal loading of 232 grams of plutonium per bundle, this equates to a net consumption of 2.1 Tonnes of Plutonium per year. This translates into net element requirements of 108,600 Ring 3 Elements (12 per Bundle) per year and 162,900 Ring 4 Elements (18 per Bundle) per year.

At the nominal ceramic weight of 578 grams per element, this equates to approximately 157,000 Kg of Mixed Oxide Fuel per year. To this, an allowance for recycle of off-specification components must be added to establish nominal running rates for the MOX fuel facility. Based on a conservative estimate of 13,000 Kg/year for recycle, the gross capacity of the CANDU MOX facility is 170,000 Kg per year. For purposes of comparison, the Zircatec Precision Industries Facility in Port Hope Ontario, has a nominal capacity of 1,200,000 Kg per year and typically produces approximately 900,000 Kg per year of CANDU fuel. The proposed CANDU MOX facility is therefore less than 20% of the size of an existing CANDU fuel fabrication facility, which verifies the reasonableness of the concept.

OVERALL FABRICATION STRATEGY

The recommended Fabrication Strategy is to house, in the new CANDU MOX fuel fabrication operation (modified FMEF Facility) only those manufacturing, test and inspection operations involving plutonium processing, handling or storage. All support operations, not involving plutonium will be performed in existing facilities, by organizations currently performing these operations. Therefore:

- (1) Zircaloy end plates, end caps, tabs, spacers and extended blanks for tubing will be produced by subcontractors, as is current practice in the CANDU program.
- (2) Cladding subassemblies, complete and ready for pellet loading, with one end cap welded in place, will be fabricated by existing CANDU fuel suppliers and delivered to

the CANDU MOX facility ready for pellet loading.

- (3) Dysprosium elements, for rings 1 and 2 will be produced in an existing CANDU Fuel Fabrication Facility and delivered to the CANDU MOX facility for bundle assembly.
- (4) Natural uranium end pellets will be will be produced in an existing CANDU Fuel Fabrication Facility and delivered to the CANDU MOX facility for element assembly.

This strategy has several advantages. First, it minimizes the capital cost of establishing the CANDU MOX facility. Second, it minimizes training and startup costs by avoiding the establishment of redundant capabilities. Third, it minimizes the amount of equipment and facility potentially requiring treatment as TRU waste at the end of the program. Fourth, it minimizes the disruption to, and impact on, the existing CANDU fuel supply infrastructure by keeping as much work as practical in existing facilities.

The recommended strategy also minimizes the square footage of new facilities required to establish the MOX fuel fabrication capability. Based on initial estimates of the square footage required for CANDU MOX fabrication, and the amount of space available within the FMEF, there will be significant extra space available within this building. This space could be used to house the processing lines for conversion of the surplus weapons components to PuO_2 powder in the same building as the CANDU fuel fabrication facility. This would allow for significant savings in capital and operating costs, by taking advantage of the full capability of this facility, and by spreading the security and landlord costs over a broader base. It would also provide for enhanced safeguards and security protection of the plutonium as there would be only one location in which all processing operations would be performed and there would be no shipment of PuO_2 powder from the conversion facility to the fabrication facility with the attendant costs and risks of such shipments.

The facility plan described in the following sections is based on performing all operations from powder weighing and blending to element seal welding in atmosphere controlled gloveboxes. Besides protecting workers and the environment from contamination, properly maintaining the integrity of these gloveboxes will minimize the eventual Decontamination and Decommissioning costs. Once the elements have been seal welded and verified free of surface contamination, they will be removed from the glovebox facility for bundle assembly. Completed bundles will be loaded into shipping container inserts, compatible with storage facilities in the FMEF, which will protect the bundles from damage and maintain their integrity under potential accident conditions.

Pellet fabrication, element fabrication and bundle assembly operations will be automated such that all routine manufacturing, inspection and test operations will be performed without direct operator exposure to the plutonium bearing materials. Manual operation will be avoided where ever practical. The primary motivation for automating is to avoid hands-on processing of plutonium bearing materials, components and assemblies to minimize worker radiation exposures. Further, automation of these operations will provide cost benefits over the life of the program. It should be noted that automation of these operations will not be a major challenge for the CANDU design. Pellet pressing operations are routinely performed in an automated manner and the compact design of the CANDU elements and bundles greatly simplifies

automated handling and processing. Most element processing and bundle assembly operations are already automated in existing CANDU production facilities.

MOX PELLET AND ELEMENT FABRICATION STRATEGY

The reference CANDU MOX fuel bundle design uses different plutonium loadings in the Ring 3 and Ring 4 Elements. There are two possible strategies for production of these components. They are:

- (1) Using a single processing line, producing the two different loadings in campaigns, separated by enrichment cleanouts. Or,
- (2) Using two separate lines, each dedicated to production of one of the loadings.

The Fuel Supply Team concluded that two lines should be used. The two lines will share a common bulk PuO_2 vault and Master Blend preparation area. Master Blends, nominally 10% PuO_2 , will be prepared and QA sampled in areas adjacent to the bulk PuO_2 vault and returned to the vault for storage until required for pellet fabrication. Each line will perform all operations from Master Blend dilution and final blending to element loading, seal welding and inspection. This strategy was selected for the following reasons:

- (1) Minimization of locations where pure PuO_2 powders are processed and handled. The single Master Blend Processing area, adjacent to the Bulk PuO_2 vault, minimizes handling of pure PuO_2 and simplifies accountability requirements.
- (2) Quality Assurance and Accountability. Use of two separate pellet and element lines provides virtually 100% assurance that there can be no inadvertent co-mingling of powders, pellets or elements and assures that each element is properly loaded.
- (3) Reduced Radiation exposure. In an automated fuel manufacturing facility, the greatest potential for radiation exposure comes during non-routine and manual maintenance operations. As enrichment cleanouts would fall into this category, and would be required frequently, this potential source of worker radiation exposures is eliminated by this strategy.
- (4) Reduced operating cost. Enrichment cleanouts are time consuming and labor intensive. Their avoidance reduces the annual operating costs for the CANDU MOX facility.
- (5) Utilization of existing FMEF equipment. While the CANDU MOX facility will have a considerably greater throughput than was originally planned for the MOX line located on the 70 foot level of the facility, preliminary analysis is that the sintering furnaces, and other portions of the existing equipment, already in place would be adequate to support the required throughput for a single element type.
- (6) Redundancy. Having two lines provides added flexibility in the event that one line suffers a major outage. While total output would suffer, having two independent lines provides the ability to switch a line from one loading to the other (after an enrichment cleanout) and maintain approximately 50% of the nominal output while the breakdown is repaired.

It should be noted that the alignment pad and spacer locations on Ring 3 and Ring 4 Element types are different. The facility will take advantage of this difference and element fixturing in the two lines will be designed so that only the correct cladding subassemblies will be able to be used in each line. Once element fabrication has been completed, similar fixturing will be used to assure proper placement of the elements into the bundles. Together, these steps provide positive assurance of proper element and bundle loading and form a key part of the overall plan for satisfaction of Material Control and Accountability requirements.

Figure 4-5 illustrates the MOX Fabrication Process Flow Sheet based on the above strategy.

4.5 Feed Materials

This section defines the requirements for feed materials, including specification requirements for DOE furnished materials (PuO_2 Powder, and DUF^6) and proposed supply arrangements and costs for fuel supply vendor acquired materials. (DUO_2 Powder conversion, Cladding components, Natural Uranium end pellets, and Dysprosium Elements.)

4.5.1 PuO_2 Powder

A specification for the PuO_2 to be provided by DOE for this program will be an essential requirement for program implementation. In consideration of the various routes for metal-to-oxide conversion currently being considered by DOE and described by Walter et al, suitable PuO_2 powder properties for the supply of high quality CANDU MOX fuel seem very attainable and not likely to be an issue. This conclusion is supported by a large base of MOX fuel fabrication experience with PuO_2 - UO_2 fuels.

In comparison with reactor grade plutonium produced by separation from spent reactor fuels, weapons grade plutonium has lower impurity levels., except for Gallium. Typical impurity levels, as described by Robbins and Wheeler¹, are presented in Table 4-2.

TABLE 4.2 TYPICAL WEAPONS GRADE PU IMPURITY LEVELS

Element	Analysis, ppm	Element	Analysis, ppm
C	260	Mn	50
Fe	380	Ni	215
Al	75	Si	70
Cr	80	Ta	-
Cu	95	W	-
Mg	10	Ga	0.96 WT %

¹Robbins, J.L. and Wheeler, A.D. (November 1965), "Some Observations on the Compressive Creep Behavior of a Delta Stabilized Plutonium - 1 w/o Gallium Alloy", Lawrence Livermore Radiation Laboratory, from preprint prepared for *The Third International Conference on Plutonium*, sponsored by The Institute of Metals.

Figure 4.5 CANDU MOX Fuel Fabrication Flow Sheet
(Source: Zircotec Precision Industries, Inc)

ASSEMBLY DWG.

FD-8861

ZIRCOTEC PRECISION INDUSTRIES INC.			
TITLE MANUFACTURING FLOW CHART			
CANDU/SAF FUEL PROCESS			
DIMENSIONS IN INCHES UNLESS OTHERWISE SPECIFIED		SCALE	
DESIGNED BY	DATE	APP. BY	DATE
CHECKED BY		APP. BY	
CHECKED BY		APP. BY	
			94C063

For the purpose of this study an assumed baseline chemical specification covering impurities levels in the PuO_2 is proposed in Table 4.3, as developed from the FFTF specification. Assumed baseline chemical and physical properties are shown in Table 4.4. Actual specification values may differ from Table 4.3 and Table 4.4 following development of the actual interface specification. Some adjustment to the particle size may be needed to achieve micro homogeneity requirements in such a dilute PuO_2 concentration as required for CANDU MOX fuel, but this will depend on bundle design requirements and blending methods. Additionally, for purposes of this study, it is assumed that the PuO_2 will be shipped to the FMEF facility in the existing design FMEF SNM canisters.

TABLE 4.3 - PROPOSED PuO_2 IMPURITY LIMITS FOR CANDU MOX FUEL MANUFACTURE

Element	Analysis, ppm	Element	Analysis, ppm
Al	500	Ni	500
Be	20	N	200
B	20	P	100
Cd	20	K	200
Ca	250	Na	500
C	150	S	300
Cl	20	Ta	400
Cr	250	W	100
Co	20	V	500
F	10	Cu,Zn,Zr, Ti	800
Fe	500	Ag,Mn, Mo,	
		Pb, Sn	200
Li	10	Sm,Eu,Gd,	
		Dy (all)	100
Mg	25	Si	150
Ga	1 wt%		

TABLE 4.4, ADDITIONAL PuO_2 PROPERTIES (BASELINE)

Property	Specification
Surface Area	2.5 m ² /gm, Min.
Particle Size	P _{avg} 2 to 5 Microns
Pu Assay	87.5 Min.
Loss on Ignition	2.5% Max by Weight when Calcined @ 950C
O/M Ratio	1.95 - 2.00
H ₂ O Content	< 0.5%

Several working specifications for cerarnic fuel grade PuO_2 powder have been used in the U.S. for DOE sponsored projects which required production of MOX fuel substantially higher in Pu content than is proposed for CANDU MOX fuel. MOX fuel with over 22% Pu was successfully

produced for the FFTF project with RDT specification #E 13-1T during the period 1971 - 1980. This specification was used as a target for PuO₂ powder supplied by Hanford, Los Alamos and Savannah River. The specification levels and typical measured properties are shown Tables 4.5 and 4.6. This specification and its successor, NE #13-1T, were developed at Hanford by Battelle and Westinghouse and both were demonstrated to be adequate with minor adjustments on surface area and volatile contents for some lots. These properties were achieved with aqueous type oxalate and peroxide precipitation from plutonium solutions in nitric acid. Calcining of some of the material was performed at 950 °C and for various times to obtain PuO₂ power with adequate morphology and Loss On Ignition (LOI) for batch make-up. This PuO₂ was used by the Babcock & Wilcox Co. to fabricate more than three core-loads of FFTF fuel pins which exceeded design goals in service.

TABLE 4.5 - TYPICAL FFTF PUO₂ CHEMICAL ANALYSIS

Element Impurity	Specification ppm	Typical Avg.	Typical Min.	Typical Max
Al	500	<20	<20	<20
Be	20	<1	<1	<1
B	20	<1	<1	<1
Cd	20	<1	<1	<1
Ca	250	<20	<20	<20
C	150	43	31	54
Cl	20	15	8	28
Cr	250	<10	<10	<10
Co	20	40	40	40
F	10	<2	<2	<2
Fe	500	62	30	100
Li	10	<1	<1	<1
Mg	25	14	10	16
Ni	500	41	40	50
N	200	<10	<10	<10
P	100	18	15	19
K	200	<20	<20	<20
Na	500	<50	<50	<50
S	300	<11	<10	<14
Ta	400	<50	<50	<50
W	100	<26	<25	30
V	500	<1	<1	<1
Cu, Zn, Zr, Ti	800	85	--	--
Ag, Mn, Mo, Pb, Sn	200	40	--	--
Sm, Eu, Gd, Dy	100	40	--	--
Si	150	34	30	50

TABLE 4.6. FFTF PUO₂ PHYSICAL PROPERTIES

	Typical Composite Sample	Typical Blended Processed Material	Specification
Surface Area	3.67 m ² /gm	4.14 m ² /gm	2.5 m ² /gm, Min.
Particle Size	.65 Microns	.63 Microns	P _{avg} <10 Microns
Pu Assay as Received	87.76%	87.90%	85.0% Min.
Pu Assay Fired at 950C	88.10	87.78	87.5 Min.
Loss on Ignition	.25%	.27%	Calcined @ 950 2.5% Max by Weight, Calcined @ 950C

Suitable MOX fuel powder has also been successfully produced by blending PuO₂ from direct oxidation of plutonium metal. Fuel rods used in the SAXTON project were successfully fabricated from plutonium metal buttons converted to PuO₂ by direct oxidation. The buttons were broken up by hydraulic pressing, and air oxidized to PuO₂ powder, which was sieved and dry ball milled to minus 325 mesh. Uranium metal balls of identical enrichment as the UO₂ in the pellets were used in the pulverizer. Pu micro-homogeneity was satisfactory in the 6.6% PuO₂ sintered pellets and the Zircaloy clad finished rods met Westinghouse specifications. These rods went into bundles that were tested very successfully to burn-ups above 50,000 MWD/MTHM.

PuO₂ for European MOX programs at Dessel (Belgium) Caddarache (France), HANAU (Germany) and in the UK has been made to specifications consistent with those developed by the US - DOE with good quality pellets resulting. From this wide base of experience, PuO₂ physical and chemical properties needed for fabrication of high quality CANDU MOX fuel pellets are well understood and can be satisfied by all of the metal-to-oxide conversion routes currently under consideration, including direct oxidation of Pu Metal.

ACCEPTABILITY OF GALLIUM - The Fuel Supply team, with the help of Oak Ridge National Laboratory, has determined that Gallium will be an acceptable impurity in the CANDU MOX fuel. An analysis was made to determine the effect, if any, of leaving the approximately 1% Gallium (GA) in the Plutonium during conversion to PuO₂. It has been concluded that the Gallium contamination is acceptably low, and does not require chemical removal during metal to oxide conversion. The reasons for this are outlined below:

- (1) The MOX powder made from this type of PuO₂ will contain only 120 to 200 ppm Gallium prior to fabrication into pellets, since the Pu is diluted by Gallium-free UO₂ during powder blending.
- (2) During sintering at 1650-1700°C (typ.) in a hydrogen-containing atmosphere, the oxygen chemical potential is low enough to reduce the initial oxidation state in Ga₂O₃ to GaO

and then to Ga(m). The vapor pressure of Ga(m) dominates since it is quite high ~ 2 to 3×10^2 atm at these temperatures. GaO is extremely unstable and is not a factor. Most of the Ga will thus boil off during sintering, leaving perhaps only a few 10's of ppm in the sintered pellets. There is a low possibility of interaction with Zircaloy.

- (3) After loading the fuel rods, what is the possibility of formation of Ga-Zr compounds? No irradiation tests were known to exist where UO_2 based fuel with Ga was tested in Zircaloy cladding.

The phase diagram for the Ga-Zr system shows a slight solubility of Ga in Zr which increases from 700°C to 850°C . The normal service temperature of Zr clad is below 500°C . The phase diagram shows no solubility data in this low temperature area.

The vapor pressure of Ga within the fuel rods is difficult to estimate, although it will be lower than during sintering, because the normal centerline temperature is relatively low.

In any case if all the Ga contained in MOX with ppm levels were to evaporate and be dissolved or reacted with the Zircaloy cladding, the average % Ga for several initial Ga levels would be as shown on table 4.7 (based on 13.08mm sheath diameter, 0.406mm wall thickness):

TABLE 4.7 - WT % GA IN ZR IF ALL TRANSFERS TO SHEATH

Initial ppm Ga in MOX pellets after sintering				
	1	5	10	20
wt% Ga in Zircaloy*	0.001	0.005	0.01	0.02
ppm Ga in Zircaloy*	10	50	100	200

The first Zr-Ga compound identified on the phase diagram is Zr_2Ga which contains $\sim 27\text{W/O GA}$. With the worst case scenario (above) in which all the Ga residue in pellets transfers to the Zircaloy, there is not enough available for significant compound formation at fuel operating temperature. The vapor pressure of Ga in the gas within the fuel rod would be extremely low, since the residual Ga would be interstitially accommodated in the UO_2 crystal lattice and its' low ppm content also means lower likelihood of release to the pellet-sheath gap and thence to the sheath. (The presence of a graphite-based layer on the sheath I.D. has not been analyzed). Even if the Ga volatility were more characteristic of elemental Ga, at the 500°C sheath I.D. temperature, its' vapor pressure is only $\sim 2 \times 10^{-12}$ atm, which is extremely low for bulk transport to occur.

Despite these analytical predictions, acceptability of Gallium in CANDU, or any other MOX fuel reactor alternative, should be confirmed by fabrication tests and design studies on fuel performance with 1 weight percent of Ga in the Pu portion of the MOX powder. Of all reactor options being considered, CANDU MOX fuel should be the least sensitive to this constituent, on the basis of lower Pu content in the fuel.

AMERICIUM: This study assumes that Americium will be present in the PuO_2 and will not be removed prior to fabrication. In comparison with reactor generated plutonium, weapons plutonium contains much lower levels of Pu^{241} which is the source of the Am^{241} . The same level of Am^{241} which would accumulate by aging reactor grade Plutonium for 6 years would require over 20 years of aging in weapons plutonium. However much weapons grade Pu has been stored from 10 - 20 yrs. Depending on the specific history of this material (i.e., original Pu^{241} content and its age when processed for Am removal) some of the plutonium to be issued towards the end of this project will have significant amounts of Am^{241} , comparable to reactor grade plutonium.

For additional assurance of fabricability, this facility will be equipped with a PuO_2 finishing station to facilitate final adjustments to the powder morphology on an as needed basis. The Pu finishing station is described later in this report.

4.5.2 UO_2 Powder

A reliable supply of depleted UO_2 powder (DUO_2) capable of producing high density, crack-free fuel pellets will be required. This will be the base material into which the requisite amounts of plutonium oxide (PuO_2) and dysprosium oxide (Dy_2O_3) will be blended. Approximately 170 MgU (as DUO_2) per year will be required for the reference case or 340 MgU (as DUO_2) per year if the volume were doubled.

The reference supply source for this study is base on drawing depleted UF_6 (DUF_6), in shippable condition from DOE stocks at Oak Ridge, Tennessee. The material (0.20 - 0.25% U^{235}) would be transported to Cameco Corporation's operations at Port Hope Ontario and there converted to high grade powder.

A new defluorination unit to convert DUF_6 to DU_3O_8 and the couple to Cameco's existing small scale UO_2 conversion plant would be built and will become an operating plant, separate from their main UO_2 conversion plant, but fully qualified and operable.

The new defluorination unit would receive and handle the large DUF_6 shipping containers, vaporize the contents, clean the cylinders, and pass them to an existing cylinder recertification operation. The DUF_6 would be treated by hydrolysis and ammoniation (to form ammonium diuranate (ADU)) and by pyrohydrolysis (heat and moisture) to form DU_3O_8 or DUO_3 .

The oxide product from the defluorination unit would be fed into existing facilities for conventional treatment consistent with present practices. This would include reforming to ADU, calcining under hydrogen atmosphere to form DUO_2 followed by conditioning and blending to assure consistent high quality material for drum shipment to the fuel fabrication sites (MOX and Dy).

Cameco has sufficient space for the additional equipment in their licensed nuclear facilities at Port Hope. Although Cameco is not converting UF_6 to UO_2 at this time, they did so from the mid-1960's until 1982 and have the necessary process knowledge. Cameco is a world scale producer of UF_6 from U_3O_8 in concentrates and is the mainstay source of natural UO_2 for CANDU reactors.

Although the reference supply does not rule out investigation of other sources, use of the Cameco plant has distinct advantages since it makes use of a proven finishing plant. Output material from this facility would reliably meet the existing natural UO_2 quality requirements, including pellet making tests which are conducted by the fuel fabricator, subject to proof testing to ensure that chemical limits can be met, particularly fluorine content.

Costs for the DUO_2 supply are estimated as follows (in 1993 dollars):

- For 170 MgU per year for 25 years. Unit cost, including construction of added facility, operation and decommissioning cost (spread over the period): \$25.00 per KgU as DUO_2
- For 340 MgU per year for 25 years. All in unit costs as above: \$17.86 per KgU as DUO_2
- The cost of shipping DUF_6 from Oak Ridge Tennessee to Cameco's Ontario facilities is estimated at: \$0.29 per Kg DUF_6
- The cost of shipping DUO_2 to Richland Washington is estimated at: \$0.283 per KgU as DUO_2
- The cost of shipping about 20% of the DUO_2 to nearby fuel fabricators (for Dysprosium element fabrication) is negligible.

For scheduling purposes, it is estimated that after a firm go-ahead, 8 - 12 months would be required for regulatory approvals and engineering plus about 2 years for construction and tests. Cameco can supply small quantities of DUO_2 in the interim.

Should the volume need to be doubled (to 340 MgU per year), at the outset or after operation at the 170 MgU per year level, it would not be difficult for Cameco to respond as the existing finishing plant has a capacity of 500 MgU per year which is ideal for the larger quantity. Only upgrading of the defluorination unit would be required and it is estimated that this could be accomplished in 6 - 9 months for regulatory approval and engineering plus a year for construction.

4.5.3 Zirconium Alloy Components

The Zirconium Alloy Components required for the production of CANDU MOX fuel bundles are: 1) Fuel Tubing, 2) Appendages, 3) End Caps, and 4) End Plates. These components will be produced for this program by existing, qualified, CANDU fuel suppliers, using existing facilities and qualified procedures. This will assure the quality of these components, avoid unnecessary costs and minimize the impact of this program on the existing CANDU fuel infrastructure.

Brief descriptions of the approaches to be used for supply of these components follow.

FUEL TUBING. CANDU fuel suppliers have manufacturing facilities for the production of the seamless zirconium alloy tubing required for the fuel sheath. The starting material for these facilities is large diameter zirconium alloy tubing which is purchased from qualified suppliers in the United States and Europe (Teledyne Wah Chang, Western Zirconium and Cezus). This starting material is reduced in diameter and wall thickness in multiple steps, using a cold pilger process, with intermediate cleaning and heat treating steps. Existing CANDU fuel suppliers have sufficient capacity to support this program.

APPENDAGES: The appendages required for assembly of the CANDU fuel bundle are manufactured by each fuel supplier to meet their own needs. Starting materials in the form of zirconium alloy strip and wire are purchased from qualified suppliers. The fuel suppliers each have their own facilities for coating the appendages with the beryllium required for the brazing process used to attach the appendages to the fuel sheath. Existing CANDU fuel suppliers have sufficient capacity to support this program.

END CAPS: The caps that are welded to each end of the fuel sheath are manufactured from zirconium alloy bar using standard machining techniques. Current practice is for the fuel suppliers to procure the bar material, perform an ultrasonic test for defects and subcontract the machining of the end caps. Existing suppliers have sufficient capacity to support this program.

END PLATES: The end plates to which each fuel element in a CANDU fuel bundle are welded are manufactured from zirconium alloy strip using a progressive die punching operation. Current practice is for the fuel suppliers to procure the strip material and subcontract the manufacture of the end plates. Existing suppliers have sufficient capacity to support this program.

COST ESTIMATE FOR SUPPLY OF ZIRCALOY BUNDLE COMPONENTS: A cost estimate has been developed for supply of all zirconium alloy components for 9050 CANDU fuel bundles per year. Fuel bundle component kits will include sheath sub-assemblies for the 30 outermost elements, with one end cap welded on, 30 end caps and two end plates. Components will meet all exiting specification requirements. The end cap weld will be subjected to both a non-destructive test of weld integrity and a helium leak test.

The estimated cost for delivery of completed fuel bundle component kits to the FMEF at Hanford is \$555.78 per kit. Allowing for a small loss rate during MOX assembly, this would mean an annual cost of \$5,107,000 per year.

4.5.4 Natural Uranium End Pellets

Natural uranium end pellets will be used at both ends of the fuel elements to reduce flux peaking. These pellets will be required in multiple lengths as they also serve to adjust pellet stack length in the elements. These pellets will be produced by existing CANDU fuel suppliers, and delivered to the MOX facility as fully qualified and certified parts, ready for element assembly. Existing suppliers have sufficient capacity to support this program. Cost of end pellets for 9,050 bundles per year is estimated at \$410,000.

4.5.5 Dysprosium Elements

The center elements of each bundle will be loaded with dysprosium containing depleted uranium dioxide pellets. It is planned that these elements will be fabricated at a site remote from the MOX fuel plant and that they will be provided to the fuel fabrication facility as fully qualified and certified parts, ready for bundle assembly. This will minimize the cost of producing these elements and will avoid possible dysprosium contamination of the MOX lines.

Previous experience with making dysprosium containing pellets has been good. The Chalk River Laboratories are currently making pellets to AECL-CANDU's latest dysprosium pellet specification, and are fabricating fuel assemblies for test irradiations in the NRU test reactor

and in a CANDU power reactor. Experience to representative burnup levels will be demonstrated well ahead of the schedule for the current program.

For this program, it is planned to locate the dysprosium fabrication in a dedicated facility, possibly at either of the two CANDU fuel fabricator's plants or a U.S. fuel fabrication plant. Although final details have to be worked out, no problems are anticipated in large scale fabrication of these pellets and elements. Several companies have indicated an interest in supplying the ceramic dysprosium oxide pellets and elements. A rough cost estimate for supply of these components, including amortized capital costs, material, and shift operations, leads to a delivered cost of about \$2,590,000 per year.

4.6 FMEF REFERENCE CASE

4.6.1 General FMEF Description

The Fuels and Materials Examination Facility (FMEF) was built during the late 1970's and early 1980's as a major addition to the breeder reactor technology development program at the Department of Energy's Hanford Reservation. The FMEF includes a facility, known as the SAF (Secure Automated Fabrication) Line, for production of MOX fuel. This facility, located on the 70 foot elevation of the FMEF, is used as the starting point for development of a CANDU MOX fuel fabrication facility at Hanford. A cutaway view of the FMEF facility is shown in Figure 4-6.

The SAF line was originally intended to receive nuclear fuel materials, process them into uranium-plutonium mixed oxide (MOX) pellets, and encapsulate the pellets in stainless steel cladding to form fuel pins for U.S. liquid metal fast breeder reactor (LMFBR) development programs. It was to supply mixed oxide fuel for the Clinch River Breeder Reactor (CRBR) and the Flux Test Facility (FFTF) and demonstrate fully automated nuclear fuel fabrication. SAF had a design capacity of 6 tonnes of MOX per year, operating 3 shifts per day, 5 days a week. The intended fuel composition was approximately 35/65 PuO_2 / UO_2 using feed plutonium oxide containing 10 - 20 wt% Pu^{240} . Installation and pre-operational testing has been completed for 75% of the equipment. The CRBR project was canceled and when FFTF was shut down it left SAF Line without a mission. The following describes the line and important process equipment as it was originally intended for use in fabricating LMFBR fuel.

POWDER PROCESSING STATION: The SAF Line Powder System is located in Room 607 on the 70 foot level. A manually operated transfer cart picks up the fissile material (in 4.1-liter inner canisters contained within French canisters) and delivers it to the Fissile Receiving enclosure. Within the enclosure, inner canisters are identified and weighed before being temporarily stored or emptied into batching equipment. Figure 4-7 is a photograph of the FMEF SNM storage vault, with an inner SNM canister being transferred by remote tooling. As explained later, this vault and tooling would be used as part of the CANDU MOX fuel fabrication operation in FMEF.

Figure 4.6 Cutaway View of Hanford FMEF (Source: Westinghouse Hanford Company)

Figure 4.7 FMEF SNM Storage Vault
(Source: Westinghouse Hanford Company)

Fertile materials (UO_2) are stored as received in 5 gallon drums in Room. 433. It is repackaged into a 4.1-liter inner canister in Room 421 for transfer in a French canister to Fertile Receiving enclosure. Then the UO_2 powder is transferred to Batching. Proper quantities of batch components are weighted in Batching before the powders are transferred to Blending. The V-shell blender mixes the batch components to produce a homogeneous macroblend in preparation for Milling. The normal batch size is 70 kg. Milling breaks up powder agglomerates and completes the blending process to achieve microhomogeneity. A nitrogen-driven jet mill is used for this process.

Milled powder is transferred to Binder Addition, where binder/pore-former material (organic powder) is added to, and blended with, the powder for fuel pellet density and Oxygen to Metal (O/M) ratio control. The blend of MOX powder and binder/pore-former is transferred to the adjacent enclosure for compaction and granulation. The powder is pre-slugged (pressed into tablets) and granulated to prepare a consistent pellet press feed material. The final operation is Lube Addition, which blends pressing lubricant (organic powder) with the granules. Turbula blenders are used for both Binder and Lube Addition. Lubricated granules, contained within inner canister/French canister assemblies, are transferred to the Pellet System in Room 605 using a manually operated transfer cart.

SAF LINE PELLET SYSTEM: Canisters of lubricated granules are transferred into the enclosure and emptied into a hopper which feeds two presses. FFTF pellets are pressed five at a time. Hydraulic presses are employed with features to equalize pressure in each die cavity. Shallow molybdenum boats are loaded with about 1.2 kg each of pressed pellets for automated transport to debinding and sintering operations.

At debinding, a low-temperature continuous furnace removes the organic material (binder/pore-former and lubricant) from the pellets. Boats are transferred to sintering where a high-temperature continuous pusher furnace densifies the pellets and produces ceramic-grade pellets with the specified density and O/M ratio. Reducing atmospheres, (N_2 - 8% H_2 for debinding and Ar-8% H_2 for sintering), are employed for heat-treating operations.

Boats of sintered pellets are transferred to Boat Unloading where a sample is taken to determine further disposition. Density gauging and chemical analysis determine whether the pellets are acceptable, require further furnace processing, or should be recycled. Acceptable pellets are emptied into storage containers and placed in Pellet Storage where up to 200 kg of MOX pellets can also be transferred to the SNM Storage Vault. Pellets determined to have high total retained gas are transferred to Property Adjustment for treatment in a low-temperature vacuum offgas furnace, after which the pellets are returned to storage.

Pellet Surface Inspection receives pellets in storage container quantities (10 Kg sub-batches). Using a fiber optic system, pellets are inspected for surface spots caused by impurities; rejected pellets are removed. After inspection, pellets are sent directly to grinding in a continuous line. The Grinding subsystem centerless grinds the pellets to a specified diameter based on information received from the density gage. Ground pellets are fed directly from the grinder to Pellet Gauging, where pellets are inspected for diameter and surface flaws (cracks, voids, and inclusions). Diameter information is required for certification on a statistical sampling plan basis. Sample pellets are selected for transfer to Chemical Analysis and weight/length (W/L) gauging. The W/L gage is in the same enclosure and adjacent to the pellet surface flaw gage.

Acceptable pellets are fed to Pellet Transport, which loads pellets into a modified inner canister placed within a French canister that is transferred to Column Makeup using a manually operated transfer cart.

Figure 4.8 is a photograph of a typical FMEF glove box station and the current pellet transfer system which would be utilized in the CANDU MOX fabrication operation, as described later in this section.

SAF LINE PIN ASSEMBLY STATION: Pin operations are performed in Room 604 and begin with receiving the cladding with preloaded lower cold components, loading the fuel pellets and the upper cold components into the cladding, and decontaminating the pin end. The pin is then evacuated and backfilled with helium. It is then transferred to the pulsed magnetic weld station where the tag gas capsule is inserted and the top end cap welded onto the cladding. The welded pin is decontaminated and checked for alpha contamination along its length. Completed pins are loaded into shipping containers, leak-checked, and transported within a shipping cask to the Fuel Assembly Area for further processing.

SAF PROCESS INSTRUMENTATION AND CONTROL SYSTEM: The original SAF instrumentation and control system consisted of a data center linked to supervisory process remote control racks (RCRs). The SAF Line was intended to be controlled from the process control racks on the Level 82-ft mezzanine with central data acquisition in the data center computer room. This instrumentation and control system was to provide real-time control, material flow coordination, criticality control, data acquisition and recording, process data analysis, and maintenance diagnostics.

The data center was to receive fuel fabrication process and certification data from the process control racks. This information was to be recorded and subsequently used to generate production and/or traceability reports on a data center computer. Displays of process and material flow data at the RCRs are provided for operators to coordinate SAF Line activities including material transfers between subsystems and chemistry sampling. The RCRs provide alarms and annunciations to aid the operators. Operators communicate with each other through a digital data network and/or an intercom system.

Process control of each subsystem occurs at the remote control racks on the Level 82-ft mezzanine and in Room 605C on Level 70-ft 0-in. The Facility and Safety RCR would remotely monitor parameters and provide alarm and annunciation to the Powder, Pellet and Pin RCR operators. Parameters monitored include criticality (fissile inventory), radiation, enclosure environment, nitrogen recycle system, utilities, security, and enclosure fire protection. (Radiation, criticality, and differential pressure monitors also alarm locally.) Local control consoles (at process enclosure locations) are available for setup, maintenance, and upset recovery when manual operation is needed.

In addition, a digital data communication link between the data center and the safeguards computer system was provided to report SNM movements within the SAF Line.

Figure 4.8 Typical FMEF Glovebox and Pellet Transfer Scheme
(Source: Westinghouse Hanford Company)

4.6.2 Additions and Modifications Required to Support Reference Case

A significant portion of the original process equipment is still in place. Some systems and gloveboxes have been removed during the last 10 years for use in other missions. The study team also determined that some equipment could not be used because of the different configuration and requirements of CANDU powder, pellets and bundles. Overall, about 10 to 20% of the original equipment, has been incorporated in the modified, CANDU MOX line as defined below: (Avoided cost of new equipment estimated at 20 to 30%.)

To produce CANDU MOX fuel in the existing FMEF, the study team proposes to use the existing SAF line area (70 foot level in FMEF) to fabricate ring 3 CANDU fuel elements with 2% PuO_2 . A new, separate, production line will be installed on the 42 foot level in FMEF to fabricate ring 4 CANDU fuel elements with 1.2% PuO_2 . CANDU MOX fuel bundles will be assembled, inspected, packed and stored prior to shipment in an area located on the 0 foot level of FMEF. The bundle assembly area will receive completed fuel elements from the production lines on the 70 and 42 foot levels along with the required Dysprosium elements supplied from an external source. A general schematic showing this layout is shown in figure 4.9.

As shown in Figure 4.9, PuO_2 powders will be received into and stored in the existing FMEF Special Nuclear Materials (SNM) storage vault on the 21 foot level. Facilities for storage of DUO_2 powders will also be established. In close proximity to the storage vault, a new, master blend powder processing station will be installed to blend and condition batches of PuO_2/UO_2 for supply to the production lines on both the 70 foot and 42 foot levels. This facility will serve as the entry point for all fissile and fertile powders into the CANDU MOX fuel fabrication line.

4.6.2.1 General Design Considerations

The combined throughput of the CANDU MOX plant, with two lines producing 157 net tonnes of MOX fuel per year, demands high reliability and maintainability. The process equipment and gloveboxes will utilize advanced design features from SAF Line combined with proven designs from other oxide pellet fuel lines, such as Zircatec's UO_2 plant. It will be essential to minimize unplanned outages for equipment repairs due to breakdowns, as these can impact deliveries, unit costs, and personnel radiation exposure. For example, rebuilding a MOX sintering furnace due to hearth plate slumping (internal deformation) required 20 days during FFTF MOX production. For the larger CANDU MOX furnaces, a month might be needed. Accordingly, such furnaces need to be over-designed and operated conservatively to prolong life and cut down on long maintenance outages. In other areas, redundant equipment and installed spares may be appropriate.

Equipment that has not yet been proven in UO_2 or MOX service will need to be tested, first at the factory, using surrogate materials as feed, then again at the plant site using surrogates, then depleted UO_2 as required to demonstrate working behavior. Vital material handling devices for powder transfers, weighing, blending, homogenization, milling, ceramic pellet fabrication and inspection will be accepted for in-line use only after operating and maintenance personnel, together with vendors, have tested and agreed on adequate demonstration of performance.

Auxiliary equipment will be tested and formally accepted in the same way. Examples are:

Figure 4.9 Schematic of FMEF Building Layout for CANDU MOX Production

- (1) Off gas treatment for removal of MOX fines downstream from milling. Removal, as used here, includes recovery and reintroduction into the process flow stream as well as polishing for reduction of dust load to HEPA's and pre-filters.
- (2) Closed circuit cooling loops, vacuum systems, gas supply systems, electrical and control systems, emergency power and safety-related systems.
- (3) Remote viewing and monitors for automated systems.
- (4) Close coupled Q.C. and process control measuring systems, especially for Pu and U assay, critical impurities and MC&A measurements.

Operator training and qualification will include methods for improving equipment performance and lifetime, spotting early warning signals for P/M scheduling, and cross-communication to aid manufacturing engineers with information from direct observation of equipment behavior.

Wherever practical, the same proven process equipment and containment designs will be used for the Ring 3 and Ring 4 lines to help standardize operating and maintenance procedures. Routine maintenance operations will be included in the hands-on training program. Enough mock-up maintenance operations will be done prior to hot startup to debug access and tooling setups and to permit a rough estimate of maintenance worker radiation exposure(s).

These design and pre-operational training measures will be rigorously pursued in order to achieve ALARA goals less than 500 mRem/year average radiation dose for operators and the extremity dose limits.

4.6.2.2 Master Blend Station

As shown in Figure 4-10, the 21 foot level of the FMEF will house a station to blend and condition batches of PuO_2/UO_2 prior to distribution to the production lines on the 70 and 42 foot levels. This station will serve as the entry point of all fissile and fertile powders into the CANDU MOX fuel fabrication line. Equipment currently located in the SAF Line on the 70 foot level will be relocated to the 21 foot level. With some modifications, this equipment will be used to receive PuO_2 from the SNM storage vault and depleted UO_2 from the FMEF storage area and prepare "master blend" batches containing plutonium at a nominal 10 wt% level. Master blend batches will be stored until required by the production lines on the 70 and 42 foot levels.

Master blend batches will be milled to ensure microhomogeneity prior to further blending with depleted UO_2 to create batches of the appropriate PuO_2 wt% content for the production lines on the 70 and 42 foot levels. Removal of PuO_2 agglomerates from the MOX powder will improve both the consistency of pellet formation during pressing and the quality of the sintered pellets. This will reduce the amount of processing scrap that must be recycled during normal operation of the production facility. In addition, good performance of the MOX fuel in the reactor dictates that there not be PuO_2 agglomerates that would create hot spots in the fuel matrix. Creation of a master blend is key to both good performance of the fuel production processes and the performance of the fuel in the reactor.

Figure 4.10 FMEF Facility Layout at 21 Foot Level
(Source: Westinghouse Hanford Company)

The master blend facility will consist of five major stations: Fissile Receiving, Fertile Receiving, Batching, Blending and Milling. (a sixth station for PuO₂ powder conditioning will be added as described below) Each of these stations is designed to operate remotely with appropriate sensors monitoring process conditions. The equipment is designed for contact maintenance consistent with personnel exposure and safety requirements. For the most part, existing equipment from SAF line will be used at these stations as described in the following paragraphs:

FISSILE RECEIVING: The Fissile Receiving station on the 21 foot level of FMEF will receive PuO₂ from the storage vault, weigh and transfer it to the batching station. The handling system will remove the lid from the double-door port, remove the inner canister containing the PuO₂ from the outer canister and locate it on the weigh station. The inner canister will have a permanent identification code which includes tare weight. Canister identification and gross and tare weights will be recorded for material balance and material control purposes. The canister will be stored or transferred to the batching operation as required.

The equipment for this station now exists on the 70 foot level of the FMEF. It will be moved to the 21 foot level and will require no modifications. The equipment has more than enough capacity to satisfy the daily requirements.

FERTILE RECEIVING: This station will receive depleted UO₂ powder from the FMEF storage area, weigh and transfer it to the batching station. The container of UO₂ will be manually loaded into an elevator which will raise it to the enclosure above the batching station. The container will be manually weighed and identified before being placed inside the enclosure airlock. A motor operated tilting mechanism (similar to the dumper used in the Fissile Receiving Station) is provided to empty the contents of the container in the batching hopper.

The equipment for this station now exists on the 70 foot level of the FMEF. It will be moved to the 21 foot level and will require no modifications. The equipment has more than enough capacity to satisfy the daily production requirements.

BATCHING: The batching station will receive PuO₂ and depleted UO₂, weigh the required quantity of each and transfer this material to the blending operation. The batches being created will contain approximately 10% PuO₂ by weight.

The equipment for this station now exists on the 70 foot level of the FMEF. It will be moved to the 21 foot level and modifications will be carried out to improve the accuracy and reliability of the existing weighing system. This equipment has more than enough capacity to satisfy the daily production requirements.

BLENDING: The blending operation will receive pre-weighed quantities of UO₂ and PuO₂ from the batching operation and blend these powder components to achieve macro homogeneity. Blended powder will be transferred to the milling operation.

The equipment for this station now exists on the 70 foot level of the FMEF. It will be moved to the 21 foot level and the blending unit will be replaced by one of the same type but of a more appropriate size. This equipment has more than enough capacity to satisfy the daily production requirements.

MILLING: The milling operation will receive MOX powder from the blending operation and prepare a micro-homogeneous powder batch for transfer to storage or to the production line on either the 70 or 42 foot levels. The MOX powder may contain agglomerates of PuO_2 and the milling operation is designed to break up these agglomerates to achieve a micro-homogeneous mixture.

The equipment for this station now exists on the 70 foot level of the FMEF. It will be moved to the 21 foot level and only minor modifications should need to be made.

The current milling station is based on the use of a "jet mill". There is some concern that the jet mill may not be appropriate due to powder holdup in this type of equipment. The existing system will be used if it can be demonstrated that: (1) the jet mill has met acceptance test requirements, including powder product transfer through the line leading to the Apitron filter over multi-shift periods, and (2) the Apitron filter/HEPA filter/hopper product discharge system meets operational and trace MOX powder effluent requirements.

If not, the deficiencies will be corrected before hot startup (with PuO_2) or another milling micro-homogenization system will be substituted. If needed, the replacement system will be a multistage "micronizer" hammer mill. This will first homogenize the master blend, followed by second-stage homogenization of the final blend after V-blending with the balance of the depleted UO_2 . Use of mechanical milling in place of jet milling will be contingent upon demonstration of adequate microhomogeneity in sintered pellets. The Pu microhomogeneity specification is the key requirement for this process equipment selection issue.

NOTE: During the SAF Line tour, it was observed that the product transfer line from jet mill to Apitron filter/hopper was very long (at least 20 feet) and included bends, which are likely powder traps; cleanouts were not provided. This will have to be addressed as part of relocation of this equipment to the 21 foot level. Data on the Apitron filter performance for removal of sub-micron dust indicated a potential for excessive material holdup, both on the Apitron filter and in the upper reaches of the hopper.

Once these concerns are addressed, the blending facility has more than enough capacity to satisfy daily production requirements for Master Blends.

PUO₂ FINISHING STATION: A PuO_2 finishing station consisting of a calcining furnace and other equipment will be provided to facilitate minor adjustment of the issued PuO_2 powder if needed. Calcining will provide capability to address powder that is high in moisture or organics. Calcining followed by milling will enhance the sinterability of some oxides which are low in surface area, such as those produced from dry metal conversion routes.

COST ESTIMATE FOR MASTER BLEND LINE: We estimate the equipment procurement and modification costs for the master blend line at \$357,000, and the cost to install and hook-up to existing services at \$143,000, for a total cost of \$500,000.

4.6.2.3 Ring 3 (2% PUO_2) Pellet and Element Line

This line will receive batches of ceramic grade mixed oxide powder from the master blend facility on the 21 foot level, process this material into sintered pellets containing 2% PuO_2 and

encapsulate them into fuel elements meeting all requirements of CANDU MOX fuel. Figure 4.11 shows the layout at this floor. This line will operate at the 70 foot level, predominately by remote control, with all process equipment contained in gloveboxes. Completed fuel elements will be transferred to the CANDU fuel bundle assembly line on the 0 foot level.

This production line will consist of 14 stations which are described in the following sections. All stations will be designed to be operated remotely with monitoring by sensors built into the operating equipment. The equipment will be designed for contact maintenance consistent with personnel exposure and safety requirements.

- (1) **BATCHING:** The batching station will receive master blend MOX powder and depleted UO_2 , weigh the required quantity of each and transfer this material to the blending operation. The batches created will contain 2% PuO_2 by weight. Closed-circuit television cameras at the enclosure will allow the operating technician to observe the operation. This will be a new station of the same type as the one used in the master blend operation on the 21 foot level but scaled up for the larger batch sizes.
- (2) **BLENDING:** The blending operation will receive pre-weighed batches of MOX powder from the batching operation and blend this material to achieve macro homogeneity. The blended powder will then be transferred to the pre-compaction and granulation operation. This will be a new station of the same type as the one used in the master blend operation on the 21 foot level but will use higher capacity equipment to process the larger batch sizes required.
- (3) **PRE-COMPACTION & GRANULATION:** Smoothly flowing MOX powder is essential to successful pressing and sintering of the fuel pellets. To increase the flowability of the MOX powder it has been found necessary to add a process to compress the powder and then granulate the resultant compacts. Batches of MOX powder will be received from the blending operation and, after processing, will be transferred to the lubricant addition and blending operation. Equipment for this purpose now exists on the 70 foot level of the FMEF. However, this unit does not have the necessary capacity and will have to be replaced by equipment with higher capacity.
- (4) **LUBRICANT ADDITION AND BLENDING:** To reduce pellet press tooling wear rates, it is necessary to add a lubricant (such as zinc stearate) to the MOX powder. This station will add the required quantity of lubricant to granulated MOX powder and blend to achieve macro homogeneity. This will be a new station using the same type of blending unit as the one used to blend the MOX powder batch earlier in this production line (item 2 above).
- (5) **PELLET PRESSING AND BOAT LOADING:** This facility will receive granulated and blended MOX powder containing the lubricant and press it into pellet form ready for sintering. Pressed pellets will be transferred onto formed molybdenum trays which will in turn be stacked onto molybdenum slabs to form the sintering boats. The sintering boats will be transferred to the lubricant removal station.

11 11 70 0"

Figure 4.11 FMEF Facility Layout at 70 Foot Level
(Source: Westinghouse Hanford Company)

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Equipment for this purpose currently exists on the 70 foot level of the FMEF. However, this unit does not have the necessary capacity and will have to be replaced by equipment with a higher capacity. The new facility will consist of hydraulic presses designed for quick change of tooling cartridges and ease of contact maintenance. An existing and proven design of pellet handling equipment will be modified for this application to improve ease of maintenance in a glovebox environment.

- (6) **LUBRICANT REMOVAL:** The lubricant added to the MOX powder to improve pellet press tooling life will be removed from the pressed pellets prior to sintering. This will reduce contamination of the sintering furnace with the condensed products off-gassed from the lubricant and therefore reduce the requirement for maintenance clean-up of the furnace. Boats of pellets will be received from the pressing operation by way of the existing boat transport system. After thermal treatment in the continuous furnace, the boats will be returned to the boat transport system for storage and transfer to the sintering operation. Equipment for this purpose currently exists on the 70 foot level of the FMEF. Only minor modifications will be required to the existing equipment.
- (7) **PELLET SINTERING:** Pressed pellets will be heated in a reducing atmosphere to achieve the high density required for the CANDU MOX fuel design. Boats of pellets that have been subjected to lubricant removal will be received from the boat transport system and after being sintered to the required density will be returned to the transport system. The existing furnaces on the 70 foot level of the FMEF will be modified to increase their capacity to meet the requirements for CANDU MOX production.
- (8) **BOAT UNLOADING:** After sintering, pellets will be unloaded from the sintering boats and transferred to the pellet grinding operation. Boats of sintered pellets will be delivered from the boat transport system and empty boats will be returned to the boat transport system for delivery to boat inspection and cleaning. Equipment for this purpose currently exists on the 70 foot level of the FMEF. However it will need to be replaced with a new unit designed to handle the different style of sintering boat that will be used in the CANDU MOX production line.
- (9) **BOAT TRANSPORT SYSTEM:** The boat transport system is a multi-functional system designed to both transport and store loaded and/or empty sintering boats throughout sintering system. It will receive loaded sintering boats from the pressing and boat loading facility and transport them to and from the lubricant removal, sintering and boat unloading systems. In addition, it will transport empty boats to the boat inspection and cleaning station and provide surge storage of both full and empty boats. Equipment for this purpose currently exists on the 70 foot level of the FMEF. However, it was designed for lower loads than will be encountered for the CANDU MOX mission. This will necessitate upgrading portions of the drive systems and supports at an estimated cost of \$50K.
- (10) **PELLET GRINDING:** Sintered pellets must be ground on their cylindrical surface to bring surface finish and diameter to within specification tolerances. Sintered pellets will be received from boat unloading and centerless ground to CANDU MOX specifications. Ground pellets will be transferred to pellet washing and drying. Equipment for this

purpose currently exists on the 70 foot level of the FMEF. However, it will have to be replaced by a unit with larger capacity for both material removal and throughput.

- (11) **PELLET WASHING AND DRYING:** Ground pellets will be washed to remove residual material from the grinding operation and then dried to remove moisture. Ground pellets will have their diameter measured to provide process feedback control. Washed and dried pellets will be transferred to the pellet stacking station. No equipment currently exists for this operation in the FMEF. A new unit, designed for remote operation, will be provided.
- (12) **PELLET STACKING:** Pellets received from the pellet washing and drying operation must be assembled into a columnar stack for loading into the fuel sheath. Stack length adjustment will be made by adding pellets at each end of the stack of an appropriate length to bring the total stack length into specification requirements. These end pellets will be made from natural UO_2 and supplied from an external source. An inventory of end pellets in a number of different lengths will be stored at this station until needed for stack length adjustment. Completed pellet stacks will be transferred to the loading station. Equipment for this purpose currently exists on the 70 foot level of the FMEF. However, it will have to be replaced by a unit designed for the pellet size and stack length of CANDU MOX fuel.
- (13) **PELLET STACK LOADING:** The pellet stack loading facility will receive prepared fuel stacks from the stacking station and empty fuel sheaths, with one end cap attached, supplied from an external source. The pellet stacks will be loaded into the fuel sheath and the assembly transferred to the end cap welding station. The fuel sheath will be weighed before and after stack loading to provide material accountability information. Equipment for this purpose currently exists on the 70 foot level of the FMEF. However, it will have to be replaced by a unit designed for the pellet size and element length of CANDU MOX fuel.
- (14) **END CAP WELDING:** The end cap welding facility will receive fuel sheaths containing the pellet stack and weld a cap to the open end of the fuel sheath to encapsulate the fuel element. This facility will also perform a minor machining operation to remove excess metal resulting from the resistance welding process. The completed weld will be tested for soundness using a nondestructive method and element integrity will be verified using a helium leak testing technique. After verification that the elements are free of surface contamination, they will be removed from the glovebox system. Completed fuel elements will be subjected to SNM assay and loaded into magazines for transfer to fuel bundle assembly. Equipment for this purpose currently exists on the 70 foot level of the FMEF. However, it will have to be replaced by a unit designed for the size of fuel element used in CANDU MOX fuel.

COST ESTIMATE FOR RING 3 (2% PU) LINE: We estimate the cost of new equipment and modifications to existing equipment at \$7,500,000; the cost of new glove boxes at \$1,500,000, and the cost of installation and hook-up to existing services at \$1,500,000, leading to a total cost for this pin line of \$10,500,000.

4.6.2.4 Ring 4 (1.2% PuO₂) Pellet and Element Line

This line will provide processes to receive batches of ceramic grade mixed oxide powder from the master blend facility on the 21 foot level, process this material into sintered pellets containing 1.2% PuO₂ and encapsulate them into fuel elements meeting all requirements of CANDU MOX fuel. This line will operate for the most part by remote control with all process equipment contained in gloveboxes. Completed fuel elements will be transferred to the CANDU fuel bundle assembly line on the 0 foot level in the FMEF.

This line will be located on the 42 foot level in the FMEF. Since there is no existing equipment at FMEF for this line, all new equipment will be obtained for this line. As much as possible, the equipment for this line will be the same as that used on the 2% PuO₂ line to allow interchangeability and reduction in the amount of spare parts that will need to be stocked.

This production line will consist of 14 subsystems configured as shown on Figure 4-12. (All equipment will be designed to be operated remotely with monitoring by sensors built into the operating equipment. The equipment will be designed for contact maintenance consistent with personnel exposure and safety requirements.) The 14 subsystems are the same as in the Ring 3 line at the 70 foot level (Section 4.6.2-3), with some exceptions. For example, due to space constraints, the boat transfer system will be more serial than in the 2% line, with less provision for surge storage.

COST ESTIMATE FOR RING 4 (1.2% PU) PIN LINE. We estimate the cost for new equipment at \$9,464,000, for new glove boxes at \$1,786,000, and for installation and hook-up to existing services at \$1,893,000 for a total cost for the ring 4 pin line of \$13,143,000.

4.6.2.5 Bundle Assembly Line

A CANDU bundle assembly line will be located on the 0 foot elevation of the FMEF. Figure 4.13 shows the layout at this floor. It will receive completed fuel elements from the production lines on the 42 and 70 foot levels. In addition, completed elements containing burnable poison (dysprosium) will be received from a qualified external source. Assembly of the elements into CANDU fuel bundles and inspection of the completed bundles will be carried out in a semi-automated line operated predominantly by remote control with all operations performed in shielded workstations. This production line will consist of four subsystems which are described in the following sections:

- (1) **BUNDLE ASSEMBLY:** The bundle assembly facility will receive completed fuel elements from the assembly lines on the 70 and 42 foot levels and burnable poison elements from an external source. These elements will be assembled into CANDU bundle form using custom designed fixturing which will assure proper loading and orientation of the elements based on the location of the appendages which are uniquely different by Ring. Assembly will be followed by welding each end of each element to Zircaloy end plates supplied from an external source. Completed fuel bundles will be transferred to the bundle final inspection facility.

This facility will be designed to be operated remotely with monitoring by sensors built into the operating equipment. The equipment will be designed for contact maintenance

Figure 4.12 FMEF Facility Layout at 42 Foot Level
(Source: Westinghouse Hanford Company)

Figure 4.13 FMEF Facility Layout at Ground Level
(Source: Westinghouse Hanford Company)

(ADDITIONAL INFORMATION)

consistent with personnel exposure and safety requirements. This will be a new facility using proven technology currently used in CANDU fuel manufacture.

- (2) BUNDLE FINAL INSPECTION: A number of nondestructive tests and inspections are required to certify a completed CANDU fuel bundle meets all specification requirements. The equipment to perform the required functions will be shielded to provide operator protection from radiation exposure and predominately remotely operated. Functions which require manual intervention will be performed by the operating technician using manipulators and other assisting devices. Bundles successfully passing all tests will be transferred to the bundle packing and storage facility. Those failing to pass a test will be transferred to the bundle repair station for processing.

This will be a new facility using proven technology currently used in CANDU fuel manufacture. The equipment will be designed for contact maintenance consistent with personnel exposure and safety requirements.

- (3) BUNDLE PACKING AND STORAGE: Bundles received from final bundle inspection will be packed for shipment and stored until their scheduled shipping date arrives.

The packaging station will be a new station designed to be operated remotely with monitoring by sensors built into the operating equipment. The equipment will be designed for contact maintenance consistent with personnel exposure and safety requirements.

Two forms of storage will be used at FMEF, storage of up to 3000 bundles in the existing FFTF tile holes, and storage of up to 750 bundles in the shipping drums, to be located in a separate area in the FAA area of FMEF. In both cases, the bundle will be surrounded by a cylindrical protective device designed to reduce radiation levels below 5 mr/hr, and to provide protection against physical hazards.

- (4) BUNDLE REPAIR STATION: Bundles rejected at final inspection will be disassembled in this station. Fuel elements that are acceptable for rebuild into bundles will be prepared and transferred to the bundle assembly station. Elements not acceptable for rebuild into a bundle will be transferred to scrap recovery.

This will be a new facility with equipment designed to be operated predominantly remotely but with some functions requiring intervention by the operating technician using manipulators and other assisting devices. The equipment will be designed for contact maintenance consistent with personnel exposure and safety requirements.

COST ESTIMATE FOR BUNDLE ASSEMBLY LINE: We estimate the cost of new equipment for the bundle assembly line at \$2,179,000, for installation and hook-up to existing services at \$429,000, and for shielded work stations at \$786,000 for a total cost of \$3,393,000.

4.6.2.6 Facility Modifications

In addition to processing equipment described above, some facility equipment previously installed in the FMEF will require upgrading to support the CANDU MOX mission. These items are described below.

- (1) **SAF LINE INSTRUMENTATION:** The original SAF instrumentation and control system consisted of a data center linked to supervisory process Remote Control Racks (RCRs). Process control of each subsystem occurs at the RCR. The SAF Line RCR's are generally located on the Level 82'- 0" mezzanine with central data acquisition in the data center computer room. The data center was to receive fuel fabrication process and certification data from the process RCR's. This information was to be recorded and subsequently used to generate production and/or traceability reports on a data center computer. Displays of process and material flow data at the RCRs was to be provided for operators to coordinate SAF Line activities including material transfers between subsystems and chemistry sampling. Operators could communicate with each other through a digital data network and/or an intercom system. In addition, a digital data communication link between the data center and the safeguards computer system was to be provided to report SNM movements within the SAF Line.

The SAF Line Remote Control Racks (RCRs) have been installed and some functional testing was completed for most subsystem. The Input/Output (I/O) interface to the subsystem equipment is provided by Computer Automated Measurement and Control (CAMAC) racks containing a variety of different I/O cards (depending upon the type of signal to be handled). The CAMAC racks are controlled by PDP-11/73 computers which also provide the interface to the data center. The original Data Center computers were Digital Equipment Co. (DEC) model VAX 1170. In the original configuration, two computers were connected as a redundant pair with the third as a watchdog.

The Data Center computers were removed and excessed since they were not in use and had become obsolete. In order to establish the approximate cost of their replacement, representatives for both IBM and DEC were contacted and questioned relative to the original specifications. The original configuration, while popular in the mid 1980s, is no longer used. It is unlikely that a source could be found capable of providing a direct replacement for the original equipment. None of the applications software for the Data Center computers had been completed at the time the SAF Line project was canceled.

In lieu of this out-of-date technology, the fuel supply team recommends that a modern local area network be considered with four work-station type computers used as the central server. The costs for such a system are expected to be \$150K for the central servers (including peripherals) plus an additional \$90K to \$125K for the software (not including applications software development). Applications software, based on the use of the newer software tools, will require an additional \$150K to \$200K to complete.

Since the existing RCR computers are also becoming obsolete, it is recommended (but not required) that they be replaced with current industrial PC's. This would cost about \$120K. It was also suggested that the existing CAMAC crates and the RCR computers

both be replaced by new programmable controllers, again due to their age. This would cost an additional \$180 K.

In summary, we estimate about \$800,000 will be required to modernize the process instrumentation systems.

- (2) **REMOVE PORTIONS OF THE SAF LINE:** In order to utilize the existing SAF Line for the new proposed mission, it will be necessary to remove significant portions of the existing SAF Line. A cost estimate was prepared to establish the scope and cost of such an undertaking including both engineering and labor. Removal of the following existing equipment was included within the scope of the estimate:

- binder addition glovebox and equipment along with electrical wiring, piping and local controls.
- Compaction and granulation glovebox and equipment along with electrical wiring, piping and local controls.
- Lube addition glovebox and equipment along with electrical wiring, piping and local controls.
- Boat unloading exit interface conveyer along with electrical wiring and local controls.
- Pellet storage glovebox and equipment along with electrical wiring, piping and local controls.
- Pellet surface inspection glovebox and equipment along with electrical wiring, piping and local controls.
- Property adjustment glovebox, equipment and interface conveyer along with electrical wiring, piping and local controls.
- Pellet grinding glovebox and dust removal system (Aplitron filter) along with electrical wiring, piping and local controls.
- Pressing and boat loading glovebox and equipment, hydraulic press units, and interface conveyers along with electrical wiring, piping and local controls.
- Boat inspection glovebox and associated boat conveyers along with electrical wiring, piping and local controls.
- Pin loading gloveboxes and equipment in room 605A along with electrical wiring, piping and local controls.
- The four gloveboxes used for fertile receiving, fissile receiving, batching, and blending/milling, along with equipment, support structure, electrical wiring, piping and local controls.

The costs for removal of this equipment is estimated at \$300K for engineering and \$900K for labor and equipment.

- (3) **PIPING AND WIRING FOR NEW SYSTEMS:** Upon removal of the above systems and equipment and reinstallation of the new systems and equipment for the two pin lines, the master blend station and the bundle assembly station, it will be necessary to reinstall utility and process control piping and wiring. We have estimated a cost of 17% of the actual equipment costs for this piping and wiring activity. This amounts to about \$5,550,000.

- (4) **SAF LINE NITROGEN RECYCLE SUBSYSTEM:** The SAF Line Nitrogen Recycle Subsystem provides high-pressure nitrogen for the jet-mill and valve actuators, and low pressure nitrogen for enclosure inerting. The Nitrogen Recycle subsystem includes two oil-free compressors which take their suction from the discharge of two vacuum pumps and a surge tank composed of four large tanks connected in parallel. The nitrogen gas discharged from the compressors is cooled, dried, and then stored in a high pressure receiver tank. Nitrogen from the receiver tank is reduced in pressure (5 psi) by a regulator and supplied to each glovebox for inerting purposes by a low pressure distribution header. The glovebox inerting nitrogen is introduced to each glovebox through a vacuum breaker and through metering valves with flow meter. The high-pressure nitrogen is distributed by a high pressure header and reduced in pressure at the individual SAF Line subsystems prior to use for valve actuators or the jet-mill. The return nitrogen from the valve actuators, jet-mill and process enclosures is collected and passes through two HEPA filter stages before being returned to the vacuum pumps. The Nitrogen Recycle Subsystem is initially filled from the FMEF Normal Inert Gas Supply and periodically purged to maintain purity within acceptable limits.

At the time the SAF Project was canceled, most equipment associated with the Nitrogen Recycle system was installed. Those items which were not installed included some of the long lead control valves (which had not yet been received at the time the project was canceled) and some of the instruments and controls for which detailed design had just been completed at the time the project was canceled. The items which were received after the project was terminated have since been excessed. In addition, some equipment, such as the dryer and compressors have been removed and redeployed elsewhere on site. A cost estimate was prepared to establish the cost of completing the existing system including engineering, labor and materials. Procurement of the following new equipment was included within the scope of the estimate:

- New control valves and over pressure relief devices.
- Compressors and local control panels (\$140,000.00 plus installation costs).
- A new desiccant dryer.
- Vacuum breakers, metering valves and flow meters at each glovebox along with nitrogen supply piping from the existing header above each glovebox.
- Nitrogen Recycle moisture analysis and oxygen monitoring equipment.
- Instrumentation, fabricate and install the local control board CP-51113 and terminal panels TB-114 and TB-115.
- System Remote Control Rack (RCR) CP-10310.

The second processing line on the 42'6" level will require similar services. In addition it is planned to relocate the jet mill and associated master blending equipment to the 21 foot level. The process lines on the 70 and 42 foot levels will not use jet mills as part of their process. The existing nitrogen recycle system has adequate capacity to operate one jet mill (which is the largest single user of nitrogen) plus the gloveboxes on all three levels of the facility. In order to utilize this system, supply and return headers and their associated valves and controls (at each glove box) will need to be added to the 42 and 21 foot levels.

The cost associated with the complete upgrade of the nitrogen recycle system, including extending it to the 42 and 21 foot levels is estimated as \$300K for engineering and \$2,980K for labor and equipment.

As mentioned above, consideration is being given to replacing the jet mill with an alternate milling system. If the jet mill is deleted from the process, consideration should be given to a once-through nitrogen inerting system to reduce the initial installation costs and complexity of the system. This would also reduce the maintenance requirements for this potentially contaminated equipment and associated person-rem exposures. A cost benefit analysis will be needed since these savings would be offset by an increase in the quantity of nitrogen consumed by the facility.

- (5) **VRC POSITIONING ARM:** At the head end of the powder systems, a "power assist positioning arm" was provided to allow the operators to remove material from the Vertical Reciprocating Conveyor (VRC) and place it in the transport cart without direct contact. The original positioning arm was installed at the time the SAF Line was terminated and has since been excessed. A replacement has been estimated to cost approximately \$20K. A second positioning arm will be required for the new line proposed for the 42'-0" Level.
- (6) **CLOSE COUPLED CHEMISTRY ANALYSIS:** The existing designs for the Close Coupled Chemistry Analysis equipment was reviewed for possible use in support of the proposed new mission. The existing designs include:

• Fuels Assay Subsystem	• O/M Ratio Subsystem
• Offgas Subsystem	• Carbon/Sulfur Subsystem
• Halides/Nitrogen Subsystem	• Moisture Subsystem
• Density Gauging Subsystem	• Manual Sample Sender
• Auto Sample Sender	• Manual Sample Receiver
• Auto Sample Receiver	

Of these, only the Fuels Assay, O/M Ratio, and Density Gauging Subsystems along with the auto and manual sample senders and receivers appear to have applicability to the new proposed mission.

The Fuels Assay equipment designed and built for use in the SAF Line no longer exists. It was originally built and tested in the 325 Building at Hanford. Since it was contaminated, it was later decommissioned as contaminated equipment. It was designed to be installed in a single glovebox along with the equipment for the Moisture subsystem. The assay equipment operated on the principle of background analysis combined with x-ray excitation fluorescence analysis to determine the ratio of Pu/(U + Pu). This information, when combined with O/M ratio obtained by the O/M Ratio subsystem would yield the weight percent of U and Pu. If this piece of equipment were to be replicated, it is estimated that between \$250K and \$300K would be required. A new design based on new assay technology would be appropriate for the proposed mission.

The majority of the sample sending and sample receiving equipment was not permanently installed at the time the SAF Line was terminated. All modules which were not installed have been excessed. At the present time, the auto sample sender located the 70' Level as part of the SAF Line still exists. There is also a manual sample sender located on the 70' Level. An auto sample receiver located on the 21'-3" Level also still exists. The design for these pieces of equipment could be easily modified to accommodate the larger pellets associated with the new proposed mission. Due to the weight of the new pellet design and the small size of the existing pellet system transport tubing, the existing sample transport system could be used to transport one pellet at a time, and perhaps two at a time, but no more. To retrofit the existing design for the sample senders and receivers to accommodate the larger pellets, the design for the pellet feeder modules would have to be modified and new modules built. The design effort would cost about \$8K and each new pellet feeder would cost about \$10K. The original auto-sample receivers cost approximately \$62.5K each. The replacement cost for an entire sample receiver (to replace those that were excessed) is estimated to be approximately \$80K.

The original O/M Ratio subsystem was designed to record the weight of a pellet while it was heated in the presence of a reaction gas designed to oxidize the sample to stoichiometry. The weight gain along with the original sample weight are then used to calculate the initial O/M ratio. The existing design would work for analysis if the larger proposed pellets if a modified pellet feeder design were used. The original equipment has been excessed. To replicate this equipment, approximately \$90K would be required for the equipment (electronic balance, pellet feeder, pellet robot, furnace, pellet elevator, glovebox, etc.) plus \$25K instrumentation and \$25K to assemble and test. When the cost of the auto-sample receiver is also included, this would come to a total of \$220K. There are other technologies which could be applied to determine the O/M Ratio, such as conductivity, which may be simpler and less expensive.

(7) **ADDITIONAL FACILITY ITEMS**

Argon Cryogenic Storage Tank: The concrete pad and buried piping needed to accommodate the argon cryogenic storage tank is in place. It will be necessary to purchase (or lease) a tank, install it, leak check it and place it in service.

Gas Mixers: Gas mixers were purchased for argon/hydrogen mixture (6% to 8% H₂) and for nitrogen/hydrogen mixture (8% H₂) but have never been installed. The building designed for handling flammable gases exists and is located between the concrete pad for the argon cryogenic tank and the existing nitrogen cryogenic tank.

Shielding: At the time the SAF Line was canceled, detail design of the hydrogenous (neutron) shielding was in progress. The design concept was to use encapsulated polyethylene or water walls. The cost of this type of shielding could be very significant. Since the characteristics of the material to be handled for the new proposed mission is totally different, this activity would have to be totally re-evaluated.

Completion of Structural and Criticality Analysis Activities: At the time the SAF Line project was terminated, there were a number of uncompleted activities on the "Open

Items List" with respect to structural analysis. Since that time, those items which affected the building itself were closed out. There are still several items relative to the SAF Line that have not been completed. These activities generally included final as-built reconciliation of enclosures and equipment with the qualification analysis and walkdowns to evaluate and resolve the interaction potential of non-safety Class 1 equipment in the vicinity of equipment which had been seismically qualified to Safety Class 1 requirements. Estimates of the effort involved with these activities were prepared in 1987 and totals approximately 15.5 man-months (\$129K). It should be noted, that with the revised feed materials and mix ratios considered for the proposed new mission, the requirements associated with the safety classification of all equipment in the SAF Line are subject to change. The existing criticality analysis will have to be revised or replaced which may eliminate many of the requirements for safety class equipment. The original Criticality Analysis was issued in draft form (but was never formally released since the SAF Project was terminated). According to the author of the report, its preparation, including analysis, required approximately one man-year.

4.6.3 License, Permit and Regulatory Requirements

This section addresses the major licensing and regulatory requirements that must be met as a prerequisite for conversion of FMEF to fabricate MOX fuel for existing water reactors. It is divided into 4 parts, addressing the NEPA requirements (Federal), State and Local Regulations, Federal NRC Licensing requirements to assure safe operation, and DOE orders that impact leasing and operation of the facility on a DOE site.

4.6.3.1 National Environmental Policy Act (NEPA) Requirements

NEPA review is required for all "...major Federal actions significantly affecting the quality of the human environment..."². Since the lease by the DOE of FMEF to a private venture for purposes of fabricating MOX fuel would be considered a major federal action, the NEPA process will apply. The NEPA review process will be initiated by preparation of an Action Description Memorandum (ADM) that would most likely be followed by an Environmental Assessment (EA). The ADM and EA will take from 12 to 18 months to complete and will cost approximately \$50,000. If the EA results in a "Finding of No Significant Impact" (FONSI), no further NEPA action is required.

It is possible that the Plutonium Disposition Programmatic Environmental Impact Statement (PEIS) being prepared by DOE Headquarters could envelope the major Federal Decision required to lease FMEF for MOX fabrication, and thus serve to meet the NEPA requirements. A legal evaluation of this approach is required to determine if it is acceptable. However, if a separate, site specific Environmental Impact Statement is required, preparation of an EIS is estimated to take 1.5 to 3 years and cost \$1 to 2 million.

²42 United States Congress § 4321 -4347 (1989)

4.6.3.2 State Regulatory Requirements

ENVIRONMENTAL POLICY ACT (SEPA) REQUIREMENTS: A SEPA checklist is prepared concurrently with and accompanies the state air permit application. The cost to prepare a SEPA checklist is approximately \$4,000.

AIR PERMITTING REQUIREMENTS: The proposed activities have the potential of emitting regulated pollutants (both radionuclides and non-radionuclides). As such, the potential emissions must be calculated using a method approved by the (State of Washington) regulators.

Once the emissions are estimated, air permits and notifications will be required. These may include National Emissions Standards for Hazardous Air Pollutants (NESHAPS), Radioactive Air Emissions Program (RAEP), Toxic Air Pollutants (TAPs), and possibly Prevention of Significant Deterioration (PSD), depending on the details of the processes to be used and their expected emissions. In addition, the Air Operating Permit (required under Title V of the Clean Air Act) will need to be completed. Registration of any new stacks, with the Department of Health, will also be required. It is estimated that it will take from 18 months to 3 years to complete air permitting, at a cost of \$350,000.

WATER REQUIREMENTS: It is believed that industrial waste water (i.e., liquid effluents) from the production operation will not be discharged to the ground, surface waters, or into municipal sewage systems. Instead, it is projected that waste water (from the pellet grinding and washing operations) will be evaporated and discharged through a stack. Therefore, additional discharge permitting requirements including the National Pollutant Discharge Elimination System, will not apply.

The existing 400 Area sanitary sewer can be utilized for sanitary sewage from FMEF. Modifications are underway to bring the existing sanitary sewer up to current requirements. If any liquid effluents are to be discharged, additional requirements may be applicable. Based on the current information, there should not be any liquid discharges so no additional cost, or impact on the schedule is projected.

HAZARDOUS WASTE REQUIREMENTS: Source, special nuclear, and by-product material are excluded from the Resource Conservation and Recovery Act (RCRA) definition of solid waste. Based upon the information obtained to date, it does not appear as though the feed material to this process will contain non-radioactive components that will require treatment or reclamation prior to processing. Additionally, it is DOE Richland's position that special nuclear materials are defined as scrap, rather than waste, until the Department specifically declares such materials to be waste. Therefore, there do not appear to be RCRA or hazardous waste impacts to this project.

State of Washington regulations contained in WAC 173-303 may apply to this project under limited circumstances. Both a Part A and Part B permit application would be required if waste, special nuclear, or source material containing RCRA-regulated non-radiological components is reclaimed or treated prior to processing such material. Additionally, any hazardous waste generated at this facility will require management in an accumulation area or in accordance with

the 90-day accumulation provisions of WAC 173-303-200. If RCRA waste materials are to be excluded, no costs will be accrued for preparation of Part A and Part B permit applications.

CULTURAL RESOURCES: The FMEF Cultural Resource Review may need further study. If required, this review takes from 3 months to 12 months, and costs \$3,000 to \$5,000.

ENDANGERED SPECIES APPROVAL: A site assessment should be made to determine whether the proposed new activities will disturb critical habitat. No impact on cost or schedule is assumed as this is performed, on a site wide basis by DOE.

EXCAVATION PERMIT: For this proposed project, excavation should not be necessary. If excavation is needed, an excavation permit will be required. This permit usually takes about 2 weeks to obtain and costs about \$2,000.

PERMITTING PLANS: A Permitting Plan would also be required for this project. Development of this plan would take about 3 weeks and cost about \$5,000. The Permitting Plan would coordinate and initiate the permit applications and NEPA documentation, and would provide more detailed schedules and costs for permitting the new process of FMEF.

SUMMARY OF STATE PERMITTING: At this time, and with the information provided in the current work scope, permitting for the FMEF proposed project is estimated to take about 3 years to complete, and cost \$400,000.

4.6.3.3 Nuclear Regulatory Commission Licensing

Normally, DOE facilities designed to process defense program materials are not licensed by the NRC and instead are subject to a series of DOE orders and oversight by the Defense Nuclear Facility Safety Board. However, because this application of FMEF would produce nuclear fuel for civilian use, the Government may decide that NRC licensing is more appropriate. This study assumes that a facility operating license will be required from the NRC. While it is expected that public health and safety would be protected adequately by either of the two regulatory alternatives (NRC versus DNFSB), the public will have more confidence in the regulatory review provided by NRC, since it has a history of close oversight and licensing at such facilities as the Naval Fuel Fabrication facility. Many facets of NRC's functions would therefore apply to a MOX facility at FMEF. In the final analysis, it will be necessary for DOE, to decide which licensing approach should be taken.

The first steps required to secure an NRC license include preparation of a detailed Safety Analysis Report (SAR), prepared in accordance with NRC's Standard Review Plan (SRG) and Standard Format and Content Guide (SF&CG) demonstrating compliance with 10 CFR Part 70. We estimate that this will require 12 months and about \$2 million. Much of the work performed by WHCO in preparing the previous FMEF/SAF SAR would be relevant.

Once the SAR is submitted to NRC, the staff reviews the analysis, interacts with the applicant where questions arise, and prepares a Safety Evaluation Report. Under current NRC regulations, the cost of this review is to be recovered from the applicant. We estimate this

review will require up to 15 months, and will require about \$5 million of NRC effort, along with an additional \$3 million of applicant effort.

Once the SER is issued, it is subjected to public review, and hearings if requested, following which the Commission may issue an operating license. The additional time for this review is estimated at 9 months.

Thus, the study team estimates that it will require 3 years, from initiation of SAR preparation to license issuance, for an NRC operating license, with a total cost of \$ 10 million. This amount has been included in our capital cost estimate. The estimate is uncertain at this time. If the project proceeds, we strongly recommend a close interaction with the Nuclear Material Safeguards and Security (NMSS) group at NRC to fashion a more definitive estimate of time and cost.

In addition to these up front licensing costs, an allowance of \$2 million/year is also made for NRC inspection and enforcement activities at the facility.

In this regard, we note that NMSS has recently begun a complete revision to their rules and regulations for licensing nuclear Materials facilities, and expects the new rules to be published by August, 1995. The new rules are intended to be more efficient and up to date, and may in fact facilitate the licensing process. NRC SECY-94-057 dated March 4, 1994, and SECY-94-057A dated April 13, 1994 set forth the schedule for these rules and RegGuide revisions. It also states that the new par 70 will take a performance-oriented, systems approach to regulation, with requirements graded according to risk, rather than the narrowly prescriptive, compartmentalized approach that comprises the present requirements." Although this rule is predicted by NRC to add six months to the schedule, it appears that the resulting regulatory regime will afford an improved basis for licensing of FMEF, and the new schedule appears consistent with the needs for the plutonium disposition program. We suggest that DOE inform NRC that DOE will be depending on the formal publication of the new rules and the associated regulatory guidance, on the schedule defined in SECY-94-057A, and request Commission support in meeting this new schedule.

4.6.3.4 DOE Orders and Standards

The above sections assume that the FMEF will be licensed by the Nuclear Regulatory Commission for operation as a MOX fabrication facility. If instead, the decision is made to operate the facility under DOE control, then specific DOE orders will apply, and the operation would be under the oversight of the Nuclear Facility Safety Board. Some of the specific DOE orders and regulations which would apply include the following:

PRE-OPERATIONAL MONITORING: Pre-operational monitoring of facilities, sites, and operations is required by DOE Order 5400.1. An environmental study shall be conducted prior to startup of a facility, site, or process which has the potential for significant adverse environmental impact.

RADIATION PROTECTION STANDARDS: DOE Order 5400.5 establishes standards and requirements which must be followed with respect to protection of members of the public and environment against undue risk from radiation. The general environmental protection program

requirements are established in DOE Order 5400.1.

4.6.3.5 National Emissions Standards for Hazardous Air Pollutants

Requirements for continuous measurement of effluent from the facility are specified in 40 CFR Part 61, National Emissions Standards for Hazardous Air Pollutants (NESHAPs), Subpart H, "National Emission Standards for Emissions of Radionuclides Other Than Radon From Department of Energy Facilities". The requirements directly address those radionuclide emission points at DOE Facilities which, according to the methods specified at 40 CFR 61.93(b)(4), are estimated to provide a potential exposure of greater than 0.1 millirem per year effective dose equivalent to any member of the public. The emission potential of the facility, if used for the proposed project, will exceed the specified 0.1 millirem per year effective dose equivalent. The original effluent monitoring system for the FMEF facility exhaust stack was designed to the requirements of ANSI N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities" and has not received the necessary "point by point" comparison needed to establish compliance with the requirements of 40 CFR Part 61. Other "significant" effluent stacks on the Hanford site have been upgraded by installation of a pre-qualified package at a cost of \$246K each (for the equipment).

4.6.4 Safeguards

Safeguarding and securing plutonium materials during storage and processing in the CANDU MOX facility will be a major consideration. Fortunately, these considerations were taken into account in the original design of the FMEF and systems were designed and developed for FMEF to perform these functions. However, these systems were originally designed for use in a DOE owned and operated facility. It is anticipated that the CANDU MOX facility would be operated under full international safeguards by the IAEA. The following sections provide an overview of the systems originally planned for these purposes and outline the steps that are anticipated to bring these systems on line and or upgrade them to current standards.

EXISTING FMEF ACCOUNTABILITY SYSTEM: A accountability computer system (called SARIS) was originally provided to administer all SNM transfers within the FMEF. This system is no longer in use. The system was designed to provide near real time records of the location and quantity of all SNM in the facility. Movements of SNM were to be traced by accumulating both automatic and manual data entries from various facility locations. SARIS was to maintain a history of transactions including a holdup account for all loose SNM for each hood, glovebox, enclosure or storage location. Periodic NDA and/or process operation calculations were to be used to verify the accuracy of holdup estimates. In addition to performing the accountability function, the accountability computer was also intended to be responsible for criticality control by limiting the amount of material outside of fixed storage locations. Secondary functions for this system include providing supervisory control of the automated SNM Storage System and interfacing with the Security System (Including the Vertical Reciprocating Conveyor) to protect fuel materials from diversion.

To accomplish these objectives, near-real-time data was to be used to reflect chemical and isotopic material changes due to processing. Information obtained from the process would be utilized to establish the composition of SNM, by isotope and the net weight of each batch, item or container. As material moved through the process, it would be mixed with other materials

(dry scrap, MOX powder, organic binder, lubricant, etc.) and changed by processes such as sintering, all of which change the composition and weight of the material. The system was designed to model these changes and track the quantity of each accountable isotope based on measured parameters (usually weight and an identification number).

The original SARIS computer system was installed over 10 years ago and was recently declared excess. The computers are currently being removed from the facility. The computers will have to be replaced with modern equipment, and the application software re-written utilizing current software products following the basic principles of the original SARIS application software. The logic imbedded in the SARIS application software relied heavily on the use of weight and container or batch identification. If additional sampling is required to satisfy IAEA requirements, provisions to incorporate such measurements would have to be included in the new software.

SUMMARY OF IAEA REQUIREMENTS: Development of an implementation scheme for safeguarding of a facility under IAEA oversight is an iterative process which involves negotiations among the facility operators, the U.S. Government and the IAEA. The following instruments govern these negotiations and the Safeguards implementation at IAEA monitored facilities:

INFCIRC/66 and INFCIRC/153
Safeguards Agreements
Subsidiary Arrangements
Facility Attachment

For facilities located in the US, additional implementing requirements apply to commercial facilities (10 CFR 75) and to DOE facilities (DOE Order 1270.2B). The IAEA inspection process will be established by the following steps:

<u>ACTIVITY/MILESTONE</u>	<u>RESPONSIBILITY</u>
1) Consultations	US Govt./IAEA
2) Offer Facility	US Govt.
3) Select Facility	IAEA
4) Facility Preparations	Facility/US Govt.
5) Initial Inventory Declaration	US Govt. (30 days after selection)
6) Prepare Design Information Questionnaire (DIQ)	Facility (45 days after selection)
7) Establish Safeguards Approach	IAEA
8) Design Information Verification	Facility/IAEA
9) Ad Hoc Inspection (Initial Inventory Verification)	Facility/IAEA
10) Facility Attachment	US Govt./IAEA
11) Install Safeguards Equipment	Facility/IAEA
12) Begin Routine Inspections	Facility/IAEA
13) Special Inspections (optional)	Facility/US Govt./IAEA
14) Reporting	Facility/USGovt./IAEA

In the case of FMEF, DOE will forward a proposal to the Senate to include the FMEF on the IAEA list. The Senate would have 60 days to act on the proposal. If ratified, the initial inventory declaration including the forms, quantities and locations of SNM located (or to be located) within the facility must be offered up within 30 days. The facility would prepare the Design Information Questionnaire (DIQ) to be submitted within 45 days of Senate ratification. The DIQ includes information such as:

- description of the facility,
- overall process parameters,
- Nuclear Materials flow,
- protection and security measures,
- proposed nuclear material accountancy, and
- key measurement points (KMPs).

The IAEA would utilize the DIQ to develop a draft Safeguards Approach. The Safeguards Approach prepared by the IAEA would include proposals for the frequency of inspection, verification measurements (both nondestructive and destructive), containment/surveillance measures (C/S) and records and reports. The Design Information Verification is performed by IAEA with the assistance of the facility operation and includes tours, consultations and review of the facility drawings. In the case of a new process line, as is the case for FMEF, it is expected that the Design Information Verification phase would be spread out over a longer period of time with the information provided to the IAEA as it is developed. The US Government (State Department) and the IAEA negotiate to agree on the Facility Attachment. The Facility Attachment specifies the provisions necessary to implement the safeguards approach at the facility. It includes a short facility description, change provisions which are required to accommodate the IAEA Safeguards (if any), MC&A and C/S description, Records and Reports system, administrative procedures, NM termination/exemptions and summarizes routine inspections.

To implement the Safeguards Approach agreed to in the Facility Attachment, the IAEA and facility will install safeguards equipment as necessary including any additional Nondestructive Assay (NDA) equipment, facilities for NDA calibration and equipment for Containment/Surveillance (C/S). The IAEA may insist on using their own equipment for measuring and sampling associated with KMPs (Key Measurement Points) or they may allow "joint use" of facility equipment with some control over certification.

There are three types of IAEA inspections. The first is the Ad Hoc inspection whose goal is to independently verify the declaration for initial SNM inventory, changes in inventory or international shipments/receipts. These inspections are done prior to completion of the Facility Attachment. After completion of the Facility Attachment, routine inspections are performed to independently verify agreement between US reports and Facility Records, operator SNM declarations (including form, quantity and composition), discrepancies/uncertainties in facility MC&A system and continuity of knowledge (C/S system). Special inspections may be performed to verify special reports (facility report of NM loss, C/S failure, etc.) or investigate unusual situations. The IAEA Safeguards measures include examination of records and reports, item counting and identification, nondestructive assay (NDA), Destructive Analysis (DA), Containment/Surveillance (C/S) and authentication. IAEA routinely takes process samples to verify declared materials and operations. These samples are sent to IAEA's Safeguards

Analytical Laboratory at Seiborsdorf, Austria, where they are either analyzed or sent on to IAEA's network of analytical laboratories. The Containment/Surveillance program is used for the Safeguarding of "items" such as stored containers, pins or bundles. The objective is to maintain continuity of knowledge during inspector absence using techniques such as high reliability cameras and seals.

Of the four MOX fuel fabrication plants under IAEA safeguards, only one, the PFPF located in Japan, utilizes Near Real Time Accountancy". In addition, the Siemens MOX II facility (not yet in operation) was designed to use near real time accountancy and is the only one of the five with a throughput capacity comparable to that of the proposed CANDU MOX facility. The majority of materials are either not contained, or are not static; therefore a Containment/Surveillance (C/S) system approach to safeguards may not be practical. Based on the required throughput and required availability, a near real time approach may be preferred. It is also possible that IAEA inspectors would have to be located in the facility on a continuous basis. The KMP's which utilize instrumentation may have to be located at various points other than the feed material receiving and fuel assembly bundle shipping locations. For the purpose of this study, it is assumed that PuO_2 feed material will not be under IAEA safeguards until it is received at the FMEF and will require assay at the point of receipt. It is also assumed that finished fuel bundles will require NDA before they are shipped from the facility and that completed elements will require NDA before they are sent from the Element lines to Bundle Assembly. It is assumed that other data requirements are accommodated using NDA and DA sampling of powder and pellets taken from the process.

It is anticipated that space will need to be provided for the following:

- 1) Space for an NDA lab for IAEA. It is possible that this could be a "Joint Use" facility, but for the purpose of the study, it is assumed that it will not be.
- 2) Office space for IAEA personnel.
- 3) Computer room for IAEA data collection and report generation.
- 4) Handling and shipping facility for shipping samples to Austria.

ACCOUNTABILITY DATA COLLECTION SYSTEM COSTS: Safeguards data collection for the facility will originate from either the process control system (the SAF Data Center computers) or from manual stations located at various point in the building. Data which originates in the facility will be collected and maintained by the FMEF safeguards computer system. At the present time, all SNM data collected on the Hanford site is maintained by the Nuclear Materials Safeguards System which would receive data from the FMEF Facility Safeguards system. The Nuclear Materials Safeguards System in turn transmits the information to the Nuclear Materials Management and Safeguards System which maintains records for the entire country. At the present time, Nuclear Materials Management and Safeguards System is being revised which may necessitate changes to the Nuclear Materials Safeguards System.

The Data Center computers will have to be replaced, along with software development for the appropriate application software. Replacement of the FMEF safeguards computer system is estimated to be approximately \$60K for the central server and approximately \$10K per station for a total of \$100K. In addition, conversion and upgrade of the applications software is expected to cost an additional \$275K.

The costs of bundle assay equipment will be about \$400K each (it is likely that at least two will be required). In addition, assay equipment will be required for feed materials, process samples, etc. Based on the costs reported for the PFPF located in Japan, a total programmatic cost of about \$10,000K should be expected. For purposes of this study, it is estimated that 80% of these costs will be incurred during the facility planning, design and licensing phases, with 20% during procurement and installation phases.

4.6.5 Physical Security

EXISTING SYSTEMS: The existing FMEF Security System was based on a layered approach to establish "security in depth". Physical barriers and intrusion detection devices were used to support this philosophy based on possible overt actions that might be taken to acquire special nuclear material (SNM). The system was computer based with closed circuit television (CCTV) assessment capabilities.

The FMEF was designed with security in mind: it has no windows, it's walls are two (2) foot thick concrete, and redundant power is provided. The interior of the building can be easily subdivided into security areas.

The FMEF Security System was electronically integrated into the 400 Area Security System. All alarm notification, CCTV assessments, and subsequent guard response deployment were to have been performed by Hanford Patrol personnel located within the hardened facility known as the 400 Area Central Alarm Station (CAS). The CAS (Building 4790 and no longer in service) is located approximately 200 feet southeast of the FMEF perimeter fence.

PERIMETER SECURITY - The FMEF has a 100-foot-wide doubled-fenced isolation zone surrounding the facility which meet the requirements for a protected area. The two 8 foot high chain link security fences have barbed wire outriggers on top. Centrally located within this isolation zone is an Israeli taut-wire intrusion detection fence system with integrated vehicle barrier and a line of microwave motion detectors. The isolation zone is lighted by two offset rows of low pressure sodium lights around the inner and outer fences. Twenty-four fixed lens cameras and two pan-tilt-zoom-focus cameras provide assessment capability of the isolation zone.

At the southeast portion of the FMEF Protected Area is the inactive perimeter guard station (PGS, Building. 4701-C). This guard station was intended to control pedestrian and vehicular traffic into and out of the FMEF Protected Area. The PGS had metal detectors and X-ray units to screen personnel. Adjacent to the PGS is a truck lock equipped with sliding gates, movable crash beam, and microwave and infrared motion detectors. Additionally, the roof of the PGS was protected with microwave and E-field motion detectors. Within the PGS is a hardened area designed to serve as the protected area control facility.

INTERIOR SECURITY - Intrusion detectors installed within the FMEF include infrared detectors, microwave detectors, ultrasonic detectors and balanced magnetic switches. These detectors, along with their associated CCTV systems (for assessment purposes), were designed to have their alarm signals forwarded to the CAS for monitoring, surveillance, and guard deployment. The entry control system allowed only authorized personnel access to the operating areas. During non-operating hours, rooms containing SNM would have been secured and the intrusion

detection equipment activated.

EVALUATION OF EXISTING SECURITY SYSTEM: At the present time, requirements for the security system are found in DOE Orders:

DOE 5632.2A	Physical Protection of Special Nuclear Material and Vital Equipment
DOE 5632.5	Physical Protection of Classified Matter
DOE 5632.7	Protective Forces
DOE 5633.3	Control and Accountability of Nuclear Materials
DOE 6430.1A	United States Department of Energy General Design Criteria

A revised DOE Security Order is soon to be issued (late June or early July '94), which may change these requirements. Therefore, there is some uncertainty regarding the upgrades required. However, based on the current Orders, there are a number of deficiencies that must be addressed to bring the existing system into compliance. These include:

- Install anti-crawl rails at the base of the Israeli taut-wire fence.
- Upgrade the existing Israeli taut-wire fence (wire and switch replacement due to environmental aging).
- relocate 13 microwave motion sensors located in the isolation zone and provide a four foot wide asphalt strip between each set of zone units for improved detection sensitivity.
- Upgrade/repair the movable crash beam in the truck lock adjacent to the PGS.
- Replace the existing truck lock sliding gates with above grade hydraulic units.
- Install access control equipment, personnel and hand carried item surveillance equipment, and a new security control console in the PGS (4701-C).
[All of the original equipment has been removed from the PGS and reassigned or scrapped to allow the building to be utilized for other purposes. The original equipment had never been operated and was obsolete.]
- Reactivate the 400 CAS or route all signals to an alternate CAS, such as the Plutonium Finishing Plant (PFP).
[The 400 CAS was shutdown on May 2, 1994, when the FFTF Protected Area was downsized. Because the equipment is approaching obsolescence, the PFP CAS would be a likely location to upgrade. To handle the FMEF, the PFP CAS would require a major upgrade.]
- Install a new access control system throughout FMEF.
[The existing entry control system, which was never used, is obsolete and never operated reliably. With technology having advanced so rapidly over the past five years, most, if not all of the equipment must be replaced.]

To accommodate the needs of a new mission, a totally new system, using current technology, should be designed and installed. The trend today is to be as automated as possible and not person-power intensive. The existing physical facilities (e.g., existing buildings, fences, conduits, etc.) would be used to reduce the installation cost.

SECURITY UPGRADE COSTS: The costs associated with upgrading/replacing the FMEF

physical security system is estimated to be between \$3,000K and \$4,000K.

The estimated annual operating costs associated with FMEF security related functions are estimated to be between \$9,000K and \$10,000K. This estimate is based on budgeted costs (FY 95 dollars) for the new FFTF Interim Decay Storage Protected Area. The FFTF now has only one Hanford Patrol rover that resides in a limited area island and provides protected area surveillance at an annual cost of \$1,333.2K (programmatic dollars for 12 full time equivalents). With FMEF requiring a perimeter protection area (PA) plus three material access areas, the security staffing is estimated at 70 FTE's at a cost of \$7,777K per year. Additional costs of \$1,032.7K are estimated to cover coordination activities, maintenance, security training, etc. Security Application Center costs adds \$300K to provide the necessary compliance activities such as a yearly vulnerability analysis and Site Safeguards and Security Plan.

4.6.6 Environment, Health and Safety Considerations

Safe and environmentally responsible handling of plutonium materials will dominate the design and operation of the proposed CANDU MOX fuel fabrication facility. The fuel supply team gave priority attention to these topics during its' study. Assessment of more common industrial risks, such as protection of workers from falls, back injuries, cuts, burns and hazards associated with process chemicals and gasses will also be addressed in subsequent project phases.

ENVIRONMENTAL HAZARDS: WASTE STREAMS EMANATING FROM FMEF: The FMEF has an extensive ventilation system, specifically designed to preclude the release of plutonium and other radioactive materials to the atmosphere. Thus, no discharge is anticipated to the atmosphere. The only significant disposal of radioactive material from the facility will be in the form of packages of TRU and/or low level wastes for disposal. (For the purposes of this report, the availability of WIPP for TRU waste disposal is assumed.)

Estimates of TRU and LSA waste generation rates were made by tallying the number of glovebox units to be used in the total facility and comparing the proposed facility to the B&W experience during FFTF fuel manufacture. The projected generation rate for CANDU MOX was adjusted to account for more volume reduction during production than was performed during the FFTF experience and for the reduction in hands-on work in the new system.

It is estimated that the CANDU MOX facility would generate a total of 260 - 390 55-gallon (US) drums of TRU waste per year, consisting of compacted solids and miscellaneous equipment. The LSA waste is estimated to be less than 50 drums per year. Hazardous waste could be another 30 - 50 drums per year, generated in non-radioactive operations. No mixed wastes would be knowingly generated, although a few drums could be generated inadvertently.

For the nominal number of 72 gloveboxes in the CANDU MOX system, this would equate to about 0.1 drums of waste per glovebox per week. The gross volume shipped to disposal will be larger than the drum volume because of overpacks, etc. as required for disposal at WIPP or elsewhere.

MINIMIZING OCCUPATIONAL RADIATION EXPOSURE: Minimization of personnel radiation exposures is of major importance in the establishment of a facility for the production of CANDU MOX fuel. Approaches and considerations which will be followed to assure personnel

exposures are as low as reasonably achievable (ALARA) include the following:

OVERALL FACILITY CONSIDERATIONS - The major process lines (Master Blend Preparation, 1.2 % Pellet and Element Line, 2 % Pellet and Element Line, and the Bundle Assembly Area) will be located on separate floors of the building and in separate rooms, with concrete floors and walls separating major operations. Neutron scattering sources will be isolated laterally and vertically by bulk concrete. Streaming paths will be minimized.

Typical whole body averages of 400 to 450 mRem/year were estimated by calculations based on EPRI NP-404³, modified by the ratios of line throughputs and % Pu.

Data from other plutonium process lines were examined in reference 1, including Rocky Flats, Los Alamos, Belgonucleaire and Exxon (now Siemens). More development data from Westinghouse Hanford Company has also been included.

It has been concluded that operator external or whole body can be kept within 500 mRem/year average if neutron shielding is designed into the equipment and operator in-glove time is closely controlled.

The product of gross tonnage times the plutonium concentration for the two pellet and element lines is within 10% of each other so the doses will be approximately equal for the Ring 3 and Ring 4 lines. For both lines, job rotation will have to be used to assist in dose leveling, but ALARA considerations will result in reducing total radiation exposure as well. Major improvements will be expected in non-scheduled access exposure control, by local shielding, specific job training, and accessory tooling, installed or plug-in spares, and equipment reliability improvements during first 1-2 years operations.

Internal dose will be controlled well below 100 mRem/year by setting tight limits on airborne alpha contamination.

MINIMIZING RADIATION EXPOSURE - POWDER PREPARATION AREA - Operator radiation dose exposure will be an over-riding consideration for engineering the powder preparation system. This system will have a high plutonium inventory and large surface area for dust accumulation. High plutonium inventory correlates to neutron dose, and large surface areas, coated with dust correlate to gamma dose. The existing plutonium canister handling mechanisms will cut operator dose significantly. The gravity flow feed hopper and weigh hopper will do likewise. The gravity transfer to the first stage V-blender and the dust tight connections for charging and discharging will help reduce dusting. The milling system's production performance must be proven before relying on it for hot operations.

With such assurances, operator doses resulting from dismantling, cleaning and filter replacements will be minimized. Otherwise, duct contamination and fissile accumulations will cause major operator and support personnel dose. If a replacement mill is required, most of the personnel dose will come from cleaning of the powder collector and off gas separator filter

³Radioactivity in Mixed Oxide Fabrication Using LWR Recycle Plutonium, Final Report, EPRI NP-404 (1977).

system. Possibly the Apitron filter (electrostatic precipitator plus filters) could be used on the micro-pulverizer exhaust, since its flow rate is lower than for the jet mill. Other types of pulsed backflow, automatic cleaning filters may also be added as polishing units upstream of HEPA filters. In any case, there must be extremely low (few ppm, maximum) solids entrained in the exhaust air to HEPA's, otherwise HEPA replacement rates and SNM losses within acceptable limits. Scrap losses and TRU waste disposal costs will be minimized by assuring that the exhaust gas is stripped of plutonium solids as part of the process.

MINIMIZING RADIATION EXPOSURE - PELLET FABRICATION AREAS - In this part of each line, gamma sources will be attenuated by self-absorption in the depleted uranium, the major component after blending. Shielding will be used locally to reduce extremity dose rate (hands, forearms) for gammas and neutrons. Mechanization/automation will reduce operator time at glove box workstations. CCTV viewing will be used to permit operators to work the process from sufficient distance to allow gammas to be attenuated by distance and neutron direct and scattered dose rates to be shielded out by structural and hydrogen-containing shielding.

The major dose contribution will be during equipment/tool replacement and setup or adjustment, instrument maintenance and setup, radiation control support, non-scheduled outages, and cleanouts.

Dust release will be minimized by process equipment design and local use of down flow ventilation sweeping across equipment such as centerless grinders, pellet cleaning, inspection, and transfer devices.

MINIMIZING RADIATION EXPOSURE - ELEMENT AND BUNDLE ASSEMBLY AREAS - The additional self shielding from bulk material and cladding will further cut gamma dose rates, as the material moves into the final stages of the process. AECL has calculated dose rates in contact with standard CANDU MOX bundles to be 15-16 mRem per hour, with the gamma contribution dominating at 13 mRem/hour. Actual measurements on CANDU test bundles were made in June, 1994. For bundles with 2.3 % PuO₂ on contact the dose was 40 gamma plus 8 neutron totaling 48 mRem/hour. For bundles with 0.5 % PuO₂ on contact the dose was 10 gamma plus 2 neutron totaling 12 mRem/hour. The ²⁴¹Pu content, as measured in 1982, was 24%.

Since the outer, ring 4 elements dominate the gamma dose rate, and the design specifies 1.2% PuO₂ in ring 4, the actual gamma contribution should be about 20 mRem/hour. The neutron contribution is more nearly a weighted average of ring 3 and ring 4 or about 5.2 mRem/hour. Thus, the total should be about 25 mRem/hour on contact. Shielding with a uniform 4 inch panel of water extended polyester (WEP) sandwiched between 1/8 inch sheets of stainless steel would drop the gammas by a factor of 100 and the neutrons by a factor of 5 - 7. In actual practice, streaming through penetrations and scatter or sky shine off roof structural surfaces would reduce the overall attenuation around materials in process, but interim storage in concrete cubicles or other good shielding units could cut the general dose contribution to fractions of a mRem/hour for the operating area.

In summary, the contact dose rates on MOX bundles will be in the range of 15 - 25 mRem/hour. At a distance of 1 foot, (without shielding) levels would be in the range of 3 - 5 mRem/hour gamma plus neutron. With a 4 inch WEP stainless steel shield, dose rates would

be a few tenths of an mRem/hour.

The potential for neutron dose will increase as bulk elements and bundles are stored and fabricated. Neutron shielding and nuclear criticality isolation designs will be developed as part of the final design to restrict total neutron dose to acceptable levels..

The work stations will be arranged so that the number of elements and bundles exposed as sources will be minimized. Others will be transported and held in process within gamma-neutron shielding preventing direct shine to the operators. Streaming through penetrations will be reduced by shadow shields and neutron absorbers. A Shielding Design, ALARA program will be implemented as part of the initial facility design and equipment procurement program:

NEW US REGULATIONS - The new (1994) 10 CFR 20 RADIATION CONTROL Requirements specify formulas for calculating combined dose from external and internal exposure. In the CANDU MOX plant, almost no internal dose is expected. The committed effective dose equivalent (CEDE) is expressed as:

$$CEDE = \sum_{ALI_i} (Intake_i ALI_i) \times 5 \text{ REM} + \text{External Exposure} \leq 2 \text{ REM/YR}$$

where ALI is the annual Limit on Intake as listed in Table 4.8.

A conservative working limit for combined dose will be about 25% of the CEDE or 500 mRem/YR average.

RECENT EXPERIENCE AT LANL- As a point of reference for current practice in DOE plutonium facilities, the following information was provided to B&W in 1992 by the HS-1 Radiation Control Section Leader at the LANL TA-55 area. (The TA-55 area chemically separates Am241 from plutonium, converts plutonium to oxide and metal (via fluoride metallothermic reduction) and recasts it into metal components.)

The average worker exposure recently has been less than 0.5 rem/year. The highest individual exposure was <1.5 rem/year. In previous years, for example 1986, the exposures peaked just below 5 rem/year due to high work loads. External whole body exposure is the main contributor to this exposure. The maximum internal exposure was 200 mRem as measured by bioassay. Fixed and breathing zone air samples have not given reliable internal dose information, so bioassay data are required.

LANL also has cumulative person-rem ALARA goals, which have been steadily reduced.

This experience demonstrates that is it possible to control operator average dose under 500 mRem/year, even in a largely manual glove box arrangement, with glove boxes sometimes surrounding the operators who are in aisles between the box lines.

In the SAF Line, where single lines of gloveboxes extend down the length of the room, and rooms are shielded from each other by structural concrete, shielding will be even more

effective.

The average airborne activity will have to be kept below $6 \times 10^{-14} \mu\text{Ci/ml}$ to stay below 100 mRem/year internal dose. In practice, control limits would be set at a fraction of this level. Routine use of respirators is not normally allowed as a means of meeting airborne activity criteria. Supplied air masks and other respiratory protection equipment is used for maintenance work as a precaution.

FUTURE TRENDS - In order to allow a margin for internal dose, and its measurement error, it would be preferable to set the maximum external dose limit at 375 mRem/YR. Over the first 5-10 years of the CANDU MOX line operation, it is reasonable to expect that this will evolve as the working limit.

Regardless of whether 500 or 375 mRem/YR is used, the limits are quite restrictive for whole body dose. The extremity dose will also be restrictive for cleanup and maintenance. A working limit of 1 to 2.5 REM/YR will apply to extremities.

TABLE 4-8

**Table of Annual Limits on Intake (ALI) and Derived Air Concentrations (DAC) for Plutonium Operations
(limits are effective January, 1994 - from 10CFR20)**

Isotope	ALI (μCi)	DAC ($\mu\text{Ci/ml}$)
Pu- ²³⁸	0.007	3×10^{-12}
Pu- ²³⁹	0.006	3×10^{-12}
Pu- ²⁴⁰	0.006	3×10^{-12}
Pu- ²⁴¹	0.3	1×10^{-10}
Pu- ²⁴²	0.007	3×10^{-12}
Am- ²⁴¹	0.006	3×10^{-12}

NOTE: ALI's are based on the most restrictive solubility class.
DAC's are the airborne concentration, which if inhaled for a year, would lead to an intake of one ALI.

4.6.7 MOX Recycle

NORMAL PROCESS GENERATED SCRAP: The proposed fuel fabrication facility will include a recycle system for recovery of non-conforming product and process wastes. A preliminary flow sheet for this system is shown on figure 4.14. The recycle system will recondition collected oxide materials from master blending and fuel fabrication lines and return them to the powder processing areas for blending with virgin materials. Although a nominal recycle rate of 13 tonnes per year is allowed in the sizing of main process equipment, (design throughput rate of 170 MT/yr versus 157 MT/yr required for reference case), the scrap recycle system will be designed to recover even more material. Specifically, it has been sized to recover MOX in the following forms and amounts:

Figure 4.14 Process Flow Diagram - Scrap Recycle Station

Transfer Mechanism Code

C	Conveyor	V	Vibratory
G	Gravity	CAN	Canister
P	Pneumatic	MN	Manual
R	Robot		

PROCESSED MATERIAL																	
Line Number	1	2	3	4	5		7	8	9	10	11	12	13	14	15	16	17
Material Form	Pellets	Pellets	Granule	Pellets	Pellets		Powder	Powder	Granule	Powder	Soln	Cake	Granule	Powder	Powder	Powder	Powder
Transfer Mechanism	Can	G	C	Can	G		Can	G	C	C	Can	R	C	C	G	TBD	Can
Qty Transferred (Kg)																	
Transfer Time (Minutes)																	
Number of Shifts																	
Flowrate (Kg/Day)	25.5	25.5	25.5	17	17		60	60	1	59	7.5	7.5	0.5	7	44	110	110

<u>Recycle Source</u>	<u>Recycle Rate</u> (Tonnes MOX/year)
Hard Sintered Pellets	5
Dry Powder from Grinding	12
Green Pellets	3.5
Wet Powder from Pellet And Filter Washing	1.5
TOTAL	<u>22 MT/yr</u>

This provides recycle system capabilities to handle peak loads and at the same time facilitates use of the recycle system for additional missions as discussed below.

Consistent with current CANDU fuel fabrication practices, green pellets will be pulverized and treated as virgin material to be recycled to the blending areas. Hard pellets and grinding residue will be reconditioned and fed into master blending. About 1 tonne/yr of dry filter cake is also anticipated (from the pressing and blending areas) and will be treated as virgin material. The scrap recycle system should enable the facility to utilize virtually all of the original material inputs. (Some slight losses, from filters, duct holdups, residues and wipes will occur.) All recycle materials will be final blended to ensure homogeneity prior to packaging and can transfer. All operations will be remotely controlled and observed over close-circuit television. All material transfers will be performed remotely. For the purpose of this study, remote automatic furnace loading and unloading is assumed in conjunction with conveyor type transfer mechanisms. Optimization of the materials handling processes will be performed in subsequent design phase of the program. Material weights will be taken before and after for accountability. Scrap reintroduction is consistent with current assumptions for lot handling and assaying requirements.

Reject sintered pellets will be recovered as hard scrap. Sintered pellets will be transferred by canisters and weighed before recycle processing is initiated. This material will then be transferred to a dedicated weigh storage bin from which it will be fed to stationary oxidation furnace(s) for 'softening'. Softening of the sintered materials through a long (i.e., 18 - 20 hour) oxidation cycle produces a finely divided powder which is readily pulverized without concern for contamination by the pulverizer impactors. This process also improves the powder properties. Oxidation cycles will be performed in air atmospheres at approximately 850 °F. Following oxidation, this material will be combined with soft scrap material and run through a pulverizer where it will be milled into a fine power (nominally 4 micron). Commercially available bantan mills will be used. The fine powder will drop by gravity to the recycle blender.

Grinder scrap will be recovered, as will powder that has been collected in vacuum systems in the main fuel fabrication lines. Powder will be transferred by canisters and weighed before recycle processing is initiated. This material will then be transferred to a dedicated weigh storage bin from which it will be fed to a 170 mesh screen for removal of chips and

agglomerations. Over-sized particles will be fed through the pulverizer to the recycle blender as described above. The fine powder will drop by gravity directly to the recycle blender.

Reject green pellets will be recovered as soft scrap. The green pellets will be transferred in canisters and weighed before recycle processing is initiated. CANDU fuel fabrication experience has found that, with controls on lubricant addition, lubricant removal is not necessary to recycle this material. The material will be transferred to a separate weigh storage bin from which it will be combined with oxidized hard scrap material for milling.

Pellet wash water and filter cleanings will be recovered as wet scrap. These liquids will be transferred in liquid containers, and each container will be weighed and assayed before processing is initiated. The material will then be transferred to a weigh storage tank from which it will be fed to a dryer for water removal. Solids from the dryer will be collected as a dry cake and sieved for removal of agglomerations. Over-sized particles will be fed through the pulverizer to the recycle blender. The fine powder will drop by gravity directly to the recycle blender.

All of the recovered material from hard scrap, grinder scrap, soft scrap, and wet scrap will be collected in the recycle blender for blending into homogeneous batches and assayed. The blender will be sized to provide conveniently sized batches of recycle material for addition into the pellet lines.

ADDITIONAL SOURCES OF PLUTONIUM FOR RECYCLE: In addition to processing internally generated scrap, the CANDU MOX recycle facility could be used to recover scrap, or excess plutonium, from other programs to assist DOE in reducing its inventory of excess plutonium. For example, DOE has estimated that some 20 to 30 tonnes of plutonium is being stored in various forms at DOE facilities, including Hanford, Rocky Flats, Savannah River, Los Alamos, Lawrence Livermore, etc. Originally planned for recycle into the weapons stream, much of this material is now considered waste and is a major liability for the DOE's Environmental Remediation program. The fuel supply team has concluded that much of this material can be blended into the CANDU MOX fuel fabrication process stream, perhaps at a rate of 10 to 20% (rate will depend on degree of scrap contamination and americium concentration).

One such opportunity is the recycle of MOX which is currently contained in unirradiated FFTF fuel left over after cancellation of the FFTF program.

FFTF Fuel Recovery: The fuel supply team briefly investigated the type of program that would be required to utilize the "green" FFTF fuel as feed material for CANDU MOX fuel manufacture. The current inventory of unirradiated FFTF fuel is:

CONTAINED PuO₂

(56) Driver Fuel Assemblies	560 Kg
(4500) unassembled Fuel Pin Assemblies	207 Kg
(200) Kg sintered pellets in storage canisters	40 Kg
(300) Kg loose PuO ₂ in storage canisters	260 Kg
Total	1067 Kg

Combined, these items account for approximately 1 tonne of Pu metal. The CANDU MOX fuel fabrication line at FMEF can be used to recover the pellets, soften them through oxidation and reduction, and store material in canisters to be fed into the recycle pulverizer with green scrap material when the facility comes on line. The recovered MOX powder will be mixed with other PuO₂ powder to be fed into master blending. These operations will be performed with a combination of interim equipment along with some portions of the normal scrap recovery system. Recovery of other MOX fuel forms will need to be addressed on a case basis.

There is a significant amount of existing equipment, in the form of tooling and fixturing, designed for the FFTF program which could be used for recovering the FFTF fuel pellets from the stored pins and driver assemblies. This tooling is currently being stored in the U.S. Army warehouse in Umatilla, Oregon. The recovered pellets can then be processed through the normal facility for processing into CANDU fuel. The following additional facilities will be needed to facilitate recycle of the FFTF fuel:

The existing storage vault in the FAA would be utilized to store the fuel assemblies and pins in carriers. Storage racks will be needed. Loose fuel pellets will be stored in the existing SNM vault.

A Duct Cutting Station will be needed to separate the upper end fitting from the fuel assemblies and expose the (150) fuel pins for extraction. These operations will be clean. This station will also be used to cut the empty ducts for disposal. A pin downloading station will be needed to remove the pins from the ducts and transfer them to storage. This clean facility will be located in the FAA where there is sufficient overhead clearance to facilitate remote crane operation.

A Pin Slitting/Pellet Removal Station will be needed to expose the fuel pellets for removal from the cladding. This facility will consist of existing special tooling to be housed in the existing Metal Fuels Glovebox containment which is available and may be redeployed for this purpose. The fuel pin cladding will be slit and the pellets collected and packaged in canisters for transfer to the SNM vault. This station would also be used to cut the emptied cladding into pieces to be packaged in drums for disposal as TRU or LSA waste. The metal fuels glove box, which measures 132" long x 48" wide x 53" high, will need to be relocated from Building 308 on the Harford Reservation where it has been used to handle uranium materials. The pin slitting/pellet removal station will be located on the 21' or 42' level of the FMEF where tie-ins to the glovebox (Zone 1) ventilation system can be made.

Estimated cost to design, procure, install, and test the entire FFTF fuel recovery operation is about \$5 million. (this is separate from the recycle process system itself, and is not included in our overall MOX fabrication estimate)

4.6.8 Operation of CANDU MOX Facility at FMEF

The actual operation of a modified FMEF facility for production of CANDU bundles will involve a dedicated team of workers and management personnel thoroughly trained in CANDU fuel fabrication skills, and all of the skills required to handle plutonium materials, including especially those important to occupational health, and material security. The fuel supply team has estimated that a work force of about 120 workers will be required to conduct operations, on a two shift per day, 5 day per week basis. The sintering furnaces will be operated on a 3 shift, around the clock basis. This estimate is based on the experience in FFTF fuel fabrication and in natural uranium CANDU fuel fabrication, with increases as needed for new requirements such as safeguards and accountability. Ten specialists in safeguards and accountability are included in this total.

The workforce will need to be augmented by site security and outside facility support personnel and these costs are treated separately as security and landlord costs in section 4.10.

We have included costs for a training program for the full 120 man staff based on the assumption of 6 months of training for the full staff. In practice, the training will occur over a much longer time period, with the actual staff building up gradually to full level at the time of commissioning.

Based on input received from Westinghouse Hanford Company, we have used an average of \$100 K annual compensation per worker, (including benefits) leading to a total labor costs for the facility of about \$12 million per year. In actual practice, there will be some workers costing less and some earning more; the \$100 K/year figure is an average.

Additional indirect costs of operation include general and administrative expenses and operating fees of about \$4 million/year, and technology transfer fees of \$5 million per year, representing 14% of the total materials and labor costs. This is an estimate of fees for transferring both the fabrication technology and the fuel performance technology from the current owners of this technology.

4.7 Alternate Fuel Supply Case (4 MT Pu/Year)

Three potential methods for achieving the goal of increasing the plutonium disposition rate from 2 tonnes per year to 4 tonnes per year have been identified and are discussed in the following sections

4.7.1 Use of Advanced Fuel Design (CANFLEX)

As discussed in section 2, the evolution of CANDU fuel bundle designs has been towards greater subdivision of elements, from the early 7-element geometry to the current 37-element bundle. To reduce element power ratings (for a given bundle power), AECL has developed the 43-element CANFLEX bundle design. This will enable the bundle to achieve higher burnups compared to the current standard design.

The effect of using this advanced design on fuel supply is as follows. As seen in Table 4-9 the amount of Pu contained in each bundle is increased from 0.23 kg in the reference case to 0.38

kg in CANFLEX. Due to a smaller number of bundles required to fuel a reactor, the amount of MOX required for fuel production is reduced from a nominal amount of 78 Mg per year per reactor to 42 Mg per year per reactor. In order to disposition 100 MT of plutonium in 25 years, four reactors will be utilized for a total MOX requirement of 168 MT per year. This amount of throughput is within the estimated capacity of the modified SAFLINE facility (nominally, 170 MT MOX per year). Therefore, FMEF fuel production capacity is expected to adequately handle the requirements of disposition of both 50 MT and 100 MT of plutonium in 25 years.

TABLE 4.9
Comparison Between 37-Element and CANFLEX 43-Element Designs

PARAMETER	REFERENCE 37-Element	CANFLEX 43-Element
Rings 1 & 2	7 elements; 5% Dy	8 elements; 6% Dy
Ring 3	12 elements; 2.0% Pu	14 elements; 3.6% Pu
Ring 4	18 elements; 1.2% Pu	21 elements; 2.1% Pu
Average Pu conc., wt.% in HE	1.52	2.70
Total Pu per bundle, kg	0.23	0.38
Total PuO ₂ per bundle, kg	0.26	0.43
Total MOX per bundle, kg	17.3	15.8
Fueling Rate, bundles/yr/reactor	4525	2630
Total MOX Required, Mg/yr/reactor	78	42

4.7.2 Increasing the Capacity of the FMEF

The team briefly examined the possibility of further expanding the FMEF to process 340 tonnes of MOX per year as might be required to dispose of 100 tonnes of weapons plutonium in 25 years using standard 37 element CANDU fuel bundles. This may be feasible and practical, but further study is required. As one example, it might be possible to deck over the shipping and receiving area, install decking in the main process cell, and install 4 additional furnaces. Another option which should be considered is to build an addition to the FMEF to provide additional floor space while still utilizing the existing electric power and HVAC systems. We estimate the additional cost of facility modifications on the order of \$50 million, and the cost of process equipment at an additional \$30 million, bringing the total costs to about \$180 million. (as compared to \$110 million for the 170 tonne/yr facility described in Section 4.6 above.)

4.7.3 Greenfield Approach

The fuel supply team received an estimate from BNFL (British Nuclear Fuels Limited, who recently completed construction of a commercial MOX facility in England) for construction of a 155 tonne/yr MOX fuel fabrication facility designed with flexibility to fabricate different types

of MOX fuel. The estimated construction costs were \$650 million. Recognizing that in a facility of this type, costs scale at less than a linear rate with throughput, we estimate the construction cost of a 340 tonne/yr facility at about \$1 billion.

4.8 Decontamination and Decommissioning Considerations

The requirements which apply to the decommissioning of DOE owned nuclear facilities are presently contained in DOE Order 5820.2A, Chapter V, "Decontamination of Radioactively Contaminated Facilities". Technical decommissioning issues are also covered in document DOE/EV/10128-1, "Decommissioning Handbook". At this time, DOE Order 5820.2A is being revised, and decommissioning will be eliminated from its scope. A separate, new order on Decommissioning is expected to be issued in the future. In addition, DOE/EV/10128-1 is in the process of being revised. A new document titled "Decontamination and Decommissioning Guidance Document" has been issued in draft form by the DOE Office of Environmental Restoration (EM-40). This document focuses on management and programmatic issues rather than technical decommissioning issues.

Some existing requirements which affect the design of a facility with respect to the decontamination and decommissioning are presented in DOE Order 6430.1A, Section 1300-11, "Decontamination and Decommissioning." They include:

- 1) Design of critical areas shall incorporate measures to simplify decontamination.
- 2) Items such as service piping, conduits, and ductwork shall be kept at a minimum in operating areas and be arranged to facilitate decontamination.
- 3) Walls, ceilings, and floors shall be finished with washable or stripable coverings or covered with metal liners, if required. All cracks, crevices, and joints shall be caulked and finished smooth to prevent contamination of inaccessible areas. Surfaces may be painted with smooth (gloss) finishes to improve decontamination ability.
- 4) Modular, separable enclosures shall be provided for radioactive and other hazardous materials to preclude contamination of fixed portions of the structure.
- 5) Glovebox and enclosure design shall account for limitations on dimensions of packing crates or other containers accepted at transuranic disposal sites.
- 6) Localized liquid transfer systems shall be used to avoid long runs of contaminated piping. Emphasis shall be placed on localized waste batch solidification. Special provisions shall be made to ensure joint integrity in buried pipelines.
- 7) Exhaust filtration components shall be located at or near individual enclosures to minimize long runs of contaminated ductwork.

- 8) Equipment and effluent decontamination equipment shall preclude, to the extent practicable, the accumulation of radioactive or other hazardous materials in relatively inaccessible areas, including curves and turns in piping and ductwork.

Additional criteria developed specifically for gloveboxes include the following.

- 9) Gloveboxes and process enclosures shall be designed with removable shielding, polished interior surfaces, and rounded corners and edges to facilitate decontamination.
- 10) All process equipment is located within a glovebox or other process enclosure and is designed to be maintained in situ or replaced. Equipment requiring removal will be decontaminated to the extent possible within the enclosure and removed for packaging and disposal.

DECOMMISSIONING COSTS: The cost estimates for decontamination and decommissioning of facilities at Hanford are presently prepared by the U.S. Army Corp of Engineers. The only facility at Hanford which is similar to the proposed operation is the 308 Building located in the 300 area which was used for the manufacture of experimental test fuel for the Plutonium Recycle Test Reactor (PRTR), the Fast Flux Test Facility (FFTF) Reactor, and the Experimental Breeder Reactor (EBR) II. At this time, however, cost estimates have not yet been prepared for this facility. Until a more formal estimate can be prepared for the proposed project, costs in the order of magnitude of \$40 million should be anticipated for decontamination and removal of all contaminated equipment including gloveboxes and ducting. An additional \$100 million could be expected for complete decommissioning of the facility should such an option be required. For purposes of this study, we have assumed the \$40 million figure, with the objective of leaving the facility with all contaminated equipment removed and surface contamination removed or stabilized. It would be up to DOE to decide whether further dismantlement was appropriate, and to fund such a program.

4.9 Ownership and Operational Responsibility

An essential prerequisite for the proposed plutonium disposition in CANDU option is that the utility (Ontario Hydro for the purposes of this study) must be assured that:

- (a) switching from their present natural uranium supply situation to this MOX supply will not disrupt their overall fuel supply program (for Bruce A and their other 16 reactor plants) which has been developed and implemented at great effort over many years;
- (b) the MOX fuel supply for the two Bruce A reactors which use MOX will be of sufficient quality and reliability (guaranteed supply) that they can fully depend on this supply to meet their customers' electricity needs, and
- (c) Arrangements are made for direct, on-site, quality assurance surveillance by utility inspectors, including qualification of fabrication processes.

To achieve this level of assurance by the utility, it is essential that the utility and its' current fuel suppliers, be directly involved in the entire MOX fuel fabrication program, from inception (now)

through development and licensing, through production and transportation.

One way to achieve this is through the establishment of a private venture to design, license and operate the production line within FMEF. A privately financed fuel supply program would require assurances from the Government on feed material supply, negotiation of a long term (25 year) fuel supply contract with the utility, and up front guarantees of licensing (by NRC) of the facility. Securing these guarantees, particularly the licensing guarantee, prior to investment in facility modifications, may be difficult to achieve. Therefore, the team considered other arrangements for ownership and operation, in addition to direct private licensing and ownership:

Government License/Own; Private Operation The first alternative would be Government ownership (including investment in the modifications and responsibility for licensing) with operational responsibility assigned to one or both of the present fuel suppliers, as agreed to in discussions among DOE and the utility. In this case, work on modifying the FMEF could proceed immediately upon the Record of Decision, and NRC licensing would proceed in parallel with equipment procurement and facility modifications. This would allow completion and startup within the 4 year period shown in Figure 4.4. A variation of this alternative would be for the Facility to be operated by a government contractor, under license to existing Canadian fuel suppliers, and with training, product and process qualification under the supervision of the existing fuel suppliers and/or the CANDU operating utility. While this arrangement would appear, on paper, to lead to lower capital costs (government vs. private financing) the ultimate costs could be higher because of the absence of the profit incentive associated with private ventures.

Private License/Own/Operate A second alternative would be to delay the schedule for process equipment procurement and modifications to FMEF (the bulk of the investment required) until after the NRC has granted a license for operation. Once that license has been issued, there would be sufficient basis for private investment into the modification and operation of the facility. This option has the advantage of retaining the entire fuel fabrication operation in private hands and under the contractual control of the utility. This would also meet the intent of recent privatization desires expressed by DOE Management. However, its' major disadvantage is a delay in initial production of MOX fuel bundles by about 2 years. Considering the "Clear and Present Danger" concern expressed by the National Academy of Sciences, this delay might prove unacceptable to the U.S. Government.

The choice between these alternatives requires further analysis and discussions among the various parties, including the operating utility and its' fuel suppliers, the U.S. and Canadian governments, and interested private parties. It is suggested that DOE continue to evaluate the pros and cons in consultation with the parties, in order to arrive at a final conclusion on the preferred approach.

4.10 Costs and Schedule

4.10.1 Preparing the Facility for Production

The proposed schedule for preparing for full production of MOX fuel at the FMEF is shown in Figure 4.4. As shown, the FMEF can be ready to produce CANDU MOX fuel bundles for operation at the BRUCE site within four years of initial efforts, and at a cost of around \$100

million. The preparatory program, and estimated costs, are divided into seven major phases as follows:

- site confirmation & conceptual design of facility modifications and process.
- trial bundle fabrication at Los Alamos (TA-55) and Canadian fuel fabricator (non Puparts)
- final design of facility modifications and equipment
- NRC licensing and federal, state and local regulation
- IAEA safeguards program
- Modifications of FMEF; procurement, installation and checkout of new process equipment
- preparations for manufacture of dysprosium pins & conversion of UF_6

Each of these program phases is discussed in the following paragraphs, along with the basis for the schedule and cost estimates:

A. SITE CONFIRMATION & CONCEPTUAL DESIGN OF FACILITY MODS AND PROCESS -

The FMEF at Hanford has been selected as the reference site for CAN DU MOX fabrication, and this study has been developed on that basis. However, several other sites have been suggested as alternative sites that could minimize the shipment of pits and plutonium oxide, including the Nevada Test Site (possible using the existing Device Assembly Facility) and the Pantex site at Amarillo, Texas. Therefore, a further study, sponsored by DOE, can be made to evaluate the cost and safeguards/security implications of using these alternative sites, and a selection made by DOE.

In parallel with these site selection and confirmation process, conceptual design of the process design, and unit operations, as well as the facility modifications required for the selected facility would be performed. These would lead to a formal set of design and process requirements, process flow sheet, and process equipment design feature selection. This work will also serve to firm up the overall cost estimate for the project to the point where firm budget estimates could be made.

It is estimated that these site evaluation, and preliminary engineering efforts will require about one year of work and cost about \$3 million.

B. TRIAL BUNDLE FABRICATION -

In parallel with Phase A, it is necessary to develop final specifications for the fuel, and initiate fabrication of up to 90 trial MOX CANDU fuel bundles. This program has four main purposes: First, about 35 fuel bundles are needed for physics testing at a zero power test facility (available at Chalk River) to confirm nuclear design parameters. Second, irradiation testing of the reference MOX fuel bundle in the NRU reactor is required to confirm fuel performance predictions, and to test the limits of fuel performance beyond the nominal power levels expected in the Bruce reactor. This will confirm the predicted fuel design margins. About 40 fuel bundles are required for this testing, but this will include the 35 fuel bundles used for the physics testing. Third, about 50 additional fuel bundles are required for trial irradiation at the

Bruce A reactor to confirm performance and the handling aspects. Both the NRU and the Bruce irradiations will provide technical support to the AECL licensing process. The fourth purpose is to finalize and qualify the fuel fabrication process parameters via a trial bundle fabrication program.

The 90 trial bundles can either be fabricated at the Los Alamos TA-55 building, or at the RFFL facility at Chalk River. The TA-55 is the preferred facility, from the standpoint of safeguards and security (U.S. weapons plutonium could be converted and used for this program) and recent experience in making MOX fuel for other programs. However, either facility could be used.

The program would involve co-operative efforts among AECL (to finalize the fuel design and prepare fabrication drawings and specifications), Canadian fuel suppliers (to provide needed processing experience and in some cases, equipment, including direct assistance where needed), and the fabrication facility operator (either Los Alamos or AECL Chalk River) to produce the Bundles. An overall project manager will be required to plan and manage the program. Non-plutonium feedstock would be procured from available suppliers. For example the zircaloy components could be acquired expeditiously from existing CANDU fuel suppliers, the uranium oxide from CAMECO, and the dysprosium pins from Chalk River.

Preparations for the actual bundle fabrication will require 9 to 12 months. Los Alamos TA-55 is capable of producing about 12 bundles (240 KG of mixed oxide) per month. We have assumed 8 bundles per month to allow other work at TA-55 in parallel with this program. Thus, the CANDU trial bundle production effort would require about 1 year. Thus a two year program is shown on figure 4.4, with the first bundles being released for irradiation testing in about 12 months from start.

We estimate the cost for this program at about \$7.2 million .include the following:

1)	Develop fabrication specifications	\$350K
2)	Fuel Supplier support & equipment	1,000K
3)	Materials and supplies (except Pu and Dy)	400K
4)	Dysprosia pins	850K
5)	Production Labor	1,900K
6)	Facility usage	2,100K
7)	Program management and engineering	<u>600K</u>

<u>TOTAL PROGRAM</u>	<u>\$7,200K</u>
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About \$3.9 million of the total is for the first 40 bundles to be used for physics and NRU tests. The remaining \$3.3 million is for the 50 bundles to be inserted in Bruce A.

C. FINAL DESIGN OF FACILITY MODIFICATIONS AND EQUIPMENT

Estimates for the final design of the equipment and the facility modifications at FMEF are based on a percentage of the total equipment costs. Our schedule assumes that this design work would proceed immediately after the site selection and preliminary design (item A above) and would take about one year. We estimate the design cost will be about 20% of the total capital

cost, or \$3.9 million for equipment and \$2.3 million for facility modifications, for a total of \$6.2 million. This work would be done in parallel with the licensing and safeguards work in D and E below and would be integrated into an overall project. Project management costs are included in Item F below.

D. NRC LICENSING AND FEDERAL, STATE AND LOCAL REGULATION

Estimates for NRC licensing, and the regulatory permits required for FMEF utilization for MOX fuel fabrication, are based on the program described in section 4.6.3 above. Major cost elements include the following:

1)	NEPA, State and local regulations	\$2,500K
2)	Safety Analysis Report preparation (owner/operator)	2,000K
3)	NRC License fees	5,000K
4)	Licensing support by owner/operator	<u>3,000K</u>
TOTAL		<u>12,500K</u>

As shown on Figure 4-4, preparation of the SAR is done in parallel with final design, and the formal application to NRC, with the SAR is made at completion of final design, about 24 months into the program. We have allowed 21 months for NRC review (including public comments). This leads to receipt of an NRC license to operate slightly less than 4 years after program initiation. Procurement of equipment, and facility modifications are accomplished in parallel with the NRC licensing review.

E. IAEA SAFEGUARDS PROGRAM

A program to formulate, design, procure and install a full IAEA safeguards system is described in section 4.6.4. This will require extensive interactions with the IAEA, assistance from technical experts at Los Alamos, and close interaction with the facility owner and designer/operator. In many cases, safeguards instrumentation must be integrated with process instrumentation. We have allowed 18 months to develop a safeguards approach and gain approval of the Facility Attachment, and an additional 18 months for procurement and installation of the equipment in the facility. Total cost for development of the system and detailed design is estimated at \$8 million, based on the details provided in section 4.6.4. An additional \$2 million is estimated for procurement and installation of equipment, and is included as part of capital costs in F below.

F. MODIFICATIONS OF FMEF; PROCUREMENT, INSTALLATION AND CHECKOUT OF NEW PROCESS EQUIPMENT

Details of the process equipment and facility modifications are presented in section 4.6.2 and are summarized below. The total estimate for equipment procurement installation and testing, including facility modifications is \$88.2 million as summarized in the following table:

1)	process equipment	\$19,500K
2)	glove boxes	4,071K
3)	equipment installation	3,965K
4)	recycle and Pu finish station	5,126K
6)	Facility modification, security and I&C	6,000K
7)	Facility Modification, wiring & piping	5,553K
8)	Engineering & QA	6,000K
9)	Checkout, Training, Startup	13,139K
10)	Project Management	7,500K
11)	Contingency (15%)	<u>15,400K</u>

TOTAL CAPITAL COSTS **\$88,200K**

We estimate that the facility modifications, and the equipment procurement and installation will require about 18 months, and an additional 6 months will be required for checkout and startup of the facility. The actual FMEF conversion project would not begin until after a final DOE Record of Decision, currently estimated in about 2 years. Thus, if the project began immediately after the ROD, the facility would be ready to begin production of CANDU MOX bundles in about 4 years. This is the schedule shown on Figure 4.4.

G. PREPARATIONS FOR MANUFACTURE OF DYSPROSIUM PINS & CONVERSION OF UF₆

As described in section 4.5.2, a new defluorination unit will be required to convert 170 tonnes per year of UF₆ to ceramic grade UO₂ powder. The cost of this investment has been included in the overall cost of UO₂ by the supplier (CAMECO). Including shipping and conversion, the cost of UO₂ conversion is estimated at \$4,348,000 per year.

As described in section 4.5.3, a new dysprosia ceramic pellet and pin line must be installed to provide completed dysprosia pins to the FMEF production facility. The cost of this investment has been included in the estimated annual cost of dysprosia pin production, which is \$2,590,000 per year.

4.10.2 OPERATIONAL COSTS - 37 PIN, 2 BRUCE REACTORS

Operating Costs include the direct labor, the costs of materials and supplies, the cost of waste disposal, landlord and security, and licensing fees. Financing costs are not included since these depend on the ownership arrangements. Several options for ownership arrangements are discussed above. Financing costs for these options are presented in Section 9 of this report.

A summary of the operational costs, except financing, is presented below. The basis for these estimates are contained in the previous sections.

<u>Materials and Supplies Subtotal</u>	\$14,442K
UO2 conversion	\$4,348K
cladding components	5,107
dysprosia pins	2,557
consumables	2,000
natural urania end pellets	410
 <u>Labor, Fees, Management Subtotal</u>	 26,560
production, direct & indirect	12,000
mgmt, g&a, and operating fees	4,000
waste disposal	3,560
NRC licensing (I&E) fees	2,000
Technology transfer fees	5,000
 <u>Facility Support Services Subtotal</u>	 20,000
security	10,000
landlord	10,000
 <u>Annual Decontamination fund</u>	 <u>2,800</u>
 TOTAL ANNUAL OPERATIONS	 \$63,800

4.10.3 ANNUAL OPERATING COSTS - CANFLEX - 4 BRUCE REACTORS

We have not performed a detailed cost estimate for the case of 4 Bruce reactors, either for the standard bundle design at 340 tonnes/yr, or the 170 tonnes/yr of CANFLEX fuel. As a ballpark figure, we estimate the cost of production will increase about 20% because of the smaller pellets and increase from 37 to 43 pins per bundle. Landlord, security and D&D costs would not change. This leads to an estimated annual cost of \$72 million to produce 164 tonnes per year of CANFLEX fuel, sufficient to fuel 4 Bruce reactors and consume 4 tonnes/year of weapons plutonium.

4.10.4 FUEL SUPPLY COSTS RELATIVE TO NATURAL URANIUM FUEL

Ontario Hydro estimates that the current fueling costs for two Bruce Reactors is about \$16.3 million per year. This is about \$47.5 million less per year less than the MOX fuel case with reference fuel. For the CANFLEX 4 reactor case, the delta is about \$39.4 million per year.

In addition, Ontario Hydro's current fuel supply contracts are sensitive to price versus volume conditions which would be affected by substituting MOX fuel for Natural Uranium. For example, if the current purchase volume were reduced by eliminating two Bruce Reactors, Ontario Hydro estimates they would incur an additional charge of about \$3.6 million per year. Some of these costs might be offset if the existing suppliers were involved in the MOX fabrication, but this would be a matter of negotiation between Ontario Hydro and the existing fuel suppliers. In the fuel cost comparison presented below, we have assumed that 50% of this penalty would be offset by fuel supplier involvement in the MOX program, so the net penalty is shown as \$1.8 million/yr.

For the case of 4 Bruce A Reactors, Ontario Hydro estimates a penalty of about \$9.3 million/yr from reducing their procurement volume for natural uranium by 4 reactors worth of fuel. Here again we have assumed that 50% of this penalty would be offset by the involvement of Canadian fuel suppliers in the MOX program, so that the net penalty would be \$4.65 million/yr.

Taking these penalty costs into account leads to a net delta for MOX fuel as compared to natural uranium fuel of \$49.3 million per year for the reference fuel (2 Bruce Reactors) and \$44 million per year for the CANFLEX case (4 Bruce reactors).

4.10.5 TWIN CANDU MOX FACILITY IN RUSSIA:

If a twin facility to an FMEF CANDU MOX facility were designed for installation at another location, and at about the same time frame, then much of the up-front design and procurement costs can be used, and substantial cost savings achieved. Therefore, if further discussions with Russian institutions leads to an interest in construction a twin facility in Russia, then the total costs to both countries can be substantially reduced. For example, most of the \$4 million for equipment design, \$12 million for licensing, and \$8 million for safeguards concept development and design could be eliminated, and the cost of equipment procurement could be reduced by 20 to 30%. Thus there are major financial incentives to a parallel program in both countries. To achieve this, it is suggested that Russian participation in the next phase of this project be considered.

4.11 Members of the Fuel Supply Team

The fuel supply team included the following organizations.

- AECL Research
- AECL CANDU
- Babcock and Wilcox Naval Nuclear Fuel Division
- Belfort Engineering and Environmental Services
- Gamma Engineering
- Ontario Hydro
- Westinghouse Hanford Company
- *Zircatec Precision Industries

* It should be noted that there are two qualified fuel suppliers in Canada, Zircatec and Canadian General Electric. Since this was an "initial study" Ontario Hydro utilized one of the fuel suppliers as a surrogate for the Canadian fuel supply industry

5.0 Packaging and Transportation

This section describes the approach taken towards providing a safe, secure method of moving the finished CANDU fuel bundles from the reference fabrication plant to the reference disposition facility. It assumes that the MOX fuel production facility (FMEF, Hanford) will provide the MOX fuel on a timely basis to meet the supply and scheduling requirements to maintain an appropriate amount of MOX fuel at the Bruce site. Based on the study work carried out elsewhere in the report, it is expected that the fuelling requirements for two reactors at Bruce will be 15.5 bundles per full power day per reactor, thus indicating that one month's supply of fuel will be about 754 bundles for the reference case.

5.1 Packaging

For the purposes of this study, the packaging system selected was based on a standard stainless steel 55 gallon (US) drum which has been used successfully for a wide range of special nuclear materials transport. This container was selected on the basis of familiarity with handling, existing infrastructure for cargo restraints, and meeting existing regulatory requirements. The container would be capable of holding 7 CANDU MOX fuel bundles in their individual packages without significant modification to the packaging. The design of the packaging system would have to be certified.

Several issues were identified with regard to the packaging requirements for Category 1 materials. The packaging to be used must be a type B package and approved for use within both Canada and the US. The package and container identified for the purposes of this study would have to undergo certification by the US DOE, US NRC, US DOT, the Canadian AECB, and the Canadian Ministry of Transport prior to any use of the container for the intended purpose. While the packaging has been approved for bulk shipments of Category 1 materials, it has not yet been approved for the transport of CANDU MOX fuel bundles. The certification process can become quite lengthy if not approached correctly. The study indicates that all design information should be supported by test results showing the integrity of the container and packaging under approved test conditions. This implies that a program should be undertaken to qualify the container and package as soon as possible. Sketches of the container proposed and the orientation of the fuel bundles inside can be found in Figures 5.1.

Several other existing containers have been identified but all would have to be qualified in either Canada or the US. Although these containers could carry varying numbers of CANDU fuel bundles, these existing containers have been designed for other purposes. The approach to this aspect of the study was to take the shortest route to the earliest start possible in carrying out the disposition mission.

International regulations governing the packaging of plutonium for transport are currently being revised. It would be important to re-evaluate the containers and packaging when the revisions have been completed to ensure compliance with all regulations.

Figure 5.1 Sample MOX Fuel Shipping Package

AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

4

of

5

5.2 Transportation

The study reviewed several methods of transportation: shipment by air, rail transport and road transport. The US DOE currently have the capability to transport SNM by air but are currently prevented from transporting plutonium by government regulation. Shipment by rail is another possibility as the US DOE Naval Reactor Program has the facilities and experience in moving SNM in this manner. US navy reactor cores are currently moved by rail. The other method is by road and is one that readily lends itself to this study, bearing in mind that the earliest start is one of the key considerations. As more information is available regarding this method of transportation, it was chosen as the basis for the study. This does not preclude the other options, which, upon more detailed examination, may provide significant advantages over road transport.

The transport of CANDU MOX fuel from the reference fuel production facility to Bruce involves crossing the US/Canada border and will be controlled by all regulations applicable to import/export of SNM. This study assumed that the transport will be under IAEA safeguards and comply with applicable sections of IAEA INFCIRC 225. In addition the transport will comply with applicable US and Canadian regulations.

Packaging, shipping and receiving of MOX fuel will be safeguarded by a combination of item accountancy, physical assay prior to packaging and after receiving, tamper-proofing of shipping containers and vehicles, and containment and surveillance measures at the Bruce receiving and storage areas.

At the fuel production shipping area, the MOX fuel ID will be verified and logged, and the fuel will be physically assayed by using an assaying device based on passive neutron counting¹. The fuel will be packaged into shipping containers (55 gal drums), 7 fuel assemblies (bundles) per drum, and the drum put under IAEA seals. The drums will be loaded into a DOE Safe Secure Transport (SST) which has the ability to carry 48 drums on pallets with a cargo restraint system. The vehicles themselves have been designed to transport SNM in various forms and to withstand attack.

After loading, the vehicle's cargo area will be sealed. It is anticipated that three such SSTs will be loaded and travel in a convoy to the Bruce site. The use of three SSTs ensures that one month's fuel supply (for both reactors) can be moved in one convoy. The transport of the MOX fuel from the fuel fabrication facility to Bruce will follow the convoy arrangements, including armed escort, normally used by DOE for transporting SNM. At the US/Canada border, the jurisdiction changes. The US DOE armed escort will be replaced by the Canadian armed escort group but the level of diligence will stay the same. While in Canadian territory the transport will retain communication links with US as well as with the Canadian escort and the Bruce central Security Monitoring Room. During the course of the study, the US DOE

¹"Safeguards Instrumentation for Continuous, Unattended Monitoring in Plutonium Fuel Fabrication Plants", H.O. Menlove and M.C. Miller, Los Alamos National Laboratory; T. Ohtani, M. Seya, and S. Takahashi, Power Reactor and Nuclear Fuel Development Corp., INC '93

Transport Safeguards Division representative surveyed the route and ensured that the route physically accommodates the transport vehicles.

At Bruce NGS, upon entering the designated receiving area, the vehicle's seal will be broken, drums containing the MOX fuel will be unloaded and their seals verified. The SST vehicles, empty or containing empty shipping drums and cargo restraint pallets, will return to the US. Prior to storage, the MOX fuel identity (ID) will be verified and the fuel physically assayed either on 100% or statistical basis (this step may be omitted if the combination of physical assay prior to shipping and the IAEA seal integrity is sufficient "proof" for accepting responsibility for the fuel).

Fuel will be stored in a newly constructed, dedicated facility with physical attributes resisting forceful entry; limited, monitored entry points and C/S (Containment and Surveillance) equipment. The fuel storage arrangements will facilitate monthly physical inventories. Detailed logs of all fuel movements will form a part of the procedures and protocols put into place for the mission.

5.3 Safeguards and Security

The security of fresh MOX fuel is ensured by the physical security arrangements at the Bruce site, which will require some modifications to the existing facilities, and by the organizational structure and capabilities. These modifications and security measures have been discussed elsewhere (ref. section 2.6). These arrangements will be based on detailed threat assessments which will be developed as the process nears fruition. The threat assessments will take into consideration the changed mission of the Bruce facility and the response mechanisms available.

The security for the transport of the fresh MOX fuel will be provided by the US DOE Transport Safeguards Division. Their methods of shipment, convoy deployment, and escort support are of necessity classified. Their record and their experience in moving such materials demonstrates that the shipments are given the required degree of protection.

Fresh MOX fuel will be kept under C/S in a dedicated facility with restricted, controlled and monitored access. Since CANDU reactors are on-power refuelled, using approximately 16 fuel bundles per full power day, the removal of fuel from storage and loading it into the reactors is a continuous process. All phases of fuel movement are tracked by using the fuel assembly-specific ID. This administrative procedure is supplemented by physical assays at critical points, providing independent verification of the item accountancy scheme. The whole facility will be under a full-scope IAEA Safeguards regime.

The critical points involving physical verification are:

- fuel receiving (Section 2.4.1), providing substantiation of Canadian responsibility for MOX fuel;

- reactor loading, confirming the integrity of fuel storage and handling arrangements; and
- reactor discharge, confirming that no interference with the freshly discharged fuel occurred.

Physical assay at the fuel shipping, receiving and reactor loading areas will use basically the same device based on passive neutron counting. Thus, at a minimum, three such devices will be required.

Monitoring of reactor discharge is performed by "spent fuel bundle counters" which track the fuel motion and are a standard safeguard device for CANDU reactors (as a part of the IAEA safeguards). At the newest CANDU multi-unit station, Ontario Hydro's Darlington NGS, the monitoring of fuel discharge is enhanced by a Core Discharge Monitor (CDM) to reconcile the gap between actual core discharge of fuel and possible diversion prior to discharge to the spent fuel bay. This type of monitor is not installed at older CANDU stations. Initial views from IAEA are that CDMs would be an essential enhancement of IAEA safeguards should MOX burning become a concrete option.

The physical security of spent fuel stored at the site is ensured by the physical security measures of the whole compound. Spent fuel is tracked by its ID throughout storage, both wet and dry. To facilitate fuel verification, spent fuel is kept in enclosed, larger storage structures under IAEA seals. The shipping of spent fuel to the disposal site will be in secure, IAEA safeguarded, sealed containers. Further discussion of this topic is covered in chapter 6.

5.4 Cost and Schedule

The estimates for cost associated with the Packaging and Transportation aspects of the dispositioning of plutonium at the Bruce Nuclear Power Development site are based on having two Bruce Units dedicated to the mission, with MOX fuel arriving on a monthly basis in sufficient quantities to meet fuelling requirements. The packaging requirements thus not only include the qualification and licensing of an appropriate package, but the manufacture and supply of a sufficient number of packages to meet the fuelling requirements. The use of US DOE SST's has been selected for the transportation of the MOX fuel for the purposes of this study. All mission costs are inclusive of start-up costs.

Packaging costs include the initial design, construction of qualification packages, testing and qualification of the packages in both the US and Canada, manufacturing and supply of the containers, and provision for replacements. Also included are the safeguards assay equipment, computer accounting station, tags and seals, and inspection costs. The start-up costs for the packaging and associated safeguards are estimated to be \$3.75M and annual operation and maintenance costs over the duration of the mission are expected to be on the order of \$0.6M/year.

Transportation costs include the use of safeguards assay equipment, tags and seals, transport and escorts in the US and Canada. They also comprise the development of the transportation program and logistics, development of threat and risk assessment data on a timely basis, emergency response plans and training, and coordination activities. Of interest, it is expected that each trip from Hanford to Bruce with fresh MOX fuel and the return with empty containers and packages will cost about \$100K, and that 300 such trips would be necessary over the life of the mission. It is anticipated that the start-up costs for the transportation of the MOX fuel will be \$1.25M and the annual operation and maintenance costs over the mission will be about \$1.5M/year. The schedule for the packaging and transportation aspects of the plutonium dispositioning project can be found in the master schedule for the reference case.

5.5 Conclusions

The proposed arrangement for plutonium disposition by using a US fuel production facility, located at a DOE site, and burning the fuel in Canada combines the advantages of inherent good security measures at the fuel production site (where SNM of the highest attractiveness is handled) with technical capabilities of CANDU reactors which could operate with full core MOX loading without modifications. All aspects of MOX manufacturing are in the US jurisdiction. After leaving the MOX plant, the plutonium is present only in form of MOX fuel, representing a material of lower strategic significance. Each fuel assembly (bundle) has its own ID and thus it could be tracked by item accounting procedures. Nevertheless, the administrative procedures are supplemented by physical assays at selected points to substantiate the accounting scheme. The spent MOX fuel can be returned to US, thus closing the cycle.

It is expected that fuel transport and burning in CANDUs will be under IAEA safeguards. The proposed transport arrangements have been discussed with the Transport Safeguards Division at DOE. They represent the best initial alternative from road/rail/air options and can be accommodated by the available existing physical means.

This study has identified a need to have two individuals assigned as a "coordinator" and a back-up for a single point of contact in all transportation, customs, and security communications between the US DOE and the Canadian end of MOX fuel movement operations. The coordinator would integrate the various Canadian organizations which would be essential to the smooth, secure transfer of the materials to the Bruce facility.

In general, the practices, equipment, transport systems, and safeguards concepts currently in use will be applicable to this mission. The existing practices are not necessarily the most cost-effective means of carrying out the mission; however, they are currently acceptable and provide the means to effect an "immediate start" to the disposition of excess weapons-grade plutonium. Significant savings may be possible obtained by revisiting the packaging, transportation, and security issues.

In summary, there are no major impediments regarding the safeguards and security issues associated with this mission. The development, testing, qualification and manufacture of

appropriate, optimized, packaging and shipping containers for the MOX fuel bundles have been identified as requiring an early start.

6.0 Spent Fuel Storage and Disposal

6.1 Overview

The implications of spent MOX fuel on the Bruce-A station existing storage facility and operations were studied. Two scenarios of transportation and ultimate disposition of spent MOX fuel were developed as illustrated in Figure 6.1. One scenario covers disposition of the spent MOX fuel in Canada and the other scenario covers disposition of the spent MOX fuel in the U.S. A variety of options for the interim storage of the spent fuel were considered for each scenario.

The processes and facilities required for the interim storage and final disposal of spent CANDU MOX fuel will be the same as those for the existing CANDU spent fuel. The physical dimension of the MOX fuel bundle is identical to the existing CANDU fuel. The spent MOX fuel bundles may be stored in CANDU spent fuel modules (equivalent to storage racks for LWR spent fuel) for wet storage at the Bruce-A site. From the point of view of storage, the spent MOX fuel will have similar decay heat generation rate and fission product concentration as compared to the existing CANDU spent fuel. Thus the spent MOX fuel may be stored or disposed of by means that are the same as the existing CANDU spent fuel.

The underlying strategy for managing the storage and disposal of the CANDU MOX spent fuel is an integrated systems design effort that covers all three phases, namely; the interim storage, transportation and final disposal activities. The system/component configurations and performance specifications should be compatible and complementary starting from the spent fuel pool storage phase, to interim dry storage concept, transportation means, and finally the ultimate repository disposal requirements. This integrated and thereby optimized design effort will not only bring savings in the costs of handling the spent fuel, but also significant differential benefits to the environment, safety and health.

6.2 Interim Spent Fuel Storage

6.2.1 Existing Storage Facilities

CANDU natural uranium fuel cannot go critical in light water, it is stored in light water filled bays (pools) with no criticality concerns. Upon discharge from the reactor in any of the four Bruce A units, the spent fuel is transferred from the fuelling machine to the primary spent fuel storage bay. After an initial residency period in the primary bay of approximately 9 months, the fuel is transferred under water through a transfer duct to the auxiliary spent fuel bay.

All four units at the Bruce A station share a common fuel handling system which extends to the spent fuel storage bays. The spent fuel, which is stored in racks, is cooled without the need for forced convection although the water is circulated through heat exchangers so that a constant temperature is maintained in the bays.

Figure 6.1 Disposition of Spent MOX Fuel

6.2.2 Interim Wet Storage of MOX Spent Fuel

Used MOX fuel can be treated much like natural uranium CANDU fuel except that any criticality issues which are not present with the natural uranium CANDU fuel must be addressed.

Storage of MOX spent fuel in the storage bays will be in modules similar in design to those already in use for storage of spent natural uranium CANDU fuel at several Ontario Hydro sites. A conceptual design for these modules is shown in Figure 6.2. Because of the height restriction imposed by the storage bay transfer duct these modules will differ slightly from the existing 96 bundle modules currently in use, but they would be compatible with existing storage racks and dry storage canisters. New machinery to load spent fuel into the modules, to transfer the modules to stacking frames in the spent fuel bay, and new stacking frames will be necessary. In all cases, however, equipment designs will be similar to existing equipment in use at other Ontario Hydro sites.

6.2.3 Interim Dry Storage of MOX Spent Fuel

On site dry storage of spent fuel is an existing feature of several CANDU sites worldwide. Two different systems have been developed in parallel for the on site interim dry storage of spent CANDU fuel. The selection of an interim facility design would be influenced by the option selected for permanent disposal of the spent fuel.

Figure 6.2 Spent MOX Fuel Storage Module

The first on site dry storage installations, constructed in 1985, utilized sealed, unventilated concrete canisters which contain nine baskets of used fuel bundles (see Figure 6.3). Fuel is transferred to the baskets in the spent fuel bays then the baskets pass through a shielded workstation where they are dried and seal welded, into a transfer flask for transport to the canister. Once the canister is loaded with nine baskets, a canister plug is inserted in the top, seal welded and two IAEA safeguards seals are installed. The canisters provide a passive and essentially maintenance free facility over their 50 year design life. Interim dry storage of spent MOX fuel destined for permanent disposal in the U.S. would probably be based on this AECL developed concrete canister system.

In parallel with the dry storage system developed by AECL, Ontario Hydro has developed a system for use at its Pickering and Darlington stations. These dry storage containers shown in Figure 6.4 were developed to be compatible with the storage modules in use in the spent fuel bays. The design is rectangular in section with high density reinforced concrete sandwiched between two carbon steel shells. The inner shell forms the primary containment boundary while the outer shell provides structural strength and facilitates decontamination. Regulatory approval to implement dry storage at the Pickering site was obtained in 1993 with the first bundles to be placed in dry storage in October 1994.

The containers which would be used for interim dry storage of spent MOX fuel destined for permanent disposal in Canada, would probably be based on the Pickering design but would be compatible with the smaller MOX modules. Although the containers are designed to receive fuel after it has spent six years in wet storage, there is sufficient capacity in the spent fuel bays at Bruce A station that the interim dry storage facility for MOX fuel would not be required for at least ten years after the first loading of the MOX fuel. This would allow more than sufficient time to license and construct an interim dry storage facility at the site.

6.3 MOX Spent Fuel Disposal

Prior to shipping to a disposal facility, spent fuel bundles may be physically assayed to verify that their fissionable material assay is consistent with their power and storage history. The spent fuel would be shipped in sealed, fully licensed and IAEA safeguards approved containers compatible with the interim dry storage facility design and handling provisions. The available infrastructure (e.g., large pool storage capacity) at the Bruce station allows considerable flexibility in the schedule to begin shipments to a permanent disposal site either in Canada or the U.S.. Consequently, there will be ample time to ensure an interim storage system design that is optimized for final disposal options of the spent fuel, as well as to ensure appropriate transport packaging and shipment.

6.3.1 Shipment and Disposal in U.S.

As described previously, if a permanent disposal site in the U.S. is selected, the interim storage system concept and component design configurations will be developed to be compatible with the shipment and disposal options planned for the U.S. repository. In particular, the shipment and disposal component designs will be consistent with Multiple Purpose Canister and high level waste repository program. The existing AECL interim storage systems and associated

Figures 6.3 Dry storage canister for interim storage and ultimate disposal in U.S.

transportation and spent fuel disposal packaging concept will be applied in this case (see Figure 6.4).

The shipment and disposal of spent CANDU MOX fuel bundles in the U.S. should present no increase the safety and environmental risks as compared to existing commercial LWR spent fuel. The federal, state and local licensing/permitting requirements and acceptance criteria for shipping and disposal of CANDU MOX spent fuel are expected to be the same as U.S. LWR spent fuel. Since the fuel burnup and decay heat levels for spent CANDU MOX fuel will be significantly lower than LWR fuel, current U.S. spent fuel transportation and repository performance evaluations are expected to envelope the case of CANDU MOX spent fuel deposited at a U.S. repository.

Figure 6.4 Dry Storage Canister

6.3.2 Shipment and Disposal in Canada

6.3.2.1 Transportation

The responsibility for regulating the safeguards and security of the transport of radioactive material rests with Canada's nuclear regulatory agency, the AECB. The AECB's Transport Packaging of Radioactive Materials Regulations are virtually identical to the IAEA Safety Series 6, with the major difference being the requirement that all foreign Type B and fissile packages must be endorsed for use in Canada, a process which requires a technical review by AECB.

A Design Approval Certificate under the AECB's Transport Packaging of Radioactive Materials Regulations is being sought for the dry storage containers developed for interim dry storage at Pickering station for the off site transportation of used fuel. This design minimizes the amount of handling of the used fuel. As the dry storage containers have already been approved for interim storage of used fuel, it is not anticipated that there would be any difficulty in obtaining the Design Approval Certificate for transportation. The containers which will be fabricated for spent MOX fuel will differ slightly from the existing design in size only. To ensure the minimum handling of individual used fuel bundles, approval for the MOX containers will be required ten years after first loading of the MOX fuel.

6.3.2.2 Disposal

In 1977 a study was commissioned by the Government of Canada to address options for the permanent disposal of nuclear waste. This report, "The Management of Canada's Nuclear Wastes" by Professor R.K Hare of the University of Toronto, formed the basis for government policy and the subsequent research into the disposal of spent nuclear fuel. In 1981, the Canadian Nuclear Fuel Waste Management Program, a ten year generic research and development program was approved. This program represents a cooperative agreement between the Government of Canada and the Province of Ontario for the development of technologies for the safe management and permanent disposal of spent CANDU nuclear fuel. Under this agreement, AECL was given the mandate to assess the concept of disposal of spent fuel in deep, stable granitic rock of the Canadian Shield.

An engineering study was carried out for a used fuel disposal center (UFDC). The concept is under review in accordance with the objectives and requirements described in the AECB Regulatory Document R71. This AECB review will provide an opinion on the acceptability of the concept. Site selection for the UFDC will be undertaken only after the concept assessment has been satisfactorily completed. As a nuclear facility the UFDC will be subject to licensing by the AECB and will be subject to IAEA safeguards.

The conceptual spent fuel disposal center is a self-contained facility where nuclear fuel waste will be disposed of in an underground vault in granitic rock in the Canadian Shield. The waste will be isolated by means of a series of natural and engineered barriers to radionuclide release. It is assumed that the spent fuel will be stored at its origin (nuclear generating station) for a minimum period of ten years before transportation to the UFDC. The UFDC will not be placed in operation until there is sufficient CANDU fuel in interim on site storage to operate the center economically at full capacity. The in service date projected for the center is 2035.

Since the volume of the spent MOX CANDU fuel will be less than that from natural uranium fuel, MOX consumption will not impose any new requirement on the facility.

6.4 Conclusions

- a) Spent MOX CANDU fuel is compatible with existing spent fuel handling and storage facilities at the Bruce A station. A minor design modification will be required to the existing storage module to address the height restriction at the transfer duct of the storage bays. The modules (or racks) are used to hold the spent fuel in the storage bays.
- b) There is no difference in the external physical dimensions between MOX CANDU fuel and natural uranium CANDU fuel.
- c) Due to slightly higher burnup of the reference MOX fuel, the quantity of spent fuel which will be produced over the 25 year duration of the MOX program will be approximately 10% less than the quantity of spent natural uranium CANDU fuel produced in generating same amount of electrical energy. If the advanced MOX fuel design (CANFLEX) is used, the quantity of spent fuel would be reduced by approximately 50% relative to natural uranium spent fuel.
- d) The technologies required for CANDU spent fuel interim storage and transportation exist and have been demonstrated in licensed, operating facilities. The technology required for permanent disposal of spent natural uranium CANDU fuel in Canada is currently under development in prototype demonstration facilities in Canada. CANDU MOX fuel can be handled with these same technologies. As such no new technology will be required for interim storage, transportation or ultimate disposal of spent MOX CANDU fuel over and above that required for spent natural uranium CANDU fuel.
- e) The spent CANDU MOX fuel is virtually indistinguishable from spent natural uranium CANDU fuel, and can be handled with the same equipment and procedures in existing facilities as spent natural uranium CANDU fuel, therefore there are no new or significant environment, safety and health issues associated with the interim storage, shipping or disposal of spent MOX fuel.
- f) Spent CANDU MOX fuel can be transported to U.S. disposal sites using packaging already developed and in use for interim dry storage of CANDU fuel at Canadian reactor sites.

6.5 Schedule and Cost for the Storage and Disposal of MOX Spent Fuel

As described earlier, an integrated systems design effort should be under taken for the interim storage, transportation and final disposal of the spent fuel. The system/component configurations and performance specifications should be compatible and complementary starting from the spent fuel pool storage phase, to interim dry storage concept, transportation means, and finally the ultimate repository disposal requirements. This integrated design effort

will bring savings in the costs of handling the spent fuel, and benefits to environment, safety and health.

Based on this design strategy, the overall systems concept for the storage and disposal of the MOX spent fuel should be established before the first loading of MOX fuel at the Bruce A station. This overall systems concept should ensure that all interim storage component designs and transportation means are compatible with the permanent disposal facility requirements. The systems and the storage modules for the spent fuel for wet storage will be required at about 1 year after the first loading of the MOX fuel. The need for dry storage containers for spent MOX fuel will occur roughly 11 years after the first loading of MOX fuel.

A cost comparison is made for the three cases; 1) burning natural uranium fuel, 2) the reference bundle MOX fuel case and 3) the advanced bundle MOX fuel case. Several factors were considered in arriving at the cost comparisons. For the natural uranium case, the number of spent fuel bundles would be higher than either MOX fuel case however, the packing density of the storage modules and interim storage canisters is higher for the natural uranium than for the MOX fuel because of the absence of criticality concerns.

To ensure consistency with the current development plan for the Canadian Used Fuel Disposal Centre, it is assumed that the entire 25 year inventory of spent fuel will be placed into on site, interim dry storage prior to removal to the permanent disposal facility. Interim on site storage would be in the Dry Storage Canisters compatible with the storage modules used in the spent fuel bays.

For the option of permanent disposal of the spent fuel at a U.S. disposal site, it is assumed that the spent fuel will be ready for shipment to a disposal site after 10 years in on site storage. On site interim dry storage would be provided in cylindrical baskets placed in concrete canisters or modules currently licensed for Canadian use. Transportation would be in cylindrical overpacks compatible with the U.S. Multi-Purpose Canister. Costs would include the cost of reusable transport casks and the transfer cost from the storage modules to the transport casks.

The analysis indicated that no significant difference in the total costs for the reference case between the Canadian and U.S options.

Disposal Cost Summary (in 1993 U.S. \$M)

Canadian Disposal
(with 25 years interim
storage on site)

1)	Natural Uranium	319
2)	Reference Mox Bundle	370
3)	Advanced MOX Bundle	215

A breakdown of the components contributing to the total costs is given in Table 6.1.

Table 6.1 Storage, Transportation and Disposal Costs (U.S. \$M)

	Interim Storage Dry Storage Canisters ¹	Transport to Canadian Disposal Facility ²	Canadian Used Fuel Disposal Centre
Natural Uranium	59	65	195
Reference MOX Bundle	70	76	224
Advanced MOX Bundle	41	44	130

¹ Storage of the entire 25 year inventory of spent fuel in interim storage, in dry storage canisters designed to be compatible with the spent fuel modules, prior to transportation to a permanent disposal site. The cost of the modules is included in the canister cost.

² Assumes a cost of U.S. \$75 K per canister.

7.0 Safeguards and Security Overview

7.1 Introduction

Specific safeguards and security requirements necessary for a fuel production facility (chapter 4); fuel packaging and shipping safeguards and security (chapter 5); and Canadian reactor site safeguards (chapter 2), security and inspection regimes have been evaluated. The purpose of this chapter is to provide an overview of the safeguards areas identified (see figure 7.1), and the security areas identified (figure 7.2). The assessment was carried out based on current experience with the transfer of materials from U.S. installations to Canadian facilities and included a review of existing IAEA regulations assigned to the Bruce facility. The approach and decisions made took into consideration the following key objectives:

- the earliest possible start date
- minimization of development work
- systems to be used had to be consistent with both US & Canadian objectives
- minimization of risk associated with both shipping and storage of MOX fuel

The study ensured that the appropriate US, Canadian, and international organizations directly affected by this mission were consulted and had the opportunity to provide comment on the study. The organizations consulted during the course of the study were:

- US DOE, Los Alamos National Laboratories
- US DOE, Transport Safeguards Division, Kirtland AFB
- Operations B, Safeguards, IAEA
- Security Assessments, AECB
- Safeguards group, AECB
- Emergency Response Section, Field Coordination Branch, Ontario Provincial Police (OPP)
- Security, Ontario Hydro BNPD
- various representatives from the Canadian security and law enforcement agencies, such as the Canadian Security Intelligence Service (CSIS), Canada Customs and Immigration, Royal Canadian Mounted Police (RCMP), Department of National Defence (DND), and the Department of Foreign Affairs and International Trade (DFAIT).

7.2 Scope of work

From an overview perspective, the Scope of Work for this topic area included:

- Identification of key safeguards and security (S&S) issues, including those that impact on jurisdiction, transportation and cross-border exchanges of nuclear materials. The review took into consideration current U.S. and Canadian government issues regarding these transfers.
- Development of a preliminary Diversion Path Analysis for this mission.

**FIGURE 7.1
AECL DOE PLUTONIUM
DISPOSITIONING STUDY
SAFEGUARDS OVERVIEW**

7-2

**FIGURE 7.2
AECL DOE PLUTONIUM
DISPOSITIONING STUDY
SECURITY OVERVIEW**

7-3

- Assessment of safeguards and security requirements consistent with the threats and risks currently identified for the MOX fuel production facility.
- Identification of the security, law enforcement, and escort jurisdiction of the transport and facilities for the disposition of the excess weapons-grade Plutonium under the terms of this study.
- Assessment of the safeguards and security requirements consistent with the threats and risks currently identified for the reference reactor facility in which the MOX fuel will be dispositioned.
- Assessment of the safeguards and security requirements for spent fuel interim storage, transport, and ultimate disposal.

In the case of new facilities, safeguards and security requirements were identified consistent with, as a minimum, the current International Atomic Energy Agency (IAEA) guidelines. For both existing and proposed facilities work necessary to implement the appropriate safeguards and security was identified. The study also identified areas where start up or development needs are required, including the scope and cost of such work.

7.3 Preliminary Diversion Path Analysis

This section summarizes the results of a preliminary diversion path analysis (DPA) for the proposed CANDU MOX fuel cycle. A diversion path is a qualitative or quantitative description of the activities performed by a hypothetical insider, or colluding insiders, to remove nuclear material from a facility (theft), transfer the material to an unauthorized location (diversion), or to alter the records of the true location and quantity of nuclear materials in a facility.

The methodology used generally follows that laid out in the "Diversion Path Analysis Handbook" [US DOE Document, November 1978, v1-4]. At this early stage in the CANDU MOX fuel cycle envisaged for the disposition of Plutonium, this discussion presents a very preliminary application of the method. The fuel cycle, from the fuel fabrication stages at Hanford through transportation to, and fresh and spent fuel storage at Bruce, and eventual spent fuel shipment out of Bruce, is covered. The purpose of the analysis is to provide a systematic overview to demonstrate that adequate protection and controls will be in place to limit risk from potential diversion incidents throughout the proposed fuel cycle, and that the controls are consistent with identified plausible threats.

The analysis does not provide specific details of security methods and features (see also section 5.3), as it is intended at this stage to ensure analysis scope completeness, so that no credible diversion paths or preventive measures have been overlooked. Malevolent acts of sabotage are discussed separately in the sabotage vulnerability assessment, as part of the overall threat risk analysis. The diversion path analysis is provided in the form of a spreadsheet (figure 7.3), listing the relevant stages of the fuel cycle operation and diversion path information noted below.

The plausibility of a diversion path is assessed for each of the generic diversion path scenarios, appropriate for each stage of the MOX fuel cycle. The specific fuel cycle stages are:

- MOX process and fuel fabrication line at FMEF Hanford,
- MOX bundle assay and storage at FMEF Hanford,
- Fuel retrieval, loading and monthly transportation from Hanford to Bruce,
- New fuel MOX fuel receiving, assay and storage in Bruce,
- Movement from fresh fuel storage to Bruce reactor fuelling and assay,
- MOX fuel reactor irradiation,
- Spent MOX fuel storage, Bruce,
- Spent fuel shipment from Bruce to final disposal facility.

Two generically-based diversion path scenarios were then defined:

- tampering/alteration of the materials balance accounting system,
- theft, either discrete or bulk, by a single insider or colluding insiders

For each of the fuel cycle stages a detailed assessment will be provided of the adequacy of the various security systems, containment/protection barriers and emergency response systems in terms of relevant alarms, detection time and emergency response time. The preliminary analysis shown on figure 7.3 provides evidence for the security systems and protection barriers that are intended to be in place. Details of these systems would be provided with the more detailed analysis.

7.4 DISCUSSION

To give some feel for the process, an overview "flow" diagram of the safeguards areas can be found in figure 7.1. Feedstock material entering the MOX fuel manufacturing facility represents the highest attractiveness for diversion to a weapons program. Thus, the facility will comply with all pertinent DOE Orders and 10CFR Sections, (mainly Part 74, as well as Part 73). In general, the facility will be divided up into several material balance and accounting areas (plutonium receiving and storage, pellet production, fuel pin manufacturing, fuel bundle assembly, and fuel bundle storage). The record keeping, inventory and accountancies will follow all the US guides and orders. Existing plutonium oxide storage vaults, in combination with physical security measures, will provide adequate defense-in-depth against diversion of this material. The safeguards systems for the original SAFlne facility are now obsolete and will have to be replaced. Several existing facilities internationally have accounting and safeguards systems which can serve as examples for the current products and techniques available. The safeguards system for the Plutonium Fuel Production Facility (PFPPF) in Japan would appear to be the most directly applicable approach to the reference CANDU MOX fuel fabrication facility at Hanford. It is anticipated that only engineering changes will be required for the installation of such a system, and that no research and development costs will be incurred.

Figure 7.3: PRELIMINARY EXAMPLE OF CANDU MOX FUEL CYCLE DIVERSION PATH ANALYSIS

FUEL CYCLE STAGE		HANFORD, MOX PROCESS AND FUEL FABRICATION LINE	HANFORD, BUNDLE ASSAY AND STORAGE	FUEL RETRIEVAL, LOADING AND TRANSPORTATION SHIPMENT TO BNGSA	FRESH FUEL RECEIVING, ASSAY AND STORAGE, BNGSA	BNGSA, FUEL LOADING AND ASSAY PROCESS	REACTOR IRRADIATION	SPENT FUEL STORAGE AT BNGSA	SPENT FUEL SHIPMENT FROM BNGSA TO FINAL STORAGE SITE
DIVERSION PATH									
GENERIC DIVERSION PATH TYPE (PLAUSIBILITY OF SCENARIO AND POTENTIAL IMPACTS)	MATERIALS ACCOUNTING TAMPERING	Insider threat only plausible. Impact on production quantity only.	Insider threat only plausible. Impact on materials accountancy system accuracy.	Insider interference possible during transportation, but threat considered remote	Insider threat plausible. Impact on materials balance accountancy if sufficient quantity involved.	Insider threat only plausible. Impact on fuel management system if sufficient quantity involved or on reactor operation if accidental or deliberate substitute fuel used.	Not credible during irradiation. Impact on fuelling if accidental or deliberate fuel substitution from previous stage.	Storage same as CANDU fuel. Accidental or deliberate storage relocation possible.	Procedures and shipping equipment similar to those used for past shipments of standard CANDU fuel.
	THEFT	Insider threat with or without collusion, possible but unlikely for bulk and discrete quantities.	Insider threat with or without collusion, possible but unlikely	Insider theft possible during transportation, threat considered remote. Discrete theft more likely than bulk quantity.	Insider threat for discrete quantity only plausible.	Insider threat for discrete quantity only plausible.	Not credible	Security available from activity and equipment needed for handling and removal. Storage same as CANDU fuel.	As for above
CONTAINMENT FEATURES/BARRIERS	MATERIALS ACCOUNTING TAMPERING	Materials accountancy system, containment /surveillance systems. Facility management systems. IAEA safeguards.	Materials accountancy system, containment /surveillance systems. Facility management systems. IAEA safeguards.	SST transportation system designed for this threat, in addition to outsider threat.	Materials accountancy, dedicated MOX fuel receiving and storage area /surveillance system. Facility management systems	Insider threat with collusion only plausible due to management systems and operational procedures.	Local reactor power instrumentation and reactivity controls would indicate unexpected deviations from incorrect fuel.	Existing spent fuel bay containment /surveillance IAEA safeguards system.	Procedures and shipping equipment similar to those used for spent CANDU fuel past shipments would be used
	THEFT	Materials accountancy, containment /surveillance system. Facility security system. IAEA safeguards system.	Materials accountancy, containment /surveillance system. Facility security system. IAEA safeguards system.	SST transportation system designed for this threat in addition to outsider threat	Unlikely due to site and facility security.	Unlikely due to site and facility security.	Not credible	Existing containment /surveillance safeguards system. IAEA safeguards system.	As for above

A significant area of concern in carrying out the study was the issue of jurisdiction. If the mission to disposition weapons-grade plutonium as MOX fuel were to be carried out using the FMEF facility at Hanford as the fuel production site, and the reference reactor site for the burning of the MOX fuel were to be the Bruce reactor complex, an international boundary must be crossed. In doing so, the US must relinquish its sovereignty of some aspects of the material transfer. Conversely, Canada must impose its sovereignty over the materials and the processes involved. For example, the US would become an exporter of SNM and Canada the importer. The transporter would fall out of US jurisdiction and becomes subject to Canadian laws and regulations on crossing the border. Such laws and regulation include the need to meet Canadian and Ontario Ministries of Transport regulations regarding status and fire arms, Canadian nuclear regulatory requirements, and armed escort services provided by the most appropriate Canadian organization.

As part of the study, such issues have been reviewed with the various groups involved. While numerous details would have to be worked out, there do not appear to be any significant barriers to carrying out the mission. Indeed, there appeared to be a willingness on all sides to overcome any perceived obstacles.

There has been a significant amount of attention paid to the actual procedures which could be employed to ensure the site security requirements are met for this fuel. The appropriate law enforcement agencies and regulatory agencies have been consulted to arrive at a basic premise for the protocols necessary to carry out the mission.

7.5 Conclusions

The proposed arrangement for plutonium disposition, using a US fuel production facility, located at a DOE site, and burning the fuel in Canada combines the advantages of inherent good security measures at the fuel production site (where SNM of the highest attractiveness is handled) with technical capabilities of CANDU reactors which could operate with full core MOX loading without modifications. All aspects of MOX manufacturing are in the US jurisdiction. After leaving the MOX plant, the plutonium is present only in form of MOX fuel, representing a material of lower strategic significance. Each fuel assembly (bundle) has its own ID and thus it can be tracked by item accounting procedures. Nevertheless, the administrative procedures are supplemented by physical assays at selected points to substantiate the accounting scheme. The spent MOX fuel can either be returned to US for disposal, or sent to a Canadian repository.

All aspects of the disposition scheme will be under national or international safeguards. It is expected that fuel transport and burning in CANDUs will be under IAEA safeguards. The assumed fuel production is at a DOE site but the US government may wish to move that activity under international safeguards as well. The initial IAEA views on the subject were mixed, implementing IAEA safeguards at the MOX fuel production facility would offer a more complete picture but at a significant cost and complexity, neither of which is attractive to the IAEA under current circumstances. Considering that the US is a nuclear weapons state and thus the state interest in diversion is low, the IAEA may leave the MOX fuel production under

US national safeguards and introduce their safeguards regime at the fuel shipping stage. Similarly, the regulatory concern with fuel diversion in Canada is low and the IAEA safeguards are deemed to be adequate for the most part.

The proposed transport arrangements have been discussed with the Transport Safeguards Division at DOE. They represent the best initial alternative from road/rail/air options and can be accommodated by the available existing physical means.

This study has identified a need to have two individuals assigned as a "coordinator" and a back-up for a single point of contact in all transportation, customs, and security communications between the US DOE and the Canadian end of MOX fuel movement operations. The coordinator would integrate the various Canadian organizations which would be essential to the smooth, secure transfer of the materials to the Bruce facility. Such organizations would include OPP, Canada Customs and Immigration, AECB, DND, DFAIT, MOT, RCMP and CSIS.

Separate material balances are verified at the fuel production facility (several material balance areas are defined within the facility), fresh fuel transport, the Bruce site (several material balance areas are defined), spent fuel transport and spent fuel disposal.

In general, the practices, equipment, transport systems, and safeguards concepts currently in use will be applicable to this mission. The existing practices are not necessarily the most cost-effective means of carrying out the mission; however, they are currently acceptable and provide the means to effect an immediate start to the disposition of excess weapons-grade plutonium. Significant savings may be achieved by revisiting the packaging, transportation, and security issues after a start is made. In summary, there are no major impediments regarding the safeguards and security issues associated with this mission.

8.0 Environment, Safety and Health Overview

8.1 Introduction

This section provides an overview of CANDU plutonium enriched, MOX fueled reactor systems from an environment, safety and health (ES&H) point of view. The analysis encompasses the entire life cycle of required design, modification, operation, and decommission of the CANDU plutonium disposition reactor systems and fuel fabrication facility including the packaging, transportation and storage of both new and spent MOX fuel required for the plutonium disposition mission.

The reference CANDU reactor systems for plutonium disposition are located in Canada. The reference MOX fuel fabrication facility is located at Hanford, Washington State. While the intent of protecting the safety and health of workers and public is the same in the U.S. and Canada, there are differences in compliance requirements and processes in the two countries. This study identified compliance considerations that are pertinent depending on the locale of the facility, operation and material. Significant technical topics addressed in the study include:

- system design safety margins
- defense in depth for containing radionuclides
- as low as reasonably achievable occupational exposure (ALARA)
- probabilistic risk assessment
- licensing maturity, and
- waste products management

The following sections present an overview of the ES&H related findings. More detailed results are presented in other chapters of this report depending on the topical area.

8.2 Technical Discussion

8.2.1 System Design Safety Margins

Based on the preliminary analyses conducted in this study, sufficient safety margins exist with a typical CANDU plutonium disposition facility (i.e. modified Bruce-A nuclear station) and a candidate fuel fabrication facility (i.e., modified FMEF) for protecting the safety and health of workers and public. There should not be any significant changes in the design safety margins of the CANDU MOX fueled reactor systems relative to the current natural uranium fueled operations.

Bruce A reactors, in common with other CANDU designs, include four special independent safety systems (see Appendix D and section 2). Two of these are the completely independent shutdown systems. Each is independently capable of shutting down the reactor for accidents exceeding the regulating capability of the reactor control system. A three stage emergency cooling system is provided to cool the reactor should coolant be discharged from the core in a loss of coolant accident. Finally, a unique vacuum containment system is provided. This

system is capable of holding containment pressure sub-atmosphere for a few days following a loss of coolant accidents. Releases from containment are thus eliminated during the period following a LOCA.

The accident analysis methodologies for CANDUs in Canada and LWRs in the U.S. are focused on demonstrating similar safety objectives. An example is the comparable dose limits applied to the most limiting accidents by the licensing regulations in the two countries (see Appendix D). The results of this study indicate that the design safety margins of a CANDU plutonium disposition facility (Bruce-A Station) and its operations should satisfy all pertinent AECB requirements.

The safety margins of a MOX fuel fabrication facility at FMEF will meet applicable U.S., DOE and NRC standards.

8.2.2 Defense in Depth for Containing Radionuclides

Similar to the U.S. LWR systems, multiple barriers exist for containing radionuclides for design basis and severe accidents in CANDU reactor systems. The barriers include fuel cladding, nuclear-grade zirconium-niobium alloy pressure tubes, the calandria, and a containment structure. The containment system includes a highly effective pressure suppression system utilizing a vacuum building fitted with a water spray dousing system. This system returns containment pressure to a sub-atmosphere condition within seconds of any large LOCA. This precludes the release of radionuclides within the first few days following an accident. The overall reactor design concept is consistent with the defense in depth principle.

The same defense in depth principle is applied in the design and operations of the MOX fuel fabrication facility and the packaging/transportation/storage provisions for new and spent fuel produced by the plutonium disposition program. For example, all operations with unconfined PuO_2 are performed in sealed glove boxes.

8.2.3 As Low as Reasonably Achievable (ALARA) Occupational Exposure

The designs and operations of CANDU plutonium disposition reactor systems and the fuel fabrication facility follow the ALARA principle which has been applied for many years at Ontario Hydro's reactor sites. Occupational exposure rates of the workers will be maintained to the current levels of CANDU commercial power stations and are essentially the same levels maintained in the U.S.. The exposure rates will be targeted at less than the pertinent U.S. and Canadian regulatory limits.

8.2.4 Probabilistic Risk Assessment

Separate probabilistic risk assessment (PRA) of commercial nuclear power generating systems is not a licensing requirement in Canada. Instead, probabilistic safety assessment are done as a routine part of design and engineering activities.

AECL has applied the generic risk assessment methodologies in the design of CANDU nuclear systems. These risk analyses are part of the process of establishing the licensing basis for

meeting AECB prescribed dose limits to the workers and the public during plant operation and postulated accident events.

Based on the safety analysis findings of existing commercial CANDU nuclear systems and this preliminary study of MOX fuel and reactor performance, AECL Technologies believes that a MOX fueled CANDU nuclear system will satisfy the safety goals set by AECB for regulatory objectives.

8.2.5 Licensing Maturity

The Bruce-A CANDU power generation station has been operating as an AECB licensed operating facility for over 20 years with routine re-licensing required every two to three years to account for new technology or licensing considerations.. Ontario Hydro will secure AECB approval of necessary modifications to the current licensing certification introduced by the use of plutonium enriched fuel.

Due to the high degree of licensing maturity of the existing CANDU power generating systems, the inherent desirable design features of the FMEF as a fuel fabrication facility, and the conservatism exercised in the CANDU MOX fuel design, corresponding core performance and packaging for handling/transportation, there should not be any major licensing issues which could cause unreasonable delays in seeking necessary regulatory certifications and permits required by the CANDU disposition option.

8.2.6 Waste Products Management

The management of waste products is focused on the spent fuel generated at the plant. In this regard, the quantity of spent fuel generated per unit of electrical power (e.g. MW-Day) in a MOX fueled CANDU station is less than an equivalent natural uranium fueled plant. While the use of MOX fuel introduces a different isotopic distribution transuranics in the spent fuel, the difference is not expected to affect the current handling, storage and eventual disposal requirements of the spent fuel.

The fabrication of the MOX fuel required by the CANDU reactors will produce waste arisings that are inherent to the fuel fabrication process. AECL has evaluated the MOX fuel supply requirements and has placed the minimization of waste products as an important criteria in the facility design and process selection. Quantitative estimates of these waste streams are presented in Section 4 of this report.

8.3 Implementation and Enforcement Strategy

Specific plans for compliance with Canadian and U.S. national environmental policies/regulations, with AECB, U.S. DOE and NRC requirements, and with Provincial and State permitting criteria are described in the detailed sections of this report based on topical area. Elements of activities during the execution phase of a CANDU disposition program include:

- Defined requirements and criteria for protecting safety and health of workers and the public during system design, testing, operation and decommission of the CANDU disposition facility and associated support facilities (fuel fabrication) and operations (fuel transportation and storage).
- Establish requirements compliance oversight authority and function
- Conduct required reviews and compliance documentation.

8.4 Conclusion

Analysis of the reference CANDU system design and a candidate MOX fuel fabrication facility conducted in this study has shown that the ES&H concerns can be properly addressed in accordance with pertinent Canadian and U.S. policies, regulations and standards. Results of this study showed that with appropriate fuel designs, system modifications, automation/shielding/safety provisions, operating procedures and administrative support, the protection of ES&H of the workers and the public can be assured to a high and acceptable standard. ALARA principle will continue to be applied.

While securing the regulatory approvals/certifications of a CANDU nuclear system for disposition and MOX fuel fabrication facility will be a high priority task, the conservatism exercised should address any inherent issue in system design and operation that may cause a unreasonable delay in the licensing approval process.

9.0 Cost and Schedules Overview

Economic evaluations have been made for the reference and alternate cases selected for the plutonium dispositioning study. Based on the marginal costing principle, incremental costs estimates were developed for the supply of MOX fuel to the Bruce reactors, as well as the modification of the reactors to receive, store, irradiate, and transfer the fuel bundles. These costs were developed based on input from engineering analysis of the modifications; safety, safeguards, security, and licensing requirements; fuel design; and the disposition of spent fuels. The evaluations were based on ORNL costing guidelines specifying costing bases and key assumptions; the guidelines were used as a checklist to ensure that all relevant cost aspects were considered.

Figure 9.1 shows the major activities and milestones for the Reference Case that disposes 50 tonnes of plutonium in the Bruce A reactors in 25 years. Similarly, Figures 9.2 and 9.3 show the activities and schedules for the MOX fuel fabrication and supply program, and the physics and fuel start up program in more detail.

9.1 Work Methods and Alternatives Considered

9.1.1 Work Methods

Prior to the commencement of the costing and economic analysis, the following costing and economic guidelines were made available by DOE:

Nuclear Data Requirements and Updated Cost Estimating Guidelines for Phase II Cost Estimates for Reactor Disposition of Weapons Plutonium (Rev.1);

Description of and Information Requirements for Economic Model for Disposition of Weapons Plutonium (Rev. 1);

Cost Estimate Guidelines for Advanced Nuclear Power Technologies, ORNL/TM-10071/R3;

Calculations of Electric Revenues for Evolutionary and Advanced Plutonium Disposition Reactor: Phase II Economic Evaluation, memo attached to April 8, 1994, letter from K. A. Williams, ORNL, to R. J. Neuhold, DOE; and

Weapons Plutonium Disposition Study: Economic Factors for Disposal in Existing Reactors.

9.1.2 Alternatives Considered

The Reference Case considers the disposition of 50 tonnes of plutonium over a 25-year period, and Alternative 1 disposes 100 tonnes over the same period. In addition, the economic implication of using 43-element CANFLEX MOX fuel to disposition 50 and 100 tonnes of plutonium is assessed in.

Fig. 9.1 AECL DOE Plutonium Dispositioning Study Schedule

Fig. 9.2 FUEL AND PHYSICS START UP PROGRAM

Figure 9.3 Program Plan and Schedule for MOX Fuel Supply

9.1.3 Economic and Other Key Parameters

Consistent with the DOE guidelines, the reference currency is January 1993 US dollars. In order to convert current (1994) cost estimates originating from Canada, an exchange rate of US\$1.00 = C\$1.40 has been used, reflecting the prevalent exchange rates in early 1994. To convert these 1994 costs to January 1993, however, they have been de-escalated by 3% to the 1993 level.

Consistent with ORNL cost estimating guidelines, a real interest rate of 5.0% per annum has been used, being the different between the 9.1% nominal debt interest rate for public utilities and the inflation rate of 4.1%.

An economic life of 25 years for the fuel fabrication plant, matching DOE's stated requirement, has been used in this study.

For the purposes of this study, the Bruce A reactors have net outputs of 769 MWe each, and expected average annual capacity factor of 80%.

9.2 MOX Fuel Design, Physics and Testing

The fuel and physics qualification program (see section 2.9) for either the reference or advanced MOX fuel entails the demonstration necessary to qualify the fuel for large-scale implementation in Bruce A. This comprises two main elements:

Confirmation of design. This involves demonstration that the fuel design meets all the requirements for normal operating conditions. This is necessary to obtain utility and the regulatory approval for a demonstration irradiation of prototype MOX bundles in Bruce A. Included would be the irradiation of elements and bundles in the NRU research reactor followed by PIE, to finalize the design features and fabrication processes, and to demonstrate acceptable irradiation behaviour; critical heat flux tests to demonstrate that the design provides acceptable Neutron Overpower Protection (NOP) trip margins; and mechanical integrity tests to demonstrate acceptable in-reactor mechanical performance. The total cost of this part of the program would be \$19.9 million for either the reference or advanced MOX cases, as tabulated below.

Prototype bundle irradiation in Bruce A. A prototype irradiation in a power reactor of about 20-50 bundles will be required before proceeding to a full core implementation of MOX fuel. This irradiation will use fuel fabricated using the same processes as in the full-scale MOX fuel fabrication plant. The cost of this is \$5.2 million for the reference fuels, as tabulated below.

The demonstration program for the reference and advanced MOX designs could be done in parallel. The schedule for qualifying the fuel design would permit the transition to a full core of the reference MOX fuel starting in 1998.

Table 9.1 summarizes the MOX fuel design, physics qualification costs.

Table 9.1 MOX Fuel and Physics Qualification Costs

Reactor physics experiments at ZED-2	0.8
Verification of fuel design in NRU	
Fabricate test bundles	3.9
Irradiate test bundles	8.3
PIE of test bundles	0.7
Planning, management, administration	0.4
Critical heat flux measurements	
Water CHF tests	2.2
Freon tests	0.7
MOX Fuel Characteristics	0.7
Final Design and Specifications	2.2
Prototype bundles irradiation in Bruce A	
Fabricate bundles	3.3
PIE of bundles	1.4
Irradiation program at Bruce A	0.5
Total MOX Fuel Design, Physics, Tests	\$25.1 million

Table 9.6 shows the cost flows for this task.

9.3 MOX Fuel Fabrication and Supply

Tables 9.2 summarizes the construction MOX fuel fabrication facility in Hanford. Table 9.7 shows the corresponding cost flows.

Table 9.2 MOX Fuel Supply Construction and Modification Costs (see Section 4.10)

Site confirmation, Preliminary Design	3.0
Final design of facility modification and equipment	6.2
NEPA, State and local regulations	2.5
Safety analysis report (SAR) preparation	2.0
NRC license fees	5.0
Licensing support	3.0
IAEA safeguards program (excl. equipment)	8.0
Process equipment procurement	19.5
Procurement of glove boxes	4.1
Recycle and Pu finish station	5.1
Equipment installation	4.0
IAEA accountability system (equipment only)	2.0
Facility modification, security, I & C	6.0
Facility modification, wiring and piping	5.6
Engineering & QA	6.0
Checkout, training, startup	13.1
Project management & administration	7.5
Contingency 15%	15.4
Total Construction and Modification Costs	\$117.9 million

9.4 Transportation and Bruce A Modification Costs

The following table summarizes the modification and operation and maintenance costs of the Bruce A reactors for the Reference Case, and the cost flows are shown in Table 9.8.

Table 9.3 Bruce A Modification Costs

AECS licensing of Bruce A for full MOX core	11.4
Provincial EIS for station and transport	0.7
Engineering changes	
Phase 1 - Preliminary design	0.7
Phase 2 - Detailed design	0.8
Phase 3 - Construction and procurement	13.6
Phase 4 - Installation and commissioning	2.3
Design and license and procure new fuel transport package	5.0
Contingencies	2.6
Total Modification Costs for Bruce A	37.1

9.5 Operations and Maintenance Costs

Table 9.4 summarizes the annual O&M costs for the MOX fuel supply program. Table 9.5 contains the incremental annual O&M costs for the reactor and transportation, and also factors in the MOX fuel supply costs, the savings in natural uranium fuel, and the penalty for the reduced volume of natural uranium fuel, and calculates the net annual O&M costs during the program.

Table 9.4 MOX Fuel Supply Operation and Maintenance Costs (see Section 4.10)

Materials and supplies

UO ₂ conversion	4.3
Cladding components	5.1
Dysprosium pins	2.6
Consumables	2.0
Natural uranium end pellets	0.4

Labour, fees, management

Production, direct and indirect	12.0
Management & general & administration	4.0
Waste disposal	3.6
NRC licensing (I & E) fees	2.0
Technology transfer fees	5.0

Facility support services

Landlord and utility costs	10.0
Security and safeguards	10.0
Annual decommissioning and decontamination fund	2.8
MOX fuel supply annual operation and maintenance costs	\$63.9 million

Table 9.5 Incremental O & M Cost of MOX Operation

Additional safeguards and security costs

Fresh MOX fuel packaging	0.6
Fresh MOX fuel transportation	1.5
Fuel receiving and storage	1.6
New fuel loading room	0.2
In-reactor requirement	0.2
Common services and training program	0.5
 MOX fuel supply	 63.9
Savings in natural uranium fuel	(16.3)
Natural uranium fuel supply penalty	1.8
 Total Net Annual Operating and Maintenance Costs	 \$54.0 million

9.6 Summary of Plutonium Dispositioning Costs

In summary, modification and licensing of the FMEF facility at Hanford to fabricate the MOX fuel is estimated at \$118 million. The estimated licensing and modification cost of the two Bruce A reactors for MOX fuel operation including the transport and packaging system is \$37 million. The incremental annual operation and maintenance cost including the extra cost of the MOX fuel, transportation, and increased O&M costs at the Bruce site, and the savings in natural uranium fuel, is \$54 million.

9.7 Alternatives

9.7.1 Alternative 1 (100 tonnes in 25 years)

For Alternative 1, 100 tonnes of plutonium can be dispositioned over a 25-year period in the four Bruce A reactors. The start up cost of the 37 element MOX fuel fabrication plant is estimated at \$200 million.

The estimated licensing and modification cost of the four Bruce A reactors for MOX fuel operation is increased by 50% to \$54 million. The incremental annual operation and maintenance cost increased by 75% to \$92 million.

9.7.2 Alternative 2-Transition to CANFLEX (50 tonnes in 15 years; 100 tonnes in 27 years)

For Alternative 2, 50 tonnes of plutonium can be dispositioned over a 15 year period in four Bruce A reactors, assuming after 5 years, a transition from the 37 element MOX fuel design to the 43-element CANFLEX MOX fuel. The transition cost of the MOX fuel fabrication plant is estimated at \$20 million and the annual operation and maintenance cost increased to \$72 million (see Section 4.10.3).

The estimated additional licensing and modification cost of four Bruce A reactors for the

advanced MOX fuel operation is \$10 to 20 million, and the incremental annual operation and maintenance cost is reduced to about \$50 million per year.

Table 9.6 MOX Fuel and Physics Qualification Program

All costs in 1993 January US\$ - millions

Exchange rate 1.4 C\$/US\$
 Real interest rate 5.0%
 1994/1993 Deflator 1.03

	1994	1995	1996	1997	1998	Total
Confirmatory testing of nuclear performance						
Reactor physics experiments at ZED-2 (including fabrication of 3 test bundles)		0.8				0.8
Verification of fuel design in NRU						
Fabricate 6 test bundles		3.9				3.9
Irradiate 9 test bundles in NRU		3.3	5.0			8.3
PIE of 9 test bundles		0.3	0.3			0.7
Planning, management, administration		0.2	0.2			0.4
Critical heat flux measurement						
Water CHF tests		1.0	1.1			2.2
Freon tests		0.3	0.3			0.7
MOX Fuel Characterization		0.3	0.4			0.7
Mechanical integrity analysis and finalize design		0.3	2.0			2.2
Prototype bundle irradiation in Bruce A						
Fabricate 50 prototype bundles			3.3			3.3
PIE of 10 prototype bundles			1.4			1.4
Irradiation program at Bruce			0.3	0.3		0.5
Total MOX Fuel Design, Physics, and Testing		10.4	14.3	0.3		25.1

(Note: Individual numbers may not sum to totals due to rounding off)

Table 9.7 MOX Fuel Supply Cost
All costs in 1993 January US\$ - millions

Exchange rate 1.4 C\$/US\$
Real interest rate 5.0%
1994/1993 Deflator 1.03

	1994	1995	1996	1997	1998	1999	Total
Construction and Modification Cost							3.0
Site confirmation, conceptual design, firm up plans	0.3	1.3	1.5				6.2
Final design of facility modification and equipment		2.0	3.0	1.2			2.5
NEPA, State and local regulations		0.3	1.3	0.9			2.0
Safety analysis report (SAR) preparation		0.3	1.8				5.0
NRC license fees			1.0	3.0	1.0		3.0
Licensing support			1.0	1.5	0.5		8.0
IAEA safeguards program (excl. equipment)		3.5	4.5				19.5
Process equipment procurement			6.5	13.0			4.1
Procurement of glove boxes				4.1			5.1
Recycle and Pu finish station				5.1			4.0
Equipment installation				4.0			2.0
IAEA accountability system (equipment)			2.0				6.0
Facility modification, security, I & C			2.5	3.5			5.6
Facility modification, wiring and piping			1.9	3.7			6.0
Engineering & QA			3.5	2.5			13.1
Checkout, training, startup				4.4	8.8		7.5
Project management & administration			1.0	3.5	3.0		
Contingency 15%	0.0	1.1	4.7	7.6	2.0		15.4
Total Construction and Modification Cost	0.3	8.4	36.1	57.9	15.2		117.9
Operation and maintenance cost							
Materials and supplies						4.3	
UO2 conversion						5.1	
Cladding components						2.6	
Dysprosium pins						2.0	
Consumables						0.4	
Natural uranium end pellets							
Labor, fees, management						12.0	
Production, direct and indirect						4.0	
Management & general & administration						3.6	
Waste disposal						2.0	
NRC licensing (I & E) fees						5.0	
Technology transfer fees							
Facility support services						10.0	
Landlord and utility costs						10.0	
Security and safeguards						2.8	
Annual decommissioning and decontamination fund							
Total annual operation and maintenance						63.9	

(Note: Individual numbers may not sum to totals due to rounding off)

Table 9.8 Reactor Modifications and Packaging and Transportation Start Up Costs

All costs in 1993 January US\$ - millions

Exchange rate 1.4 C\$/US\$
 Real interest rate 5.0%
 1994/1993 Deflator 1.03

	1994	1995	1996	1997	1998	Total
AECB licensing of Bruce A for Full MOX Core	0.2	4.0	6.0	1.2		11.4
Provincial EIS for station and transport		0.7				0.7
Engineering changes						
Phase 1 - Preliminary design	0.1	0.6				0.7
Phase 2 - Detailed design			0.8			0.8
Phase 3 - Construction and procurement			5.0	8.6		13.6
Phase 4 - Installation and commissioning				2.3		2.3
Contingency 15%	0.0	0.1	0.9	1.6		2.6
Package Design, Licensing and Procure	0.2	0.4	0.4	2.0	0.8	3.8
Transportation Start-up Costs		0.4	0.4	0.5		1.3
Total Modification Costs for Bruce A	0.4	5.3	12.7	13.7		37.1

(Note: Individual numbers may not sum to totals due to rounding off)

10.0 Other CANDU and Fuel Supply Deployment Strategies

Sections 1 through 9 present the results of our analyses of DOE's reference and alternate cases, for burning 50 or 100 tonnes of weapons plutonium in 25 years. The deployment strategy developed for these cases are described in detail and involve fabrication of the fuel at an existing DOE site at Hanford (the FMEF), transporting by road to the Bruce Site in Canada, consuming the fuel over a 25 year program, storing the irradiated fuel at wet and/or dry interim storage for about 10 years at the Bruce site, and then transporting the spent fuel for permanent disposal at either a U.S. or a Canadian geologic repository. This is the basic deployment strategy which is analyzed, based primarily on assumptions specified by the DOE, but also based on our own choice of a reference site in Canada (the existing Bruce Reactor), and a reference site for MOX fabrication in the U.S. (the FMEF).

In performing the detailed study we developed several additional options which are described in detail in the various sections. These include:

- Consumption of much of the plutonium scrap presently in storage at various DOE sites such as Hanford, Savannah River, Rocky flats, and elsewhere. Both the MOX fabrication facility, and the Bruce reactor, are capable of consuming these waste materials. Although this study did not involve an economic analysis of this disposal option, including avoided costs, such an economic analysis can easily be accomplished and would show considerable cost savings to DOE as compared to the present approach of indefinite storage. DOE has estimated that over 20 metric tonnes of plutonium is being stored at present at these sites at considerable annual expense.
- The CANFLEX option is described in detail in chapter 2. At present this option is being developed by AECL for other customers, and is not directed at the MOX weapons plutonium use. One option would be for DOE to support the completion of this development specifically for MOX fuel, in parallel with implementation of the reference case at Bruce. This could take from 4 to 6 years, including full qualification for utility use. Once qualified, the MOX fabrication facility could be switched over from reference fuel to CANFLEX fuel and the rate of utilization of plutonium at Bruce could be doubled. For the 50 ton reference case, this would reduce the time required to utilize the plutonium from 25 to about 15 years, with considerable savings to the U.S. Government. A rough estimate of the cost savings is about 1/2 billion dollars, as compared to the reference use of a 25 year disposal program using only the reference fuel design.

In addition to these two alternative program options, which are presented elsewhere in this report, three new deployment strategies are presented below for DOE's consideration:

10.1 A Joint CANDU Fuel Program with Russia

The 100 tonne alternate case is based on the assumption that about 50 tonnes of Russian weapons plutonium would be shipped to the U.S. for conversion to CANDU fuel and utilized in 2 of the Bruce reactors. While this approach to converting the Russian plutonium into spent fuel would accomplish the goal of reducing the proliferation risk of dismantled nuclear

weapons, it is unlikely to be acceptable to the Russians. It also involves shipment of concentrated plutonium which poses an additional, albeit controllable, proliferation risk. An alternate deployment strategy which should be considered is to convert the Russian weapons plutonium to MOX CANDU fuel in an existing facility in Russia, and then ship it to Canada. In time, this could be a CANDU reactor in Eastern Europe, either one of the existing CANDU reactors in Romania, or a new CANDU reactor which would be constructed by the Russians in Russia. This approach has three main advantages as compared to the reference strategy.

First, converting the plutonium to MOX fuel in Russia would be more acceptable to Russian Government authorities, since it maintains control of the material by the owner country, and involves significant use of Russian nuclear technology, resources and labour. Second, it avoids the shipment of relatively pure plutonium; shipment of CANDU MOX fuel containing only about 2% plutonium would be better from a safeguards and security point of view. And third, it opens up the possibility that Russia would gain energy value from the plutonium in Russia or in Eastern Europe. This had been an important objective of Russian policy.

An attractive approach towards implementing this idea would be to suggest a joint Russian-U.S. Canadian effort on the MOX fabrication plant itself. By involving the Russians in the development of the U.S. MOX facility design, we would begin a technology transfer process which could lead to earlier endorsement of a joint program. This could be followed by a similar, Russian led program to convert an existing Russian fuel facility for CANDU MOX fuel fabrication, and the similarity of the two programs could lead to substantial savings in both programs.

10.2 Single Site Conversion, Fabrication and Consumption

The reference case may require that plutonium pits be shipped once to a pit to oxide conversion facility, and then again to a MOX fabrication facility, and then again to the Bruce site. An alternative deployment strategy with some merit would be to co-locate all these facilities at one site. This could either be the Nevada Test Site or the Hanford Reservation. In either case, a new CANDU reactor would be designed and built at the site for utilization of the CANDU MOX fuel and production of electric energy. AECL Technologies is in the process of licensing a CANDU reactor under 10 CFR52, and expects to have a design certification by about 1999. That in turn could be used as a basis for financing and constructing a CANDU power plant for utilizing the plutonium and producing electric energy. This option has the advantage of total containment of the plutonium at a single site. It has the disadvantage of delaying the start of plutonium consumption for an additional 5 years or more. An economic evaluation of this alternative has not been made, but could be developed should DOE be interested in pursuing this option.

10.3 The Dual Usage Option; Actinide Burning

One of the advantages of using CANDU reactors to dispose of weapons plutonium is that such a program has major side benefits to the U.S. and other countries related to the problem of treatment and disposal of high level nuclear waste. Specifically, the CANDU reactor system is being increasingly considered as a vehicle to treat high level waste and burn many of the actinides in-situ. The idea is very similar to the "annihilation" option presented

elsewhere in this report. Non-uranium matrices are used in the oxide fuel elements to transmute more of the actinides than would be the case with normal uranium or MOX fuel. This results in a major reduction of long lived actinides that must be vitrified or otherwise treated or passivated for long term storage. This fuel cycle has major potential because it does not require reprocessing of the spent fuel and hence is more acceptable from a non-proliferation point of view than other actinide burning options which require reprocessing or pyrometallurgical recycle of spent fuel. The ability of CANDU reactors to shuffle fuel to different parts of the core, on-line, affords a great deal of flexibility to achieve this extra actinide burning. The concept can be applied to existing reactors and does not require development of a new reactor system.

One deployment strategy which could be of interest to the U.S. DOE is to pursue the non-fertile matrix development program for plutonium annihilation, as described in section 2.10, with parallel focus on using that cycle as an actinide burner.

11.0 Conclusions

The United States Department of Energy (DOE) is investigating methods for long term dispositioning of weapons grade plutonium. One such method is to utilize the plutonium in mixed oxide fuel assemblies in existing CANDU (CANadian Deuterium Uranium) reactors. The resulting spent fuel assemblies would be similar to spent fuel assemblies currently produced using natural uranium as a fuel, thus rendering the plutonium in them as equally as unattractive of that in existing stockpiles of spent natural uranium CANDU fuel. This report presented the results of a DOE sponsored study into this method of plutonium dispositioning.

Major conclusions follow:

Reference Reactor: Ontario Hydro's 4 x 769 MW(e) Bruce A generating station was chosen as a reference for the study. This four reactor generating station is located, along with its four unit sister station, Bruce B, on Lake Huron, about 300km north east of Detroit. These, as all CANDU reactors, offer many options for fuel cycles. Bruce A is particularly suited for this mission because of the details of the core design, its base load operating mode, its proximity to the US border, and its existing safeguards and security infrastructure.

Nuclear and Fuel Design Reference Option: The study defined the optimum nuclear and MOX fuel design parameters, within the overall study requirements to disposition 50 tonnes of plutonium in 25 years, and the alternate study case of 100 tonnes in 25 years. An optimum nuclear and fuel design utilizes a standard CANDU fuel bundle which contains 37 pencils or elements. The outer 30 elements contain MOX fuel composed of plutonium and depleted uranium oxide. The inner 7 elements contain a mixture of depleted uranium oxide, and a burnable poison, dysprosium oxide, which compensates for the extra reactivity introduced by the plutonium. This fuel would operate within the same burnup and power rating envelope as standard CANDU fuel, and would have nuclear parameters that allow the reactor to operate within its existing licensing envelope. The study team evaluating fuel supply concluded that the required quantities could be fabricated at the existing Fuel and Materials Examination Facility (FMEF) at the Hanford reservation. Two reactors would be used for the reference case, and four for the alternate case of dispositioning of 100 tonnes in 25 years.

Nuclear and Fuel Design CANFLEX Option: An alternate fuel design utilizing the 43 element CANFLEX fuel bundle was also studied. This fuel bundle, which is currently being qualified for commercial use, has smaller diameter elements which operate at a lower linear power rating, thus permitting higher plutonium concentrations and higher burnups. Use of this design would reduce the quantity of required MOX fuel bundles by almost half, a significant economic advantage. An attractive alternative may be to launch a 2 phase program beginning with the standard CANDU fuel bundle which would allow for earliest implementation, and switch to the CANFLEX bundle once it has finished commercial qualification.

Safeguards and Security: The goal was to maximize security in fabrication, transport,

and receipt at Bruce, and to keep the number of sites used to produce the MOX fuel and to disposition the plutonium to a minimum. Methods of transportation had to be proven and reliable. For the reference case, one site will be used to fabricate the fuel, and one site will be used to dispose it. Both road and rail transportation are already routinely used to ship special nuclear materials in the U.S. For the purposes of this study road shipment in existing shipping containers and vehicles was used to evaluate the number of trips and the cost. CANDU reactors are fuelled on power and would require about one shipment of fuel per month for the reference case.

Environment, Safety, and Health (ES&H): There had to be no adverse, and preferably a positive impact on any ES&H concerns. The study compared the current fuel cycle using natural uranium to a MOX cycle which disposes 2 tonnes per year of plutonium. Instead of mining and refining over 6000 tonnes uranium ore per year, the MOX fuel cycle consumes 2 tonnes of plutonium and close to 250 tonnes of depleted uranium waste. The resulting quantity of spent fuel is actually about 10% less, due to the slightly higher average burnup which can be achieved with the full MOX core. Fuel handling and packaging methods have been developed to ensure that radiation exposure to power plant workers is equal to or less than the current levels. The fuel fabrication facility design and operation will be licensed by the US Nuclear Regulatory Commission.

Economics, Schedule, and Risk: The objective was to develop a low cost, low risk method for plutonium dispositioning which, in response to the "clear and present danger"¹ indicated by the National Academy of Sciences, could be implemented as quickly as possible. This has been achieved by designing the fuel and nuclear performance to operate within current experience including safety and licensing envelopes, by limiting changes to the existing reactors to provision for storage and handling of the new fuel, and by utilizing present infrastructures for fuel fabrication and transportation. Costs were broken out into: Qualification of the fuel, and confirmation of the nuclear design parameters (\$25M)², modifications to the existing reactor for handling and storage of MOX fuel, including licensing, and design and qualification of packaging and transportation (\$37M), modifications and licensing of FMEF to produce CANDU MOX fuel to disposition 50 tonnes of plutonium in 25 years (\$118M), and the incremental operation and maintenance cost for producing MOX fuel, transportation, and dispositioning at the reactc. site of \$54M per year for the duration of the program.

Plutonium Annihilation: Options for a more complete destruction or annihilation of the plutonium were also briefly studied. The CANDU reactor has a superior ability to annihilate plutonium and the minor transuranic actinides. This ability of CANDU is related to the basic design features of using heavy water as coolant and a moderator leading to low neutron absorption rates in the CANDU lattice. For plutonium

¹See the National Academy of Sciences Report Management and Dispositioning of Excess Weapons Plutonium, p. 1, National Academy Press, Washington D.C., 1994.

²All costs in this report are in 1993 US dollars.

annihilation, the fuel consists of a mixture of the plutonium isotopes in a non-fertile matrix. The absence of uranium 238 eliminates the source for further creation of plutonium and over 40% of the neutron absorption in the lattice, resulting in a remarkable improvement in neutron economy. The absence of the neutron absorption reduces the fissile requirement of the plutonium annihilator relative to the natural uranium fuelled core. A lower fissile inventory requires a correspondingly higher operating neutron flux level to produce the rated power which is the basis for the CANDU superiority in the annihilation process. This superiority is evident from the high (> 80%) annihilation rate that is achievable per pass through the reactor. By using the on-power fuelling machines to "shuffle" the fuel through one or two additional passes, essentially all of the plutonium may be annihilated. While this option would require substantially more development than the MOX option, it resolves the question of ultimate disposal without requiring highly advanced technology or reprocessing.

Deployment Options: Alternative deployment options were explored and the potential impact on plutonium dispositioning in the former Soviet Union was studied. For example one such alternative would be to set up a parallel CANDU MOX fuel fabrication facility in Russia. The fuel would either be shipped to Canada or utilized in existing or new CANDUs in Europe and Asia. There is potential for many flexible and attractive arrangements. Another alternative deployment option is to incorporate much of the scrap plutonium currently stored as waste at many DOE sites (> 20 tonnes) into the process stream at the MOX fuel fabrication facility. The proposed MOX process has incorporated an oversized scrap recycle system to permit this to occur at a throughput rate in addition to the disposition of the weapons plutonium.

In summary, this study performed for the US DOE has identified practical and safe options for the consumption of 50 tonnes and an alternate of 100 tonnes of plutonium in 25 years in existing CANDU reactors. By designing the fuel and nuclear performance to operate within existing experience and safety and licensing envelopes, and by utilizing existing fuel fabrication and transportation facilities and methods, a low cost, low risk method for long term plutonium dispositioning has been developed. The integrated system can be ready to begin plutonium consumption in 4 years. No changes to the existing reactor system, other than for provision of safe and secure storage of new fuel are required. Alternative deployment strategies for the reactors, fuel fabrication, and transportation have been investigated. An annihilation option which uses the unique features of the CANDU system to achieve high levels of utilization of the plutonium offers an attractive option for ultimate disposition without requiring reprocessing of the spent fuel or advanced technology.

11.1 Advantages and Challenges

The DOE requested that AECL identify, in its opinion, the top advantages and challenges to dispositioning of weapons grade plutonium using CANDU technology. These are presented below:

Challenges

Public Acceptance: Acceptance by the public may be difficult due to the perceived dangers of plutonium. Special interest groups from both pro and anti nuclear organizations may not wish to see a dispositioning scheme involving an international arrangement. Although the plan presented in this report calls for dismantling of the weapons and the conversion of the plutonium into dilute oxides incorporated into CANDU fuel bundles by the country of origin, there still may be reluctance by the Canadian public to receive this material, and by the country of origin to ship it to Canada.

Jurisdictional and Regulatory Approvals: Within Canada and the U.S. there will be many levels of government approvals required, and many processes and regulations to satisfy. The challenge here is to develop an acceptable and coordinated program to satisfy all of the requirements in a timely fashion so that dispositioning of the material can begin as soon as possible.

Developing the Annihilation Option: Only a small amount of work has been done to investigate the performance of fuels in a non fertile matrix. A major development and qualification program would be required to satisfy a utility that the special fuel would perform as reliably as existing fuels. Dedicated reactors may have to be built, as a commercial utility may not find this approach acceptable

Advantages

Global Acceptance: Dispositioning of weapons plutonium in existing CANDU reactors offers a high degree of visibility and transparency to all nuclear and non nuclear states. Canada is known for its peacekeeping role and has always promoted the use of nuclear technology for peaceful purposes.

Bilateral Participation: This method of dispositioning offers the opportunity for equal participation by both the United States and the former Soviet Union states to eliminate the threat of diversion or re-construction of its weapons materials. Both countries could minimize the direct costs of dispositioning by building parallel fuel fabrication facilities, and there may be opportunities by both groups for regional or macro economic benefits.

Utility and Private Sector Participation: Ontario Hydro, one of the world's largest electrical utilities and the largest owner of CANDU reactors has indicated an interest in participating in this program. This interest has also extended to other private sector companies, including major suppliers of CANDU fuel.

CANDU Technology Maturity: No modifications, other than those associated with safe, secure handling of the new fuel are required at the reactor site. The reactor can utilize MOX fuel in the same way as the current natural uranium fuel. There is no shutdown period required for conversion of the reactors to use MOX fuel; once the facilities and licensing process are complete, the fuel can start to be loaded into the on power fuelling system.

Annihilation without Reprocessing: Briefly studied during this program was an option to completely fission the plutonium, by incorporating it into a special fuel design which contained no fertile material. Utilizing the unique characteristics of CANDU's heavy water moderated reactor, and on-power fuelling it is possible to achieve very high levels of plutonium destruction by passing the fuel bundles through the core two or three times. This offers a method for ultimate dispositioning of the plutonium without encouraging fuel reprocessing technologies or entailing any of the environment safety and health risks associated with it.

APPENDIX A

Uniform Descriptive Technical Data (Existing Reactor)

Lawrence Livermore National Laboratory

	Reactor Plant Identification:
	I. Plant Retrofit and Operations Schedule (relative to 10/1/94 project start):
48	1) time MOX burning operations start (mo)
2 x 2832	2) thermal power (MWth)
2 x 850	3) net equivalent electrical output (MWe)
80%	4) capacity factor used in study
N/A	5) remaining plant life (y)
	II. Fresh MOX Fuel Bundle Characteristics:
1,448 (*)	1) reactor Pu inventory - first core (kg Pu)
N/A (**)	2) reactor U inventory - first core (kg U)
0.232	3) Pu content (kg Pu/fuel bundle)
0	4) Pu-238 fraction (KG Pu-238/kg Pu)
0.9402	5) Pu-239 fraction (KG Pu-239/kg Pu)
0.0582	6) Pu-240 fraction (KG Pu-240/kg Pu)
0.0014	7) Pu-241 fraction (KG Pu-241/kg Pu)
0.0002	8) Pu-242 fraction (KG Pu-242/kg Pu)
21.33	9) fuel bundle mass (kg)
0.495	10) fuel bundle length (m)
0.102	11) fuel bundle diameter (m)
18.4	12) U-mass in MOX (kg U/fuel bundle)
0.2%	13) U-235 enrichment in MOX (kg U-235/kg U)
N/A	14) U-mass in UO2 fuel (kg U/fuel bundle)
N/A	15) U-235 enrichment in UO2 fuel (kg U-235/kgU)

* this is based on total Pu in 6240 fuel bundles which is not meaningful since Bruce A will not be operated with a fresh Pu-MOX core

** this is not applicable in Bruce A since on-power refuelling will be used to insert Pu-MOX fuel bundles to displace the existing natural fuel bundles

	III. Equilibrium MOX Fuel Characteristics :
3.6 kg Pu (***)	1) Pu recharge (kg Pu)
N/A (****)	2) U recharge (kg U)
0.1828	3) Pu content (kg Pu/fuel bundle)
0.0002	4) Pu-238 fraction (KG Pu-238/kg Pu)
0.7181	5) Pu-239 fraction (KG Pu-239/kg Pu)
0.2390	6) Pu-240 fraction (KG Pu-240/kg Pu)
0.0384	7) Pu-241 fraction (KG Pu-241/kg Pu)
0.0044	8) Pu-242 fraction (KG Pu-242/kg Pu)
18.4	9) U-mass in MOX (kg U/fuel bundle)
0.2%	10) U-235 enrichment in MOX (kg U-235/kg U)
N/A (****)	11) U-mass in UO ₂ fuel (kg U/fuel bundle)
N/A (****)	12) U-235 enrichment in UO ₂ fuel (kg U-235/kgU)
6240	13) number of MOX fuel bundles in reactor
15.5	14) MOX bundles used per full power day
	IV. MOX Fuel Bundle Characteristics at discharge:
0.1535	1) Pu mass per bundle in MOX fuel elements (kg Pu)
N/A (****)	2) Pu mass per bundle in UO ₂ fuel elements (kg Pu)
0.0008	3) Pu-238 fraction in MOX fuel (kg Pu-238/kg Pu)
0.5134	4) Pu-239 fraction in MOX fuel (kg Pu-239/kg Pu)
0.3753	5) Pu-240 fraction in MOX fuel (kg Pu-240/kg Pu)
0.0860	6) Pu-241 fraction in MOX fuel (kg Pu-241/kg Pu)
0.0244	7) Pu-242 fraction in MOX fuel (kg Pu-242/kg Pu)
N/A (****)	8) Pu-238 fraction in UO ₂ fuel (kg Pu-238/kg Pu)
N/A (****)	9) Pu-239 fraction in UO ₂ fuel (kg Pu-239/kg Pu)
N/A (****)	10) Pu-240 fraction in UO ₂ fuel (kg Pu-240/kg Pu)
N/A (****)	11) Pu-241 fraction in UO ₂ fuel (kg Pu-241/kg Pu)
N/A (****)	12) Pu-242 fraction in UO ₂ fuel (kg Pu-242/kg Pu)
25,200	13) Heat generation in one fuel bundle (W)
3.25E+07	14) gamma radiation field (rem/h at surface, midplane of fuel bundle)
360	15) average full power days (FPD) in reactor
9.7	16) burnup (MWd/kgHM)

*** 15.5 bundles per full power day

**** only Pu-MOX bundles will be used

	V. MOX Fuel Characteristics one year after Discharge:
0.1542	1) Pu mass in MOX fuel elements (kg Pu)
N/A (****)	2) Pu mass in UO ₂ fuel elements (kg Pu)
0.0016	3) Pu-238 fraction in MOX fuel (kg Pu-238/kg Pu)
0.5056	4) Pu-239 fraction in MOX fuel (kg Pu-239/kg Pu)
0.3855	5) Pu-240 fraction in MOX fuel (kg Pu-240/kg Pu)
0.0826	6) Pu-241 fraction in MOX fuel (kg Pu-241/kg Pu)
0.0248	7) Pu-242 fraction in MOX fuel (kg Pu-242/kg Pu)
N/A (****)	8) Pu-238 fraction in UO ₂ fuel (kg Pu-238/kg Pu)
N/A (****)	9) Pu-239 fraction in UO ₂ fuel (kg Pu-239/kg Pu)
N/A (****)	10) Pu-240 fraction in UO ₂ fuel (kg Pu-240/kg Pu)
N/A (****)	11) Pu-241 fraction in UO ₂ fuel (kg Pu-241/kg Pu)
N/A (****)	12) Pu-242 fraction in UO ₂ fuel (kg Pu-242/kg Pu)
86.9	13) heat generation in one fuel bundle (W)
6.26E+04	14) gamma radiation field (rem/h at surface, midplane of fuel bundle)
360	15) average full power days (FPD) in reactor
9.7	16) burnup (MWd/kgHM)
0.0083	17) fraction of Pu in heavy metal (kg Pu/kgHM)
	VI. MOX Fuel Characteristics ten years after Discharge:
0.1498	1) Pu mass in MOX fuel bundle (kg Pu)
N/A (****)	2) Pu mass in UO ₂ fuel (kg Pu)
0.0017	3) Pu-238 fraction in MOX fuel (kg Pu-238/kg Pu)
0.5204	4) Pu-239 fraction in MOX fuel (kg Pu-239/kg Pu)
0.3968	5) Pu-240 fraction in MOX fuel (kg Pu-240/kg Pu)
0.0556	6) Pu-241 fraction in MOX fuel (kg Pu-241/kg Pu)
0.0255	7) Pu-242 fraction in MOX fuel (kg Pu-242/kg Pu)
N/A (****)	8) Pu-238 fraction in UO ₂ fuel (kg Pu-238/kg Pu)
N/A (****)	9) Pu-239 fraction in UO ₂ fuel (kg Pu-239/kg Pu)
N/A (****)	10) Pu-240 fraction in UO ₂ fuel (kg Pu-240/kg Pu)
N/A (****)	11) Pu-241 fraction in UO ₂ fuel (kg Pu-241/kg Pu)
N/A (****)	12) Pu-242 fraction in UO ₂ fuel (kg Pu-242/kg Pu)
5.26	13) Heat generation in one fuel assembly (W)
6.05E+03	14) gamma radiation field (rem/h at surface, midplane of fuel bundle)
360	15) average full power days (FPD)
9.7	16) burnup (MWd/kgHM)
0.0080	17) fraction of Pu in heavy metal (kg Pu/kgHM)

	VII. MOX Fuel Characteristics One Hundred Years After Discharge:
0.1405	1) Pu mass in MOX fuel (kg Pu)
N/A (****)	2) Pu mass in UO ₂ fuel (kg Pu)
0.0009	3) Pu-238 fraction in MOX fuel (kg Pu-238/kg Pu)
0.5528	4) Pu-239 fraction in MOX fuel (kg Pu-239/kg Pu)
0.4183	5) Pu-240 fraction in MOX fuel (kg Pu-240/kg Pu)
0.0009	6) Pu-241 fraction in MOX fuel (kg Pu-241/kg Pu)
0.0272	7) Pu-242 fraction in MOX fuel (kg Pu-242/kg Pu)
N/A (****)	8) Pu-238 fraction in UO ₂ fuel (kg Pu-238/kg Pu)
N/A (****)	9) Pu-239 fraction in UO ₂ fuel (kg Pu-239/kg Pu)
N/A (****)	10) Pu-240 fraction in UO ₂ fuel (kg Pu-240/kg Pu)
N/A (****)	11) Pu-241 fraction in UO ₂ fuel (kg Pu-241/kg Pu)
N/A (****)	12) Pu-242 fraction in UO ₂ fuel (kg Pu-242/kg Pu)
2.47	13) heat generation in one fuel assembly (W)
6.80E + 02	14) gamma radiation field (rem/h at surface, midplane of fuel bundle)
360	15) average full power days (FPD) in reactor
9.7	16) burnup (MWd/kgHM)
0.0075	17) fraction of Pu in heavy metal (kg Pu/kgHM)

**** only Pu-MOX bundles will be used

	VIII. Pu Inventory and Handling at Fuel Fabrication
	Facility (for identified reactor) (Reference,
	Management and Disposition of Excess
	Weapons Plutonium, National Academy of
	Sciences, 1994, Table 3.1)
C	1) DOE attractiveness of level of PuO ₂ feed material
C	2) DOE attractiveness level of MOX fuel pellets
1,000	3) average inventory of PuO ₂ feed material (kg Pu)
2,100	4) throughput of PuO ₂ feed material (kg Pu/y)
0.5	5) PuO ₂ feed material residence time or duration (y)
3 to 10	6) number of manual movements of materials (e. g. between gloveboxes)
< 50	7) average inventory of MO fuel pellets (kg Pu)
2,100	8) throughput MO fuel pellets (kg Pu/y)
3 days	9) MOX fuel pellet residence time or duration (y)
0 to 3	10) number of manual movements of MOX fuel pellets (e.g. between gloveboxes)
700	11) average inventory of MOX fuel assemblies (kg Pu)
0.1 year	12) MOX fuel assembly residence time or duration prior to fuel load (y)
3	13) number of movements of MOX fuel assemblies
120	14) staffing level of MOX fuel fabrication facility (number of persons)
240,000	15) effort at MO fuel fabrication facility (person-h/y)

	IX. Environmental, Safety and Health
no	1) requires Am removal (respond with "y" or "n")
<50/year	2) dose for fuel fab operations and intrasite transport (person-rem)
170/years	3) dose for reactor operations (person-rem)
260	4) low level waste generated for fuel fab operations (m ³)
12,000	5) low level waste generated for reactor operations during Pu disposition (m ³)
none	6) mixed waste generated for fuel fab operations (m ³)
1875	7) mixed waste generated for reactor operations during Pu disposition (m ³)
220,000**	8) spent fuel generated (number of spent MOX fuel assemblies)
1,720	9) TRU waste generated, excluding packaging (m ³)
172,000	10) TRU waste generated, excluding packaging (kg)
2 to 3	11) number of spent fuel movements to end of disposition operations (including last ten years)
	X. Other Measures
yes	1) physical protection system meets standards in DOE Orders (respond with "y" or "n")
yes	2) material control and accountability system meets standards in DOE Orders ("y" or "n")
yes	3) is IAEA monitoring practical (i.e), nothing to prohibit monitoring) ("y" or "n")
0	4) number of reactors of similar design under construction in the former republics of the USSR
0	5) number of reactors of similar design in operation in the former republics of the USSR

** CANDU fuel bundle (4" Dia, 20" Long each)

APPENDIX B

Summary of Nuclear Engineering Codes

Listed below are brief summaries of nuclear engineering codes used to evaluate reactor performance and fuel cycle parameters.

1. Reactor Physics

- A. POINTSIM (*POINT SIMulation*) is a point-kinetics code used to simulate power transients. The system reactivity is an input to the code and is written as the sum of components: the first corresponding to coolant void reactivity as a function of time, the second corresponding to the action of the shutdown system versus the degree of insertion (following its actuation) and a third term for the fuel-temperature feedback.

The code also simulates the response of neutron flux detectors and the associated electronics and trip logic. This is used to determine reactor trip time.
- B. ORIGIN (*Oak Ridge Isotope GENeration*) is an isotope generation and depletion code used to determine inventories of radionuclides. It calculates the buildup, decay and processing of radionuclides.
- C. WIMS (*Winfrith Improved Multigroup Scheme*) is a multigroup transport code for reactor lattice calculations, including burnup. It was created at the United Kingdom Atomic Energy Establishment, Winfrith, Dorset (AEEW), where it continues to evolve. The code has also been widely distributed and exists in various versions and many countries.
- D. RFSP - The computer code RFSP performs fuel management calculations for CANDU reactors by calculating the neutron flux and power distributions using two group, three dimensional neutron diffusion theory. The code is capable of calculating the time average and instantaneous flux distributions and of simulating reactor operations including refueling and burnup steps.
- E. MCNP4A - This is a general purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate eigenvalues for critical systems. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori.
- F. ORIGIN-S - The code computes time-dependent concentrations and source terms of a large number of isotopes which are simultaneously generated or depleted through neutronic transmutation, fission, radioactive decay, input feed rates, and physical or chemical removal rates.

- G. QAD-CGGP - The code uses a point kernel ray-tracing technique for gamma-ray calculations and either a SP modified Albert-Welton kernel or kernels obtained from the moments method solution of the Boltzmann equation for neutron penetration calculations.

2. Thermal Hydraulics

- A. NUCIRC is a steady-state thermal-hydraulic model used to design and analyze the primary heat transport system of a CANDU reactor for normal operating conditions.

The model simulates half of a primary heat transport system loop with one pump, one boiler, one reactor inlet header and one reactor outlet header. The core can be represented by the actual number of channels in the reactor. The circuit model uses a weighted average channel representation of the core.

In support of development, an extensive series of experiments has been undertaken to provide accurate hydraulic loss correlations for primary heat transport system headers, fuel bundles, orifices and end fittings.

In addition to its application to establish initial conditions for accident analysis, the code has been used to tune the geometric parameters of average channels used to simulate several parallel channels in models used with the other computer codes.

- B. ASSERT (Advanced Solution of Subchannel Equations in Reactor Thermalhydraulics) code is developed to address the computation of flow and phase distribution within the horizontal subchannels of CANDU PHWR fuel channels and to provide a detailed prediction of critical heat flux (CHF) distribution throughout the fuel bundle.

The ASSERT computer program is the result of a number of coordinated experimental and analytical projects, each providing information essential for the code modeling.

The code uses the subchannel approach similar to the development of the U.S. COBRA-IV computer code. Subchannels are defined by the coolant flow areas between rods, bounded by the rods and imaginary planes linking adjacent rod center lines. Subchannels are divided axially into control volumes, which communicate axially in the same subchannel, and laterally across fictitious boundaries with control volumes in neighboring subchannels.

- C. TUF (Two Unequal Fluids) is an advanced two-fluid system thermalhydraulics used to simulate CANDU heat transport system, secondary steam supply system and plant process and control systems for both operational transient and accident conditions.

The code solves liquid phase and two-phase mixture mass, energy and momentum equations within a centrally differenced nodalized representation of thermalhydraulic networks. Constitutive relationships are employed for closure of the set of equations by specifying flow regime dependent interfacial relationships between the liquid and vapor phases. Specific models are incorporated for major components such as heat transport system pumps, reactor end-fittings, pressurizer and steam generators. Detailed emulation of the control systems (process control and the direct digital controllers that provide reactor and overall plant control) and safety systems (the two shutdown systems and emergency coolant injection system) of Ontario Hydro's nuclear generating stations is incorporated in the code.

3. Fuel and High Temperature Fuel Channel Behavior

- A. ELESIM (ELEment SIMulation) is a digital computer code used to simulate fuel element thermomechanical behavior under normal operating conditions. The full element is considered to be an axi-symmetric cylinder. The code calculates: temperatures from the center of the fuel to the surface of the sheath, sheath strain, fission product gas release and the fuel-to-sheath gap.

The calculations are based on the following well established works.

- The fuel-to-sheath heat transfer coefficient is calculated based on test results (Campbell, 1977).
- MATPRO-9 (MacDonald, 1976) fuel thermal conductivity is used except at low temperatures where it is held constant to simulate the effect of irradiation damage.
- MATPRO-9 fuel thermal expansion is also used.
- Flux depression within the element is based on results obtained with the HAMMER code (Suich, 1967).
- Stable fission product gas releases to the gap are based on a physically realistic model, adequately verified by experiment, which considered diffusion to grain boundaries, bubble precipitation and interlinkage and release by fuel cracking (Notley, 1980).
- The gap inventory of radioactive fission products is calculated using the empirical approach of the ANS 5.4 Standard (USNRC, 1982 January).
- Sheath deformation is based on work done at AECL (Coleman, 1976, Holt, 1976, Hosbons, 1975).

- B. ELESTRES (ELEment STRESSs) is an extension of the ELESIM computer code based on the use of finite element modelling techniques. Its major application is in the modelling of normal operating conditions. It too is capable of establishing initial conditions for accident analysis.

APPENDIX C

Supplementary Data: Safety Analysis and Licensing

C.1 INTRODUCTION

This appendix provides the following supporting information.

- a. A description of salient features of CANDU reactors relevant to safety analysis and assessment.
- b. A brief introduction to Canadian reactor safety and licensing practice to establish differences from U.S. NRC requirements in the context of U.S. licensing practice.
- c. Identification and evaluation of safety issues arising from the introduction of plutonium into fuel at enrichments beyond natural uranium fissile content.

C.1.1 THE CANDU REACTOR

A brief description of CANDU reactor design is presented to provide context for discussion of unique aspects of accident analysis methods developed in Canada.

The most distinctive component of the CANDU reactor is the reactor vessel, referred to as the calandria vessel. This is a large horizontal cylindrical tank with flat ends containing several hundred small diameter tubes (calandria tubes) penetrating the flat ends and arranged parallel to the axis of the cylindrical shell. The tubes are fastened, with leak tight joints to the flat ends of the large tank. The calandria vessel contains recirculating cooled heavy water moderator at close to atmospheric pressure.

The reactor primary heat transport system is comprised of a parallel array of pressure tubes located within the calandria tubes and connected through reactor end-fittings to small diameter feeder pipes which, in turn, connect to large diameter reactor headers. The headers connect by large diameter piping to steam generators and heat transport pumps. Two passes of fuel channels, headers, steam generators and pumps are arranged in a series connected network, referred to as a "figure-of-eight" loop. This arrangement provides some redundancy of pumping and cooling capability.

The outer surface of the pressure tube is separated from the inner surface of the surrounding calandria tube by a gas filled annulus which limits heat losses from the hot heat transport fluid to the cooler moderator fluid. A pressure tube, calandria tube, the associated supporting internal structures and end-fittings is referred to as a fuel channel assembly.

The fuel is arranged in short bundles, consisting of small diameter, Zircaloy-4 clad elements containing uranium dioxide fuel pellets. Depending upon the reactor type, either twelve or thirteen bundles are located within each fuel channel. Fuelling machines are provided to load new fuel bundles and remove spent fuel bundles from the channels with the reactor operating at-power. Design variants of CANDU fuelling systems are based on either using one machine connected to one end of a fuel channel with fuel movement due to coolant flow, or using two machines, one inserting new fuel from one end of a channel and the other receiving spent fuel discharged from the other end.

Two independent reactor shutdown systems are provided. These access the reactor core through tubes penetrating the reactor from the top and sides. Shutdown System 1 utilizes neutron absorber shut-off rod units which are dropped, with spring-assisted acceleration, into the core from above. Shutdown System 2 utilizes a solution of neutron absorbing material in heavy water which is injected through horizontally oriented nozzles into the moderator by high pressure gas.

CANDU reactors, in common with other reactor types, are fitted with emergency core cooling systems and containment systems. The emergency cooling systems are typically comprised of three subsystems which come into operation at high, intermediate, and low pressure as coolant is discharged from the core following a loss of coolant accident.

Containment systems for CANDU reactors are either of the multi-unit reactor design with a vacuum building providing pressure suppression and water dousing, or single unit systems which rely on a reinforced concrete containment vessel without pressure suppression.

Other systems of importance in accident analysis include the steam supply system, the reactor regulating system and the fuel handling system. All of these systems are described and illustrated in general detail elsewhere¹.

The foregoing discussion is essentially generic to all CANDU reactors. The Bruce A reactors, which are the reference reactors for this study, are described in detail in section 2.1 of the main report.

C.1.2 LICENSING AND SAFETY ANALYSIS PRACTICE

In Canada and the United States, the assessment of reactor safety in design and operation proceeds through the application of broadly similar licensing processes which do have some significant differences at the detailed application level. Utilities and designers operate within a set of federal government prescribed guidelines which have been developed and refined over many decades. Safety analyses and operational practice are documented and submitted by the utilities to federal government regulatory agencies for review (the NRC in the case of the U.S., and the AECB in the case of Canada). These submissions are then reviewed with respect to regulatory requirements and established practice prior to the granting of an operating license by the regulatory authority.

Within this broad framework there are differences in the way that licensing requirements are applied. The regulations and guidelines in the U.S. include substantial detail. The Canadian requirements have traditionally been much less detailed, but tend to be all-encompassing through the focus on assuring demonstrable independence and separation between process system functions and special safety system functions.

C.1.2.1 Overall Safety Objectives

The key nuclear safety objectives, which dictate the scope and content of safety analysis and assessments, relate to assuring acceptable reactor shutdown capability, effective cooling of the fuel, adequate transport and rejection of heat from the reactor, and effective containment of radioactive material released from the reactor and associated systems. Underlying these objectives is the requirement that the design meet appropriate quality and reliability standards to assure that failure frequencies are maintained acceptably low.

For all reactors the need to use high quality design standards has concentrated on the design of the pressure boundary of systems carrying radioactive, or potentially radioactive, fluid. The ASME Code, Section III was developed in the United States for this purpose and has been used, generally modified and supplemented by national standards, by most other countries, including Canada.

The CANDU reactor system does have unique features and components which have resulted in the development of supplementary standards in Canada. The Canadian Standards Association (CSA) has issued a series of standards for design, materials, fabrication and inspection of pressure retaining components which, for the most part, utilize the rules of the ASME Code, Section III, by directly referring to them. Special standards, such as those on periodic inspection, have been specifically generated for application to the CANDU system. Other standards provide requirements for special components of the CANDU reactor whose characteristics are not covered by the ASME Code, for example, pressure tubes.

The light water reactor approach to reliability for safety is based on a deterministic single failure criterion set in U.S. regulations and standards. This criterion requires that the plant be capable of responding to anticipated transients and accidents, assuming the independent failure of an active component in a safety system. Specifically, the design of each safety system must allow for the single failure of an active component at any time following an initiating event and (but not together) for the single failure of a passive component during the long-term recovery period. Compliance with the criterion is generally achieved by providing one system for each of a set of safety functions, with redundancy of active and passive components built within the system to the extent required by the criterion.

In Canadian safety practice reliability objectives are realized by providing independent diversified systems (either safety systems or combinations of process and safety systems) that meet separation of function requirements, and which fulfil each of the required safety functions. Every safety system and certain important process systems are provided with redundancy of active components. Safety systems are also designed to be readily

testable during normal operation to demonstrate an unavailability less than 10^{-3} . It is also standard practice to assess the expected frequency of each postulated initiating event and the probability of impairment of the safety functions required for mitigation. The purpose of these probabilistic safety assessments is to verify that the expected frequency of accidents with the potential for large releases of radioactivity is acceptably low.

C.1.2.2 Some Details of Accident Analysis Methodology

The following discussion is intended to illustrate the points of similarity between Canadian and U.S. safety analysis practice. The objective of accident analysis is to demonstrate that the plant is capable of responding to anticipated transients and accidents without endangering public health and safety. In the U.S. typical key assumptions for accident analysis are as follows:

- a. the first occurring reactor trip signal can be credited (only overpressure protection analyses are required to credit the second trip signal).
- b. the maximum worth control rod or control rod assembly is assumed to remain stuck out of reactor, and
- c. the most limiting failure of a single component is assumed, in addition to the failure causing the initiating event.

The main acceptance criteria are specified in terms of limit values of certain parameters which are used as sufficient, but not necessary, conditions for preventing fuel cladding departure from nucleate boiling (anticipated transients) and for maintaining core coolable geometry (accidents). Examples of these parameters are a departure from nucleate boiling ratio for assessing margin to fuel sheath heat transfer degradation, and a maximum peak clad temperature of $\sim 1200^{\circ}\text{C}$ for limiting degradation of the core cooling capability. The potential impact on public safety is typically assessed by considering a very conservative source term in conjunction with a specified leakage from containment². The implied assumptions for such an assessment are an ineffective emergency core cooling system, an overestimate of time integrated containment pressure, and an intact containment. This "design basis" loss of coolant is one of the accidents used to evaluate the adequacy of systems with respect to public health and safety. The assessment is required to show compliance with public dose limits established by the Code of Federal Regulations (10CFR 100).

Canadian accident analysis, based on separation and independence requirements for process and safety systems, has traditionally considered two categories of events: (i) single failures, i.e., the failure of a process system requiring actuation of one or more of the safety system; and (ii) dual failures, i.e., the failure of a process system combined with the unavailability or impairment of any one of the special safety systems. (Special safety systems include the reactor shutdown systems, the emergency core cooling system and containment.)

A loss of coolant accident and a loss of reactor control are typical single failures, while a loss of coolant accident, combined with the unavailability of the emergency core cooling system, is a typical dual failure. Typical analysis assumptions are as follows:

- a. the second reactor trip parameter of the least effective shutdown system is generally assumed to ensure a conservatively delayed shutdown in the analysis;
- b. two shut-off rods of Shutdown System 1 or one poison injection line of Shutdown System 2 are conservatively assumed to be inoperable to make the credited shutdown system capability less effective than expected.

Although the acceptance criteria are formally laid down in terms of limits for the radiological doses to the public³, specific safety design objectives for fuel integrity and core coolability, similar to those used for light water reactors, are employed in the analyses. For the most frequent single failures (i.e., anticipated transients), the design objective is to minimize potential challenges to the integrity of fuel and fuel channels by preventing fuel elements operating with sustained post-dryout convective heat transfer. For the most serious single failures (e.g. loss of coolant accident) and for dual failures, the objective is to maintain core coolability. The difference with light water reactors is that these objectives are accomplished by analyzing in detail the time the fuel cladding is exposed to post-dryout or high temperature conditions, rather than by using pre-determined limits for selected parameters.

Accident analysis methodologies for CANDU reactors and light water reactors, though based on somewhat different assumptions and acceptance criteria, are focused on demonstrating similar safety objectives. This is further confirmed by similar dose limits applied to the most limiting accidents, i.e., maximum credible accident for light water reactors and dual failures for CANDU, as shown in Table C.1. Large conservatism margins are used in the analyses of both reactor types.

Additional discussion of licensing regulations and safety analysis can be found in the literature,^{4,5} and AECL reports⁶.

C.1.3 CANDU SAFETY CHARACTERISTICS AND MOX FUEL

CANDU reactors, by virtue of their natural uranium fuel and heavy water moderator, have some unique safety characteristics. The neutron spectrum contains a relatively large number of delayed neutrons and photo-neutrons. This leads to a relatively slow response to changes in reactivity. On the other hand a reduction of coolant density which could be initiated by a loss of coolant leads to an increase in the reactor's power output. In recognition of this dual, independent fast-acting shutdown systems, which monitor the level and rate of change of neutron flux and other process parameters, are provided to rapidly shut the reactor down should any parameters deviate beyond pre-set limits.

The introduction of plutonium into the fuel modifies the response characteristics

of the reactor. The proportion of delayed neutrons will be decreased, resulting in more rapid response to any reactivity change that is introduced. A goal of this study has thus been to design the fuel composition to emulate as much as possible the behaviour of the reactors with natural uranium. In order to achieve this end, additional material is introduced into the fuel to modify the reactivity characteristics. The faster kinetics introduced by the addition of plutonium is compensated by adding a poison material which decreases the reactivity increase brought about by coolant density reduction. The amount and location of the poisoning material is, however, adjusted to avoid the introduction of large degrees of reactivity increase which could be introduced by coolant density increase. This approach is intended to avoid introducing new safety and control issues.

Table C.1
Radiation Dose Guidelines for Accident Conditions

CANADA

	Single Failures		Dual Failures	
	Effective	Thyroid	Effective	Thyroid
Individual	5 mSv (0.5 rem)	30 mSv 3 rem	250 mSv (25 rem)	2.5 Sv (250 rem)
Population	10 ² person-Sv (10 ⁴ man-rem)	10 ² person-Sv (10 ⁴ man-rem)	10 ⁴ person-Sv (10 ⁶ man-rem)	10 ⁴ person-Sv (10 ⁶ man-rem)

UNITED STATES

	Maximum Credible Accident	
	Effective	Thyroid
Individual	250 mSv (25 rem)	3.0 Sv (300 rem)
Population	Not Specified	Not Specified

CANDU reactors operating with natural uranium fuel can only attain criticality in the presence of heavy water moderator⁷. Enrichment with plutonium raises the possibility that fuel may reach criticality in the presence of light water. Conditions leading to criticality must thus be established and measures taken to avoid those conditions.

Natural uranium fuel used in CANDU reactors is only slightly radioactive and is manually handled during the fuel loading process. Addition of plutonium to the fuel increases radioactivity to the extent that modified handling procedures will be necessary. It is thus necessary to establish shielding- and fuel-operator separation requirements for handling of fresh fuel at the reactor site.

Enrichment of the fuel with plutonium can also allow higher fuel burnup to be attained than is characteristic of CANDU reactors operating with natural uranium. This, in turn, increases fission product loading and decay heat levels which must be taken into account in the evaluation of emergency cooling, containment system capability, and consequences to the public should an accident occur.

The following sections address these issues, in turn, for both the standard 37-element and CANFLEX fuel designs.

C.2 SAFETY ASPECTS OF PLUTONIUM FUEL DEFINITION

C.2.1 GENERAL STUDY PARAMETERS

The basic constraint of the study, provided by the Department of Energy, is the disposition of 50 tonnes of plutonium in 25 years. Two nuclear generating stations, each consisting of four reactor units, are sited at the Bruce Nuclear Power Development in close proximity to the United States border. Four of these reactors at Bruce NGS A were chosen as the basis for the reference study as no reactor design changes were anticipated for safe utilization of plutonium enriched fuel. Fuel supply has fuel fabricated at a single existing United States facility and transported by truck to Canada.

C.2.2 REACTOR PHYSICS AND FUEL DESIGN

On-power refuelling and the use of heavy water as moderator and coolant enable existing CANDU reactors to adapt to a variety of nuclear fuel cycles without requiring major modifications. The neutronic behaviour of CANDU reactors is dominated by heavy water such that all reactivity perturbations in CANDU reactors are relatively small and slow. The reactor excess reactivity is small. Criticality is maintained with the on-power refuelling system.

CANDU reactors normally operate with natural uranium based fuel having fissile content of 0.71% U-235. The reference MOX fuel design contains substantially more fissile material (on average 1.5 wt% per bundle) than natural uranium fuel. A burnable neutron absorber (dysprosium) was added to the central 7 elements of the bundle

to minimize differences which might result in a need to modify reactor systems. The dysprosium is arranged to suppress excess reactivity and to reduce the amount of reactivity change associated with changes in coolant density. System changes are avoided by ensuring the fuel looks and behaves essentially similar to natural uranium fuel with respect to heat generation and re-fuelling operations.

The major physics parameters of the reference depleted uranium - plutonium oxide fuel as compared to natural uranium fuel are presented in Table C.2. Although the delayed neutron fraction, the prompt neutron lifetime and the fuel temperature reactivity coefficient for the plutonium fuelled reactor are smaller than those in the natural uranium reactor, these parameters are still within the regime of a well thermalized heavy water reactor and result in transient behaviour similar to that of the natural uranium reactor.

C.2.3 REACTOR PHYSICS AND LOCA SAFETY

In CANDU reactors operating with natural uranium fuel reactivity increases following loss of coolant from the pressure tubes. The energy deposited in the fuel during the transient increase in power, prior to termination by one or both of the shutdown systems, has the potential to affect the integrity of the fuel and fuel channel. Rapidly acting, reliable shutdown is therefore provided to limit the challenge to the integrity of fuel and fuel channels.

Table C.2 provides physics data needed for comparison of natural uranium and MOX fuelled Bruce A reactors at equilibrium conditions. Introduction of plutonium alone into the fuel results in somewhat more rapid reactor kinetics response due to a lower total delayed neutron fraction and a shorter neutron lifetime for the MOX core. Adding dysprosium neutron absorber to the inner fuel elements eliminates the power increase on voiding through the introduction of negative void reactivity. When coolant is voided, the scattering effect of the coolant in the fuel channel is removed. This allows more neutrons to reach the centre of the fuel bundle, i.e., the poisoned region. The negative reactivity generated by the higher neutron absorption in the poisoned inner region more than compensates the increase in reactivity in the MOX region of the bundle.

An analysis of LOCA response was undertaken with the POINTSIM code to compare the power transient in the natural fuel core with that in the MOX fuel core. The coolant density transient calculated with the TUF code, corresponding to an 100% double-ended guillotine failure of a pump discharge pipe documented in the 1994 Bruce A safety report, is used in the simulations. Table C.3 gives the core-averaged coolant density transient, and the corresponding coolant void reactivity insertion in the natural core and in the MOX core used in the POINTSIM calculation. Shutdown systems are triggered by neutronic signals in the natural uranium core while process trips occur first in the MOX core. These trips will be delayed relative to neutronic signals by at most three seconds. The results are shown in Figure C.1. The results indicate a much smaller energy deposition in the MOX core than that in the natural core for the same LOCA. Temperature increase of the fuel during the few seconds following initiation of a large LOCA will thus be substantially less for the reactor operating with the reference MOX fuel.

Table C.2
Neutronic Parameters for Natural Uranium and Plutonium/Uranium Fuel

	Natural Uranium		Plutonium/Uranium	
Pellet Material Composition (% based on weight of heavy metal in fuel)	Natural UO ₂ in all rings (37 pins)		Ring 4 (18 pins) 1.2% Pu Ring 3 (12 pins) 2% Pu Ring 2 (6 pins) 5% Dy Ring 2 (1 pin) 5% Dy	
Bundle Material Composition (fuel material only)	U ²³⁵ 0.13 kg U ²³⁸ 18.67 kg O ₂ 2.53 kg TOTAL: 21.33 kg		Pu 0.23 kg U ²³⁵ 0.04 kg U ²³⁸ 18.36 kg Dy 0.17 kg O ₂ 2.53 kg TOTAL: 21.33 kg	
Average Burnup	8,300	MWd/te	9,700	MWd/te
Maximum Burnup	15,000	MWD/te	15,500	MWd/te
Bundle/FPD	18		15.5	
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
Delayed Neutron Fraction(Beta)	0.006		0.0035	
Prompt Neutron Lifetime	0.9	ms	0.5	ms
Fuel Temperature Coefficient	-6	μk/°C	-3	μk/°C
Full Core Void Reactivity	11	mk	-4.7	mk

Table C.3
Reactivity insertion due to 100% Pump Discharge LOCA in Bruce A

Time (second)	Coolant Density (g/cc)	Natural Fuel Void Reactivity (mk) Inserted	Ref. MOX Fuel Void Reactivity (mk) Inserted
0.0	0.800	0.0	-0.0
0.125	0.775	0.34	-0.15
0.250	0.675	1.72	-0.73
0.375	0.625	2.41	-1.02
0.500	0.575	3.09	-1.31
0.625	0.510	3.99	-1.69
0.750	0.450	4.81	-2.04
0.875	0.375	5.84	-2.48
1.000	0.340	6.33	-2.69
1.125	0.290	7.01	-2.97
1.250	0.250	7.56	-3.20
1.500	0.200	8.25	-3.50
1.750	0.175	8.59	-3.64
2.000	0.155	8.87	-3.76
2.250	0.150	8.94	-3.79
2.500	0.145	9.01	-3.82

Figure C.1 Reactor Power Transients for a Large Break LOCA in Natural Uranium and MOX Cores

C.3 FRESH AND IRRADIATED FUEL DOSE CHARACTERISTICS

The influence of the revised fuel on current practice for handling of fresh and irradiated fuel is assessed and documented. MOX fuel would generate somewhat higher dose rates than natural uranium, all other factors being equal. The design goal for fresh fuel handling is to maintain dose to operators at or below the level of current operations in order to demonstrate compliance with licensing requirements. Information needed to establish the fresh fuel handling process is based on contact dose rates from gamma and neutron radiation from the fresh natural and MOX fuel. Chapters 2 and 4 of the main report discuss shielding introduced to ensure that the fuel handling safety design goal is met.

The dose information developed also includes contact dose from spent fuel. Although this is of little relevance to safety during remote handling of spent fuel, it is needed to show that the CANDU spent fuel standard is maintained.

C.3.1 CONTACT DOSE RATE FROM FRESH MOX FUEL BUNDLE.

Plutonium removed from warheads is in metallic form containing americium and gallium. Existing warheads were manufactured at different times over the past 50 years. Therefore the isotope content of the plutonium varies significantly. Most were manufactured more than twenty years ago. Because the half life of Pu-241 is about 14 years and it decays to Am-241, the accumulation of Am-241 over the storage time of the warheads can approach the initial amount of Pu-241.

Pu-241 produces no gammas in its decay process. However, its daughter product Am-241 generates low-energy gammas during decay. Thus the accumulation of Am-241 over the storage time of the warheads introduces a source of gamma which did not exist when it was newly constructed. The following analysis is to quantify the expected dose rates one would expect from a fresh MOX-fuel bundle made from decayed warhead material.

C.3.1.1 Contact Dose Rate from Gammas

(1) Plutonium Concentration - In calculating the contact dose rate for a MOX-fuel bundle, the plutonium going into the fuel bundle is assumed to have already decayed for twenty years. The initial composition of the weapon-grade plutonium is listed in Table C.4. Based on this composition, the ORIGEN-S computer code (see Appendix B) was used to calculate the plutonium composition after twenty years of decay. Table C.5 lists the new composition in weight fraction.

(2) Photon Source Spectrum - To calculate the gamma contact dose rate, the photon release rate of the bundle was extracted from the ORIGEN-S output. In this analysis, 35 energy groups were used and are listed in Table C.6. Most of the source photons are the low-energy X-rays, characteristic of plutonium decay.

(3) Dose Rate Calculation - To calculate the contact dose rate, the QAD-CGGP three-dimensional point-kernel gamma transport code (see Appendix B) was used. The

calculation was done in r-z geometry. Variable-mesh structure was used to ensure the meshes close to the bundle surface are small enough to accurately predict the contact dose rates.

To calculate the contact dose rate, a quarter of the fuel bundle was modelled (see Figure C.2). The inner two half cylinders represent the centre two and outer two rings of fuel. The outermost half cylinder represents the fuel sheath. The material composition of the model is listed in Table C.7 while the dimension of the model is listed in Table C.8. The source photons are assumed evenly distributed in the outer two rings of fuel, that is, in the homogenized fuel region 2. Due to symmetry, the dose rates calculated at points C, F, and M are the dose rates on contact, 1 foot away, and 1 metre away from the fuel bundle respectively (see Figure C.2).

(4) Results - The dose rates calculated for the MOX fuel bundle are listed in Table C.9. Dose rates from the natural uranium fuel bundle are for a bundle that contains uranium which is six months old from the time it was milled.

C.3.1.2 Contact Dose Rate from Neutrons

The contact dose rate from neutrons was estimated based on a line-source assumption. Dose rates are calculated from the unattenuated flux on the bundle surface.

Both the contact dose rates for MOX and natural uranium fuel bundle were estimated. Table C.10 lists the results. The value for the natural uranium fuel bundle was cross-checked with experimental measurement which is 20 $\mu\text{Sv/h}$.

C.3.2 CONTACT DOSE RATE FROM SPENT MOX-FUEL BUNDLE

The methodology used to calculate the contact dose rate is analogous to that used to calculate the contact dose rate for a fresh MOX-fuel bundle. The photon source data was calculated using the ORIGEN-S computer code based on an average bundle power of 492 kW (nominal for Bruce A reactor). The bundle was irradiated to its average exit-burnup and then discharged from the core.

The cross section used in the ORIGEN-S calculation is a series of CANDU burnup-dependent libraries at intervals through a fuel irradiation. This set of cross section data takes account of changes in fuel composition and flux spectrum to give a more accurate fuel isotopics in the depletion analysis. The photon source was then extracted from the ORIGEN-S output and placed in the model. Using the QAD-CGGP computer code, the dose rate from a spent MOX-fuel bundle was then obtained.

Contact dose rates for spent MOX- and natural uranium fuel bundles for different decay times are shown in Figure C.3. Dose rates for the two different bundles are close in magnitude up to 250 years of decay time. Hence, from a radiological point of view, handling spent MOX-fuel bundles is no different than handling spent natural uranium fuel bundles.

Table C.4
Composition of Weapon-Grade Plutonium

Isotopes	Weight Fraction
Pu-239	0.93825 0
Pu-240	0.05801 7
Pu-241	0.00350 0
Pu-242	0.00021 7

Table C.5
Composition of a 20-Year Old Weapon-Grade Plutonium

Isotopes	Weight Fraction
He-4	1.18E-5 *
U-235	5.32E-4
U-236	1.21E-4
Np-237	3.87E-5
Pu-239	9.38E-1
Pu-240	5.79E-2
Pu-241	1.36E-3
Pu-242	2.17E-4
Am-241	2.10E-3

* Read 1.18E-5 as 1.18×10^{-5}

Table C.6
Gamma Energy Groups in ORIGIN-S

Group Number	Mean Energy, MeV	Photons/Second
1	2.49-3*	2.50 + 8
2	7.21-3	4.97 + 8
3	1.44-2	4.08 + 10
4	2.47-2	1.66 + 9
5	3.48-2	1.31 + 8
6	4.48-2	1.14 + 8
7	5.49-2	2.52 + 10
8	6.49-2	2.85 + 7
9	7.49-2	1.60 + 7
10	8.49-2	4.80 + 6
11	9.49-2	4.38 + 7
12	1.05-1	3.20 + 7
13	1.15-1	1.42 + 7
14	1.34-1	4.39 + 7
15	1.74-1	1.37 + 7
16	2.47-1	1.67 + 7
17	3.48-1	3.09 + 7
18	4.48-1	1.11 + 7
19	5.49-1	1.30 + 6
20	6.49-1	1.59 + 6
21	7.49-1	1.64 + 6
22	8.49-1	5.79 + 5
23	9.49-1	1.18 + 6
24	1.07 + 0	8.41 + 5
25	1.18 + 0	6.01 + 4
26	1.34 + 0	1.99 + 5
27	1.74 + 0	2.29 + 5
28	2.24 + 0	3.91 + 3
29	2.74 + 0	2.05 + 3
30	3.24 + 0	1.17 + 3

* Read 2.49-3 as 2.49×10^{-3}

Table C.7
Material Composition of the QAD-CGGP Model

Regions	Density g/cc	Elements	Partial Density g/cc
Homogenized Fuel	5.52	Oxygen	0.654
		Uranium	4.853
Fuel Sheath	6.55	Zirconium	6.55

Table C.8
Physical Dimension of the QAD-CGGP Model

Regions	Outer Radius, cm
Homogenized Fuel Region 1	2.182
Homogenized Fuel Region 2	4.985
Fuel Sheath	5.027
Overall Length	24.765

Figure C.2 QAD-CGGP Model of One Quarter of a Fuel Bundle

Table C.9
Gamma Dose Rates from Fresh MOX and Natural Uranium Fuel Bundle

	Dose Rates, mrem/h	
Dose Location	MOX	NU
on contact	13.0	4.2
1 foot away	0.8	0.25
1 metre away	0.1	0.03

Table C.10
Neutron Dose Rates from Fresh MOX and Natural Uranium Fuel Bundle

	Dose Rates mrem/h	
	MOX	NU
On Contact	2.8	0.03

Figure C.3 Contact Dose Rates from Spent MOX and Natural Uranium Fuel Bundles

C.4 CRITICALITY SAFETY

The natural uranium fuel used in CANDU reactors can only become critical in the presence of heavy water in a fuel/moderator core configuration close to the designed lattice arrangement. There is thus no need to take special precautions to prevent criticality outside the reactor as with reactor designs which depend on fuel enriched in fissile content.

Accident scenarios with MOX fuel may be postulated which could lead to criticality concerns. These range from accidental immersion in light water during transport, water at the plant, to fuel bundles being discharged at low burnup (because of a suspected fuel defect) into the irradiated fuel bay and coming into close contact due to a handling accident. The complete range of such accidents would be analyzed as part of the licensing basis for MOX operation, however, based upon a review of some scenarios no difficulties are foreseen.

C.4.1 CRITICALITY SAFETY OF THE REFERENCE MOX-FUEL BUNDLE

C.4.1.1 Analysis Basis

The configuration of MOX fuel bundles considered was the storage of fresh MOX fuel bundles in the irradiated-fuel bay using the Darlington fuel handling modules that are proposed for handling MOX fuel bundles at Bruce NGS A. (See Section 2.4)

The method selected to calculate the k_{eff} of the system consists of the MCNP4a computer code modelling the two configurations exactly using the continuous neutron cross-section library. Temperature for the cross section is 300°K. The thermal $S(\alpha, \beta)$ tables for hydrogen and zirconium were used to account for thermal neutron scattering by molecules and crystalline solids.

The accepted k_{eff} from MCNP4a is the least squares combination of results generated by the collision, absorption and track-length estimators. The standard deviation quoted is in the normal 68% confidence level.

C.4.1.2 Fuel Handling Module

The Darlington fuel handling module is shown in Figure C.4. The module consists of 48 parallel stainless steel tubes in a 4x12 arrays attached to two stainless-steel sheets and a stainless-steel frame. Each tube has a nominal thickness of 1 mm. The length of each tube is 997 mm, enough to hold two fuel bundles back to back. In the analyses that follow the handling modules are fully loaded with 96 fresh MOX-fuel bundles with the bundle composition as indicated in Table C.2.

Five scenarios were looked at.

- a. The fuel handling module is completely immersed in light water.
- b. Like case (a) but with thicker stainless-steel tube.
- c. An infinite assembly of fuel handling modules completely immersed in light water.
- d. Like case (c) but the stainless-steel tubes have a larger pitch.
- e. Like case (c) but the light water has a density of 0.8 g/cm^3 .

Results from scenario (a) tells us whether a fuel handling module filled with fresh MOX-fuel bundles can go critical if it is immersed in the spent-fuel bay. For scenario (b), the result gives us an indication as to how the stainless-steel tube thickness affects the criticality of the assembly. Results from scenario (c) tell us whether stacking up a series of fuel handling modules filled with fresh MOX-fuel bundles in the spent fuel bay can go critical. And the result from scenario (d) tells us how the arrangement of the stainless-steel tubes affects the criticality of the assembly. As for scenario (e), the result tells us how the criticality of the assembly changes with the water density.

Calculations for all five scenarios were completed by modelling the geometries as described and then the k_{eff} was evaluated. All scenarios were assumed to be in room temperature. Because temperature has a negative reactivity effect, the scenario for a higher temperature case was not considered.

(1) Simulation Model

Figures C.5 to C.8 illustrate the simulation model. Figure C.5 is the end view of the fuel handling module. The rectangular outer region represents the light water region. This region is more than five neutron mean free paths in light water and is sufficient to account for all scattering effect at the boundaries. Each stainless-steel bundle holds two MOX 37-element fuel bundle end-on. A cross-section of the bundle in finer detail is listed in Figure C.6.

Figure C.6 is the cross section view of the 37-element fuel bundle in a stainless-steel tube. Each fuel element is modelled with two regions, the inner fuel region and the outer fuel sheath. The air gap which exists in a fuel element is not included in the model.

Figure C.7 is the side view of the fuel handling module. Shown in the figure are stainless-steel tubes, each with two fuel bundles. Exact details of the endplate were not modelled. The end regions of each fuel bundle are represented as homogeneous regions of the endplate and light water. The frame of the fuel handling module is shown as the small box at the centre. The rectangular box that surrounds the stainless steel tubes is an artifact of the modelling. This box is used to describe the spacing between the stainless-steel tubes.

Figure C.8 is the plan view of the fuel-handling module through the centre.

(2) Material Composition

The ring composition of the MOX 37-element fuel bundle is listed in Table C.2. The density of the fuel was modelled at 10.4 g/cm^3 . The compositions of the fuel sheath and the stainless steel are listed in Table C.11. The density of light water for scenarios (a) to (d) is 1.0 g/cm^3 . The temperature for all materials is 300°K .

(3) Infinite Assembly

To simulate an infinite assembly of the fuel handling module, i.e., scenarios (c), (d), and (e), specular reflective boundaries were used on all six surfaces of the module, see Figure C.8. These reflective boundaries are just beyond the physical boundaries of the fuel-handling module to simulate a closely packed array.

(4) Results

Simulation results for the different scenarios are summarized in Table C.12. Case (e) has the highest k_{eff} . It is an infinite assembly of the fuel-handling modules submerged in light water at a density of 0.8 g/cm^3 . Nonetheless, none of the scenarios poses a criticality problem.

C.4.1.3 Conclusion

There is no criticality concern for the fuel handling module which consists of all fresh MOX-fuel bundles and is immersed in light water. There is also no criticality concern if we assumed an infinite array of these fuel-handling modules containing all fresh MOX-fuel bundles stored in the spent fuel bay.

Table C.11
Material Composition used in the Simulation Model

Materials	Density g/cm ³	Element	Weight Fraction
Fuel Sheath	6.23	Zirconium	9.97E-1 *
		Iron	1.60E-3
		Chromium	1.10E-3
		Nickel	6.00E-4
		Natural Boron	3.10E-6
Stainless Steel	7.90	Carbon	3.00E-4
		Silicon	8.75E-3
		Phosphorous	4.50E-4
		Sulphur	2.84E-4
		Chromium	1.90E-1
		Manganese	2.00E-2
		Iron	6.79E-1
		Cobalt	1.00E-3
		Nickel	1.00E-1

* Read 9.97E-1 as 9.97×10^{-1} .

Table C.12
 K_{eff} for the Different Scenarios of the Fuel Handling Module

Scenarios	k_{eff}	Standard Deviation
(a)	0.85022	0.00080
(b)	0.76051	0.00199
(c)	0.89320	0.00077
(d)	0.85676	0.00083
(e)	0.90545	0.00084

Figure C.4 The Darlington Fuel Handling Module

Figure C.5 MCNP Model of the Fuel Handling Module - End View

Figure C.6 MCNP Model of a Fuel Bundle in the Stainless Steel Tube of the Fuel Handling Module

Figure C.7 MCNP Model of the Fuel Handling Module - Side View

Figure C.8 MCNP Model of the Fuel Handling Module - Plan View

C.5 REACTOR THERMALHYDRAULIC PARAMETERS

The introduction of MOX fuel is accompanied by some changes, relative to natural uranium fuel, in the distribution of heat generation within fuel bundles and along the string of fuel bundles in a channel. The variations in power distribution exist radially because of the introduction of enrichment in the outer two rings of 30 elements of the MOX fuel bundle and the reduction of enrichment in the inner two rings of 7 elements. As a result the power generated in the inner 7 elements is much less than for the natural uranium bundle. Axial variations in power along the fuel bundle string are introduced through the flattening of the core power distribution associated with the MOX fuel and two bundle shift fuel management scheme employed.

These factors were studied extensively as reported in Sections 2.2 and 2.3. of the main report to determine their impact on thermalhydraulic parameters. The studies show that the impact on critical heat flux performance of the 37-element MOX bundle is not significant relative to the 37-element natural uranium bundle. Calculated critical channel powers, (the channel power level at which CHF first occurs in any bundle in the channel), are essentially identical for MOX and natural uranium fuelled channels.

C.6 SHUTDOWN AND CONTROL SYSTEM EVALUATION

The Bruce reactors are each fitted with two completely independent shutdown systems which are intended to ensure shutdown in the event of any malfunction. An independent control system control system regulates reactivity during normal operation to control reactor power. These systems are described in more detail in Chapter 2 of the report. The introduction of MOX fuel modifies reactor response so that a systematic review of the performance of these systems is required. The following sections address the ability of the shutdown systems and of the control system in turn.

The introduction of a MOX fuelled core into a Bruce NGS A reactor results in changes in the worth of reactivity devices and the neutron kinetics of the core, relative to a natural uranium fuelled core. These changes are such that the worth of reactivity devices is decreased and total delayed neutron fraction and prompt generation time also decrease. However, an offsetting factor is that the coolant density reactivity coefficient changes from positive to negative, thereby removing coolant void formation as a source of rapid positive reactivity insertion for events such as loss of coolant accidents and severe flow-power mismatch.

Because of these changes in the reactor physics characteristics of the core the capabilities of the two independent shutdown systems must be demonstrated. The effectiveness of the shutdown systems is governed in part by the ability of trip parameters to provide timely signals to trip the reactor, which assure that existing safety design objectives are met, and in part by provision of adequate depth and speed of reactivity insertion. This effectiveness has been assessed for the MOX fuelled core and the results of this assessment are documented below.

The major impact on shutdown system trip coverage, relative to that for the natural uranium core, occurs for accidents involving reactivity excursions in which neutronic trip parameters provide protection, such as loss of coolant accidents (LOCA) and loss of

reactivity control events (LORC). Because of the negative coolant density reactivity coefficient in the MOX fuelled core, an event involving a rapid increase in coolant density, such as liquid refill of voided channels, will give rise to a positive reactivity insertion. This class of events is analyzed below as a loss of reactivity control accident to demonstrate effective shutdown capability is maintained.

C.6.1 Safety Design Objectives

The effectiveness of trip parameters is assessed in relation to a number of safety design objectives. These objectives are conservative criteria that, when met, provide assurance of a timely and effective reactor shutdown by either one of the two shutdown systems acting alone. The general safety design objectives applied to shutdown system action are:

- a. Fuel breakup during rapid power excursions is prevented;
- b. Fuel channel integrity is not impaired by fuel overheating;
- c. Heat transport system integrity is not impaired (applies to a non-LOCA event);
- d. An adequate water inventory is maintained in the steam generators to assure heat sink capability over a sufficiently long period of time to allow establishment of an alternate sink.

These general objectives are reduced to specific, conservative criteria applicable to particular accident scenarios. The specific criteria employed in this assessment are described below in the various sections dealing with specific classes of accidents.

C.6.2 Loss of Coolant Accidents (LOCA)

The system response and effective trips following a LOCA may vary depending on the size of the discharge from the break. The range of possible discharge rates is therefore divided into three categories, termed small, intermediate and large LOCAs.

A small LOCA is defined as having a maximum discharge between zero and 1,000 kg/s. This range covers breaks in pipes up to and somewhat larger than a pressure tube. It can be seen from Figure C.9 that breaks or leaks in the vast majority of pipes in a unit will fall into this category.

Large LOCAs are defined as having a discharge rate from about 2,000 kg/s up to the maximum possible discharge from a single large pipe break of about 40,000 kg/s. As can be seen in Figure C.9, this range covers the discharge from a guillotine break of the relatively small amount of large diameter piping in a unit.

The intermediate range corresponds to initial discharges of between 1,000 and 2,000 kg/s, in which the system response progresses from the slow depressurization behaviour characteristics of a small LOCA to the rapid depressurization and flow reduction behaviour observed in a large LOCA. As shown in Figure C.9, discharges in this range are relatively improbable and would be produced by partial failure of a large diameter pipe.

APPROXIMATE LENGTHS OF VARIOUS DIAMETER PIPES IN A BRUCE UNIT

Figure C.9

C.6.2.1 Large Break LOCA

Large break loss-of-coolant accidents are characterized by rapid coolant voiding in the fuel channels which induces a rapid reduction in heat transfer from fuel to coolant. In a natural uranium fuelled core the coolant voiding introduces a positive reactivity insertion which, in turn, induces a transient increase in core power that is terminated by operation of either one of the two independent fast-acting shutdown systems. In the MOX core, coolant voiding introduces negative reactivity which would act to reduce the reactor power. For conservatism in this assessment the mitigating effect of coolant void reactivity is not credited (i.e., assumed to be essentially zero), nor is the small negative reactivity associated with fuel heatup, thereby resulting in unchanged reactor power during the early stages of the accident.

The safety design objective for shutdown system action is to assure integrity of the fuel channels which can be met by preventing the onset of fuel centreline melting as a consequence of the increase in fuel stored heat (increase in fuel enthalpy) associated with the degraded heat transfer from fuel sheath to coolant.

The limiting break is a rupture of an inlet header resulting in a break area that is 35% of the area of a double-ended guillotine break. This break, the so-called "critical" break, poses the greatest potential for fuel overheating as a consequence of the fact that on pass of the heat transport loop experiences an early period of sustained low flow during which there are very low driving pressure differentials across the channels in the pass.

The effectiveness of the trip parameters to provide timely reactor shutdown signals is assessed by comparing the calculated maximum fuel bundle enthalpy in the core during the first five seconds following the break, assuming adiabatic heatup of the fuel, with corresponding values from the current Safety Report¹⁰, which are well below the threshold values necessary for the onset of fuel centreline melting. This is a valid conservative comparison because the credited process trips initiate earlier than five seconds and the fuel enthalpy calculated for adiabatic conditions at the fuel sheath is a measure of the energy deposition in the fuel that must be removed by degraded heat transfer. By demonstrating that the maximum fuel enthalpy is less than that calculated in existing Safety Report analyses the peak fuel and pressure tube temperatures are assured to be lower than those in the existing analyses for natural uranium fuelled core and, therefore, the consequences with respect to fission product releases and fuel channel integrity will be bounded by the existing analyses.

The protection provided by the two shutdown system trip parameters in the event of a large loss of coolant accident is shown in Table C.13, which also compares the protection provided in a natural uranium core. As can be seen, the neutronic trips that occur in a natural uranium core are replaced with the process parameters trips on Heat Transport Low Pressure and Heat Transport Low Flow and will be effective in lowering the enthalpy addition in the fuel compared to the natural uranium core.

Utilizing the above methodology to evaluate maximum fuel enthalpy over time periods extending five seconds beyond the reactor trip, and progressively increasing the reactor trip time, it can be shown that the maximum fuel enthalpy can be limited to values lower than the highest value already licensed in the Safety Report. It should be noted that this assumes that no credit is taken for the inherent negative reactivity inserted by the reducing coolant density and the increasing fuel temperatures.

Table C.13
Trip Parameter Effectiveness for Large Break LOCA

	MOX Core		Natural-U Core	
	SDS1	SDS2	SDS1	SDS2
Credited trip parameters (Nominal Trip Setpoint)	HTLP (<i>Note 1</i>) (8.0 MPa(a)) HTLF (19.5 kg/s)	HTLP (7.75 MPa(a)) HTLF (<i>Note 2</i>) (18.24 kg/s)	NOP (124%FP) NHLR (10 %/s)	NOP (118 % FP) NHLR (15 %/s)
Max. initial bundle power Reactor Power change	800 kW Constant	800 kW Constant	1035 kW Increases	1035 kW Increases
<i>35 % RIH Break</i> Credited Backup Trip Reactor Trip Time, (sec) Max Bundle integrated Energy to 5 seconds, (kJ/kg)	HTLP 1.5 462	HTLP 2.5 516	NHLR 0.412 682	NHLR 0.441 633
<i>100 % PD Break</i> Credited Backup Trip Reactor Trip Time, (sec) Max Bundle integrated Energy to 5 seconds, (kJ/kg)	HTLP 0.9 429	HTLP 1.8 478	NHLR 0.365 767	NHLR 0.384 712

LEGEND:

NOP = Neutron Overpower
 NHLR = Neutron High Lograte
 HTLP = Heat Transport Low Pressure (High power setpoint > 70 %FP)
 HTLF = Heat Transport Low Flow (Setpoint for 4 HT pumps operating)

NOTES:

1. Nominal setpoints are given in the table, the effective setpoints credited in analyses are more conservative to account for measurement and calibration uncertainty.
2. Implementation of a heat transport low flow trip is assumed based upon electronic logic changes to an existing trip that utilizes flow orifices in a six outer flow zone channels.

C.6.2.2 Small Break LOCA

The general safety design objectives are achieved for this class of accidents by having a reactor trip before the onset of fuel centreline melting in any fuel element, which will be met by preventing sustained dryout on any fuel element. Short periods of transient dryout will not violate the criterion.

There are at least two effective trips over the entire small break spectrum for both SDS1 and SDS2¹¹. The SDS1 trip parameters which provide protection over the small break spectrum are pressurizer low level and heat transport low pressure. Pressurizer low level and heat transport low pressure are also the effective parameters over most of the small break spectrum for SDS2. There is no change in trip coverage for small LOCA in a MOX core relative to a Natural Uranium core.

Manual reactor trip of both shutdown systems, credited at 15 minutes after the first indication to the operator of the upset conditions, is also effective over a large proportion of the range of small breaks.

C.6.2.3 Pressure Tube/Calandria Tube Rupture

In the event of a pressure tube/calandria tube failure, high temperature and pressure coolant will discharge into the moderator. The discharged coolant will flash to form a mixture of steam and water, raising the calandria pressure and eventually bursting the pressure relief discs and forcing fluid into the containment.

For a small fraction of the life of the reactor, the moderator may contain a relatively large amount of soluble neutron poison, to control the reactivity of new fuel or compensate for xenon after a shutdown, and a pressure tube/calandria tube failure at these times would add positive reactivity to the system by diluting the poisoned moderator with purer heat transport D_2O . The expected rate of reactivity insertion is such that the reactor power will be controlled by the regulating system using zone controller filling and control absorber insertion, and enough negative reactivity depth is available to regulate the power until after process trips occur. Apart from this potential reactivity insertion, the course of events following PT/CT break is no different from that following the same size of break at any other location in the HT system.

The maximum expected initial discharge rate from a PT/CT break is about 130 Kg/s. Effective coverage for both SDS1 and SDS2 is provided by the pressurizer low level trip and by the heat transport low pressure trip. This coverage is maintained in the MOX fuelled core.

Although the reactivity depth of the shutdown systems are reduced in the MOX core, the reactivity worths of other reactivity devices and feedbacks, such as Xenon poison, are also reduced by concomitant amounts. This means that the reactivity worth of poison in the moderator necessary to compensate for reactivity deviations, such as Xenon decay, is also reduced. Analysis of the scenarios considered in the Safety Report for in-core LOCA has

established that the shutdown system depth remains as acceptable in the MOX core as it is in the natural uranium core.

C.6.3 Loss of Regulation

The fundamental safety objective following a loss of regulation event is to maintain the integrity of the heat transport system. In the event of fuel failure, radioactive fission products would then be retained in the heat transport system and would not be released to the containment. Shutdown system effectiveness is determined on the basis of whether the reactor is tripped before fuel overheating or coolant overpressure could compromise heat transport system integrity by imposing the following safety design limits¹²:

- prevention of the onset of fuel centreline melting
- prevention of excessive overpressure in the heat transport system

C.6.3.1 Loss of Reactivity Control

The reactor regulating system computer controls short term reactivity primarily by varying the volume of light water in the liquid zone controller compartments. Mechanical control absorber rods are used to augment the range of negative reactivity control. Long term reactivity control is achieved by on power fuelling and the addition or removal of moderator poison. Withdrawal of shutoff rods is also a control function, which occurs only after a reactor trip signal has been cleared following shutdown. Under normal operating conditions, zone controllers are partially filled and mechanical control absorber and shutoff rods are withdrawn.

The reactivity depths and the maximum average reactivity insertion rates for the reactivity control devices are given in Table C.14. The total number of mechanical control absorbers that can be driven simultaneously is restricted to two. In addition, the devices are subject to a system of hard-wired interlocks, external to the control computers, which limit the number of devices that can operate simultaneously. For example, these interlocks inhibit simultaneous withdrawal of more than one of the two banks of shutoff rods.

Under normal operating conditions, the mechanical control absorbers and shutoff rods are parked outside the core and, therefore, cannot add positive reactivity. Therefore, computer malfunctions can, at most, result in a positive reactivity rate of approximately .097 mk/s associated with all zone controllers draining, a channel being fuelled and the simultaneous removal of moderator poison load.

When the devices are not in their normal positions, maximum average positive reactivity rates of up to approximately 0.39 mk/s can arise due to regulating system computer failure coincident with the failure of one hardware interlock. It is noted that shutoff rod withdrawal alone is not normally a concern because the reactor cannot be made critical before the shutoff rods are fully withdrawn except for the case in which sufficient poison has not been added to the moderator to compensate for the decay of xenon following a prolonged shutdown.

Table C.14
Maximum Nominal Reactivity Rates Available from Reactivity Devices in a MOX Core

Control Device	Reactivity Depth (mk)	Nominal Reactivity Rate (mk/s)
Liquid Zone Controllers	4.5	0.086
Mechanical Control Absorbers (MCA)	5.25	
- typical		.0035
- one bank, 2 rods (max)		.011
- both banks, 4 rods (max)		.035
Moderator Poison Removal		0.0008
Refuelling		0.0008
Shutoff Rod (SOR) Withdrawal	27	
- one bank (15 rods)		.09
- both banks (30 rods)		.18
All zone controllers draining, and all MCAs driving out		0.121
All SORs driving out, all MCAs driving out and all zone controllers draining		0.422

A maximum average reactivity rate of approximately 0.422 mk/s can be obtained by postulating the simultaneous withdrawal of both shutoff rod banks, all four control absorbers (two banks) and draining of the zone controllers following, for example, a reactor trip. Under these circumstances, the reactor would initially be highly subcritical at some decay power level. In order to produce such large reactivity rates, the computer system failure must coincide with the highly improbable failure of two separate hardware interlocks.

Transient loss of reactivity control is analyzed parametrically, considering various linear ramps of positive reactivity insertion at rates ranging from 0.001 mk/s up to 2 mk/s, a rate well in excess of that associated with the most improbable system failure. The parametric analysis assumes constant net positive reactivity rates.

Trip coverage in the MOX core for loss of reactivity control events remains essentially unchanged from that in the natural uranium core. The effectiveness of the neutron high lograte trips is slightly enhanced due to the smaller delayed neutron fraction and neutron prompt generation time in the MOX core. However, this difference is not sufficiently great to make any material change in the transient behaviour of the core. This is demonstrated in Figures C.10 and C.11 which compare the neutron flux and power to coolant transients for events involving draining all the zone controllers and the combination of draining zone controllers and withdrawing all the mechanical control absorber (MCA) rods in either a MOX core or a natural uranium core. As can be seen from these transients, there is not a significant change in the nature of the events, with the primary difference being that the slightly more rapid increase in neutron flux in the MOX core is compensated by slightly earlier reactor trips.

Additional protection against onset of fuel centreline melting is provided by the heat transport high pressure trip on both shutdown systems. Pressure relief through the liquid relief valves, which have the same setpoint as the SDS1 high pressure trip and share the SDS1 pressure transmitters, will not delay the SDS1 high pressure trip.

C.6.3.2 Transients Involving Increases in Coolant Density

Positive reactivity insertion in the MOX core will accompany any transients involving increased coolant density. For events involving increased coolant density due to temperature changes in single phase liquid coolant, such as secondary side depressurization, the rate and magnitude of the induced positive reactivity will be low and will remain within the capability of the reactor regulating system.

Larger and more rapid positive reactivity insertion is theoretically possible if coolant void (steam) conditions in the channels were to be displaced by liquid heavy water coolant entering the channels. However, no realistic mechanisms can be identified by which such an event, involving the situation of a highly voided core with cold liquid suspended in headers and piping above the reactor, could occur. Nevertheless, this situation is postulated as a bounding scenario for purposes of analyzing coolant density induced positive reactivity insertion. It is assumed that all channels in the core are fully voided, with the reactor at a low power level such that the fuel does not overheat, and that the channels are arbitrarily refilled

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with cold water at flow rates consistent with either maximum natural circulation flows (approximately 1 kg/s per channel) or with forced flow from the maintenance cooling pumps being instantaneously started (approximately 2 kg/s per channel). The corresponding reactivity insertion rates associated with these void displacement rates is 0.03 mk/s to 0.06 mk/s, which is within the range of the reactor regulating system. Even if the reactor regulating system action is not credited, the resultant reactivity insertion rates are within the range of analyzed rates for which effective trip coverage has been demonstrated.

C.6.3.3 Distorted Flux Shapes

A loss of reactivity control initiating with an initial distorted flux shape followed by a slow, uniform increase in reactivity throughout the core is an incident which serves as the design basis of the Neutron OverPower (NOP) systems - also referred to as the Regional OverPower (ROP) systems. Because there is no significant difference in the critical channel powers for MOX and natural uranium fuelled cores, for very slow increases in reactor power from distorted flux shapes the NOP trip would be equally effective in preventing the onset of fuel centreline melting in a MOX core as it is in a natural uranium core.

For fast power increases from high and intermediate power levels, trip coverage is provided by both the high log rate and NOP trips. For intermediate rates of power increase, protection in addition to the NOP trip is provided by the heat transport high pressure trip. The extent of this additional coverage is governed by the initial bulk power level and the corresponding initial margin to dryout in the highest powered channel as determined by the initial distorted flux shape.

The effectiveness of the high neutron power, high pressure and high log rate trip parameters in a MOX core will not change relative to the effectiveness for a natural uranium core.

C.6.3.4 Loss of Reactor Power Regulation - Linear Power Ramps

The reactor regulating system is designed to manoeuvre the reactor power at rates of up to 0.8 percent of full power per second. In analyzing these events, it is postulated that unspecified failures cause the power to increase at constant rates both within this normal manoeuvring range and at rates of up to 20 percent per second. Regulating system setbacks and stepbacks are not credited and the incident is terminated by shutdown system action.

These events involve power increases at rates controlled rates that are not dependent upon the kinetics of the core. Therefore, the effectiveness of the high neutron power, high pressure and high log rate trip parameters in a MOX core will not change relative to the effectiveness for a natural uranium core.

C.6.4 Loss of Flow Accidents

Loss of flow events considered in the Safety Report analysis¹³ include total and partial loss of Class IV power to the electrical buses supplying the main heat transport pumps,

trip of a single heat transport pump and mechanical failure leading to seizure of the impeller of a pump. Coverage by the heat transport high pressure and low flow trips will remain essentially unchanged from that currently reported for the natural uranium core. The major difference is that neutron overpower trips will not occur in the MOX core for conditions which result in the transient development of significant coolant void in channels. Conversely, the absence of an overpower transient will result in greater transient margins to onset of dryout in the MOX core, thereby marginally increasing the effectiveness of the process trips for these events, relative to the natural uranium fuelled core.

C.6.5 Secondary Side Accidents

The primary safety design objective for failures in the secondary side feedwater and steam systems¹⁴ is to ensure that a timely reactor trip occurs such that the steam generators contain sufficient liquid inventory to provide time for the establishment of an alternate heat sink for the heat transport system. Typically, decision and action times for operator action to establish an alternate heat sink, evaluated from the initiation of a credited trip, are taken as 15 minutes for action from the control room and 30 minutes for actions in the field. Because the decay heat powers in the MOX fuel are approximately 11 % lower than those in natural uranium during the first 30 minutes following a reactor shutdown, the existing trip effectiveness for secondary side accidents will be as good, or better than those currently demonstrated by Safety Report analysis for the natural uranium fuelled core.

LOSS OF REACTIVITY CONTROL COMPARISON OF MOX vs Nat U CORES

Figure C.10

LOSS OF REACTIVITY CONTROL COMPARISON OF MOX vs Nat U CORES

Figure C.11

C.7 DECAY HEAT AND RADIONUCLIDE GENERATION

Information on decay heat levels and radionuclide content of the MOX fuel following full burnup or equilibrium accident conditions is needed to assess the performance of safety systems and accident consequences in terms of dose. Information on decay heat is also needed for the assessment of irradiated fuel storage options. The following sections establish decay heat at exit burnup and mid burn up for the assessment of spent fuel handling and average accident conditions respectively.

C.7.1 DECAY POWER FROM MOX-FUEL BUNDLE

This section documents a calculation of the decay power of a MOX-fuel bundle discharged at mid- and exit-burnup. The decay power will be presented as both a power-rundown and an integral-power curve. The data is compared with natural uranium fuel discharged at the corresponding mid- and exit-burnup as reference.

C.7.1.1 Reactor and Fuel Properties

The CANDU reactor used in this analysis is a Bruce NGS A reactor with 480 fuel channels. The rated fission power is 2832 MW. Each channel holds 13 bundles, however, because only 12 of the 13 bundles, the 11 centre bundles plus two half-bundles from each end, are irradiated in the core, the average bundle power is therefore 492 kW.

The fissile content of the MOX-fuel bundle considered is listed in Table C.2. The plutonium used in the fuel bundle is assumed to have decayed for twenty years from weapon-grade plutonium. The average exit-burnup of a MOX fuel bundle is 9,700 MWD/teHE and the average mid-burnup is 5,000 MWD/teHE.

C.7.1.2 Analysis

Calculation of the decay heat of a MOX-fuel bundle was performed with the aid of the ORIGEN-S code. The cross-section libraries used in ORIGEN-S are the CANDU burnup-dependent cross-section libraries. They contain all isotopes in the standard ORIGEN-S cross-section and nuclear decay data but have been updated with collapsed multi-group cross sections derived from a WIMS-AECL neutronics analysis of a CANDU 37-element fuel assembly.

Two ORIGEN-S runs were done. One was to simulate a MOX-fuel bundle in a full-power core up to its discharge at exit-burnup. The bundle was then allowed to decay for 350 years. The second ORIGEN-S run simulated a MOX-fuel bundle in a full-power core up to its mid-burnup to simulate a fuel bundle in the equilibrium fuelled core. For this case, the fuel bundle was then allowed to decay for 20 years.

(1) Decay Power Rundown - Figure C.12 shows the decay power rundown for a MOX-fuel bundle discharged at exit-burnup. Also shown in the figure is the power rundown curve for a natural uranium fuel bundle discharged at exit-burnup. At early decay times, the power from a MOX-fuel bundle is less than that from a natural uranium fuel bundle. This trend continues to about one month when the decay power of the MOX-fuel bundle becomes greater. Both power rundown curves are close in magnitude to about 30 years when the decay power from the MOX-fuel bundle becomes 30 percent greater than that from the natural uranium fuel bundle. This is due to the higher actinide quantities in a MOX-fuel bundle which dominate when the high-energy decay gammas from fission products die away.

In Table C.15 the decay power for MOX- and natural uranium fuel bundles for the first ten years is tabulated. As seen in the table, the power output from the spent MOX-fuel bundle is about 20% higher than that from the spent natural uranium fuel bundle.

Figure C.13 shows the integrated-power curve for a MOX-fuel bundle irradiated at full power and discharged at exit-burnup. The integral curve starts at discharge (time zero) and continues for 350 years.

(3) Spent Fuel Bay Heat Load - Heat load in this section is defined as the energy release rate a particular system will experience at a particular time. Fuel bundles discharged from a CANDU reactor are stored in the spent-fuel bay for cooling before they are transferred to a concrete canister for dry storage. Table C.16 lists the integrated decay power from spent MOX- and natural uranium fuel bundles discharged at exit-burnup and Table C.17 lists the decay power from spent MOX- and natural uranium fuel bundles discharged at mid-burnup. Based on these two tables, the spent fuel bay heat load for different cooling time for MOX and natural uranium fuel bundles is calculated and listed in Table C.18.

C.7.1.3 DECAY POWER FROM THE ADVANCED MOX-FUEL BUNDLE

Decay power from a 43-element advanced MOX-fuel was calculated. Calculation method is analogous to that used for the reference MOX fuel bundles. The fuel composition is listed in Table C.19 and the results of the calculations are shown in Figures C.12 and C.13.

For short decay times, up to four days, the decay power from both the reference and advanced MOX bundles are slightly lower than that of equivalently irradiated natural uranium fuel. For long decay times the decay power from both the reference and advanced MOX bundles starts to deviate from that of the natural uranium fuel bundle because nuclear heating from actinides starts to dominate. At 350 years after discharge, the decay heat for the advanced MOX fuel bundle and the reference MOX-fuel bundles are four and three times higher than that of natural uranium, respectively.

C.7.2 FUEL RADIONUCLIDE CONTENT

To address the impact of using MOX fuel on the potential radiation exposure to public arising from accidents, the radionuclide inventory in a MOX fuel bundle at

mid-burnup (representative of core average equilibrium burn-up) to the inventory in a natural uranium fuel bundle at a corresponding mid-burnup.

The ORIGEN-S code was used to calculate the radionuclide inventory. Two ORIGEN-S runs were analyzed - one for a MOX fuel element and the other for a natural uranium fuel element. Both elements were irradiated under constant maximum linear power up to mid-burnup. The maximum linear powers for MOX and natural uranium fuel elements are given in Table C.20. Note that the values in the table account for the maximum-to-average power ratio.

The inventory of radiologically important radionuclides in a MOX fuel element is compared with that in a natural uranium fuel element in Table C.21.

Table C.15
Decay Power at Different Time Steps for
Natural Uranium and MOX-Fuel Bundle Discharged at Exit-Burnup

Cooling Times, years	Decay Power, watts	
	MOX-Fuel Bundle	Natural Uranium Fuel Bundle
0.25	2.90E2*	2.67E2
0.5	1.69E2	1.45E2
0.75	1.16E2	9.35E1
1.0	8.69E1	6.81E1
1.5	5.65E1	4.34E1
2.0	3.96E1	3.04E1
2.5	2.88E1	2.22E1
3.0	2.15E1	1.67E1
3.5	1.66E1	1.30E1
4.0	1.33E1	1.05E1
4.5	1.09E1	8.77E0
5.0	9.25E0	7.56E0
6.0	7.31E0	6.11E0
7.0	6.31E0	5.33E0
8.0	5.77E0	4.90E0
9.0	5.46E0	4.62E0
10.0	5.26E0	4.44E0

* Read 2.90E2 as 2.90×10^2 .

Table C.16
Integrated Decay Power at Different Time Steps for
MOX- and NU-Fuel Bundles Discharged at Exit-Burnup

Cooling Times	Integrated Decay Power, Joules	
	MOX-Fuel Bundle	NU-Fuel Bundle
2 hours	4.79E7 *	5.29E7
3 hours	6.38E7	7.05E7
5 hours	9.20E7	1.02E8
8 hours	1.29E8	1.43E8
12 hours	1.74E8	1.92E8
18 hours	2.33E8	2.58E8
1 day	2.88E8	3.18E8
5 years	1.21E10	1.07E10
6 years	1.23E10	1.09E10
7 years	1.25E10	1.11E10
8 years	1.27E10	1.13E10
9 years	1.29E10	1.14E10
10 years	1.31E10	1.16E10

* Read 2.79E7 as 2.79×10^7 .

Table C.17
Decay Power in the First Day of Cooling for
MOX- and NU-Fuel Bundle Discharged at Mid-Burnup

Cooling Times	Decay Power, watts	
	MOX-Fuel Bundle	Natural Uranium Fuel Bundle
Discharge	2.46E4 *	2.91E4
30 seconds	1.59E4	1.82E4
1 minute	1.42E4	1.62E4
5 minutes	1.07E4	1.23E4
15 minutes	8.44E3	9.84E3
30 minutes	6.91E3	8.12E3
1 hour	5.49E3	6.47E3
2 hours	4.37E3	5.17E3
3 hours	3.88E3	4.58E3
5 hours	3.37E3	3.97E3
8 hours	2.97E3	3.50E3
12 hours	2.65E3	3.12E3
18 hours	2.36E3	2.78E3
1 day	2.15E3	2.54E3
30 days	4.61E2	4.64E2

* Read 2.46E4 as 2.46×10^4 .

Table C.18
Spent Fuel Bay Heat Load from Spent MOX-Fuel Bundle in Different Years

Cooling Time	Spent Fuel Bay Heat Load MW	
	MOX	NU
6 years	4.66	4.42
8 years	4.73	4.47
10 years	4.79	4.52

Table C.19
Mass Composition for an Advanced MOX-Fuel Bundle

	mass, g
^{235}U	37
^{238}U	17,700
Pu	375
Dy	253

Table C.20
Maximum Linear Power Experienced by the MOX-Fuel Element

Fuel Type	Maximum Linear Power	Mid-Burnup, MWD/teHE
MOX	52.6 kW/m	5000
NU	58.5 kW/m	4000

Table C.21
Radionuclide Content of
MOX- and NU-Fuel Elements Irradiated under Maximum Linear Power

Radionuclides	MOX-Fuel Element (Ci)	NU-Fuel Element (Ci)	Ratio MOX/NU
CS137	8.51E+00	6.72E+00	1.27
I131	7.49E+02	7.47E+02	1.00
I132	1.08E+03	1.13E+03	0.96
I133	1.41E+03	1.57E+03	0.90
I135	1.30E+03	1.46E+03	0.89
KR 83M	6.62E+01	9.54E+01	0.69
KR 85	5.09E-01	6.84E-01	0.74
KR 85M	1.30E+02	2.11E+02	0.62
KR 87	2.32E+02	4.14E+02	0.56
KR 88	3.52E+02	5.81E+02	0.56
KR 89	3.68E+02	6.46E+02	0.57
SR 89	3.15E+02	5.64E+02	0.56
SR 90	3.34E+00	5.14E+00	0.65
TE131	6.47E+02	6.67E+02	0.97
TE131M	1.25E+02	1.12E+02	1.12
TE132	1.04E+03	1.08E+03	0.96
TE133	7.64E+02	8.77E+02	0.87
TE133M	4.78E+02	5.95E+02	0.80
XE133	1.42E+03	1.56E+03	0.91
XE133M	4.71E+01	4.94E+01	0.95
XE135	1.30E+02	7.32E+01	1.78
XE135M	3.24E+02	3.15E+02	1.03
XE137	1.19E+03	1.28E+03	0.93
XE138	1.04E+03	1.27E+03	0.829

Figure C.12 Decay Heat as a Function of Time for Spent MOX and Natural Uranium Fuel Bundles Discharged at Exit Burnups

Figure C.13 Integrated Decay Heat as a Function of Time for Spent MOX and Natural Uranium Fuel Bundles Discharged at Exit Burnups

C.8 FUEL SOURCE TERM AND DOSE CONSEQUENCE ASSESSMENT

Canadian licensing requires calculated dose consequences to be less than regulatory dose limits for a wide range of postulated accidents. This, in part, depends on analysis to estimate fractional releases of radionuclides from overheated fuel. Accidents of relevance to fission product release from fuel include large LOCA and events which subject fuel from a single channel to overheating; for example, it is postulated that the 13 bundles in a channel could be discharged from the reactor into containment in the event of an end-fitting failure.

A detailed qualitative assessment was performed to evaluate fission product release from fuel following postulated accidents. The assessment involved comparing MOX fuel behaviour with that of natural uranium fuel under accident conditions. These comparisons were then used to evaluate the impact on the quantity of fission product releases from the fuel.

It is concluded that the fraction of fission product inventory released from MOX fuel under accident conditions will be reduced relative to natural uranium fuel. This reduction is primarily attributed to the reduced decay heat and lower local maximum operating temperatures of the MOX fuel. The basis for this conclusion is discussed in detail below.

C.8.1 LOCA

Only a small range of breaks in large pipes of the primary circuit are predicted to result in fuel sheath failures. These breaks, usually referred to as critical breaks, are characterized by rapid depressurization of the heat transport system and significantly reduced flows to many channels downstream of the break. Furthermore, coolant voiding in the natural uranium core results in increased reactor power that is terminated by shutdown system action. Fuel temperatures rise in the channels experiencing reduced flow. A rise in fuel temperature increases the internal fuel element pressures, whereas a rise in sheath temperature reduces the sheath strength. Increased internal fuel element pressures along with the decreased channel pressure increase fuel sheath stresses which, in turn can result in sheath failure. Operation of the reactor shutdown and the emergency coolant injection systems limits the period of fuel heatup and therefore, only a small number of fuel sheaths of high powered elements may fail.

The fission products within each fuel element are located within the (UO₂) grains, on the grain boundary, and as free inventory. Sheath failure will result in release of the free inventory, but does not imply release of fission products from the grain boundaries or from grains. Releases from grains and grain boundaries are temperature and time dependent. For events of short duration, such as LOCA, where temperatures are limited by blowdown cooling, a significant release of fission products from the grain boundaries and the grains is not expected.

The maximum linear power for the MOX fuel is lower than that for natural uranium fuel, which is important because only high powered element sheaths are at risk of failure.

Calculations have shown that the highest powered MOX fuel element operates at lower centreline and average fuel temperatures as compared to natural uranium fuel. Lower average operating temperatures result in lower fission product gas release within the fuel. In addition, the MOX fuel design ensures no power increase due to coolant voiding during the accident, further limiting any increase in fuel temperature. Lower gas release, together with lower temperatures ensure that the sheath will experience lower stresses than for natural uranium fuel. Therefore, there will be few fuel failures, possibly none at all, during a LOCA in a MOX fuelled core. Certainly, the dose consequences will be bounded by existing safety report analysis for the natural uranium fuelled core.

C.8.2 LOCA with Loss of Emergency Coolant Injection

The consequences following LOCA will change significantly if impaired operation of the emergency coolant injection system is postulated (i.e. lower probability dual failure). The coolant will be discharged through the break from the higher pressure primary system to the lower pressure reactor building. If it is assumed that the channels are not refilled with the emergency coolant, further increase in the fuel temperatures will be driven by decay heat. Fuel temperatures will be limited by heat rejection to the moderator through the pressure tube and the calandria tube. Under these high fuel temperature conditions larger fission product releases from the fuel are expected.

Lower fractional fission product release from the MOX fuel are expected as a result of lower decay heat and lower initial power of the outer ring (ring 4) fuel elements of high power bundles. Consequently they will experience lower temperatures during the transient heatup. A large fraction of the fission product inventory is contained in these 18 elements. The temperatures of the intermediate ring (ring 3) fuel elements will tend to be lower than for natural uranium fuel bundle heatup due to the significantly lower decay heat power in the central 7 elements and the lower temperature of the outer ring of elements. Fission product inventory in the 7 inner elements in the MOX fuel bundle is small because of the low initial fissile content and burnup. All these factors contribute to expected lower fission product releases from MOX fuel during a LOCA with Loss of Emergency Coolant Injection. Therefore, the dose consequences will be bounded by existing safety report analysis for the natural uranium fuelled core.

C.8.3 SINGLE CHANNEL FAILURE EVENTS

Accidents involving a single fuel channel can be divided into two categories: End Fitting Failure type and Flow Blockage type. End fitting failure type events are those where the fuel string is assumed to be ejected from the channel into the fuelling machine vault. Flow Blockage type events are those where the flow in a single channel is significantly reduced resulting in fuel heatup at high power. For both types of events the fuel behaviour in the unaffected channels is similar to that for a small LOCA and, therefore, no fuel failures are predicted. It is only the fuel in one channel which is expected to fail and release fission products.

If any fuel bundles are ejected from the channel following the failure of an end-

fitting, then they are expected to be damaged due to impact with the walls and floor of the fuelling machine vault. The bundles could break into single elements or small clusters of elements. Depending on the extent of damage and the magnitude of any subsequent heatup, the sheath of some or all of the fuel elements in an ejected bundle may fail and release their gap inventory of fission products. If the damage to the fuel element is severe, the fuel itself may be exposed to air and will oxidize. Fuel oxidation results in an enhancement of the release of fission products which are trapped within grains of the fuel. The oxidation rate and the resultant enhanced release rate depend strongly on the temperature of the fuel fragment. In accident analysis all fuel element sheaths are assumed to fail immediately at the time of the accident. All pellets are assumed to be ejected from the sheaths and break into fragments of varying sizes to obtain bounding fractional releases. Fuel fragment sizes are determined by performing sensitivity analysis under assumed extremely severe degraded cooling conditions. Assuming that the enhanced fission product release due to oxidation in air is the same for MOX and natural uranium fuel, then the dose consequences will be bounded by the existing safety report analysis for natural uranium fuel.

For Flow Blockage type events, fuel sheath failures are predicted only for severe flow reductions i.e. more than 50%. Following the assumed blockage, the reactor initially operates at full nominal power with significantly reduced flow in the channel. For a "severe blockage" (i.e. 90% to 100% flow reduction), the released fission products along with the primary coolant, fuel and hydrogen will be discharged into the moderator following the channel failure. A large percentage of the channel fission product inventory will be released from the fuel. No significant differences are expected in the fractional releases from MOX and natural uranium fuel. Therefore dose consequences for these events are expected to remain similar to those predicted in existing safety report analysis.

C.8.4 CONCLUSION - REFERENCE MOX FUEL

It is concluded that the fraction of fission product content released from MOX fuel will, for most accident categories, be reduced relative to natural uranium fuel. This reduction is attributed to a combination of factors such as, reduced heat input to the fuel in the first few seconds following a large LOCA, reduced local maximum operating temperatures of the MOX fuel and reduced decay heat for several days post-accident. Therefore, the dose consequences will be bounded by existing safety report analysis for the natural uranium fuelled core.

C.8.5 ADVANCED CANDU FUEL BUNDLE

A more highly enriched MOX fuel bundle has also been evaluated for potential use in CANDU. The purpose of that bundle is primarily to reduce fuel production requirements. The implications on fission product source terms and dose consequences for accidents are discussed briefly here. The bundle contains nearly twice as much plutonium, commensurate with doubled burnup.

Decay heat and radionuclide content of the advanced fuel is greater than for the reference fuel. The increased decay heat will tend to increase fission product release fraction

from the fuel. However, it is expected that existing safety margins will accommodate potential increases in fission product releases without dose limits being exceeded.

C.9 SUMMARY AND CONCLUSIONS

Reactor safety issues arising from the use of fuel enriched with plutonium beyond the fissile content of the natural uranium fuel used in CANDU reactors have been reviewed. The MOX fuel has been designed to avoid introducing new safety issues. This is accomplished by the introduction of dysprosium poison to the inner fuel elements which makes void reactivity negative for the MOX fuelled core rather than positive as for the natural uranium fuelled core. As a result, reduced dose consequences are expected for large break LOCA events.

It is concluded that detailed consequence analysis will demonstrate that AECB dose limits will be met for the entire spectrum of accidents which must be considered for Canadian licensing.

The introduction of enriched fuel to CANDU reactors introduces a need to consider the potential for criticality outside the heavy water moderated core. Natural uranium fuel normally used in CANDU reactors cannot achieve criticality outside the core. Analysis of representative scenarios indicates that arrays of the reference fuel in handling modules in the irradiated fuel bay cannot become critical. Although our analysis is not as extensive as will be required for licensing we believe it provides a strong indication that criticality issues can be managed through use of the fuel handling modules adapted from CANDU fuel storage practise.

The conclusion of this study is that reactor safety, in the overall context of risk to the public and to in-plant personnel arising from accident situations, is unaffected by the introduction of plutonium enriched MOX fuel.

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