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MARTIN MARIETTA

Letter Report

FMDP CANDU PEIS Data Report Rev. 1

June 5, 1995

Fissile Materials Disposition Program Reactor-Based Technologies Facility Project

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FOR THE U.S. DEPARTMENT OF ENERGY

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February 19, 1996

Mr. T. E. Magette
Tetra Tech, Inc.
4700 King Street, Suite 500
Alexandria, Virginia 22303

Dear Tom:

The purpose of this letter is to transmit to you 20 copies of FMDP PEIS Data Reports issued in response to your Data Calls: the LWR Data Report and the CANDU Data Report.

The LWR submittal contains the response to Appendix A of the Data Call *Commercial LWR Operational Characteristics* and Appendix B *Commercial LWRs Using Mixed Oxide Fuel (Operational Deltas)* modified as appropriate to include data for four large BWRs and eight large PWRs currently in operation.

A markup of Sections 1.0 through 3.0 of the original Tetra Tech FMDP LWR PEIS Data Call was sent to you on February 22, 1995, that provided our comments and suggested revisions to the sections to match the information being provided herein.

The CANDU submittal contains the ORNL response to the FMDP CANDU PEIS Data Call. This data report was prepared with extensive input from, and interaction with, Mr. John Luxat of Ontario Hydro Nuclear and Mr. Michael Fletcher of AECL Technologies, Inc. This revision contains a variety of technical assumptions required to facilitate the aggressive schedule requested in the CANDU Data Call document. Many of the assumptions incorporated in this data report warrant follow-on confirmatory analysis prior to and following the Record of Decision. Assumptions specific to the subject under discussion are contained in each section of the report.

This revision incorporates revised Section 3.8 *Accidents*, which includes the design basis and severe accident basis for the Bruce Reactors, plus the accident analysis methodology employed by Ontario Hydro Nuclear for the Bruce units. Other minor editorial changes are also included in this revision.

Please call me or Sherrell Greene if you have any questions.

Sincerely yours,

George Murphy

George Murphy

GAM:lek

Enclosure

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FOREWORD

A variety of technical assumptions were required to facilitate the aggressive schedule requested in the CANDU Data Call document. One critical assumption incorporated throughout this document is that data supplied by AECL Technologies, Inc., and Ontario Hydro Nuclear from the Plutonium Disposition Study (PDS) is valid in all cases. Thus, the PDS vendor information was generally employed with little or no independent confirmation. In a similar manner, existing CANDU operating data were taken from AECL and Ontario Hydro sources. Many of the assumptions incorporated in this data report warrant follow-on confirmatory analysis prior to and following the Record of Decision. Assumptions specific to the subject under discussion are contained in each section.

FISSILE MATERIALS DISPOSITION PROGRAM (FMDP) PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT (PEIS) CANDU REACTOR USING MIXED OXIDE FUEL OPTION

DATA REPORT

1.0 REFERENCE REACTOR SITE

The Ontario Hydro Nuclear Bruce-A Generating Station was chosen for the reference reactor site for evaluation of the Canadian Deuterium Uranium (CANDU) option to dispose of excess fissile materials. This station, containing four 769 MW(e) generating stations along with its four-unit sister station, Bruce-B, is located on Lake Huron about 300 km northeast of Detroit, Michigan. All data provided is based on the Bruce-A station. The level of detail is the same as that provided in Revision 2 of the *FMDP LWR PEIS Data Call* (ORNL/MD/LTR-9) prepared by Oak Ridge National Laboratory.

2.0 IMPACT DRIVERS

The impact drivers identified in this report include personnel, core changes, waste effluent, worker exposure, transportation and handling, and storage. These seven impact drivers were evaluated to determine the impact "deltas" for a CANDU Reactor using the mixed oxide (MOX) fuel option. An examination of CANDU reactor documents provided by AECL Technologies, Inc., and Ontario Hydro Nuclear led to an assessment of these environmental impacts that may affect the Bruce-A generating station. Section 3 contains information necessary to evaluate environmental impacts associated with each of the listed impact drivers, which are described below.

- **Operational Employment Changes**. The change to a MOX fuel may result in additional or modified fuel handling and shipping activities. These activities could require that additional personnel be hired at the reactor site. Section 3.1 contains projected operating employment changes.

- **Core Changes**. Physical changes required to the reactor core which derive from replacing natural uranium fuel elements with MOX fuel elements. Core changes could alter the accident basis and will alter the source term used in accident analysis. Section 3.2 discusses core changes and Section 3.8 addresses changes to accident basis.

- **Waste Generation Changes**. Additional or modified activities associated with the handling, processing, storage, and shipping of MOX spent fuel may affect waste generation rates. Section 3.3 contains an assessment of mixed waste generated, and Section 3.4 contains spent fuel generation rates.

- **Effluent**. Gaseous and liquid effluents may change as a result of using the MOX fuel. Sections 3.5 and 3.6 provide an assessment of gaseous and liquid radioactive effluents, respectively.

- **Worker Exposure**. The use of MOX fuel may affect refueling frequency and change possible radiation exposure for some categories of workers. Section 3.7 provides an assessment of personnel exposures that may result from operation with MOX fuel.

- **Transportation and Handling.** Section 3.9 addresses the additional risk that may be incurred due to th transport and handling of the MOX fuel elements.

- **Storage.** Use of MOX fuel in a CANDU reactor could result in changes in spent fuel volume. The changes could affect requirements for wet and dry storage space at the generating site. Section 3. contains estimates of the impact on spent fuel storage space required.

3.0 CANDU REACTOR USING MIXED OXIDE FUEL (OPERATIONAL DELTAS)

This section provides descriptions of the physical and operational changes that would be made for Bruce- generating station reactors using MOX fuel in lieu of natural uranium fuel.

3.1 Operational Employment Changes

Provide estimates of the operational employment changes for the Bruce-A generation facility.

Based upon conclusions that may be reached by a study completed by AECL Technologies, Inc., of the Bruc A generating station (Ref. 1) to utilize MOX, no significant changes to the existing CANDU reactors a required. Changes to the physical plant required to store and handle new mixed oxide fuel are described Section 3.4. No changes would be required to the current spent fuel storage and handling methods since t decay heat generation rates and radioactivity of the spent fuel is similar to the natural uranium fuel current used.

Personnel required to handle and provide security for the shipment of new fuel from a fuel fabrication facil to Bruce-A are not included in the estimate of the number of additional personnel needed at the site. Howev added safeguards steps at the Bruce-A site would have to be provided for an accounting scheme whi identifies and tracks the MOX fuel from receipt at the site through loading into the reactor, through the c power cycle, and to subsequent discharge to the spent fuel storage bays. Upon delivery of the MOX fuel the site, the IAEA would be required to verify that the seals placed on each container by the IAEA at t shipping site are intact. The IAEA may delegate this responsibility to national authorities subject to perio IAEA verification. A device based upon passive neutron counting, developed by Los Alamos Natio Laboratory (LANL) (Ref. 2), would then be used to verify that the container did indeed contain the ri amount of MOX fuel. A second passive neutron counting device would also be required in the new f loading room prior to insertion into the new fuel loading magazine to ensure that no diversion of MOX f had occurred.

A secure facility for receiving, inspecting, and storing new fuel would be necessary. Existing facilities procedures used for natural uranium fuel may not be adequate for mixed oxide fuel, mostly due to increased level of security required for the MOX fuel. A new fuel loading room which contains the equipm that loads fuel into the fueling machines and a new fuel transfer room, which, in turn, carry it to the react would need to be built. This building and transfer room must be hardened to IAEA Category 1 requireme Reinforced concrete floors, walls and roof, alarms, and other security features, such as controlled acc would require additional security, new fuel, and spent fuel handling personnel. In addition to the cur personnel at Bruce-A, five security and five new and spent fuel handling personnel would need to be hi

With natural uranium fuel, the Bruce-A refueling machines visit four fuel channels per full power day (F The refueling rate for MOX fuel would require six channels to be visited per FPD, about 1.5 times the cur rate in order to accommodate uniform fuel burnup. Fuel handling experts at Bruce-A estimate that the exis fuel trolleys could be modified to handle the increased tasking. Five additional operating and mainten

personnel would need to be hired to handle the additional fuel handling tasks and provide adequate preventive and corrective maintenance.

Staffing levels at Bruce-A generating station for a four-unit station at the end of 1994 (Ref. 3) and the changes anticipated due to the use of MOX fuel at two of the units are given in Table 3.1.1.

Table 3 1.1. Operational employment changes

	Current Bruce-A generating station employment	Bruce-A generating station employment using mixed oxide fuel
Operations (incl. security)	1004	1019
On-site technical support	476	476
Admin & general	252	252
Total	1732	1747

References

1. AECL Technologies, Inc., *Plutonium Consumption Program, CANDU Reactor Project, Final Report*, July 1994.

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

3. Bruce Nuclear Generating Station A Quarterly Technical Report, Fourth Quarter 1994, Ontario Hydro Nuclear.

3.2 Core Changes

3.2.1 Background

This section provides information concerning the isotopic inventory of the Bruce-A CANDU reactor utilizing natural uranium fuel in an equilibrium core. In addition, this section summarizes expected changes in those inventories due to the introduction of MOX fuel and an increase in discharge burnup from 8300 MWd/MT to 9700 MWd/MT. Unlike U.S. reactors, the CANDU undergoes continuous on-line refueling. The fuel element consists of 37 pins positioned in 3 rings around a central pin and are approximately 0.5 m in length. This section provides estimates of MOX/UO_2 total core inventory ratios for various isotopes. These ratios can be utilized with later sections of this report to help quantitatively assess the differences in emissions or accident source terms between a MOX-fueled plant and a natural uranium-fueled CANDU plant.

Source terms for quantification of the consequences of accidents in Ontario Hydro's CANDU reactors may be expressed in terms of the percentage of the full core inventory of specific radionuclides released in the course of an accident. Typically, dose consequence assessments are performed for short-term and long-term, or chronic, releases utilizing this source term information, together with atmospheric dispersion calculational methodology developed by Ontario Hydro.

3.2.2 Core inventory of radionuclides in a natural-U core

The full core inventory of selected radionuclides in a natural-uranium fueled Bruce-A reactor are given in Table 3.2.1. The radionuclide inventory in this table was calculated with the ORIGEN-S computer code for

a Bruce-A reactor operating at a thermal power of 2832 MW(t). An average discharge burnup of 8300 MWd/MT (see Ref. 2) was used for these equilibrium conditions.

Table 3.2.1. Full core fission product inventory in a natural-U fueled Bruce-A reactor (Ref 2)

Isotope	Curies	Isotope	Curies
Am-241	3.73E+02	Pu-241	1.44E+06
Ba-139	1.43E+08	Rb-86	3.02E+04
Ba-140	1.40E+08	Rh-105	7.49E+07
Ce-141	1.32E+08	Ru-103	1.18E+08
Ce-143	1.23E+08	Ru-105	8.46E+07
Ce-144	4.80E+07	Ru-106	1.49E+07
Cm-242	5.39E+04	Sb-127	7.65E+06
Cm-244	2.75E+02	Sb-129	2.61E+07
Cs-134	7.78E+05	Sr-89	7.76E+07
Cs-136	1.55E+06	Sr-90	1.48E+06
Cs-137	2.03E+06	Sr-91	9.43E+07
I-131	7.87E+07	Sr-92	1.03E+08
I-132	1.15E+08	Tc-99m	1.29E+08
I-133	1.63E+08	Te-127	7.25E+06
I-134	1.76E+08	Te-127m	6.58E+05
I-135	1.51E+08	Te-129	2.55E+07
Kr-85	1.97E+05	Te-129m	3.66E+06
Kr-85m	2.05E+07	Te-131m	1.22E+07
Kr-87	3.98E+07	Te-132	1.13E+08
Kr-88	5.58E+07	Xe-131m	5.3E+05[a]
La-140	1.50E+08	Xe-133	1.55E+08
La-141	1.32E+08	Xe-133m	2.3E+0.7[a]
La-142	1.29E+08	Xe-135	1.59E+07
Mo-99	1.47E+08	Xe-135m	3.2E+07[a]
Nb-95	1.13E+08	Xe-138	5.0E+06[a]
Nd-147	5.03E+07	Y-90	1.57E+06
Np-239	2.14E+09	Y-91	9.64E+07
Pr-143	1.19E+08	Y-92	1.03E+08
Pu-238	3.56E+03	Y-93	1.20E+08
Pu-239	1.65E+04	Zr-95	1.25E+08
Pu-240	1.79E+04	Zr-97	1.36E+08

[a]Reference 3.

3.2.3 Core inventory of radionuclides in a MOX core

The ORIGEN-S computer code has been employed to calculate the radionuclide inventory for a MOX-fuel Bruce-A core. The reactor thermal power is the same as used for the natural-uranium core described in subsection 3.2.2 above. The average discharge burnup for the equilibrium core is 9700 MWd/MT. T

reference fuel design is described in Ref. 1 and is based on a 37-element fuel bundle with an initial plutonium content of 0.232 kg/bundle. Table 3.2.2 shows the core inventory for the MOX-fueled CANDU core (Ref. 2).

Table 3.2.2 Full core fission product inventory in a MOX-fueled Bruce-A reactor (Ref. 2)

Isotope	Curies	Isotope	Curies
Am-241	1.31E+03	Pu-241	3.91E+06
Ba-139	1.32E+08	Rb-86	1.78E+04
Ba-140	1.31E+08	Rh-105	1.06E+08
Ce-141	1.24E+08	Ru-103	1.49E+08
Ce-143	1.08E+08	Ru-105	1.13E+08
Ce-144	3.54E+07	Ru-106	2.76E+07
Cm-242	1.03E+05	Sb-127	1.03E+07
Cm-244	1.63E+02	Sb-129	3.11E+07
Cs-134	2.96E+05	Sr-89	4.46E+07
Cs-136	2.36E+06	Sr-90	7.27E+05
Cs-137	1.84E+06	Sr-91	6.61E+07
I-131	8.38E+07	Sr-92	7.70E+07
I-132	1.20E+08	Tc-99m	1.25E+08
I-133	1.59E+08	Te-127	9.91E+06
I-134	1.66E+08	Te-127m	1.02E+06
I-135	1.45E+08	Te-129	3.07+E07
Kr-85	1.10E+05	Te-129m	4.69E+06
Kr-85m	1.46E+07	Te-131m	1.41E+07
Kr-87	2.61E+07	Te-132	1.17E+08
Kr-88	3.65E+07	Xe-131m	5.6E+05[a]
La-140	1.40E+08	Xe-133	1.53E+08
La-141	1.24E+08	Xe-133m	2.3E+07[a]
La-142	1.18E+08	Xe-135	2.88E+07
Mo-99	1.43E+08	Xe-135m	5.9E+07[a]
Nb-95	8.97E+07	Xe-138	9.0E+06[a]
Nd-147	4.82E+07	Y-90	7.40E+05
Np-239	1.05E+09	Y-91	6.02E+07
Pr-143	1.06E+08	Y-92	7.77E+07
Pu-238	5.63E+02	Y-93	9.75E+07
Pu-239	5.13E+04	Zr-95	1.04E+08
Pu-240	6.18E+04	Zr-97	1.28E+08

[a]Reference 3.

3.2.4 MOX/natural-U core inventory ratios

The ratios of the inventories of selected nuclides in the MOX core to the inventory in the natural-uranium core are given in Table 3.2.3. With respect to tritium it should be noted that the tritium content is not expected to

change significantly with the use of MOX fuel since the dominant component is the tritium content of heavy water moderator which is managed to target levels by periodic removal of tritium at the Darling Tritium Removal Facility (TRF). It is important to note that the ratios quoted here are based on w performed by Ontario Hydro. The results of independent work performed by ORNL have yielded so specific MOX/UO_2 ratios that are significantly different. These differences are currently under investigati

References

1. AECL Technologies, Inc., *Plutonium Consumption Program, CANDU Reactor Project*, July 31, 19 U.S. DOE Contract DE-ACO3-94SF20218.

2. Ontario Hydro, faxed writeup to ORNL from John Luxat, dated April 6, 1995, and faxed writeup to OR from John Luxat, dated April 3, 1995.

3. Ontario Hydro, faxed writeup to ORNL from John Luxat, dated June 2, 1995.

Table 3.2.3. MOX/natural-U core inventory ratios for the Bruce-A reactor (Source: Ref. 2)

Isotope	MOX/natural-U ratio	Isotope	MOX/natural-U ratio
Am-241	3.51	Pu-238	0.16
Ba-139	0.92	Pu-239	3.11
Ba-140	0.94	Pu-240	3.45
Ce-141	0.94	Pu-241	2.72
Ce-143	0.88	Rb-86	0.59
Ce-144	0.74	Rh-105	1.42
Cm-242	1.91	Ru-103	1.26
Cm-244	0.59	Ru-105	1.34
Co-58	1[a]	Ru-106	1.85
Co-60	1[a]	Sb-127	1.35
Cs-134	0.38	Sb-129	1.19
Cs-136	1.52	Sr-89	0.57
Cs-137	0.91	Sr-90	0.49
I-131	1.06	Sr-91	0.70
I-132	1.04	Sr-92	0.75
I-133	0.98	Tc-99m	0.97
I-134	0.94	Te-127	1.37
I-135	0.96	Te-127m	1.55
Kr-85	0.56	Te-129	1.20
Kr-85m	0.71	Te-129m	1.28
Kr-87	0.66	Te-131m	1.16
Kr-88	0.65	Te-132	1.04
La-140	0.93	Xe-133	0.99
La-141	0.94	Xe-135	1.81
La-142	0.91	Y-90	0.47
Mo-99	0.97	Y-91	0.62
Nb-95	0.79	Y-92	0.75
Nd-147	0.96	Y-93	0.81
Np-239	0.49	Zr-95	0.83
Pr-143	0.89	Zr-97	0.94
Xe-131m	1.04[b]	H-3	1.00[c]
Xe-133m	1.00[b]	Kr-83m	0.69
Xe-135m	1.81[b]	Xe-138	1.81[b]

[a]It is assumed that there is no difference in the cobalt inventories since these isotopes are primarily activation products.

[b]Reference 3.

[c]See preceding description regarding the selection of a ratio of 1 for tritium.

3.3 Waste Generation Changes

Possible changes in frequency of fuel assembly handling and processing are expected to change waste generation rates. Provide estimates of waste generation per year if the generating station were to use mixed oxide fuel.

Although the contents of the MOX fuel bundles differ from the natural uranium fuel bundles, externally the bundles are identical except for the level of radiation that would be associated with new MOX fuel compared to natural uranium. This difference will not result in any increase in the quantity or hazard level of waste produced at the Bruce station. The Bruce Nuclear Power Development has facilities on-site for storage of low-, medium-, and high-level radioactive waste. Table 3.3.1 shows the amount of LLW and mixed wastes that were generated by Bruce-A averaged over the past 4 years and notes that there will be no anticipated changes due to the use of MOX fuel.

Table 3.3.1. Waste generation at Bruce-A (m^3/year)

Waste type	Bruce-A generating station with natural-U fuel	Bruce-A generating station with MOX fuel
LLW	1352	No change
Mixed waste	233.6	No change

Management of waste products is primarily affected by the fuel fabrication process. Fabrication of the MOX fuel required by the CANDU reactors will produce waste inherent to the fuel fabrication process. AECL Technologies, Inc. (Ref. 1) evaluated the MOX fuel supply requirements and placed the minimization of waste products as an important criteria in the fuel facility design and process selection. Quantitative estimates of these waste streams are presented in Fig. 3.3.1.

As shown in Fig. 3.3.1, the use of MOX fuel instead of natural uranium fuel in two CANDU reactors would allow for a reduction in overall waste generation for the entire fuel cycle. This reduction would first manifest itself in a reduction of materials currently stored as waste in the United States to be used in the fuel fabrication process. Two tonnes/year of plutonium metal and 170 tonnes/year of depleted uranium in the form of UF_6 may now be utilized as fuel. Additionally, the MOX fuel fabrication process would eliminate the need to mine approximately 6,300 tonnes/year of uranium ore with its associated mill tailings.

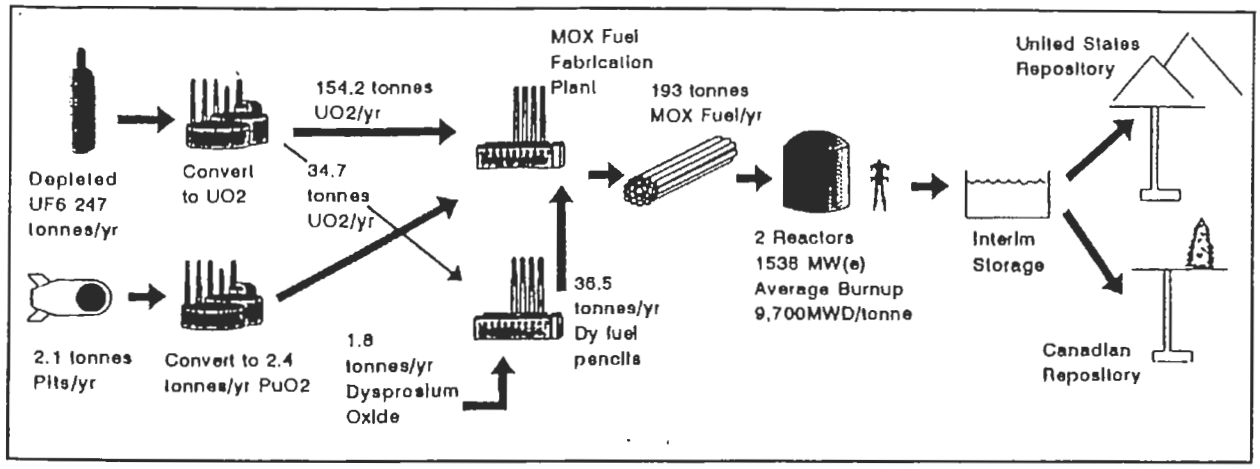

Fig. 3.3.1. Annual material flow balance—two Bruce-A reactors

References

1. AECL Technologies, Inc., *Plutonium Consumption Program, CANDU Reactor Project, Final Report*, July 1994

3.4 Spent Fuel Handling and Storage

Provide estimates of potential increases to spent fuel generation rates. Also, provide estimates of additional wet and dry storage space that might be required to accommodate the additional spent fuel.

CANDU reactors have excellent neutron economy because of the use of heavy water as moderator and coolant. On-power refueling enables 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 0.7 wt% U-235. The requirement of disposing 50 tonnes of weapons-grade plutonium over a period of 25 years can be met by using two of the reactors at Bruce-A station. This plutonium disposition rate requires that the MOX fuel contains approximately 1.6% fissile plutonium and depleted uranium using the existing 37-element fuel design operating within current burnup limits. Table 3.4.1 compares the nuclear characteristics of the MOX fuel design with those for the standard natural uranium fuel design currently utilized in Bruce-A reactors.

Table 3.4.1. Comparison of natural fuel and MOX fuel characteristics

	Existing Bruce-A reactor fuel	Bruce-A reactor with MOX fuel
Fuel bundle dimension	102 mm (4.02") diam 495 mm (19.5") long	Same as existing fuel bundle
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	U^{235} 0.04 kg U^{238} 18.36 kg O_2 2.53 kg Dy 0.17 kg Pu 0.23 kg Total: 21.33 kg
Average burnup	8,300 MWd/tonne	9,700 MWd/tonne
Maximum burnup	15,000 MWd/tonne	15,500 MWd/tonne
Maximum bundle power	950 KW	780 KW
Bundles used per full power day	18	15.5

Information important to note from the above table is that using the same fuel bundle composition of 21.33 kg, the maximum fuel element burnup of the MOX fuel is calculated to be 15,500 MWd per metric tonne, which is essentially the same as the maximum fuel burnup attained by natural fuel elements in the current Bruce-A fuel bundles. The MOX fueling rate is 15.5 bundles per full power day, which is lower than the present fueling rate of 18 bundles per full power day with natural uranium fuel. Due to the slightly higher burnup of the MOX fuel, the quantity of spent fuel that will be produced over the 25-year duration of the MOX program will be approximately 14% less that the quantity of spent natural uranium CANDU fuel produced in generating the same amount of electrical energy.

The processes and facilities required for the interim storage and final disposal of spent MOX fuel will be the same as those for the existing natural uranium 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. During 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.

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 Darlington and Pickering stations. Because of a height restriction imposed by the storage bay transfer duct at Bruce station, these modules would need to be modified to a reduced height of two modules vs four modules. However, they would be compatible with existing storage tanks 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 would be necessary. In all cases, however, equipment designs will be similar to existing equipment in use at other CANDU sites.

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 dry storage cask design would be influenced by the option selected for permanent

disposal of the spent fuel. The containers which would be used for interim dry storage destined for permanent disposal in Canada would probably be based on the Pickering design but would be compatible with the smaller MOX modules. If the spent fuel were returned to the United States, concrete cylinders would be used. Although the containers are designed to receive fuel after it has spent 6 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 10 years after the first loading of the MOX fuel. This would allow sufficient time to license and construct an interim dry storage facility at the site.

The spent CANDU MOX fuel is virtually indistinguishable from spent natural uranium CANDU fuel except that fewer modules would be expended per reactor year. The MOX fuel 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 environmental, safety, or health issues associated with the interim storage, shipping, or disposal of spent MOX fuel as compared to CANDU natural uranium fuel.

Table 3.4.2 shows the expected amounts of spent fuel that would be generated and stored based upon two Bruce-A units operating at 80% capacity factor (typical for the last four years).

Table 3.4.2. Spent fuel generation and storage requirements

	Current Bruce-A generation station	Bruce-A generation station using MOX fuel
Spent fuel bundles generated per year at 80% capacity factor	10,512	9,052
Wet storage (m^3)	9,430	Same as current operation
Dry storage (m^3)	None	Requirements accommodated under current natural uranium planning for dry storage.

References

1. AECL Technologies, Inc., *Plutonium Consumption Program, CANDU Reactor Project, Final Report*, July 1994.

3.5 Gaseous Radioactive Effluents

Provide estimates for expected increases (changes?) of gaseous radioactive effluent, if any, for a reactor using mixed oxide fuel.

3.5.1 Background

Reference 1 describes the approach used to obtain estimates of gaseous and liquid source terms for normal operation for existing light water reactors. That approach was adapted to obtain effluent estimates for MOX-fueled CANDU reactors.

3.5.2 Approach

The effluents from a MOX-fueled CANDU unit are estimated from the measured effluents from Bruce-A, the calculated MOX fuel to natural uranium (NU) fuel radioisotope ratios, and a number of simplifying assumptions.

The current effluent data for Bruce-A is presented in Ref. 4.

The baseline emissions for the NU data are adjusted to account for differences in the fission product isotopic yields for MOX vs NU and the differences in the amount of fission gas released by the fuel. This approach is used for both the gaseous effluents in this section and the liquid effluents in Section 3.6. This is the same approach used in the LWR PEIS data report (Ref. 1).

3.5.3 Major assumptions

Most radioactive effluents originate in the fuel pellets; therefore, the release into the primary coolant throug the cladding is the flowpath of interest. An estimate of the concentration of radionuclides in the primar coolant is necessary, which leads to the first assumption: *The leakage from MOX fuel pins will be equivale to that found in current NU cores.* Historically, leakage from CANDU fuel has been a function manufacturing defects. Although a different manufacturing facility is proposed for MOX fuel, quality contr requirements will be at least equivalent, and any leakage should be the same. This assumption is reasonab since, in general, fuel performance for all reactors improves with time due to operating experience a improved designs. A concern is inhomogeneity of the Pu within the MOX fuel pins would cause worse fu performance. This is consistent with LWR MOX experience and is not unique to a CANDU unit.

Because the fission yield curve for MOX fuel is different from NU fuel, the inventory of fission products w differ. The MOX/NU inventory ratios are listed in Table 3.2.3. *It is assumed that the MOX/NU invent ratios can be uniformly applied to isotopes that are normally gaseous (and to gases dissolved in liq effluent, with the exception of Argon 41 and tritium, which are discussed below.)*

Reference 2 discusses fuel performance in Section 2.3 under normal operating conditions. Table 2.3.3 in R 2 summarizes computer predictions of the fuel for NU and MOX under a high-power envelope, which is m severe than the typical power history. Due to the lower peak element power ratings for MOX fuel, the amo of fission gas released is considerably lower for MOX fuel bundles compared to NU fuel.

Three of the parameters given in Table 2.3.3 of Ref. 2—(1) volume of fission product gas released percentage of fission gas released, and (3) internal gas pressure—can be used to estimate the differ between MOX and NU fuel performance. The percent of fission gas released is used for two reasons—(1 is the same parameter used in the LWR report to estimate similar radioactive effluents (Ref. 1) and (2 results in a bounding value for MOX fuel effluents.

Using the percent of fission gas released to the fuel rod plenum, a MOX bundle will release 43%[1] of that of a NU bundle. *These calculations lead to the assumption that the 43% figure can be used to quantify primary coolant inventory concentrations of gaseous nuclides for a full MOX core as compared to a NU core.*

Because the current emissions and the projected emissions are both for Bruce-A, no scaling is necessary. *It is assumed that there is a direct linear relationship between the primary coolant concentration and the final gaseous or liquid emissions.* This is reasonable since the values are of a similar magnitude but lower, and the same plant systems are used.

3.5.4 Activation products

The approach in Section 3.5.3 is not valid for four isotopes resulting from neutron activation: (1) Ar-41 (gaseous effluent), (2) Fe-55 (liquid effluent), (3) Co-58 (liquid effluent), and (4) tritium (both gaseous and liquid effluent.) These effluents are proportional to the neutron flux in the reactor core. The ratio of average thermal flux in the primary coolant (pressure tubes) for MOX/NU cores is 0.58 (Ref. 3). Argon 41 is produced by activation of argon found in air. Iron 55 and cobalt 58 are neutron activation products of reactor structural materials. The materials in the fuel bundles are planned to be the same, and the core components will remain the same. Therefore, the production rate for these three isotopes should be proportional to the flux ratios in Table 3.5.1.

Tritium is principally produced by thermal neutron absorption by the deuterium in the heavy water primary coolant and moderator. Although tritium will be produced at a lower rate, the average concentration should remain the same. The heavy water is sent to the Darlington de-tritiation plant when the tritium content becomes excessive. Since the average concentration will be a function of the limiting value for de-tritiation and not its production rate, the average will not change.

3.5.5 Results and conclusions

Table 3.5.1 provides the predicted annual gaseous emissions for Bruce-A Station using MOX fuel. In all cases, the predicted releases from a MOX core are less than current releases for a NU core. This is mainly due to the lower element power limits for a MOX core and is reasonable based on the *assumptions* given above.

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[1]This is calculated as follows:

$$\left(\frac{\text{MOX intermediate ring \% gas release}}{\text{NU outer ring \% gas release}} * \frac{\text{no. fuel pins intermediate}}{\text{total no. fuel pins}}\right) + \left(\frac{\text{MOX outer ring \% gas release}}{\text{NU outer ring \% gas release}} * \frac{\text{no. fuel pins outer ring}}{\text{total no. fuel pins}}\right) =$$

$$\left(\left(\frac{7}{19} * \frac{12}{30}\right) + \left(\frac{9}{19} * \frac{18}{30}\right)\right) = 0.4315$$

Note: In the MOX fuel, only the intermediate and outer rings, containing 30 fuel pins, produce any significant power.

Table 3.5.1. Gaseous radioactive effluents for Bruce-A station (Ci/year)

Isotope	Current Bruce-A releases[a]	MOX/NU ratio[b]	Bruce-A releases using MOX fuel[c]	Change (MOX vs NU)
Ar-41	1.6E-03	0.58[d]	0.93E-03	–(0.67E-03)
Kr-85	8.1	0.56	1.95	–6.15
Kr-85m	842.4	0.71	257.2	–585
Kr-88	2292.9	0.65	640.9	–1652.0
Xe-131m	22.0	1.04	9.8	–12.2
Xe-133m	939.7	1.0	404.1	–535.6
Xe-133	6369.2	0.99	2711.4	–3657.8
Xe-135	653.4	1.81	508.5	–144.9
Total Noble Gas (Ci-MeV)	1.43E+04	n/a	n/a	n/a
Tritium	3.54E+04	1.00	3.54E+04	0

[a]Emissions for 1994 from Ref. 4 for all four units at Bruce-A.
[b]Ratio = MOX isotopic inventory/NU isotopic inventory from Table 3.2.3, except for Ar-41 and tritium which are proportional to the thermal neutron flux.
[c]MOX value = (current release)(MOX/NU ratio)(0.43) except for Ar-41 and tritium.
NOTE: To calculate the estimated MOX releases, the noble gas release was converted to Ci, which allows an estimate of CANDU MOX release based on the estimated isotopic distribution of the NU releases. Since Xe-133 dominates the release in a manner similar to LWRs, the MOX noble gas release is a function of the fission product migrating to the primary coolant which is estimated at 43% of that for a NU fueled reactor.
[d]Reference 3.

References

1. *FMDP LWR PEIS Data Report,* Rev. 2, ORNL/MD/LTR-9, Oak Ridge National Laboratory, February 28, 1995.

2. AECL Technologies, Inc., *Plutonium Consumption Program CANDU Reactor Project Final Report,* July 31, 1994.

3. Private communication from R. T. Primm, April 3, 1995.

4. Bruce Nuclear Generating Station A Quarterly Technical Report, Fourth Quarter 1994, Ontario Hydro Nuclear.

3.6 Liquid Radioactive Effluents

Provide estimates for expected releases of liquid radioactive effluent, if any, for a reactor using mixed oxide fuel.

The approach for predicting liquid effluents for Xe-133 and tritium for a MOX core is described in Section 3.5. Fe-55 and Co-58 are activation products and are proportional to the average core flux. The predicted effluents are given in Table 3.6.1 below. The table shows that the fission product related effluents are less than a NU core, and the activation products are essentially the same.

Table 3.6.1. Liquid radioactive effluents (Ci/year)

Isotope	Current Bruce-A releases[a]	MOX/NU ratio[b]	Bruce-A releases using MOX fuel[c]	Change (MOX–NU)
Tritium	4.48E+04	1.00	4.48E+04	0
Fe-55[c]		0.58		
Co-58[c]		0.58		
Xe-133	See Note below	0.99	See Note below	See Note below
Gross Gamma	0.78	n/a	n/a	n/a

[a]Emissions for 1994 from Ref. 4 for all four units at Bruce-A.
[b]Ratio = MOX isotopic inventory/NU isotopic inventory, from Table 3.2.3 for Xe-133.
[c]Ratio = MOX thermal flux/NU thermal flux for Fe-58 and Co-60 (Ref. 3, Sect. 3.5).
NOTE: To calculate the estimate MOX releases, the release was converted to Ci, which allows an estimate of CANDU MOX release based on the LWR releases. Since Xe-133 dominates the release in a manner similar to LWRs, the MOX noble gas release is a function of the fission product migrating to the primary coolant which is estimated at 43% of that for a NU-fueled reactor.

3.7 Additional Worker Exposure

Provide estimates for expected increases of exposure in person-rem, if any, for a reactor using mixed oxide fuel. Any additional integrated exposure estimates should be attributable to (1) additional employees required for mixed oxide fuel operations and (2) the additional or different activities required to be executed.

At Bruce-A, as at all Ontario Hydro nuclear generating stations, the limitation of external and internal exposures to plant personnel is accomplished by a combination of facilities incorporated in the station design and by adherence to a set of approved operating procedures and regulations. A radiation protection program is in place in each station under the station health physicist. The health physicist has the responsibility in parallel with the operating staff for the approval of station radiation protection procedures and the training and qualification of personnel for radiation protection.

The regulatory limits for the maximum permissible exposure for a radiation worker at Ontario Hydro nuclear facilities are as follows:

	Quarterly (rem)	Annual (rem)
Whole body dose	3	5
Skin, thyroid, bone	15	30
Extremities	38	75
Other single organs	8	15

Station data (Ref. 1) indicates that no regulatory limits were exceeded in the 1994 calendar year. The occupational whole body dose information by job category for Bruce NGS A personnel during 1994 is shown in Table 3.7.1.

Based on the design for new fuel handling of the MOX fuel bundles and the similar fission product content of discharged MOX bundles relative to that for natural uranium bundles, no increase in occupational exposure is expected from operation with MOX fuel. It is possible that dose exposure may actually decrease as a consequence of the use of shielding sleeves for handling the new MOX fuel bundles; however, this possibility is not credited. Therefore, the estimated exposures to plant personnel are the same as for natural uranium fuel operation, as shown in Table 3.7.1.

Table 3.7.1. Worker exposure for all operations (person-rem)

Job category	Current Bruce-A exposures	Bruce-A exposures using MOX fuel
Maintenance	233.23	233.23
Operations	88.6	88.6
Others	15.4	15.4
Attached personnel	217.7	217.7
Fuel handling	58.3	58.3
Total	613.23	613.23

Note: Since the CANDU reactors are refueled continuously while operating at power, a separate table for refueling exposures is not applicable. Also, due to differences in grouping job categories, these are different than requested in the data call.

References

1. Bruce Nuclear Generating Station A Quarterly Technical Report, Fourth Quarter 1994, Ontario Hydro Nuclear.

3.8 Changes to Accident Basis

The information presented in this Section is based upon a discussion of CANDU accident sequences and the effects of MOX fuel provided by Ontario Hydro and based upon the Bruce-A CANDU reactors. The Bruce NGS A units are licensed in accordance with the AECB Siting Guide requirements (Ref. 1), which define reference individual and total population dose limits for single failure (a process system failure) and dual failure events (a process system failure plus failure of one of the special safety systems), and specify maximum frequencies of occurrences of these failure events. These limits are shown in Table 3.8.1 below.

The design basis accident categories considered in the accident analysis for the Bruce NGS A Safety Report may also be classified separately from the Siting Guide classification employed in licensing on the basis of frequency of occurrence of initiating events. For example, the most recently licensed Ontario Hydro reactors at the Darlington station were licensed using the AECB Consultative Document C-6 (Ref. 2) on a trial basis. This consultative document, which has not yet been made into a regulatory document, employs five classes of events based upon frequency, with an associated dose consequence limit for each class, as shown in Table 3.8.2.

For the purpose of conforming to the manner in which accidents are classified in U.S. environmental reports, a set of design basis accidents in the Bruce NGS A Safety Report have been categorized as *nonsevere accidents*. These accidents generally correspond to the single failure events defined by the Canadian Siting Guide. Furthermore, the nonsevere accidents are further classified on the basis of initiating failure event frequency.

A second set of design basis accidents are categorized as *severe accidents*. These accidents generally correspond to the dual failure accidents analyzed for the Bruce NGS A Safety Report. These accidents involve either substantial physical deterioration of the fuel due to overheating or an impairment of the containment structure envelope or a subsystem such that increased releases of radioactivity from the containment can occur.

For both the nonsevere and severe accident categories, a spectrum of accidents is identified. The fission product source terms for releases from the fuel and for releases from containment for selected accidents in a natural uranium-fueled core are summarized based upon information contained in the Bruce NGS A Safety Report. Specific features of the multiunit CANDU design at Ontario Hydro's nuclear generating stations which influence accident consequences are discussed, as appropriate.

Subsequently, the impacts of operating Bruce NGS A reactors with a full core of MOX fuel on the fission product source terms for nonsevere and severe accidents are estimated.

Table 3.8.1. Regulatory Siting Guide limits applicable to Bruce NGS A

Situation	Maximum frequency	Individual dose limit (see note)	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.

Table 3.8.2. AECB consultative document C-6 event class and dose consequence limits (Applied on a trial basis in Darlington licensing)

Class	Initiating event frequency (per year)	Whole body dose limit (rem)	Thyroid dose limit (rem)
1	$1 \geq f > 10^{-2}$	0.05	0.5
2	$10^{-2} \geq f > 10^{-3}$	0.5	5
3	$10^{-3} \geq f > 10^{-4}$	3	30
4	$10^{-4} \geq f > 10^{-5}$	10	100
5	$10^{-5} \geq f$	25	250

3.8.1 Nonsevere accidents in natural uranium-fueled CANDU reactors

Typical nonsevere accidents analyzed as part of the design basis for CANDU reactors are listed in Table 3.8.3 together with the associated initiating event frequencies associated with the failures. The event frequencies are representative values taken from the draft Bruce NGS B (sister four-unit plant to Bruce-A) risk assessment, since the Bruce NGS A risk assessment is in the early stages of preparation and quantification of initiating event frequencies has not been completed.

Table 3.8.3. Nonsevere event frequency categorization analogous to categorization employed in U.S. environmental reports

Frequency categorization	Range of frequency (events/year)
Moderate frequency accident	1 to 2×10^{-2}
Infrequent accident	2×10^{-2} to 2×10^{-3}
Low frequency accident	$<2 \times 10^{-3}$

Radioactive releases for the design basis accidents are reported in the Bruce NGS A Safety Report (Ref. 3) in terms of the dose consequences to the most critical individual member of the public as well as the total population dose. Of the nonsevere accidents listed in Table 3.8.4, two have been selected based upon the severity of consequences for events in the infrequent to low frequency classifications. The two selected accidents are a rupture of a single steam generator tube (infrequent) and a large break loss of coolant accident (low frequency). These two accidents reflect a conservative bound on releases in the nonsevere accident spectrum since other accidents either have insignificant releases of radioactivity or have lower dose consequences than the two selected accidents.

The steam generator tube rupture accident does not result in fuel overheating and, therefore, has no fission product source term due to consequential fuel failures. Activity in the heat transport coolant is released into the affected steam generator by the coolant discharge from the ruptured tube. Subsequently, release of activity outside of containment can occur from the secondary side through such pathways as the atmospheric steam discharge valves and steam generator blowdown lines. The airborne releases are primarily radioiodine, some noble gases and tritium from pre-existing concentrations in the coolant and are strongly dependent upon

whether or not steam release occurs from the secondary side system. Typically, in accident analyses the tritium concentration in the coolant is assumed to be at a conservative level of 3 Curies/kg and a conservative level of radioactive iodine (925 Curies, corresponding to the shutdown limit), is assumed to be present in the coolant due to pre-existing fuel defects. The isotopic activity release for a steam generator tube rupture in which the condenser steam discharge valves are assumed to be unavailable to assist the cooldown of the secondary side is summarized in Table 3.8.5.

The large break loss of coolant accident is the most limiting of the nonsevere accidents in that consequential fission product releases due to limited transient fuel overheating are calculated to occur. The containment response to this accident exerts a governing role in determining the potential releases to the environment. Fission product releases from the fuel and from containment for this accident are summarized in Table 3.8.5. The isotopic information on source terms from containment for nonsevere accidents in the existing natural uranium-fueled Bruce A reactors serve as an appropriate baseline for evaluation of differences between natural uranium and MOX-fueled cores in these reactors.

3.8.2 Nonsevere accident considerations resulting from MOX fuel

The characteristics of MOX-fueled CANDU reactors, as described in Ref. 4, are such that lower fission product releases are expected for nonsevere accidents due to a number of factors.

One major factor is that the core power distribution is significantly flattened relative to the current power distribution in the natural uranium-fueled Bruce A reactors. Consequently, the maximum powered elements of fuel bundles in the high-powered regions of the core operate at significantly lower power ratings than with natural uranium fuel. This significantly reduces the fission gas release from MOX fuel pellets into the free volume within the fuel elements. In addition, there is a reduction in the period during the bundle irradiation where maximum element ratings exceed 40 kW/m, above which higher fission gas release rates occur.

Heterogeneity of the mixed plutonium and uranium oxide powder in MOX fuel is a potential factor which can cause enhanced fission gas releases from the fuel. Experience with CANDU MOX fuel test irradiations conducted in the prototype Nuclear Power Demonstration Reactor (NPD) during the 1970s has indicated no significant increase in fission gas release despite heterogeneity of the mixed oxides. The irradiations were performed at maximum element linear power ratings (40 to 50 kW/m) and burnups in excess of 17,000 MWd/tonne Heavy Element that are in similar ranges to those expected in a MOX-fueled Bruce A core. This would suggest that no increased fission gas releases are expected for CANDU MOX fuel, and that the ratios of isotope core inventories derived from the information in Table 3.2.3 can be applied directly to estimate the changes in fission product source terms for a MOX core.

An additional factor affecting the large loss of coolant accident source term in a MOX core is the change in the coolant void reactivity coefficient to a negative value as opposed to the positive void coefficient in a natural uranium-fueled core. This occurs as a result of the use of dysprosium as a burnable poison in the central cluster of seven elements of the 37-element fuel bundle. As a result, less energy generation occurs in the MOX fuel during the early phase of a loss of coolant accident, thus resulting in lower fuel temperatures when fuel cooling degrades. In turn, the lower fuel temperatures will result in less fission product release during the accident. This beneficial effect has not been credited in estimating the change in fission product source term for this nonsevere accident.

The change in coolant void reactivity coefficient from a positive value in a natural uranium core to a negative value in MOX core introduces the possibility of different accident response in a MOX core for events involving an increase in coolant density, such as overcooling of the heat transport system and collapse of coolant void due to liquid refill of voided fuel channels. These have been addressed in Ref. 4 and determined

to be within the existing analyzed rates of positive reactivity insertion accidents. In addition, injection of light water emergency coolant into voided fuel channels following a loss of coolant accident poses no hazard because in CANDU reactors light water is a neutron poison, and thus emergency coolant injection adds negative reactivity to the core.

3.8.3 Nonsevere accident source terms for a MOX-fueled core

The source terms for nonsevere accidents are estimated from the natural uranium-fueled core source terms given in Table 3.8.5 multiplied by the full core inventory MOX/UO2 isotopic ratios given in Table 3.2.3.

3.8.4 Severe accidents for CANDU reactors

Severe accidents may be defined as very low frequency accidents for which there is the possibility of larger releases of radioactivity into containment or from containment to atmosphere. Typically, the potential for increased activity release arises as a consequence of either impairment of the containment boundary or extensive overheating of the fuel resulting in severe damage to the fuel and increased fission product release from the fuel.

The phenomena of fuel melting and relocation and other related severe accident phenomena have been studied extensively for existing LWRs in the United States. As a result, severe accident progression events have been generally treated more mechanistically than have severe accident event progression considerations in Canada, and the amount of source term information on the accident profiles (particularly for severe accidents) for U.S. LWRs is quite extensive compared to what is available for the CANDU reactors. It is important to recognize that the approaches to severe accident considerations and the analyses performed are different for the CANDU plants in addition to basic differences from LWRs in the design of the reactors themselves.

A number of severe accidents in CANDU reactors are routinely analyzed as part of the licensing safety analysis due to the Siting Guide requirement to analyze dual failure events (a process system failure coincident with failure of one of the special safety systems). Such events include a loss of coolant accident with a coincident pre-existing impairment of the containment boundary and a loss of coolant accident with coincident failure of the emergency coolant injection system (LOCA/LOECI accident). Typical severe accidents analyzed in the Bruce A Safety Report and the associated event frequencies are listed in Table 3.8.6.

Severe accidents resulting in major fuel damage do not directly lead to significantly increased releases of radioactivity to the environment if containment integrity is maintained. Consequently, for CANDU, as for other reactor systems, identification and limitation of challenges to containment integrity is a major factor with respect to limited releases to the environment. Source terms for two of the severe accidents listed in Table 3.8.6, a large loss of coolant with failure of emergency coolant injection, and a large loss of coolant with a large pre-existing impairment of containment (deflated airlock seals on the accident unit and an airlock door open on an adjacent unit) are presented in Table 3.8.7.

3.8.5 Severe accident considerations resulting from MOX fuel

Based upon information presented in Ref. 4, the consequences of severe accidents in a MOX-fueled core are not expected to differ substantially from those associated with a natural uranium-fueled core. In particular, severe accident sequences leading to fuel overheating and damage to fuel will have similar consequences based upon the fact that there are not significant differences in the decay heat characteristics for the two fuel types and, furthermore, as shown in Table 3.2.3, the full core inventory of the majority of radioisotopes do not differ by significant amounts.

The higher content of heavy element isotopes (plutonium and americium) in the MOX core does not alter the source term since these isotopes are not significant components of the release source term.

Since the use of MOX fuel does not materially impact the primary phenomena occurring during severe accident sequences, there are no additional challenges to containment as compared to accidents occurring in a natural uranium-fueled reactor. With similar decay heat levels, the timing of the progression of events, such as core heatup, hydrogen generation, heat rejection to the moderator, etc., will be essentially unchanged.

3.8.6 Severe accident source terms for MOX-fueled CANDU reactors

Source terms for selected severe accidents have been estimated for Bruce NGS A operating with a MOX core. The source terms for a natural uranium-fueled core, given in Table 3.8.7, were used as the basis, and the releases are scaled by the MOX/UO2 ratios for individual isotopes presented in Table 3.2.3.

Since the source term for severe accidents involving significant degradation of fuel cooling, such as the LOCA/LOECI event, is generally dominated by relatively slow heatup of fuel from decay power levels and involves releases driven by thermal diffusion processes, no additional factors to modify the release estimates are considered necessary.

The dose consequence to members of the public and environmental impacts of these severe accident releases from MOX-fueled reactors at Bruce NGS A will not differ significantly from the consequences of the same accident occurring in a natural uranium-fueled reactor.

References

1. Hurst, D. G., F. C. Boyd, "Reactor Licensing and Safety Requirements," AECL 1059, Paper 72-CNA-102, 12th Annual Conference of the Canadian Nuclear Association, Ottawa, June 1972.

2. Requirements for the Safety Analysis of CANDU Nuclear Power Plants, AECB Consultative Document C-6, Revision 0, Atomic Energy Control Board of Canada, June 1980.

3. Ontario Hydro, Bruce NGS A Safety Report, 1994 printing.

4. AECL Technologies, Inc., "Plutonium Consumption Program, CANDU Reactor Project," July 31, 1994, U.S. DOE Contract DE-AC03-94SF20218.

Table 3.8.4. Typical CANDU design basis nonsevere accidents

Failure event	Initiating event frequency	Frequency category	Fission product release consequences
Loss of regulation			
– Loss of reactivity control	2×10^{2}	Moderate	No releases
– Loss of bulk power regulation	2×10^{-2}	Moderate	No releases
– Loss of spatial power regulation	1×10^{-1}	Moderate	No releases
– Loss of heat transport pressure control	3×10^{-2}	Moderate	No releases
– Loss of secondary side pressure control	1×10^{-1}	Moderate	No releases
Loss of Coolant Accidents (LOCA)			
– Small break LOCA	2×10^{-3}	Infrequent	Coolant activity release into containment
– Large break LOCA	2×10^{-4}	Low	Limited releases from fuel
– Pressure tube rupture	8×10^{-3}	Infrequent	Releases from fuel in failed channel
– Single steam generator tube rupture	3.6×10^{-3}	Infrequent	Coolant activity + pre-existing SG tube leak outside containment
Steam supply system failures			
– Small steam line break	1×10^{-1}	Moderate	Activity release due to pre-existing SG tube leak
– Large steam line break	2×10^{-4}	Low	Activity release due to pre-existing SG tube leak
Feedwater system failures			
– Total loss of feedwater	2×10^{-3}	Infrequent	No releases
– Small feedwater line break	4×10^{-3}	Infrequent	Activity release due to pre-existing SG tube leak
– Large feedwater line break	1×10^{-3}	Low	Activity release due to pre-existing SG tube leak
Total loss of Class IV electrical power to a unit	3×10^{-3}	Infrequent	No releases
Loss of moderator inventory	1×10^{-2}	Infrequent	Moderator tritium release into containment
Service water systems failures			
– Total loss of low-pressure service water	1×10^{-2}	Infrequent	No releases
– Total loss of high pressure recirculating water	1×10^{-2}	Infrequent	No releases

Table 3.8.5. Source terms for releases from containment for selected CANDU nonsevere accidents (Natural Uranium Core)

Isotope	SG tube rupture (*Note 1*) release (Ci)		LOCA release (Ci)	
	Short term	Long term	Short term	Long term
Kr-83m	0	0	4.2E+00	4.2E+00
Kr-85	0	0	1.6E-01	5.2E+03
Kr-85m	0	0	9.2E+00	1.0E+01
Kr-87	0	0	1.7E+01	1.7E+01
Kr-88	0	0	2.5E+01	2.5E+01
I-129	0	0	9.6E-05	9.6E-05
I-131	3.7E-01	5.4E-01	1.6E+00	2.0E+00
I-132	1.7E-01	2.5E-01	1.9E+00	1.5E+01
I-133	1.9E-01	2.7E-01	2.2E+00	2.2E+00
I-134	1.4E-02	2.0E-02	2.1E+00	2.1E+00
I-135	7.9E-02	1.1E-01	1.8E+00	1.8E+00
Xe-131m	1.5E-01	2.1E-01	6.8E-01	8.9E+03
Xe-133	1.8E+01	2.6E+01	7.5E+01	4.2E+05
Xe-133m	1.2E+00	1.7E+00	1.1E+01	1.7E+04
Xe-135	9.5E+01	1.4E+02	4.6E+00	6.6E+02
Xe-135m	5.2E+02	7.6E+02	1.2E+01	1.3E+01
Tritium	7.5E+04	1.1E+05	1.2E+02	1.2E+02

NOTE: 1. Coincident failure of the condenser steam discharge valves (CSDVs) of the secondary side steam supply system is assumed.

Table 3.8.6. Typical CANDU severe accidents analyzed in safety analysis

Accident event	Event frequency	Overpressure period(s) *(Note 2)*	Repressurization time (h) *(Note 3)*
Large Loss of Coolant with coincident containment impairment:			
– Failure of all instrumented pressure relief valves (IPRVs) *(Note 1)*	$< 2 \times 10^{-7}$	56	46
– Deflated airlock seals on accident unit	$< 2 \times 10^{-7}$	56	3.4
– Failure of isolation dampers to close	$< 2 \times 10^{-7}$	56	3.2
– Deflated airlock seals on accident unit and open airlock door on a nonaccident unit	$< 2 \times 10^{-7}$	56	1.4
Large loss of coolant and failure of emergency coolant injection system (LOCA/LOECI)	$< 2 \times 10^{-7}$	56	46

Notes:

1. IPRVs are a set of pneumatically controlled relief valves on the manifold connecting the containment pressure relief duct to the vacuum building.
2. Overpressure period refers to the short period following the pipe rupture during which the containment pressure is above atmospheric prior to discharge to the vacuum building returning the pressure subatmospheric.
3. Repressurization time is the time taken for containment pressure to return to atmospheric following discharge into the vacuum building.

Table 3.8.7. Source terms for releases from containment for selected CANDU Severe accidents (Natural Uranium Core)

Isotope	LOCA with deflated airlock seals and open airlock door on nonaccident unit release (Ci)		LOCA/LOECI *(Note 1)* release (Ci)	
	Short term	Long term	Short term	Long term
Kr-83m	1.3E+01	1.3E+03	4.2E+00	4.2E+00
Kr-85	1.1E+00	5.3E+03	1.6E-01	4.2E+04
Kr-85m	3.0E+01	3.9E+03	6.3E+00	9.8E+00
Kr-87	5.6E+01	1.1E+03	1.7E+01	1.7E+01
Kr-88	7.9E+01	5.4E+03	2.5E+01	2.5E+01
I-129	1.0E-03	1.0E-03	9.5E-05	9.6E-05
I-131	1.8E+01	2.4E+01	1.5E+00	5.6E+00
I-132	2.1E+01	2.3E+02	1.9E+00	1.4E+02
I-133	2.2E+01	2.3E+01	2.2E+00	2.3E+00
I-134	1.8E+01	1.8E+01	2.1E+00	2.1E+00
I-135	1.7E+01	1.7E+01	1.8E+00	1.8E+00
Xe-131m	4.5E+00	1.4E+04	6.8E-01	9.4E+04
Xe-133	3.7E+02	9.5E+05	7.5E+01	4.4E+06
Xe-133m	5.4E+01	7.2E+04	1.1E+01	1.8E+05
Xe-135	1.5E+01	3.8E+04	4.6E+00	1.0E+04
Xe-135m	3.8E+01	7.5E+03	1.2E+01	1.4E+01

NOTE: 1. The LOCA/LOECI source terms are evaluated from the 1989 issue of the Bruce NGS A Safety Report. Analysis is planned to update this section of the Safety Report in 1996.

3.9 Transportation and Handling of Mixed Oxide Fuel

Provide a description of the transportation system to be used for the transport of the mixed oxide fuel from the FMEF facility at Hanford to the Bruce-A generating station. Identify the curie content of a typical shipment and the approximate number of required shipments per year. Identify any different or additional loading/unloading/handling facilities that might be required at the Bruce-A generating station because of the mixed oxide fuel.

This section provides a description of the transportation system to be used for the transport of mixed oxide fuel from the FMEF facility at Hanford to the Bruce-A generating station in Canada. CANDU reactors feature on-power refueling. While the reactor is operating, a pair of fueling machines latch onto opposite ends of one

of the fuel channels. One fueling machine then pushes fuel bundles into the reactor while the other machine receives the ejected fuel bundle.

The disposition of 50 MT of weapons-grade plutonium will require two reactors at the Bruce generating station to carry out the disposition mission in 25 years. Each reactor is expected to have fueling requirements of 15.5 bundles per full power day. The reactors are expected to perform at an 80% capacity factor. Based on two reactors, 9052 fuel bundles per year would be required for approximately 25 years to disposition the 50 MT of plutonium. It is estimated that a total of 220,000 fuel bundles would be required over the lifetime of the project. Based on the annual requirements for fuel, the disposition mission could be accomplished in 24.3 years. Use of MOX fuel results in a 15% savings in the total number of fuel bundles needed in relation to natural uranium fuel (requiring 18 bundles/day). Each fuel bundle will have a core residence time of approximately 1 year.

Transport of fresh MOX fuel to Canada will be the responsibility of the U.S. Department of Energy (DOE) Transportation Safeguards Division (TSD) and will be coordinated with the Canadian government. Transport will utilize a TSD Safe Secure Trailer (SST). Fresh MOX fuel bundles will be packaged in a standard stainless steel 55-gal (U.S.) drum. The container would be capable of holding seven CANDU MOX fuel bundles. The container would have to be certified as a type B package and approved for use within both Canada and the United States. The package would have to undergo certification by the U.S. DOE, U.S. Nuclear Regulatory Commission (NRC), and/or the U.S. Department of Transportation (DOT), as well as the Canadian AECB and the Canadian Ministry of Transport. While the above package has been approved for bulk shipments of Category 1 materials, it has not yet been approved for the transport of CANDU MOX fuel bundles. [Editor's note: The AECL report does not specifically identify the package by name or certificate of compliance number].

The SST is capable of carrying 48 drums on pallets with a cargo restraint system: six pallets of eight drums, with each pallet having four drums stacked two high. Each drum (seven bundles) is estimated to weigh between 400–450 lb. Because the SST payload capacity is approximately 12,000 lb, only about 24 drums will be transported per SST shipment (168 bundles). Based on the annual fuel requirement of 9052 bundles, approximately 54 SST shipments per year will be required (slightly more than one per week). Information about fresh CANDU MOX fuel shipments is summarized in Table 3.9.1. This table also lists the CANDU natural uranium fuel cycle for comparison purposes.

Table 3.9.1 Fresh MOX fuel shipments

Reactors needed	Curies per package (7 bundles)	Curies per shipment (24 bundles)	Shipments per year	Assemblies needed per year	Assemblies to disposition 50 MT
CANDU MOX (two reactors at Bruce station)	961.0	23,070	54.0	9,052	220,000
CANDU natural U cycle	0.09	NA	NA	10,512	NA

Gamma radiation fields from new MOX fuel are about four times higher than for natural uranium (20 mrem/h contact for MOX fuel; 5 mrem/h contact for natural uranium) (Ref. 1). In addition, MOX fuel will have a 5 mrem/h neutron field (Ref. 1).

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 loaded into the reactor fueling machine. This will allow MOX fuel to be handled with radiation doses to personnel equivalent to, or lower than, natural uranium. Shipping containers and fuel storage facilities will provide additional shielding. A criticality analysis of seven new MOX fuel bundles (without shielding sleeves) in the 55-gal drum, filled with water, shows that these is no danger of criticality.

References

1. AECL Technologies, Inc., *Plutonium Consumption Program CANDU Reactor Project, Final Report*, July 31, 1994.