FUEL HANDLING ACCIDENT ANALYSIS FOR THE UNIVERSITY OF MISSOURI RESEARCH REACTOR'S HIGH ENRICHED URANIUM TO LOW ENRICHED URANIUM FUEL CONVERSION INITIATIVE

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The undersigned, appointed by the dean of the Graduate School, have examined the thesis entitled;

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Abstract

In accordance with the 1986 amendment concerning licenses for research and test reactors, the MU Research Reactor (MURR) is planning to convert from using High-Enriched Uranium (HEU) fuel to the use of Low-Enriched Uranium (LEU) fuel. Since the approval of a new LEU fuel that could meet the MURR's performance demands, the next phase of action for the fuel conversion process is to create a new Safety Analysis Report (SAR) with respect to the LEU fuel. A component of the SAR includes the Maximum Hypothetical Accident (MHA) and accidents that qualify under the class of Fuel Handling Accidents (FHA). In this work, the dose to occupational staff at the MURR is calculated for the FHAs.

The radionuclide inventory for the proposed LEU fuel was calculated using the ORIGEN2 point-depletion code linked to the MURR neutron spectrum. The MURR spectrum was generated from a Monte Carlo Neutron transPort (MCNP) simulation. The coupling of these codes create MONTEBURNS, a time-dependent burnup code. The release fraction from each FHA within this analysis was established by the methodology of the 2006 HEU SAR, which was accepted by the NRC. The actual dose methodology was not recorded in the HEU SAR, so a conservative path was chosen. In compliance to NUREG 1537, when new methodology is used in a HEU to LEU analysis, it is necessary to re-evaluate the HEU accident.

The Total Effective Dose Equivalent (TEDE) values were calculated in addition to the whole body dose and thyroid dose to operation personnel. The LEU FHA occupational TEDE dose was 349 mrem which is under the NRC regulatory occupational dose limit of 5 rem TEDE, and under the LEU MHA limit of 403 mrem. The re-evaluated HEU FHA occupational TEDE dose was 235 mrem,

which is above the HEU MHA TEDE dose of 132 mrem. Since the new methodology produces a dose that is larger than the HEU MHA, we can safely assume that it is more conservative than the previous, unspecified dose.

Chapter 1

Introduction

1.1 Background

The University of Missouri Research Reactor (MURR) is a pool-type reactor that is currently fueled by 93% enriched U-Al_x curved-plate fuel elements arranged radially around the MURR's flux trap. Standard operation is at 10 MW thermal, making MURR the largest research reactor on a University campus in the United States of America. The Curators of University of Missouri-Columbia commissioned the reactor to be built to provide a new field of research and advancement for the University while endeavoring to provide beneficial products to the local community. Since coming online in the fall of 1966, the MURR has served its mission in conducting research and selling medical isotopes to hospitals, both local and abroad. As a part of the Reduced Enrichment for Research and Test Reactors (RERTR) program, the MURR will be converting to low-enrichment monolithic U-10Mo fuel developed at Argonne National Laboratory. A multitude of different analyses must be conducted to determine the implications of the conversation. One of these analyses is a study of the radiological consequences of a fuel handling class accident.

The fuel handling accidents (FHA) are a class of accidents caused by events or scenarios that could cause a breach in the fuel cladding. Events which could cause an accident in this category include: a fuel handling event where a fuel plate is damaged or scratched severely enough to breach the cladding, simple failure of

the fuel cladding due to a manufacturing defect or corrosion, or overheating of a fuel element with subsequent potential cladding failure due to a loss of primary coolant accident (LOCA) or loss of primary coolant flow (LOFA).

The Nuclear Regulatory Commission (NRC) provides guidelines and regulations that define the methodology for calculating radiation dose from a fuel handling accident. The methodology from the MURR SAR is the implementation of NRC Guide 1.183, "Alternative Radiological Source Terms for Evaluation Design Basis Accidents at Nuclear Power Reactors" [1] and NRC Guide 1.195, "Methods and Assumptions for Evaluating Radiological consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors" [2]. Reg. Guide 1.183 require licensee to calculate the Total Effective Dose Equivalent (TEDE) dose, which is the combination of the Committed Effective Dose Equivalent (CEDE) from inhalation and the Deep Dose Equivalent (DDE) from immersion in a semi-infinite cloud of nuclides for an alternate source term. Reg. Guide 1.195 requires the licensee to calculate the CEDE for designed based accidents with accepted source terms. Both regulations agree that public dose does not need to be considered if containment integrity is held.

1.1.1 Previous Work

Accident	**Dose [mrem]**
MHA	132.28
FHA: FHA	Dose < MHA
FHA: 60 Day Decay FHA	0.78
FHA: Fuel Element Malfunction	Unspecified

Table 1.1: Dose to Operational Personnel from 2006 SAR

The MURR applied to renew its 40 year operation license in 2006, which was accepted by the NRC. The renewal required a re-evaluation of the Safety Analysis Report (SAR) for its 10 MW license in 1974. The 2006 SAR provides a detailed report about facility and considered multiple accident scenarios including but not limited to: LOCA, LOFA, FHA, and Maximum Hypothetical Accidents (MHA).

The dose to operation personnel from these scenarios is tabulated in Table 1.1 It should be noted that when the SAR is referenced within this thesis, it is referring to the HEU SAR, as this report is a component for the LEU SAR.

The HEU to LEU conversion team, a collaboration of the MURR facility, Argonne National Laboratory, and the University of Missouri-Columbia, has developed an in-depth radionuclide inventory for the proposed LEU fuel. The provided inventory of radionuclides was the basis for the dose from fission release calculations done within the each accident analysis.

1.1.2 MURR LEU Conversion Project

The Reduced Enrichment for Research and Test Reactors (RERTR) program was initiated by the U.S. Department of Energy (DOE) in 1978. The program's mission was to reduce the proliferation risk from fuel for non-power reactors by converting HEU fuel to LEU fuel. The new fuel is to maintain the performance of the MURR with LEU fuel as it was with HEU fuel. [3]. RERTR became a domestic program in 1986, when the NRC amended 10 CFR Part 50, which required all current, and future licensees to convert their reactor fuel from HEU to LEU once a suitable fuel was available [4]. With the success of RERTR, in 2004 the DOE's National Nuclear Security Administration created the Global Threat Reduction Initiative (GTRI), which incorporated RERTR program as well as nuclear material protection and removal programs. Argonne National Labs, under the GTRI Reactor Conversion Program, has developed a high-density, low-enrichment monolithic U-10Mo plate fuel to replace MURR's low-density, high-enrichment U-Al_x dispersion plate fuel. This fuel will be used for the MURR conversion. The fuel has a total density of 17.02 g/cm^3 compared to the current density of 6.4 g/cm^3. To achieve the same performance with LEU as from HEU, MURR will operate at a 20% power uprate with LEU fuel to 12 MW, compared to the 10 MW with HEU fuel. The U-238 content in the reactor will increase by 955%. The changes of power, inventory, radial distribution, and radial geometry merited an evaluation in the source term

for Accident Analyses in the LEU core to update the methodology used.

1.2 Motivation and Scope of Work

This thesis updates the radiation dose to the operation personnel within containment during a FHA class accident at the MURR for both the LEU and HEU fueled cores. The major aspects of dose calculations are establishing the core inventory, calculating the release, dispersion, and resultant dose from the nuclides within containment.

The difference between this thesis and the previous work for the MURR SAR include inventory calculations and dose methodology. The dose paths were not recorded in the previous SAR for the FHA. A dose path with the conservativeness of the MHA was chosen for the Fuel Handing Accident Analysis while Fuel Malfunction Analysis keeps the MHA path.

The core inventory was developed by using the MONTEBURNS code. MONTEBURNS couples the neutronic Monte Carlo Neutron transPort Code (MCNP) and the burnup and production code ORIGEN2.2. The MONTEBURNS code was compared to ANL's REBUS-DIF3D burnup and production code and one-group cross sections generated by WIMS-ANL. The inventories for LEU and HEU were generated for comparison and standardization.

The dose calculated will be conservative to provide an upper limit of radiological effects to operational personal. The radiological effects to the environment are not required by Reg. Guides 1.183 and 1.195 while containment integrity is maintained. Containment integrity or Containment closure is assumed under standard operation and shutdown. Fuel handling accident analysis involving a fuel element with 60 days of decay outside of containment integrity is evaluated to justify its exception in the current procedure.

1.3 Regulatory Limits and Guidance for Research Reactors

The NRC provides instructions for accident analysis in licensing applications for fuel conversion projects at research reactors in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors". NUREG-1537 states:

- New or revised analytical procedures may be provided if the information demonstrates that the conversion from HEU fuel could introduce new, unanalyzed accidents, could cause significant different consequences from a previously postulated accident, or previously used analytical methods are not appropriate to compare the accidents with acceptable validity. If significant different analytical methods are used for the LEU-fueled reactor, they may have to be validated by reanalyzing some HEU-fueled accidents.

Reg. Guides 1.183 and 1.195 contain the same assumptions for fuel handling accidents within containment:

- If the containment is isolated, no radiological consequences need to be analyzed.

- If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is radioactivity is released to the environment, no radiological consequences need to be analyzed.

Occupational dose limits for adults are stated in 10 CFR 20.1201 [5]. Dose to occupational staff should not exceed the following annual limits:

- 5 rem (0.05 Sv) whole body total effective dose equivalent

- 50 rem committed dose equivalent to the thyroid.
- 50 rem (0.5 Sv) from the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue except for;
- 15 rem (.15 Sv) from the sum of the deep-dose equivalent and committed dose equivalent to the lens of the eyes.

The LEU fuel conversion does not introduce any new, unanalyzed accidents, nor would it cause significantly different consequences from previously postulated accidents. Some of the analytical methods have been updated for this analysis including the radiological source term for the core and the analytical methodology for the FHA scratch analysis. The HEU FHA scrath analysis was re-evaluated with the new methodology.

1.4 Organization of Report

- Chapter one provides an introduction to MURR and the HEU to LEU conversion team. Previous work is discussed and a summary of the radiological release source term is given to justify the proceeding accident consequences within this report.

- Chapter two summaries the MURR HEU and LEU core inventory calculations. The in containment occupational dose during the MHA demonstrates the accepted paths of radiological release. The discrepancy in results from different Derived Air Concentrations libraries used in the SAR methodology are described in detail, and justification of the preferred library is described.

- Chapter three calculates the three fuel handling accident dose consequences to operational personal along with the re-evaluation of the SAR FHA. The Request of Additional Information by the NRC from the previous SAR is updated in regard to not needing containment integrity to handle fuel with a decay time over 60 days.

- Chapter four provides a summary of results and compliances to regulations.

Chapter 2

Radiological Source Term

2.1 Background

Many types of accidents are considered in the SAR for the operation of MURR. In all accidents, safety systems have been designed such that the likelihood of an accident involving the release of a significant amount of fission produces has essentially been eliminated. The safety systems take the form of automatic reactor shutdown circuits and process systems designed to ensure, through redundancy, that the reactor will shut down upon a significant deviation from normal operations conditions. The reactor is housed within a containment building, providing a third barrier with integrated decontaminations systems in the ventilation. Additional filtration systems exist in each of the coolant loops, but no credit is given to any of these systems in the accident analysis. The Maximum Hypothetical Accident (MHA) postulates conditions leading to consequences worse than those from any credible accident. The accepted source term in the SAR hypothesized that in event of a cladding break from fuel melting or physical-chemical deterioration, that only the radioiodines and noble gases would escape the reactor pool. All other particles released would remain entrapped in the primary coolant or in the reactor pool. The MURR LEU core radiological inventory was calculated to serves as the basis for the MHA and FHA analysis.

The initial evaluation of the LEU source term was conducted as a part of the neutronic technical basis for the conversion project, completed by John A. Stillman in 2012 [6]. In those analyses, DIF3D-REBUS, a neutron diffusion and transport code for fuel management, was utilized to conduct a fuel cycle simulation that produced the depleted fuel characteristics. WIMS-ANL is a deterministic code system for lattice calculations, and was used to model the MURR HEU and LEU cores to create the 10 group cross section inputs for the DIF3D-REBUS simulations. WIMS-ANL was utilized to provide MURR specific cross section library to accurately model the neutron spectrum and reaction rates in the DIF3D-REBUS calculations. DIF3D-REBUS was coupled with MCNP to produce the power distribution within the core for fresh, mixed burnup, and depleted fuels. However, the isotopic details in the DIF3D-REBUSs calculations were not sufficient for calculation personal dosage. Specifically, only 11 actinides and 4 fission products were tracked in the DIF3D-REBUS model.

2.2 LEU Radionuclide Inventory

The LEU radionuclide inventory was simulated with a MONTEBURNS code, a coupling of ORIGEN2 and MCNP. The code was created and executed by John A. Stillman (ANL) and Nickie J. Peters, Ph.D. (MURR) in 2004. ORIGEN2 is a zero-dimensional isotope depletion and generation analysis code that utilizes one-group cross sections that represent over 100 actinides and activation products, and over 800 fission products. ORIGEN2 provides a more in-depth radioisotope inventory than DIF3D-REBUS code. ORIGEN2 has several built in cross section libraries for different reactor systems. It appears that the libraries for pressurized water reactors (PWR) and "Thermal" libraries were the best libraries to serve as a base for the MURR's ORIGEN2 simulation. The *pwrus* library is for a core fueled with UO_2 pins containing low-enriched uranium (5%) and cooled with pressurized light water. A specific description of *thermal* library could not be found, but it

is the one-group cross section library for neutrons at thermal conditions at room temperature (0.0254 eV). ORIGEN2's prebuilt libraries are not reactor specific, but system specific cross sections can replace any of the ORIGEN's cross sections until the library is customized to the reactor system it is analyzing.

To determine an appropriate base cross section library and cross section replacement strategy for calculating end of life isotopic inventories for the MURR with an ORGIEN2 model, the isotopic inventory of an end of life element from the MURR fuel cycle was calculated by DIF3D-REBUS. DIF3D-REBUS used its base 10 group cross section library along with the MURR WIM-ANL model's generated cross sections to generate the cores inventories. ORIGEN2 was implemented using the *thermal* and *pwrus* libraries. The U-235 fission cross section in the thermal library is 585 barns, which is the cross section at 0.0253 eV, whereas in the pwrus library the U-235 fission cross section is 47.5 barns. The *pwrus* U-235 cross section was calculated as a one group cross section that would go into an ORIGEN2 simulation that was built to distribute weighting factors based on the neutron energy distribution of a pressurized water reactor. Consequently, the neutron flux needed to meet the specified element power when the *thermal* library is used is much lower than that calculated by DIF3D-REBUS. The lower flux results in a significant under prediction of the Pu-239 inventory in the end of life fuel elements.

The *pwrus* library was chosen to be the base library for the ORIGEN2 model. The actinides and fission products used in the DIF3D-REBUS simulation replaced those used in the *pwrus* library to create the *pwrus+* library. The replaced cross sections were prepared by collapsing the 69-group WIMS-ANL library data to one group with the spectrum averaged over all plates in the fuel element at the middle of life (100 MWd for HEU, 120 MWd for LEU). ORIGEN2 run with the *pwrus+* library provided a good match for the DIF3D-REBUS model for end of life inventory of Pu-239. Thus, ORIGEN2 with the *pwrus+* libraries were used to generate the core's radiological inventory for dose calculations. The HEU and LEU fuel's ORIGEN2 outputs along with the SAR HEU source is displayed in

Table 2.1 [7]. Whole-core inventories in Table 2.1 were calculated by depleting the core in twelve 10-day cycles with 15 days of cooling between cycles that simulates the time an element is typically out of the reactor before being reloaded. The HEU and LEU simulations operated at 10 MW and 12 MW, respectively. The SAR was calculated using an ORIGEN2 model, operating at 10 MW, and the discrepancies between the SAR and HEU are unexplained. The discrepancies are few and are within 15% deviation, meriting no further investigation.

Table 2.1 shows the HEU SAR Source term used in the HEU MHA, the re-evaluated HEU Source term, and the LEU Source term. The majority of the fission products inventories between the ORIGEN2 outputs between the HEU and LEU sources match the 20% ramp up in power. However, for the Xe-135 and Sm-149, the inventories in the discharged LEU element are a factor of 2.3 and 3.0, respectively, of the HEU element at the end of simulated irradiation. This is due to two factors. First, the fission rate in the LEU element is 20% higher as previously stated. This increases the production rates of the Xe-135 and Sm-149. At the same time, these fission products are destroyed through parasitic capture. Since the LEU neutronic spectrum is harder than HEU's spectrum, the effective cross sections for capture for Xe-135 and Sm-149 are much lower than they were in the HEU simulation. The combined effects of higher production and decreased destruction rates results in a ratio of fission products above the expected 1.2 ratio from the power increase. Lastly, since the LEU fuel is loaded with 30 times the U-238 than the HEU fuel, the LEU fuel has a higher abundance of Pu-239 and the other actinides.

	SAR	**HEU** *pwrus+* MOL	**LEU** *pwrus+* MOL	**LEU/ HEU**
Iodines (Ci)				
I-131	1.7×10^5	1.6×10^5	1.90×10^5	1.2
I-132	3.3×10^5	3.2×10^5	3.90×10^5	1.2
I-133	5.1×10^5	5.6×10^5	6.70×10^5	1.2
I-134	6.3×10^5	6.3×10^5	7.60×10^5	1.2
I-135	5.2×10^5	5.2×10^5	6.20×10^5	1.2
Kryptons (Ci)				
Kr-85	4.7×10^2	4.7×10^2	5.50×10^2	1.2
Kr-85m	1.1×10^5	1.0×10^5	1.20×10^5	1.2
Kr-87	2.1×10^5	2.1×10^5	2.50×10^5	1.2
Kr-88	3.0×10^5	3.0×10^5	3.50×10^5	1.2
Kr-89	3.8×10^5	3.8×10^5	4.40×10^5	1.2
Kr-90	3.8×10^5	3.8×10^5	4.40×10^5	1.2
Xenons (Ci)				
Xe-133	4.2×10^5	3.8×10^5	4.50×10^5	1.2
Xe-135	9.6×10^4	8.1×10^4	1.90×10^5	2.3
Xe-135m	9.4×10^4	9.4×10^4	1.20×10^5	1.2
Xe-137	4.9×10^5	4.9×10^5	5.90×10^5	1.2
Xe-138	5.2×10^5	5.2×10^5	6.10×10^5	1.2
Xe-139	4.3×10^5	4.2×10^5	6.00×10^5	1.2
Actinides (g)				
Pu-238		0.293	0.627	
Pu-239		7.436	213.100	
Pu-240		0.767	14.88	
Pu-241		2.71×10^{-1}	4.54×10^0	
Pu-242		1.47×10^{-2}	1.67×10^{-1}	
Am-241		2.22×10^{-3}	3.92×10^{-2}	
Cm-242		2.07×10^{-4}	2.63×10^{-3}	
Lanthanides (g)				
Sm-149		3.09×10^{-2}	9.13×10^{-2}	3
Pm-149		3.00×10^{-2}	$3.60 \times 10-2$	1.2

Table 2.1: Radioactive Source Term for Whole Core Inventory [7]

2.3 LEU MHA within Containment

It is important to gauge each accident analysis against the MHA during the analysis process. The MHA includes both occupational dose and public dose calculations. If another accident's TEDE dose were to surpass the MHA dose, then that accident would be treated as the MHA, requiring a re-evaluation of assumptions and the calculation of public dose. In the MHA for the MURR, it is assumed that an accident condition has caused the melting of the core, releasing fission product into the primary coolant system. For plate type fuel, it is accepted that the maximum hypothetical accident fission product release will be from the melting of 4 fuel elements in the same radius around the core. The inventory released is the product of the base fission product inventory multiplied with the peak power density for the plate. The power density was the highest in plates number 1 and number 23. Plate 23 produced the larger, more conservative inventory and was chosen to be evaluated as the source for the MHA (Ref. [8]).

Because this postulated accident is considered worse than any credible accident, the conditions that lead to this event are immaterial to the analysis. While one might postulate that the MHA could result from a partial flow blockage to the fuel, mitigating features such as the primary coolant system strainer, the fuel element end-fittings, and the pre-operational inspection of the reactor pressure vessels and core region following any fuel handling evolution, all prevent an accident of this type from occurring. In addition, it has been shown that a 75% blockage of coolant flow to the hot channel is insufficient to cause cladding failure [9]. The assumptions made in the MHA are the following [10]:

- 100% of the radioiodines and noble gases are released from 4 number 23 plates into the primary coolant.
- A peaking factor of 2.01 is assumed within the number 23 plates, producing a total core inventory release of 5.47%
- The primary coolant leaks through the pressure vessel head packing and

flange gasket into the reactor pool at a rate 40 gallons (151 l) per week or 4×10^{-3} gpm. In the MHA, a leak rate of 80 gallons (303 l) per week is assumed.

- The noble gases that leak from the primary coolant are assume to leave the reactor pool instantaneously to form a homogenous cloud within containment.

- The radioiodine that would leak from the primary coolant over 10 minutes is assumed to instantaneously and homogenously be mixed in the reactor pool for 10 minutes.

- The reactor pool water evaporation rate is approximately 80 gallons per day. In the MHA, it is assumed that 40 gallons of reactor pool water, that contains the radioiodine concentrations at 10 minutes, evaporates over the 10 minutes into containment and instantaneously forms a uniform concentration within the containment building air. This provides a 700% over assumption of radioiodines. It should be noted that containment air at 100% humanity would contain H_2O vapor equal to 40 gallons (151 l). Recognizing that under normal operation, the containment air is at normally at 50% relative humidity, the pool water evaporation rate overestimates by a factor of two.

- The average operation personal is evacuated within the first 5 minutes, but a conservative 10 minutes is assumed to give operational personal time to secure the primary coolant relief value. The MHA dose will be the average dose of a 10 minute exposure time. Since the leak rate from the primary coolant to the reactor pool is linear, the radionuclide inventory is divided by 2.

- The CDE, DDE, and CEDE doses from the radioiodine are calculated radioiodines dose conversion factors.

- The DDE dose from noble gases are calculated with DAC values.

- The TEDE dose is the sum of the DDE dose from the noble gases and CEDE dose from the radioiodine.

2.3.1 DACs

Since the dose from the noble gases is only an external dose due to submersion, and since the Derived Air Concentrations (DACs) for these radionuclides are based on this type of exposure, the individual noble gas doses for 10 minutes in containment are based on their average concentration in the containment air and the corresponding DAC value in Appendix B of 10 CFR 20 or Appendix C of 10 CFR 835 (2006 edition). Appendix B of 10 CFR 20 does not list DAC values for Kr-89, Kr-90, and Xe-137. Consequently, the MURR SAR for the HEU fuel used the DAC values from Appendix C (2006 edition) for Kr-89, Kr-90, and Xe-137. These values were used by the DOE and taken from ICRP publications. On June 8, 2007 the Rule was revised and on April 13, 2011, 10 CFR 835 Appendix C was amended, at which point Kr-89, Kr-90, and Xe-137 became unlisted in 10 CFR 835 Appendix C. However, the ICRP values have not changed, and the NRC accepted the pre-2007 10 CFR 835 values for Kr-89, Kr-90, and Xe-137 in the previous SAR. Thus, the SAR DACs are used as for the dose for operation personnel. The default DACs are also included below to show a comparison between the two, and could be considered for an additional degree of conservativism.

2.3.2 LEU MHA Dose

In collaboration with the Phase II report for the MURR conversation team [11] and in the dedicated thesis for the MHA [12] we state that the dose to operations is 403 mrem. Until the Phase II report is accepted, in regard to the NRC accepting the analysis using the MURR's the previous DAC values, the dose from the noble gases calculated with the default DAC are recorded. This is continued in this report for the FHA accidents at MURR SAR and the FHA section of the Phase II report. Table 2.2 contains the dose to occupation personnel during the MHA.

MHA	HEU	LEU
SAR DACs	132	403
Default DACs		1403

Table 2.2: MHA Dosage in mrem

Chapter 3

Fuel Handling Accident Analysis

3.1 Background

Fuel Handling Accidents are events or scenarios that could cause a breach in fuel cladding. The fuel handling events considered within this document are classified as an event where a fuel plate is damaged or scratched severely enough to breach the cladding or the malfunction of the fuel cladding from a manufacturing defect or corrosion. The FHA does not include the Loss of Coolant or Loss of Flow Accidents, LOCA and LOFA respectively. The LOCA and LOFA analyses concluded that there would be no realistic damage to the fuel [11]. Any hypothetical damage done is enveloped by the Maximum Hypothetical Accident analysis which includes the melting of 4 end-of-life, maximum peak heat flux fuel plates.

3.2 Fuel Handling Accident

All fuel handling is performed in accordance with Special Nuclear Material (SNM) Control and Accounting Procedures as outlined in the Operations Procedures. The procedures will remain the same between HEU and LEU fuel. In accordance with procedures, containment integrity is held during fuel handling events. Thus under the appendices of Reg. Guides 1.183 and 1.195 results that the no radiological consequences to the public and environment need to be analyzed. Irradiated fuel is handled with a specially designed remote tool. The normal fuel handling tool is designed to provide a positive indication of latching prior to movement of a fuel

element. This feature is tested prior to any fuel handling sequence. Fuel elements are always handled one at a time so that they are maintained in a criticality-safe configuration. New or irradiated fuel may be stored in any one of 88 in-pool fuel storage locations (not including the core). These storage locations are designed to ensure geometry such that the calculated K_{eff} is less than 0.9 under all conditions of moderation, thus allowing sufficient convection cooling and providing sufficient radiation shielding.

Irradiated fuel does not leave the facility until it is loaded into a U.S. Nuclear Regulatory Commission (NRC) approved cask for shipment. Transfer of spent fuel from the in-pool storage locations to the cask is done manually with the cask underwater and resting on the shelf behind the weir wall. The 15-ton capacity overhead rectilinear crane is used to move the cask from the reactor pool. A spent fuel element is not loaded into a shipping cask for shipment until a predetermined cooling period has elapsed from the time the element was last removed from the reactor core. Cooling times are based on a thermal analysis of the decay heat generated by a spent fuel element and by the storage requirements at the Department of Energy (DOE) site. The cooling time ensures that a fuel element has decayed to a level where air cooling, in the horizontal position, is adequate to maintain fuel temperature below the design limits. Thus, in the event of a dropped loaded cask or a loss of coolant water from the cask, the fuel element would not release fission products by a meltdown. The handling and storage of this fuel is discussed in greater detail in Section 9.2 of the Safety Analysis Report (SAR) [10].

As described above, the fuel handling system provides a safe, effective, and reliable means of transporting and handling reactor fuel from the time it enters the facility until it leaves. All cask lifting equipment, including the 15-ton capacity crane, is rigorously maintained, including preventive maintenance and magnetic particle testing, as appropriate. A dye penetrate inspection is also performed on the shipping cask. Therefore, no specific accidents regarding the handling of fuel have been identified for the MURR. The probability of dropping a fuel element

while underwater and damaging it severely enough to breach the fuel cladding was considered. From 1976 to 2010, there have been over 35,000 fuel element handlings. Only once during that interval was a fuel element slightly damaged from a drop, which did not release any measurable fission products. The plate 24 was slightly bent inward, reducing coolant channel width between plates 23 and 24, so it was retired before complete burnup. A conservative potential radionuclide release and calculation of the occupational exposure are included below for worse drop.

3.2.1 Fuel Handling Accident Source Term

The following calculations determining the postulated dose from a potential release from a fuel element handling accident mirror the MHA calculations for personal exposure due to a release of fission products. The objective of these calculations is to present a worst-case dose assessment for a person who remains in the containment building for 5 minutes following the release from a breached fuel element. The evacuation of the containment building would occur within approximately 2 minutes for research staff, and about 5 minutes for Operations personnel.

The two outer fuel plates of the low-enriched uranium (LEU) fuel element, number-1 and -23, are the plates most likely to be damaged during fuel handling. The number-1 fuel plate has the peak power density in the core and contains 18.08 grams of uranium-235 (^{235}U) with a peaking factor of 3.14. The number-23 fuel plate has a peak power density of 2.01, contains 81.97 grams of U-235, and has the most surface area to be damaged. To be conservative, the analysis assumes that 0.248 grams of U-235 is exposed from plate number-23 during the FHA, which corresponds to removing a section of fuel meat from a plate that is 1 inch square and 5 mils thick, and NRC accepted value for the MURR's plate type fuel [13]. A power peaking factor of 2.01 is also applied. The calculation procedure is the same as used in the highly-enriched uranium (HEU) SAR. It should be noted that the cladding on the HEU fuel plates is 15 mil thick, while the cladding thickness

$^{131}I - 2.31 \times 10^{1}\ Ci/gm$	$^{85}Kr - 4.52 \times 10^{2}\ Ci/gm$	$^{133}Xe - 3.87 \times 10^{1}\ Ci/gm$
$^{132}I - 3.13 \times 10^{1}\ Ci/gm$	$^{85}Kr_m - 1.27 \times 10^{1}\ Ci/gm$	$^{135}Xe - 1.37 \times 10^{1}\ Ci/gm$
$^{133}I - 5.48 \times 10^{1}\ Ci/gm$	$^{87}Kr - 2.02 \times 10^{2}\ Ci/gm$	$^{135}Xe_m - 8.54 \times 10^{0}\ Ci/gm$
$^{134}I - 6.15 \times 10^{1}\ Ci/gm$	$^{88}Kr - 2.85 \times 10^{2}\ Ci/gm$	$^{137}Xe - 4.84 \times 10^{1}\ Ci/gm$
$^{135}I - 5.10 \times 10^{1}\ Ci/gm$	$^{89}Kr - 3.62 \times\ 10^{2}\ Ci/gm$	$^{138}Xe - 5.00 \times 10^{1}\ Ci/gm$
	$^{90}Kr - 3.60 \times 10^{2}\ Ci/gm$	$^{139}Xe - 4.06 \times 10^{1}\ Ci/gm$

Table 3.1: Activity in Core in Curies per Gram

on the LEU fuel plate number-23 will be 16 mils thick. While the thicker cladding decreases the likelihood of a fuel plate breach that would lead to a release of radioactive material in the event of a mishandling accident relative to the HEU fuel, this was not considered in the analysis.

The release of radioisotopes of krypton, xenon and iodine are the major sources of radiation exposure to personnel and will, therefore, serve as the basis for the source term for this dose calculation. MURR operations with LEU fuel at 12 MW were simulated in twelve 10-day cycles over a 300-day period with 12.054 kg of U-235. This results in a conservative total core burnup of 1,440 MWd. Normal operations are expected to be over a 6.5-day cycle with an anticipated weekly burnup of 78 MWd, and a typical total core burnup of 750 MWd. The core inventory is shown in Table 3.1.

The radionuclide inventory decays during the time it would take to move a fuel element from the reactor core into the pool. Typically, refueling activities after reactor operation do not commence until one to two hours after shutdown, if not later. If the control rods are fully withdrawn prior to shutdown, it will take 13 minutes for the control rod drive mechanisms to fully insert (insertion speed is 2 inches per minute from a fully withdrawn position of 26 inches). By procedure, the primary coolant system must remain in operation (forced circulation) for at least 15 minutes to remove decay heat. After the primary coolant system is se-

cured and depressurized, special tools and the 15-ton capacity overhead rectilinear crane must be used to remove the reactor pressure vessel head to access the fuel. The pressure vessel head is bolted to the upper spool piece using 12 bolts and is under approximately 17 feet of water. All of these tasks must be performed sequentially. Therefore, the quickest an individual can handle fuel from the core after reactor operations would be approximately one hour. To be conservative, the FHA analysis uses a decay time of 30 minutes in evaluating the dose to Operations personnel.

3.2.2 Radiological Release

The specific activity by radioisotope following decay of 30 minutes and the dose calculations of this analysis was obtained from a spreadsheet model and confirmatory hand calculations. The calculated values are presented in Table 3.2.

$^{131}I - 2.30 \times 10^{1}\ Ci/gm$	$^{85}Kr - 4.52 \times 10^{-2}\ Ci/gm$	$^{133}Xe - 3.86 \times 10^{1}\ Ci/gm$
$^{132}I - 2.69 \times 10^{1}\ Ci/gm$	$^{85}Kr_{m} - 1.17 \times 10^{1}\ Ci/gm$	$^{135}Xe - 1.32 \times 10^{1}\ Ci/gm$
$^{133}I - 5.38 \times 10^{1}\ Ci/gm$	$^{87}Kr - 1.53 \times 10^{2}\ Ci/gm$	$^{135}Xe_{m} - 2.20 \times 10^{0}\ Ci/gm$
$^{134}I - 4.14 \times 10^{1}\ Ci/gm$	$^{88}Kr - 2.53 \times 10^{2}\ Ci/gm$	$^{137}Xe - 2.09 \times 10^{0}\ Ci/gm$
$^{135}I - 4.84 \times 10^{1}\ Ci/gm$	$^{89}Kr - 4.91 \times 10^{-2}\ Ci/gm$	$^{138}Xe - 1.14 \times 10^{1}\ Ci/gm$
	$^{90}Kr - 4.20 \times 10^{-16}\ Ci/gm$	$^{139}Xe - 1.15 \times 10^{-12}\ Ci/gm$

Table 3.2: Activity in Core after 30 Minutes of Decay

The radioiodine and noble gas nuclides that are released from 0.248 gm of U-235 are assumed to instantaneously and homogenously distribute in the reactor pool. After release from the damaged fuel plate, it is assumed that no further decay of the radionuclides occurs, resulting in a conservatively calculated occupational dose. Equation 3.1 states the amount of radio-nuclides are present within the reactor pool, the results are in Table 3.3.

$$A_w = A_c \times Gm \times P \times 1000 \frac{mCi}{Ci} \tag{3.1}$$

A_w : Amount of Activity present in the Reactor Pool in mCi

A_c : Activity of the Fuel in Ci per gram

Gm : Amount of Fuel Released in grams

P : Peaking Power Factor of Fuel Plate

$^{131}I - 1.15 \times 10^4\ mCi$	$^{85}Kr - 2.25 \times 10^1\ mCi$	$^{133}Xe - 1.93 \times 10^4\ mCi$
$^{132}I - 1.34 \times 10^4\ mCi$	$^{85}Kr_m - 5.86 \times 10^3\ mCi$	$^{135}Xe - 6.57 \times 10^3\ mCi$
$^{133}I - 2.69 \times 10^4\ mCi$	$^{87}Kr - 7.65 \times 10^3\ mCi$	$^{135}Xe_m - 1.09 \times 10^3\ mCi$
$^{134}I - 2.06 \times 10^4\ mCi$	$^{88}Kr - 1.26 \times 10^4\ mCi$	$^{137}Xe - 1.04 \times 10^2\ mCi$
$^{135}I - 2.41 \times 10^4\ mCi$	$^{89}Kr - 2.45 \times 10^1\ mCi$	$^{138}Xe - 5.71 \times 10^3\ mCi$

Table 3.3: Radioiodine and Noble Gas Activities Released Into Pool

The radioiodine released into the reactor pool is conservatively assumed to be instantaneously and uniformly mixed into the 20,000 gallons (75,708 l) of bulk pool water, which then results in the following pool water concentrations for the iodine isotopes. The krypton and xenon noble gases released into the reactor pool are assumed to pass immediately through the pool water and into the containment building air volume where they instantaneously form a uniform concentration in the isolated structure. When the reactor is at 12 MW and the containment building ventilation system is in operation, the evaporation rate from the reactor pool is approximately 80 gallons (303 l) of water per day. For the purposes of this calculation, it is assumed that a total of 20 gallons (76 l) of pool water containing the previously listed radioiodine concentrations evaporates into the containment building over the 5 minute period. Containment air with the temperature of $75°F(25°C)$ and 100% relative humidity contains H_2O vapor equal to 36.5 gallons (138 l) of water, but to maintain a conservative bias, it is assumed that at containment will hold 40 gallons (151 l) at 100% relative humidity. Since the air

in containment is normally at about 50% relative humidity, thus containing 20 gallons (76 l) of water vapor, the assumed addition of 20 gallons (76 l) of water vapor will not cause the containment air to be supersaturated. It is also conservatively assumed that all of the iodine activity in the 20 gallons (76 l) of pool water instantaneously forms a uniform concentration in the containment building air. When distributed into the containment building, this would result in the following radioiodine concentrations in the 225,000 ft^3 air volume. Eq. 3.2 is used to generate Table 3.5 from Table 3.4.

$^{131}I - 5.74 \times 10^4 \ \frac{\mu Ci}{gal}$	$^{134}I - 6.70 \times 10^4 \ \frac{\mu Ci}{gal}$
$^{132}I - 1.34 \times 10^4 \ \frac{\mu Ci}{gal}$	$^{135}I - 1.03 \times 10^4 \ \frac{\mu Ci}{gal}$
$^{133}I - 2.69 \times 10^4 \ \frac{\mu Ci}{gal}$	

Table 3.4: Radioiodine concentrations in the Pool Water

$$A_i = I_c \frac{\mu Ci}{gal} \times E \ gal \times \frac{1}{V} \frac{}{ft^3} \times 35.3147 \frac{ft^3}{m^3} \times \frac{1}{10^6} \frac{m^3}{ml} \tag{3.2}$$

A_i : Amount of Radioiodine Activity present in the Containment Building's Air in microcuries per ml

I_c : Radioiodine Concentrations in the Pool Water in microcuries per gallon

E : Evaporated Pool Water (20 gallons)

V : Volume of Air in the Containment Building (225,000 ft^3)

$^{131}I - 1.80 \times 10^{-6} \ \frac{\mu Ci}{ml}$	$^{134}I - 3.24 \times 10^{-6} \ \frac{\mu Ci}{ml}$
$^{132}I - 2.10 \times 10^{-6} \ \frac{\mu Ci}{ml}$	$^{135}I - 3.24 \times 10^{-6} \ \frac{\mu Ci}{ml} l$
$^{133}I - 4.21 \times 10^{-6} \ \frac{\mu Ci}{ml}$	

Table 3.5: Radioiodine concentrations within containment

The krypton and xenon noble gases released into the reactor pool are assumed to pass immediately through the pool water and enter the containment building air volume where they instantaneously form a uniform concentration in the containment. This assumption is conservative since it ignores the known solubility of krypton and xenon noble gases in the 100 °F pool water, which would reduce their release into the containment building. From Eq. 3.3 and considering that the volume of the containment building is 225,000 ft^3, the maximum noble gas concentrations in the containment building are calculated in Table 3.6.

$$A_n = N\ mCi \times \frac{1}{V}\frac{}{ft^3} \times 35.3147\frac{ft^3}{m^3} \times 1000\frac{\mu Ci}{mCi} \times \frac{1}{10^6}\frac{m^3}{ml} \tag{3.3}$$

A_n : Amount of Noble Gas Concentration in the Containment Building's Air in microcuries per ml

N : Noble Gas Concentration in the Pool Water in microcuries

V : Volume of Air in the Containment Building (225,000 ft^3)

$^{85}Kr - 3.54 \times 10^{-6}\ \frac{mCi}{ml}$	$^{133}Xe - 3.02 \times 10^{-3}\ \frac{mCi}{ml}$
$^{85}Kr_m - 9.20 \times 10^{-4}\ \frac{mCi}{ml}$	$^{135}Xe - 1.03 \times 10^{-3}\ \frac{mCi}{ml}$
$^{87}Kr - 1.20 \times 10^{-3}\ \frac{mCi}{ml}$	$^{135}Xe_m - 1.72 \times 10^{-4}\ \frac{mCi}{ml}$
$^{88}Kr - 1.98 \times 10^{-3}\ \frac{mCi}{ml}$	$^{137}Xe - 1.63 \times 10^{-5}\ \frac{mCi}{ml}$
$^{89}Kr - 3.85 \times 10^{-6}\ \frac{mCi}{ml}$	$^{138}Xe - 8.96 \times 10^{-4}\ \frac{mCi}{ml}$
$^{90}Kr - 3.29 \times 10^{-20}\ \frac{mCi}{ml}$	$^{139}Xe - 9.03 \times 10^{-17}\ \frac{mCi}{ml}$

Table 3.6: Noble Gas Concentrations in the Containment Building Air

3.2.3 Occupational Dose from Radioiodines

The objective of this calculation is to present a worst-case dose assessment for a person who remains in the containment building for 5 minutes following the FHA. Therefore, as noted previously, the radioactivity in the evaporated pool water is assumed to be instantaneously and uniformly distributed into the building once released into the air. The evacuation of the containment building would occur within

about 2 minutes for research staff, and about 5 minutes for Operations personnel. The slowest evacuation time is used to calculate dose. As previously noted, the primary coolant system will already be secured at the time of any handling of a fuel element, therefore operations personnel will not remain in containment for the 10 minutes as assumed the MHA. The following are the common abbreviation definitions and conversation factors used in calculating the radioiodine dose: CDE - Committed Dose Equivalent, CEDE Committed Effective Dose Equivalent, DDE - Deep Dose Equivalent, and TEDE - Total Effective Dose Equivalent. The following notations are used throughout the dose calculations:

$$
\begin{aligned}
Sv = &\ \text{Sieverts} \\
Bq = &\ \text{Becquerel} \\
\text{Curie}(Ci) = &\ 3.7 \times 10^{10} Bq \\
\text{Microcuries}(\mu Ci) = &\ 3.7 \times 10^{4} Bq \\
\text{Breathing Rate} = &\ 3.3 \times 10^{-4}\ m^3/sec
\end{aligned}
$$

Based on the source term data provided, it is possible to determine the radiation dose to the thyroid from radioiodine and the dose to the whole body resulting from submersion in the airborne noble gases and radioiodine inside the containment building. Table 3.7 and Table 3.8 were used with Eq.s (3.4 3.5) respectively to generate the Thyroid CDE in Table 3.9 and the DDE in Table 3.10.

$^{131}I - 2.92x10^{-7} Sv/Bq$	$^{134}I - 2.88 \times 10^{-10} Sv/Bq$
$^{132}I - 1.74x10^{-9} Sv/Bq$	$^{135}I - 8.46 \times 10^{-9} Sv/Bq$
$^{133}I - 4.86 \times 10^{-8} Sv/Bq$	

(in Sieverts per Becquerel)

Table 3.7: Committed Dose Equivalent Per Unit Intake to the Thyroid (Ref [14])

$^{131}I - 1.82 \times 10^{-14} Sv/(Bq \cdot sec \cdot m^{-3})$	$^{134}I - 1.30 \times 106{-}13 Sv/(Bq \cdot sec \cdot m^{-3})$
$^{132}I - 1.12 \times 10^{-13} Sv/(Bq \cdot sec \cdot m^{-3})$	$^{135}I - 7.98 \times 10^{-14} Sv/(Bq \cdot sec \cdot m^{-3})$
$^{133}I - 2.94 \times 10^{-14} Sv/(Bq \cdot sec \cdot m^{-3})$	

(in Sieverts per (Becquerel-second-meter^{-3}))

Table 3.8: Deep Dose Equivalent from Air Submersion to the Whole Body (Ref. [15])

$$\mathbf{Dose}_{Thyroid} = \frac{\mathrm{I}_c}{2} \frac{mCi}{ml} \times \mathrm{BR} \frac{m^3}{sec} \times E\ min \times \mathrm{CDE} \frac{Sv}{Bq} \times C \frac{sec \cdot Bq \cdot ml}{min \cdot mCi \cdot m^3} \quad (3.4)$$

I_c : Iodine concentration in containment air from Table 3.5

BR : $3.3 \times 10^{-4}\ m^3/sec$

E : Exposure time (5 minutes)

CDE : CDE Value from Table 3.7

C : Conversion constant $= 2.22 \times 10^{22}$

$$\mathbf{Dose}_{WholeBody} = \frac{\mathrm{I}_c}{2} \frac{mCi}{ml} \times \mathrm{E}\ min \times \mathrm{DDE} \frac{Sv}{Bq \cdot sec \cdot m^{-3}} \times C \frac{sec \cdot Bq \cdot ml}{min \cdot mCi \cdot m^3} \quad (3.5)$$

I_c : Iodine concentration in containment air from Table 3.5

E : Exposure time (5 minutes)

DDE : DDE Value from Table 3.8

C : Conversion constant $= 2.22 \times 10^{22}$

$^{131}I - 9.64 \times 10^{-4} Sv$	$^{134}I - 1.71 \times 10^{-6} Sv$
$^{132}I - 6.70 \times 10^{-6} Sv$	$^{135}I - 5.87 \times 10^{-5} Sv$
$^{133}I - 3.75 \times 10^{-4} Sv$	Total Thyroid Dose $- 1.41 \times 10^{-3} Sv$

Table 3.9: Dose to the Thyroid from Inhalation
(in Sieverts)

$^{131}I - 1.82 \times 10^{-7} Sv$	$^{134}I - 2.34 \times 10^{-6} Sv$
$^{132}I - 1.31 \times 10^{-6} Sv$	$^{135}I - 1.68 \times 10^{-6} Sv$
$^{133}I - 6.88 \times 10^{-7} Sv$	Total Thyroid Dose $- 6.19 \times 10^{-6} Sv$

Table 3.10: Dose to the Whole Body from Submersion
(in Sieverts)

By converting these totals to current dose quantities, where rem $= 10^{-2} Sv$ and millirem $= 10^{-5} Sv$, the following values have been derived and will represent the dose from radioiodine to an individual remaining inside of the MURR containment building for 5 minutes after the release of fission products from a fuel element. The CDE is multiplied with the weighting factor of the thyroid, .03, to calculate the CEDE dose from the thyroid to the body. The CEDE and DDE doses are summed together to calculate the TEDE [17].

CDE (thyroid)	=	140.6	mrem
CEDE (thyroid)	=	4.22	mrem
DDE (radioiodines)	=	0.619	mrem
TEDE (whole body)	=	4.84	mrem

Table 3.11: 5-Minute Dose from Radioiodines in Containment Building

3.2.4 Occupational Dose from Noble Gases

Dose from the kryptons and xenons that are present in the containment building is assessed in much the same manner as the iodines, and the dose contribution from each individual radionuclide is calculated and then added together to arrive at the final noble gas dose. Since the dose from the noble gases is only an external dose due to submersion, and since the Derived Air Concentrations (DACs) for these radionuclides are based on this type of exposure, the individual noble gas doses for 5 minutes in containment are based on their average concentration in the containment air and the corresponding DAC value in Appendix B of 10 CFR 20 [16] or Appendix C of 10 CFR 835 (2006 edition) [20]. Appendix B of 10 CFR 20 does not list DAC values for Kr-89, Kr-90, and Xe-137. Consequently, the MURR SAR for the HEU fuel used the DAC values from Appendix C (2006 edition) for Kr-89, Kr-90, and Xe-137. These values were used by the DOE and taken from ICRP publications. On June 8, 2007 the Rule was revised and on April 13, 2011, 10 CFR 835 Appendix C was amended, at which point Kr-89, Kr-90, and Xe-137 became unlisted in 10 CFR 835 Appendix C. However, the ICRP values have not changed, and the NRC accepted the pre-2007 10 CFR 835 values for Kr-89, Kr-90, and Xe-137 in the previous SAR. Thus, the SAR DACs are used as for the dose for Operational personnel. The default DACs are also included below to show a comparison between the two, and could be considered for an additional degree of conservativism.

The DAC value is the concentration of a radionuclide that a radiation worker can breathe in during 2000 hours of work and reach a 5 Rem annual dose limit.

Below is an example of calculating a dose from using DACs.

$$
\begin{aligned}
&= \text{Dose of } {}^{133}\text{Xe } mrem \\
&= 5\ Rem\ \frac{1000\ mrem}{1 Rem}\ \frac{{}^{133}\text{Xe } \frac{\mu Ci}{ml}}{\text{DAC value } \frac{\mu Ci}{ml}}\ \frac{\text{Exposure Time } hr}{2000\ hr}\quad \frac{\text{Breathing Rate}}{\text{Breathing Rate}} \\
&= 5000\ mrem \frac{3.02 \times 10^{-3} \frac{\mu Ci}{ml}}{1 \times 10^{-4} \frac{\mu Ci}{ml}}\ \frac{\frac{5}{60}\ hr}{2000\ hr} \\
&= 6.30\ mrem
\end{aligned}
$$

The noble gas concentrations in the containment building during the 5-minute period following the release of fission products from a fuel element and the corresponding dose for 5-minute occupancy are given below:

Concentration of Noble Gas	Dose form MURR SAR DACs	Dose from Default DACs
85Kr - 3.54 x $10^{-6} \frac{\mu Ci}{ml}$	0	0
85Kr_m - 9.20 $x10^{-4} \frac{\mu Ci}{ml}$	10	10
87Kr - 1.20 x $10^{-3} \frac{\mu Ci}{ml}$	50	50
88Kr - 1.98 x $10^{-3} \frac{\mu Ci}{ml} l$	206	206
89Kr - 3.85 x $10^{-6} \frac{\mu Ci}{ml} l$	0	8
90Kr - 3.29 x $10^{-20} \frac{\mu Ci}{ml} l$	0	0
133Xe - 3.02 x $10^{-3} \frac{\mu Ci}{ml} l$	6	6
135Xe - 1.03 x $10^{-3} \frac{\mu Ci}{ml} l$	21	21
135Xe_m - 1.72 x $10^{-4} \frac{\mu Ci}{ml}$	4	4
137Xe - 1.63 x $10^{-5} \frac{\mu Ci}{ml}$	0	34
138Xe - 8.96 x $10^{-4} \frac{\mu Ci}{ml}$	47	47
139Xe - 9.03 x $10^{-17} \frac{\mu Ci}{ml}$	0	0
DDE for Noble Gases: (in mrem)	345	386

Table 3.12: 5-Minute Dose from Noble Gases in Containment Building

3.2.5 FHA Occupational Dose

To finalize the occupational dose in terms of TEDE for a 5-minute exposure in the containment building after the FHA, the doses from the radioiodines and noble gases are added together, and result in the following values:

CDE (thyroid)	=	141	mrem
CEDE (thyroid)	=	4.2	mrem
DDE (radioiodines)	=	4.84	mrem
DDE (noble gases)	=	344	mrem
TEDE (whole body)	=	349	mrem

Table 3.13: 5-Minute Dose from Radioiodines and Noble Gases with the SAR DACs

CDE (thyroid)	=	141	mrem
CEDE (thyroid)	=	4.2	mrem
DDE (radioiodines)	=	4.84	mrem
DDE (noble gases)	=	386	mrem
TEDE (whole body)	=	391	mrem

Table 3.14: 5-Minute Dose from Radioiodines and Noble Gases with Default DACs

It is also worth noting that individuals exposed in the containment building for only 2 minutes after the FHA (the expected evacuation time for most occupants of the building) would receive doses about 2.5 times lower than those shown above and would receive a TEDE of only about 139 millirem from the MURR SAR DACs and 155 millirem from default DACs. In comparison to the MHA, the FHA dose for a 5 minute exposure at 30 minutes after shutdown is far less than the MHA dose for both respective DAC sources.

3.3 HEU FHA

Considering the procedure and the dose to occupational personal were not included in the previous SAR for the FHA accident, the methodology used in the LEU FHA is new, meriting an re-evaluation of the original HEU FHA under NUREG-1537.

The objective is to determine the maximum occupational dose for a 5 minute exposure. The number-24 fuel plate has the peak power density in the core and contains 45.32 grams of uranium-235 with a peaking factor of 1.67 [10, 7]. To be conservative, the analysis assumes that 0.125 grams of U-235 is exposed from plate number-24 during the FHA, which corresponds to removing a section of fuel meat from a plate that is 1 inch square and 5 mils thick. The core inventory used is SAR inventories in Table 2.1 but multiplied by the factor 2.016×10^{-5}, which is the ratio of core released. The calculation procedure and radiologic pathology to occupation personal are held the same as in the above accident. The power peaking factors for both accidents were calculated on the mixed burnup core with xenon equilibrium and Beginning Of Life fuel elements on MCNP. The control blades in the HEU core were at 24.0 inches while the control blades in the LEU core were at 24.3 inches extracted.

The HEU FHA in the SAR was encompassed by the MHA dose of 132 mrem. With the new methodology, the HEU FHA dose of 235 mrem surpasses the MHA dose unless 2 hours of fuel decay takes place prior to the FHA. This means that the new methodology is more conservative than the previous unrecorded methodology used in the SAR.

CDE (thyroid)	=	76.2	mrem
CEDE (thyroid)	=	0.42	mrem
DDE (radioiodines)	=	2.71	mrem
DDE (noble gases)	=	233	mrem
TEDE (whole body)	=	235	mrem

Table 3.15: 5-Minute Dose from Radioiodines and Noble Gases from the HEU SAR

3.4 FHA at 60 Days of Decay

Our analysis confirms that containment integrity is not required when handling fuel with a decay time of greater than 60 days. Reactor containment integrity exists when the following conditions are satisfied:

- The truck entry door is closed and sealed;
- The utility entry seal trench is filled with water to a depth required to maintain a minimum seal of 4.25 feet;
- All of the reactor containment building ventilation system's automatically-closing doors and automatically-closing valves are operable or placed in the closed position;
- The reactor mechanical equipment room ventilation exhaust system, including the particulate and halogen filters, is operable;
- The personnel airlock is operable (one door shut and sealed); and
- The recent reactor containment building leakage rate test was satisfactory.

The analysis derives the potential dose from a fuel handling incident, with an element that has decayed for 60 days. The source term is derived in the same fashion as in Chapter 2 of this report, but with 60 days decay taking place before the fuel handling accident. The fuel was still simulated to run at 12 MW for 1,440 MWD in twelve 10-day cycles over a 300-day period with 12.054 kg of U-235 (normal operating cycle is 6.5 days with an approximate total core burnup of 750 MWD). Table 3.16 shows the radioiodine, krypton and xenon activities per gram of U-235 will be present in a fuel element with 60 days of decay. Note that all other short-lived radioiodines and noble gases used in the MHA/ FHA analysis have decayed to background levels and thus were not used in this analysis.

$^{131}I - 2.30 \times 10^{1}\ Ci/gm$	$^{85}Kr - 4.52 \times 10^{-2}\ Ci/gm$	$^{133}Xe - 3.86 \times 10^{1}\ Ci/gm$

Table 3.16: Activity in Core after 60 Days of Decay
(in curies per gram of ^{235}U)

The only difference from the procedure described in Section 3.2 is the exposure time and the evaporation of pool water within containment for this analysis. The exposure time is set to 60 minutes and it is assumed that 80 gallons (303 l), approximately the daily evaporation of pool water, will evaporate during the 60 minute period. It is also assumed that the evaporation of the pool water is constant, so that the average exposure to Operations personnel is from 40 gallons (151 l) of pool water.

Following the same procedure as in Section 3.2 with a scratch release of, 0.248 grams of U-235 from plate number-23, which corresponds to removing a section of fuel meat from a plate that is 1 inch square and 5 mils thick, core inventory from Table 3.16, and a peak-power factor of 2.01, the occupational dose was calculated and shown in Table 3.4.

CDE (thyroid)	=	25.9	mrem
CEDE (thyroid)	=	0.78	mrem
DDE (radioiodines)	=	0.78	mrem
DDE (noble gases)	=	0.11	mrem
TEDE (whole body)	=	0.90	mrem

Table 3.17: 60-Minute Dose from Radioiodines and Noble Gases in Containment

The assumptions used to calculate the dose were different from the HEU SAR in respect to gallons of pool water evaporated and the timing of evaporation. Previously, 40 gallons (151 l) of pool water was assumed to evaporate instantaneously into the containment air at the beginning of the accident, and was used to calculate the dose for a 60 minute exposure. Now 80 gallons (303 l) of reactor pool water is assumed to evaporate at a constant rate over the period of 60 minutes, and the average exposure received from the evaporated pool water, 40 gallons (151

l), is used to calculate the dose to Operations personnel. Containment air with the temperature of 75 °F (25 °C) and 100% relative humidity contains H2O vapor equal to 40 gallons (151 l) of water. Since the air in containment is normally at about 50Comparing the maximum TEDE and CDE for those occupationally-exposed during this release to applicable NRC dose limits in 10 CFR 20 shows that the final values are well within the published regulatory limit and, in fact, less than 1% of any annual occupational limit.

3.5 Fuel Cladding Malfunction

Corrosion can result in aluminum cladding failure in two ways: pitting and oxide film formation. Oxide film forms as a result of essentially uniform corrosion, while pitting occurs when the corrosion rate is accelerated in a local area. The MURR has experienced no fuel element failures due to pitting [19], but retired one fuel element early after it had been used for 126 MWD of the planned 150-MWD-usage because of a suspected manufacturing defect which caused a slight increase in I-131 level in the primary coolant ([18]). Pitting corrosion is not catastrophic in nature and can be detected by conventional monitoring techniques in place at the MURR as demonstrated in this example. To place an upper limit on a pitting event, a worst-case pit release scenario was analyzed. The MHA addresses a release of fission products by assuming the melting of four number-23 fuel plates, which results in the melting of 2.72

Based on the statement in Reference [21] that the pit diameter is six times the depth, the surface area of a pit where it penetrates the 0.016-inch thick fuel plate cladding would be 0.096 inches in diameter. To be conservative, the analysis assumes a pit diameter of 0.20 inches, and an instantaneous release of fission products (100% release fraction) from the fuel meat volume equal to the pit area times the fuel meat thickness (0.017 inches). This volume contains 0.027 grams of U-235 compared to the 327.87 grams contained in the number-23 plate of four fuel elements. Based upon these assumptions, the release would be equivalent to 8.23

x 10-5 times the release assumed in the MHA. Assuming instantaneous dilution in the 2,000 gallons (7,571 l) of primary coolant and an equilibrium 131I activity of 2.78 x 105 curies in the core at 12-MW operation, the primary coolant I-131 concentration would be calculated as follows ([8]):

$$^{133}\mathrm{I} = \frac{(2.78 \times 10^{11}\ \mu Ci) \times (8.23 \times 10^{-5}) \times 0.0547}{2000\ gal \times 3,785\ \frac{ml}{gal}} = 1.65 \times 10^{-1} \frac{\mu Ci}{ml}$$

where:

$2.78 \times 10^{11} = {}^{131}I$ activity in the core;

$8.23 \times 10^{-5} = 0.027gm/327.87gm$, the ratio of the amount of fission products activity assumed released by the bit compared to the MHA release;

$0.0547 =$ the fraction of the total fission product activity released in the MHA assuming a 100% fission product release fraction.

Mirroring the conditions of the MHA but including the 8.23×10^{-5} release multiplier, the dose received from the iodine and noble gasses total 0.03 mrem to occupational workers for the first 10 minutes of release within the containment building. This pit release is within the detectable ranges of Fuel Element Failure Monitoring System.

Chapter 4

Conclusions

The MURR is complying with the 1986 amendment to 10 CFR Part 50, requiring that all current and future licensees of non-power reactors to convert from HEU to LEU fuel once a suitable substitute is found. MURR is currently in the process of evaluating the consequences of the new U-10Mo plate fuel proposed by ANL for the conversion initiative. This thesis satisfies the safety evaluation for the FHA class of accidents for the new SAR for the HEU to LEU fuel conversion project. All fuel handling class accidents occupational doses were calculated, and unspecified HEU fuel handling accident doses were re-evaluated as shown in Table 4.1.

Accidents [mrem]	**SAR**	**LEU**	**HEU**$_{re\text{-}evaluated}$
MHA	132	403	
FHA	Dose < MHA	349	235
60 Day Decay FHA	0.78	0.90	
Fuel Element Malfunction	Unspecified	0.03	0.3

Table 4.1: Final Occupational Dose from each addressed accidents

The primary FHA is assumed to come from a dropped fuel element. Operation Procedures during fuel handling dictate that the core will be shut down and containment integrity be secured. Under Regulatory Guides 1.183 and 1.195, the radiological consequence of an FHA accident to the public or environment does not need to be evaluated. The dose of the FHA is encompassed by the MHA. The MHA is within NRC annual limits to the public, hence the FHA too. Under NuReg 1537, since the methodology of the FHA was not specified, the previous accident for HEU fuel was re-evaluated. Since the re-evaluated HEU FHA is not

encompassed by the HEU SAR as it was in 2006, it shows that the new methodology provides a more conservative dose limit for the accident. In the event of a FHA with a fuel handling element with 60 days decay outside out of containment integrity, an hour of exposure would result in a dose less than 1% of the annual limit of a radiation worker. In the event that a fuel element malfunctioned to form a pit within the cladding, the radioiodine would be detected from conventional monitoring of the primary coolant.

The FHA analysis satisfies a section of the MURR's responsibility to evaluate a new LEU SAR and has shown it's compliancy to 10 CFR 20 occupational staff annual limits.

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