Attachment 4

Oregon State University Application for Special Nuclear Material License
Request for Additional Information

Revised Application (Redacted)

APPLICATION FOR LICENSE

OF

SPECIAL NUCLEAR MATERIAL

Submitted to

Director, Office of Nuclear Materials Safety and Safeguards

Fuel Cycle Safety Branch

Division of Industrial and Medical Nuclear Safety

U.S. Nuclear Regulatory Commission

Washington D.C. 20555

by

Oregon State University

Corvallis, OR 97331

June 2010

1) **10 CFR 70.22(a)(1)**

Oregon State University
Corvallis OR 97331

Rainier Farmer, Radiation Safety Officer
127 Oak Creek Building, Oregon State University
U.S. Citizen

Scott Menn, Senior Health Physicist
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U.S. Citizen

Steve Reese, Director
Radiation Center, Oregon State University
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312 Kerr Administration Building, Oregon State University
U.S. Citizen

Mark McCambridge, Vice President for Finance and Administration
640 Kerr Administration Building, Oregon State University
U.S. Citizen

2) **10 CFR 70.22(a)(2) Activity and location for which Special Nuclear Material License is requested:**

The Department of Nuclear Engineering and Radiation Health Physics of Oregon State University (OSU) located within the Radiation Center (RC) will use the special nuclear material to experimentally acquire hydro-mechanical properties of single fuel elements. The fuel elements are from five U.S. high performance research reactors. The fuel elements are not and will not be irradiated at this facility.

The licensed materials are to be used in experiments in the Radiation Center located at Oregon State University. Associated facilities include a research reactor with appropriate Nuclear Regulatory Commission license (R-106) and a broad-scope State of Oregon radioactive materials license (ORE90005).

A diagram of the facility location and floor plan are included in Appendix A-1, 4. The primary location for storage and use of the special nuclear material will be [redacted], respectively.

3) **10 CFR 70.22(a)(3) Requested duration of license is for <u>10 years</u>.**

4) 10 CFR 70.22(a)(4) Description of Special Nuclear Material:

I. Description of Fuel Element Test Sections

Each fuel element has the same type LEU-Mo alloy source material. The fuel is enriched to [redacted]. In each case, the fuel element has an aluminum cladding. The fuel elements are not and will not be irradiated at this facility. The fuel for each of the reactors will be described below. The mass content for each fuel element listed below was estimated from the nominal dimension off the fuel elements design drawings for each element. However, as stated in the safety analysis, criticality calculations were based upon maximum fuel loading envelope for each element. Therefore, the criticality calculations will overestimate the actual mass loading of the fuel.

A. Advanced Test Reactor (ATR)

The ATR will have an estimated [redacted] in a single fuel element. The ATR element has an overall envelope of 66.25 inches in length by 2.5 inches in depth and 4.25 inches in width. There are nineteen (19) fuel plates which make up an ATR element. Each plate is 49.5 inches in length by 0.05 inches in thickness and the width varies between 2.37 inches and 4.28 inches in arc length. The fuel meat within each fuel plate is centered in all directions of the plate with a length of 48 inches, a thickness of 0.03 inches, and varies in width between 1.88 inches and 3.95 inches in arc length.

B. High Flux Isotope Reactor (HFIR)

The HFIR will have an estimated [redacted] in the section of the fuel element for testing. The fuel region of the HFIR is composed of two concentric, cylindrical fuel elements containing vertically oriented curved plates extending in the radial direction. The inner element contains 171 plates. The inner diameter is 5.067 inches, and the outer diameter is 10.590 inches. The outer element contains 369 plates. Its inner diameter is 11.250 inches, and the outer diameter is 17.134 inches [2]. The fuel element test section of the HFIR received for the purpose of this project will consist of a 45° azmuthal section of a complete HFIR fuel element.

C. Massachusetts Institute of Technology Reactor (MITR)

The MITR will have an estimated [redacted] in a single fuel element. The MITR element is rhomboid shaped element, with an overall length of 26.25 inches by 2.405 inches in depth and 2.405 inches in width. There are fifteen (15) fuel plates which make up a MITR element. Each plate is 23 inches in length by 0.08 inches in thickness and 2.525 inches in width. The fuel meat within each plate is centered in all directions of the plate with a length of 22.75 inches, a thickness of 0.03 inches and a width of 2.165 inches.

D. Missouri University Research Reactor (MURR)

The MURR will have an estimated [redacted] in a single fuel element. The MURR element has an overall envelope of 32.5 inches in length by 4.5 inches in width and 3.16 inches in depth. There are twenty four (24) fuel plates which make up a MURR element. Each plate is 25.5 inches in length by 0.05 inches in thickness and the width varies between 1.993 inches and 4.342 inches in arc length. The fuel meat within each fuel plate is centered in all directions of the plate with a length of 24.75 inches, a thickness of 0.03 inches, and varies in width between 1.643 inches and 3.992 inches in arc length.

E. National Bureau of Standards Reactor (NBSR)

The NBSR will have an estimated [redacted] in a single fuel element. The NBSR element has a square cross sectional shape and is a Material Test Reactor (MTR) type fuel element. There are seventeen (17) fuel plates which make up an NBSR element. The fuel is contained in curved fuel plates approximately 13 inches in length by 2.793 inches in width by 0.050 inches in thickness. The dimensions of the fuel meat in each plate are 11 inches in length by 2.436 inches in width by 0.020 inch in thickness, and the cladding thickness is 0.015 inches. The radius of curvature is 5.5 inches. Each fuel element contains an upper and a lower fuel section separated by a 7 inch, non-fueled gap. Each plate has a 0.5 inch un-fueled region in this gap, and a 1.5 inch un-fueled region at its opposite end. The overall length of the fuel element assembly is approximately 68.8 inches.

II. Usage

The study will utilize a single fuel element from each of the above listed five U.S. high performance research reactors to experimentally acquire hydro-mechanical fuel element properties. The test facility will provide high fidelity plate vibration, local fluid pressure, local fluid temperature, and local plate deformation measurements within the test section of the test facility during operation. All testing will be non-destructive. The material will be used for research and development only.

5) 10 CFR 70.22(a)(6) Technical Qualifications of Applicant:

I. Administrative Structure

Staff qualifications for responsible utilization of licensed special nuclear materials at Oregon State University include the administration of a nuclear reactor operating license and a broad-scope state radioactive materials license. The administrative structure consists of a Radiation Safety Committee, Radiation Safety Officer, Reactor Operations Committee, RC Director, RC reactor staff. Radiation Center staff includes reactor operators and health physics personnel. The organization charts for Radiation Safety and the Radiation Center are included in Appendix A-5, 6. Specific data on key personnel is contained in Appendix A-7.

II Radiation Safety Committee (RSC)

The Radiation Safety Committee members are appointed by the Vice President for Finance and Administration. It shall consist of not less than five nor more than ten members including the University Radiation Safety Officer. The members shall be appointed on the basis of knowledge of the principles and practices of the control of hazards from the use of radiation, and on experience in the use of radioisotopes and/or radiation producing machines. Committee membership shall reflect the diversity of the scientific disciplines using ionizing radiation on campus.

A. Responsibilities of Radiation Safety Officer (RSO)

The University Radiation Safety Officer, who is appointed by and reports administratively to the Environmental Health and Safety Manager, is responsible for managing the day-to-day affairs of Radiation Safety and providing secretariat for the Radiation Safety Committee. The RSO will administer the special nuclear materials license.

Responsibilities include operating efficient programs for radioactive waste disposal, package receipt surveys and delivery, preparation of radioactive materials for shipment, personnel dosimetry, workplace surveillance, records management, and basic personnel training. The RSO also administers the University's radioactive materials licenses and maintaining liaison with federal, state, and local regulatory agencies.

To be clear, the RSO has primary responsibility for assuring license and regulatory compliance at a working level for the State of Oregon general license and this proposed SNM license. Because of the unique facilities and expertise within the Radiation Center, we have our own organizational structure but the work performed under this proposed SNM license will still report to the RSO. It is true that the Radiation Center Senior Health Physicist and the Health Physicist do not directly report to the RSO. The administrative structure for the Radiation Center was established because work in this facility is unique and occurs under the jurisdiction of two separate licenses (i.e., USNRC reactor license and broad-scope state radioactive materials license). Just like current use of radioactive material in the Radiation Center and elsewhere on campus, use of this material radioactive material will be controlled and regulated by the RSO. However, in the interest of continuity, the RSO serves as a voting member of the Reactor Operations Committee (see below). Appendix 6 of the application shows the recommended dotted-line report from the Senior Health Physicist to the Reactor Operations Committee, and thus the RSO. The RSO shall be the individual delegated overall responsibility for the health, safety and environmental protection functions and will have the authority to shut down operations if they appear to be unsafe and, in that case, must approve restart of shutdown operations.

B Qualifications of Radiation Safety Officer

Qualifications of the RSO require a Bachelors of Science in health physics or radiation protection, or in a physical or biological science or engineering; however, a Master of Science is preferred. A minimum of three years' professional experience is required involving the radiation safety aspects of using radiation sources and radioactive materials, including the management and administration of a radiation safety program which, preferably, involved a wide range of applications of radiation sources and radioactive materials under a broad scope institutional license.

III. Reactor Operations Committee (ROC)

A Reactor Operations Committee responsible to the Vice President of Research with a least five members knowledgeable in fields which relate to reactor engineering and nuclear safety shall review and evaluate the safety aspects associated with the operation and use of the reactor facility. The RSO and RC Director are members of the committee. However, RC Staff shall never comprise a majority number of the voting membership.

IV. Radiation Center Reactor Personnel

All Radiation Center reactor personnel meet the qualifications in ANSI/ANS 15.4-1988; R1999, *Standard for the Selection and Training of Personnel for Research Reactors*.

A. Responsibilities of the RC Director

The RC Director is responsible for ensuring that all regulatory requirements, including implementation, are in accordance with all requirements of the USNRC and the Code of Federal Regulations for the R-106 license. For the State of Oregon general license and this proposed SNM license, the RC Director is also responsible for ensuring activities under those licenses are performed as required by the licenses.

B. Responsibilities of the Senior Health Physicist

The Senior Health Physicist is responsible for directing the activities of health physics personnel and implementation of the radiation safety program at the RC. Some of the specific responsibilities include implementing all applicable federal, state, and university radiation control regulations and all RC policies involving radiation protection. Responsibilities include operating efficient programs for radioactive waste disposal, package receipt surveys and delivery, preparation of radioactive materials for shipment, personnel dosimetry, workplace surveillance, records management, and basic personnel training.

6) 10 CFR 70.22(a)(7) Facilities and Equipment for Handling Special Nuclear Material

I. Areas of Storage and Use

The special nuclear materials described in this license will be stored in the Radiation Center, [redacted]. Construction of the room consists of fireproof exterior walls, provisions for continuous radiation monitoring and controlled access monitoring.

The special nuclear material will be used in the Radiation Center [redacted], under positive control at all times. [redacted]. Only reactor personnel and personnel specifically trained to use the testing facility will be allowed to handle the special nuclear materials. Construction of the room will consists of fireproof exterior walls and have radiation monitoring equipment available. Only one fuel element shall be in use (i.e., out of the storage rack) at any given time that will be administratively controlled via a lock out/tag out procedure. When not in use, all fuel elements shall be placed in their appropriate position in the storage rack.

II. Shields, Equipment and Handling Devices

The low specific activity of the fuel elements and the fact that all materials will have appropriate cladding allows for direct handling of the material. The fuel elements will be handled in accordance with proper written procedures.

III: Measuring and Monitoring Devices

Personnel monitoring devices are required of all persons working in the Radiation Center with radiation sources if the individual is likely to exceed 10% of their allowable annual limits in accordance with 10 CFR 20.1502. Monitoring of additional individuals for particular environments is at the discretion of the Senior Health Physicist. Personnel dosimetry appropriate for the material being used are provided by a vendor as required. Direct reading dosimeters such as ionization chambers and electronic dosimeters are available for gamma radiation if necessary. All individual monitoring results, incidents, and exposures exceeding the dose limits in 10 CFR 20 will be submitted consistent with the applicable criteria of 10 CFR 20, Subpart M.

Portable radiation monitors utilized in the reactor facility capable of detecting alpha, beta, gamma, and neutron radiation are also available for use with the special nuclear material. The RC maintains and calibrates these instruments or has the means for the instruments to be calibrated. Radiation monitors will be calibrated annually in accordance with ANSI N323A using NIST traceable standards. Procedures will require that check sources will be utilized to verify correct instrument operation prior to use at the beginning of the day. Dosimetry will be processed by a National Voluntary Laboratory Accreditation Program (NVLAP) accredited vendor.

Specialized detection systems are available for analytical radiation measurements. These include an HPGe spectroscopy systems, proportional counters, a liquid scintillation counter and other miscellaneous detectors and equipment to analyze radioactive materials.

IV. Radioactive Waste Disposal

There is little or no waste associated with this license. The fuel assemblies are sealed and unirradiated and will not be unsealed or irradiated. If low-level waste were created, provisions exist through Environmental Health and Safety to collect and dispose the material.

7) 10 CFR 70.22(a)(8) Safety Procedures to Protect Health and Minimize Danger to Life or Property

Procedures are applied to establish safe conduct of activities with radioactive materials and radiation sources. The procedures in effect satisfy various requirements of federal USNRC and state licenses for radioactive materials. Procedures are reviewed by staff, researchers and students during initial training as radiation workers and at regular intervals following initial training. Creation of new procedures or modification of existing procedures will occur through the process described in 10 CFR 50.59 and existing procedures. A criticality safety plan should not be necessary because the fuel assemblies will not be altered nor destructively tested and all of the accident scenarios (i.e., worst case involving an infinite water media or complete melting as a result of a fire) analyzed in the attached safety analysis show a k_{eff} of less than 0.9. The SNM will be a metallic U-Mo alloy between layers of aluminum cladding. The SNM will not be in a soluble or readily dispersible form. As such, personnel are not expected to receive 10% of the applicable limit and therefore will not be routinely monitored for internal exposure. However, if there is evidence of dispersible material and suspect that an uptake has occurred, appropriate bioassay would be performed to determine the uptake and dose.

The radiological hazards for these elements are minimal. The radioactive material, enriched uranium, will be inside cladding designed to withstand the environment of a nuclear reactor. Although contamination monitoring will occur, it is extremely unlikely. Additionally, the external dose rate is expected to be on the order of 1.0 mrem h^{-1} on contact. This estimate is based upon surveys of enriched uranium fuel recently received under a separate license. Use of the materials will be limited to non-destructive testing and analysis of the hydro-mechanical performance of the fuel elements themselves (i.e., they will not be used in a reactor). The fuel elements will be verified to be unirradiated during the receipt survey that will performed once we have received each element and verification of NRC Form-741.

No gaseous effluents are anticipated from this effort. Liquid effluents are also not expected. However, a procedure will be put into place to periodically sample the water of the test loop that the elements will be used in. The water the elements will be used in will be sampled before release to the sanitary sewer to ensure there is no radioactivity present above 10 CFR 20 Appendix B limits. If results are not below the limits found in 10 CFR 20, Appendix B, the water will be pumped to a storage tank for further evaluation and disposal.

I. Monitoring Procedures

A. Access to laboratory areas is controlled by staff personnel or by individuals having keys to specific rooms.

B. Personnel dosimetry badges are required in radioactive material laboratories.

C. Status of special nuclear material will be verified by annual inventory (12 months)

D. Status of the fuel assembly will be monitored by leak tests of the sources (6 month cycle). The wipe sample will be taken of the entire assembly. The wipe sample will be analyzed for radioactive contamination with the appropriate instrument capable of detecting the presence of 0.005 μCi of radioactive material. If the test reveals the presence of 0.005 μCi or more of removable radioactive material, the assembly will be removed from testing for further evaluation.

II. Operating Procedures.

A. Operation of the test facility will be done only by personnel specifically trained to operate the facility and only according to specific written procedures.

B. A portable radiation monitor will be available at all times in the facility. Personnel will be required to perform a contamination survey after handling the material.

III. Emergency Procedures

A. Basic emergency procedures and the Radiation Center emergency plan (approved by NRC under R-106) in effect for radiological emergencies in the Radiation Center complex. As required by the Emergency Response Plan, the Radiation Center has an annual training program for staff and building residents. Annually, we perform building evacuation tests/exercises, refresher training, and emergency drills. Biennially, as part of the Emergency Response Plan refresher training, the Radiation Center staff undergoes on-site fire extinguisher training provided by the Corvallis Fire Department.

The Emergency Response Plan requires annual building evacuation tests and exercise, refresher training and emergency drills. The Plan also requires that the emergency drills occur with an outside first responder agency at least biennially. Both the Corvallis Fire Department and Samaritan Regional Medical Center receive annual training on the use and unique nature of the Radiation Center. This training usually consists of an hour lecture on the Emergency Response Plan and their own procedures on handling radiological incidents, followed by a "nuts-and-bolts" tour of the entire facility. Additionally, both have current copies of the Emergency Response Plan for the Radiation Center.

B. Precautions for material storage are used to minimize the potential for airborne radioactivity from exposure to fire hazards. Storage of the fuel assemblies when not in use and during testing will be away from flammable materials. All materials licensed under this application will be stored within the confines of [redacted] which is locked and alarmed when the facility is not occupied. [redacted]. Structural floors are designed for superimposed loads of 100 pounds per square foot and roofs for 25 pounds per square foot. The ventilation system provides fresh [redacted] at the rate of 4.4E6 cm^3 s^{-1} and is independent of the attached buildings and laboratories. The conditioned air then discharges into [redacted] through four outlet ducts near the ceiling. The exhaust air exits [redacted] through four outlet ducts; three near the ceiling and one near the floor. The floor duct exhausts half of the total volume of effluent to help facilitate mixing within the bay. The stack extends approximately 7.2 meters above the roof of the building, which places the exhaust approximately 20 meters above the ground. The air is discharged at approximately 1.97E3 cm s^{-1}, which ensures that the exhaust air carries to higher elevations and mixes rapidly with the surrounding air. The room is maintained at a negative pressure in relationship to outside static air pressure by controlling the amount of air pumped [redacted]. The exhaust is monitored for both particulate and gaseous radioactive effluent. Emergency lighting is checked annually and covers the room and all stairwells and corridors. The building is inspected annually by the City of Corvallis Fire Department for compliance with applicable building a fire codes. [redacted] has four rate-of-rise/fixed temperature detectors located on the ceiling along with a smoke detector located in the exhaust ducting. All detectors are connected to the Radiation Center building fire alarm distribution panel. Each of the detectors is fully tested annually. The fire extinguishers are inspected annually. There is no appreciable combustible loading within the storage room. Movement of combustible materials into the storage room is strictly controlled/inventoried but generally prohibited. There is current nothing on the inventory of combustible materials into the storage room. There are no hazardous chemicals or processes which may contribute to a fire hazard in this room. In this case, the fire areas are defined as the inside areas (rooms) of the buildings because they are each physically separated from other areas by space, barriers, walls, or other means in order to contain fire within that area. All electrical systems are installed in accordance with NFPA 70, National Electrical Code. Lightning protection is not required; however, the building is grounded in accordance with the National Electrical Code. The [redacted] (International Building Code, 2003, Section 721.2.1.1). The physical barrier for the storage rack is stainless steel with a minimum thickness of 0.125 inches. The barrier does not have a fire rating.

[redacted]

[redacted] there will be no ventilation system for the use area. Emergency lighting will be provided in the new building and implemented in a similar methodology as the present system. The building will be inspected upon construction and annually by the City of Corvallis Fire Department for compliance with applicable building a fire codes. This room will have four rate-of-rise/fixed temperature detectors. All detectors will be connected to the Radiation Center building fire alarm distribution panel. Each of the detectors will be fully tested annually. The fire extinguishers will be inspected annually. There is no appreciable combustible loading anticipated to be [redacted]. [redacted] (International Building Code, 2003 Section 720, Table 720.1 or Northwest Concrete and Masonry Association, "Concrete Masonry Fire Resistance" February 2005.). There is no sprinkler or automated fire suppression system in either room.

NFPA 45 does not apply since neither room will contain flammable or combustible liquids equal or greater than 4 L nor contain greater than 2.2 standard m^3 of flammable gas. Each room complies with most, but not all, of that described in NFPA 801.

IV. Training Program

The Radiation Center is the Oregon State University's institutional facility for the accommodation of teaching, research and statewide service programs involving the use of ionizing radiation. The Radiation Center orientation program is mandatory for all personnel who desire unescorted access to rooms or laboratories posted as radiation areas or containing radioactive materials in accordance with 10 CFR 19.12. This program consists of material on radiation interactions, radiation hazards, dose measurements, and laboratory procedures.

V. As Low As Reasonably Achievable (ALARA) Program

The Radiation Center ALARA program shall consist of a review by the Senior Health Physicist of proposed new uses of radiation or radioactive materials, and major modifications of facilities which could change personnel exposures or radioactive material releases. The ALARA program will be documented with summaries of the reviews or descriptions of actions taken. Dose investigations will be performed when necessary as described in specific procedures. Periodic review of radiation doses of staff, students and visitors is carried out by the Senior Health Physicist and the ROC.

8) 10 CFR 70.53 and 70.54, Material Control and Accountability

The Radiation Center maintains a special nuclear material inventory and reporting program in accordance with 10 CFR 74.13 for Oregon State University. The facility Reporting Identification Symbol (RIS) is ZRH. Annual material status reports are made to the Nuclear Materials management and Safeguards Systems (NMMSS) within sixty calendar days of the beginning of the physical inventory as required by 10 CFR 74.19(c). Transfers and receipts of special nuclear materials are reported in accordance with 10 CFR 70.54 and 10 CFR74.15 to the NMMSS. Specific procedures for these reports are contained in NUREG/BR-0006 and NUREG/BR-0007.

The total inventory (this material and R-106) of special nuclear material at the Radiation Center would exceed the definition of moderate strategic significance per 10 CFR70.4. However, other than the material in this license application, the special nuclear materials are TRIGA LEU fuel elements in use within the TRIGA research reactor core. The University controls the total quantity of materials such that the facility does not exceed the Category III quantity of special nuclear material of low strategic importance.

9) 10 CFR 73, Physical Protection of Plants and Materials

The Radiation Center implements the requirements of 10 CFR 73 through its NRC approved Physical Security Plan. The Physical Security Plan (as amended) implements additional security procedures required after September 11, 2001. A report shall be made to the NRC within one hour of the discovery of a loss or theft of special nuclear material in accordance with 10 CFR 74.11.

10) 10 CFR 70.25 Financial Assurance and Recordkeeping for Decommissioning

The five sealed source fuel elements are U. S. Government owned material. As such, the disposal costs for retrieval and final disposition of these materials will be the responsibility of the U. S. Department of Energy (DOE). Additional decommissioning costs associated with laboratory cleanup or decontamination from the use of these materials shall be the responsibility of Oregon State University. The University is a public university and an institution of the State of Oregon. The Oregon State Board of Higher Education (OSBHE) conducts and controls the Department of Higher Education [ORS 351.010]. OSBHE has jurisdiction over Oregon University System, which includes OSU [ORS 352001(2)]. The OSBHE is an agency of the state [State ex rel Eckles V. Wooley, 302 Or 37, 45 (1986), citing State ex rel Kleinorge v. Reid, 221 Or 558, 570 (1960)]. Since OSBHE is a state agency and it has jurisdiction over OSU, OSU is a state agency. Estimated decommissioning costs for this license are estimated to be less than $10,000. OSU's financial obligations are backed by the full faith and credit of the State of Oregon. The Legislature has to raise sufficient revenue to pay for the expenses of the State each fiscal year and to pay any interest on debt incurred by the State [Oregon Constitution, Article IX, Section 2]. Every agency receives a state appropriation from the Legislature [Oregon Constitution, Article IX, Section 4]. That appropriation is allotted to each agency by the Department of Administrative Services [ORS 291.234]. Most of the legislative appropriations come from the General Fund [ORS 293.105]. If there is insufficient money available in the General Fund, the Treasurer borrows money to pay expenses [Oregon Constitution, Article IX, Section 6; ORS 293.165]. The Treasurer can issue bonds to finance current expenses of the State, if the General Fund is insufficient. The bonds are backed by the tax revenues and full faith and credit of the State of Oregon [ORS 293.173(3)(b)].

The Radiation Center has a fully implemented health physics monitoring and survey program in accordance with 10 CFR 20 that includes documentation of spills or other contamination events. All contamination events, personnel radiation exposure and facility effluent release are tracked and records are retained for the lifetime of the facility and materials license. The HP program is inspected every other year by the U.S. Nuclear Regulatory Commission as a part of the monitoring of the reactor facility license (R-106). Recordkeeping commitments are and will be consistent with the requirements of 10 CFR 20, Subpart L.

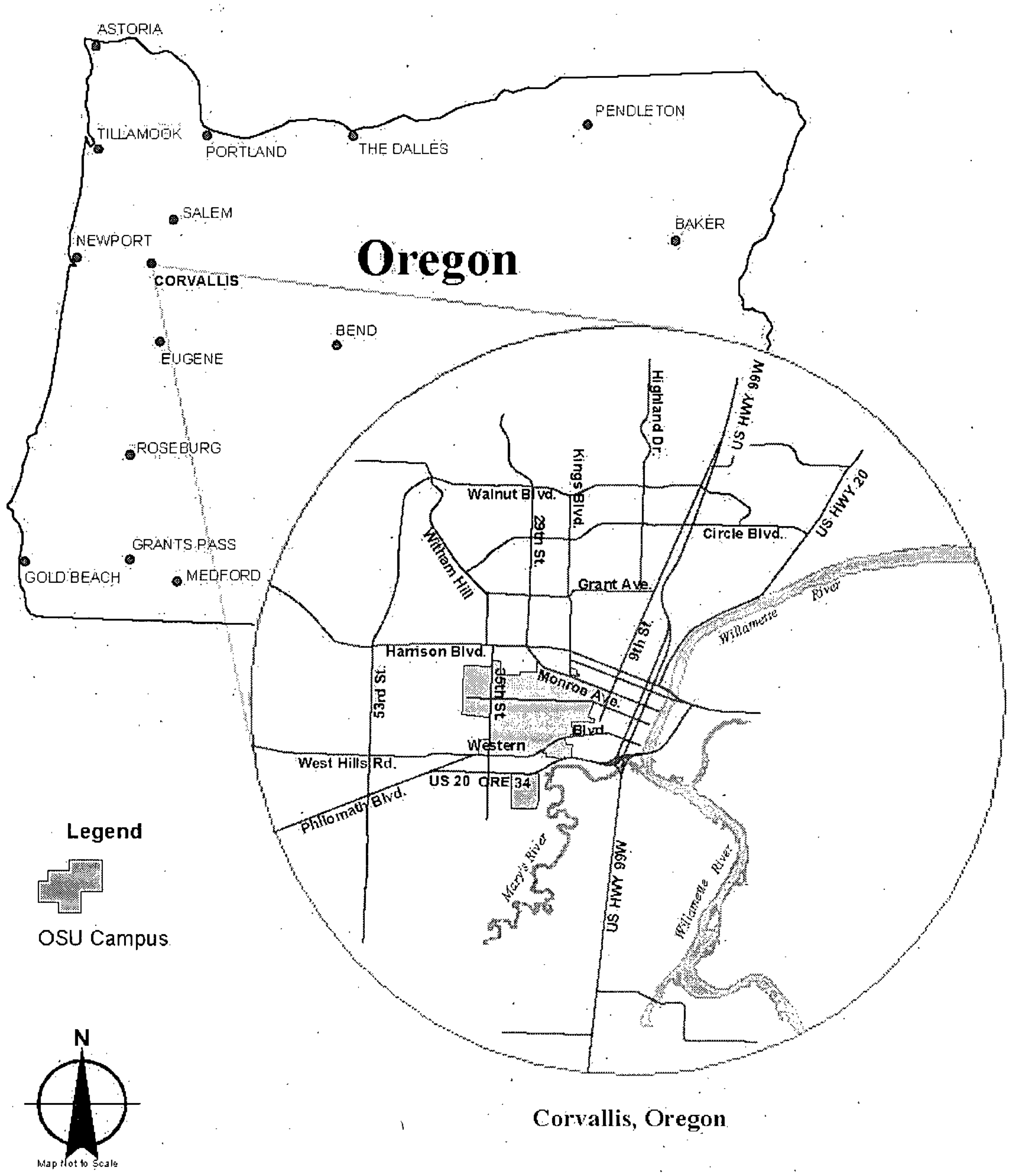
APPENDIX

Application for Special Nuclear Material

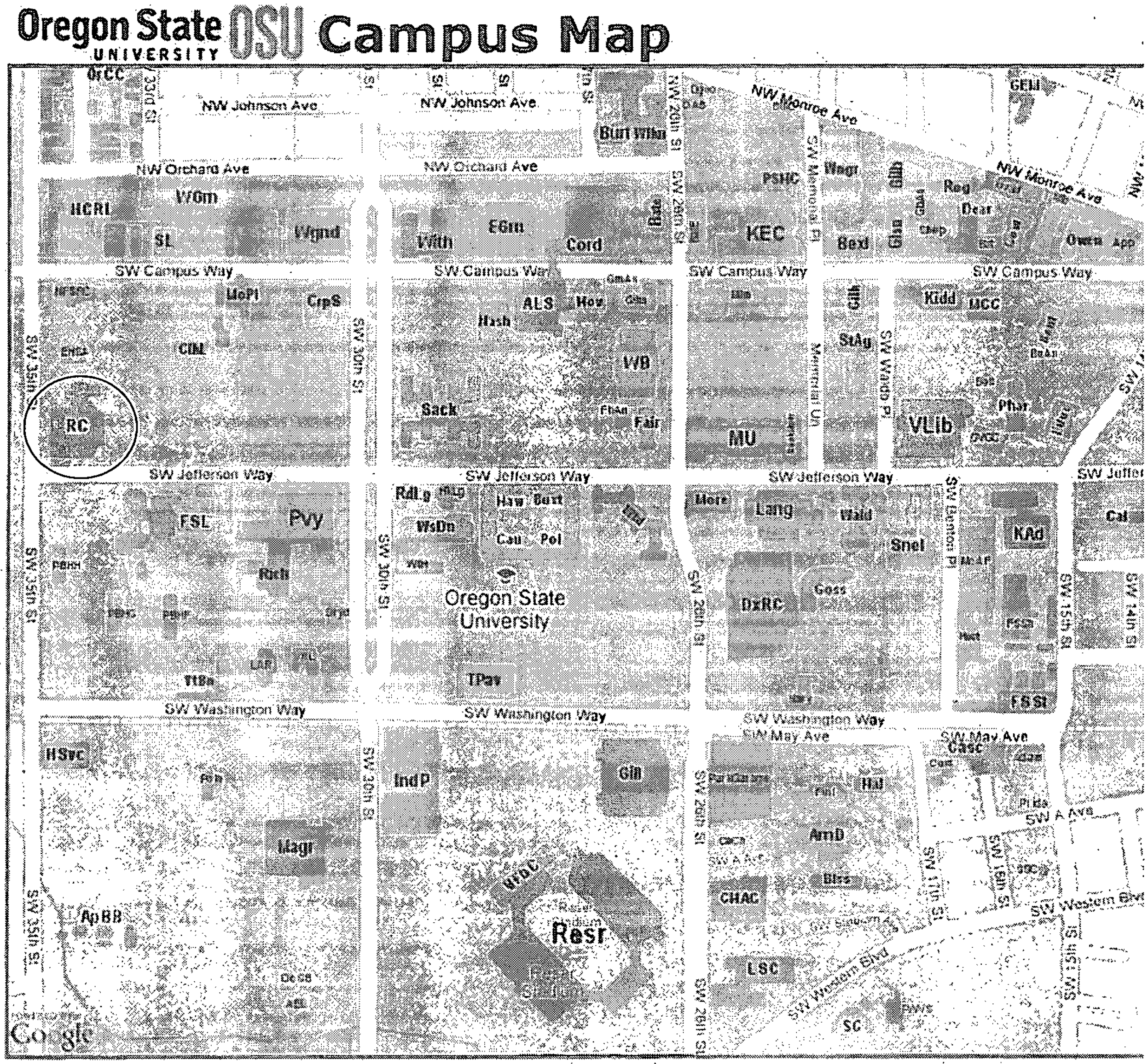
Table of Contents

1. Location of Oregon State University
2. Location of the Radiation Center at Oregon State University
3. Floor plan of the Radiation Center
4. Floor plan of [redacted]
5. Radiation Safety Organization Chart
6. Radiation Center Organization Chart
7. Vita/Resume of Radiation Safety Officer, RC Director and RC Senior Health Physicist
8. Fuel Assembly Description and Safety Report

Appendix 1: Location of Oregon State University



Appendix 2: Location of Radiation Center at Oregon State University



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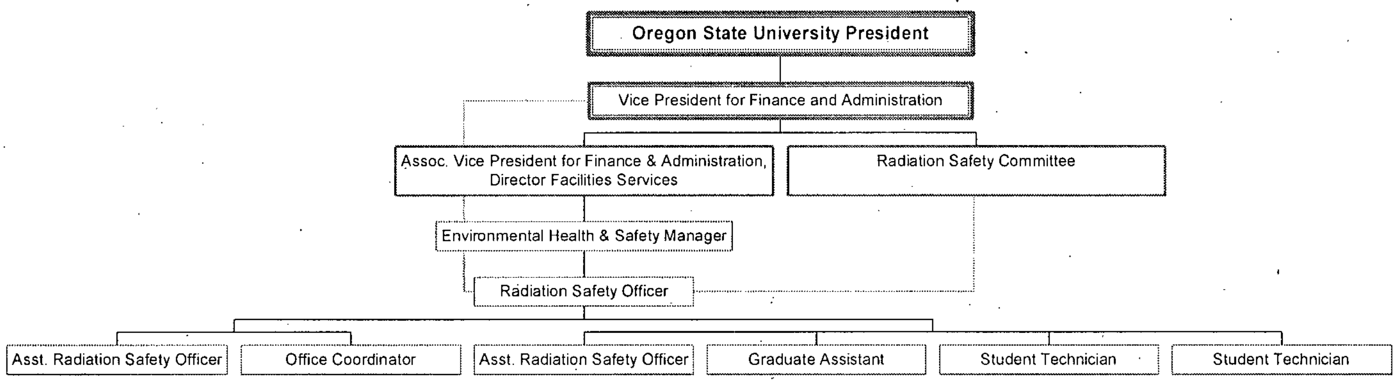
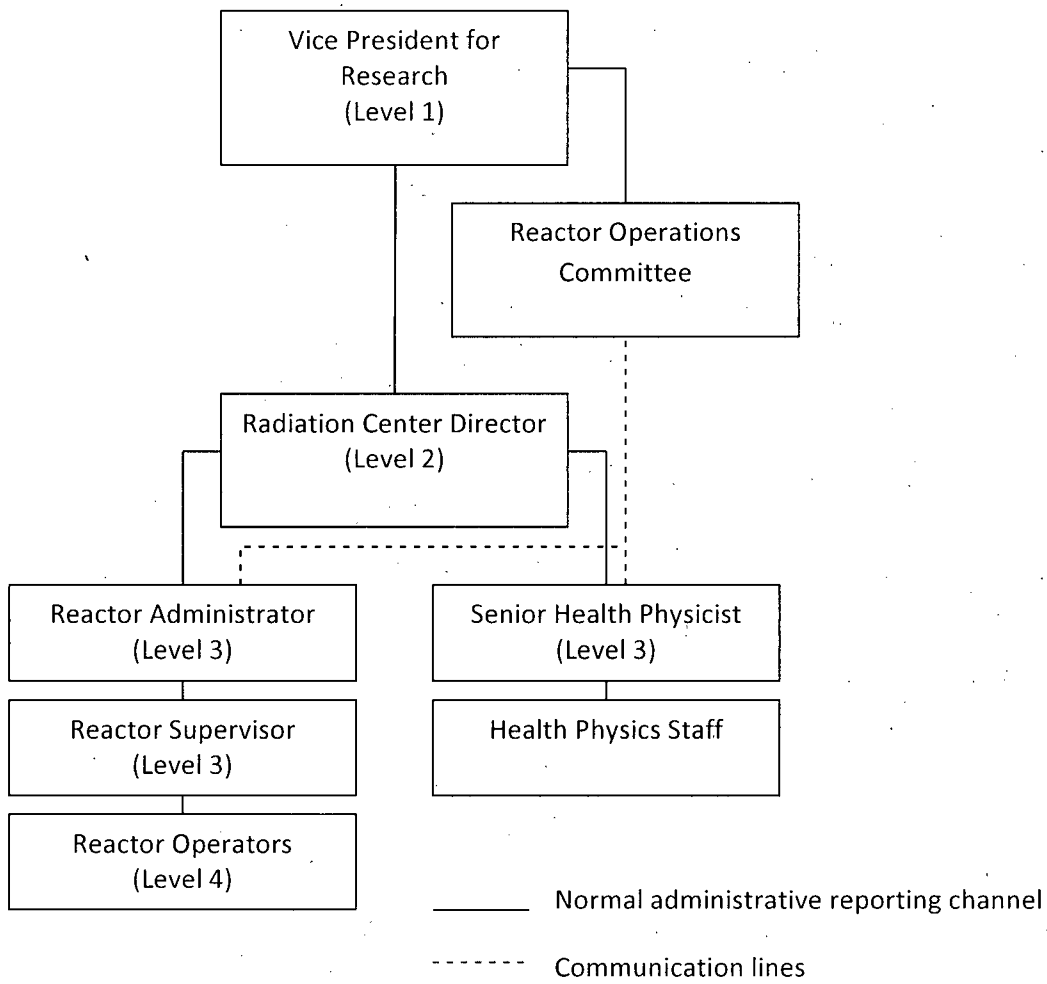
Appendix 3: Floor Plan of the Radiation Center

Appendix 4: Proposed Floor plan of [redacted]

Appendix 5: OSU Radiation Safety Organization Chart



Appendix 6: Radiation Center Organization Chart



Appendix 7:

RESUMES OF KEY PERSONNEL

Resume

Rainier H. Farmer
4153 SE Shortridge Street
Albany, Oregon 97322
(541) 926-3185

Experience

1991 to present	**Consultant**. Provide consulting services to private and public sector clients. Areas of specialization include training, licensing, radiation safety program assessments, radioactive materials shipping, procedure development and review. Licensed by State of Oregon, Health Division, License # 95018A.
5/90 to present	**Radiation Safety Officer**, Oregon State University Radiation Safety Office. Responsibilities include managing all aspects of a radiation safety program in support of a institutional broad license for a large research university, involving radiological surveillance; radioactive waste management; personnel monitoring; inventory, package receipt, and shipment of radioactive materials; instrument calibration; maintaining the radioactive materials license; coordinating review of applications to use sources of radiation; maintaining emergency response capability; orientation, training of radiation users; providing consulting services to radiation users.
3/85 to 4/90	**Radiation Specialist 2**, Oregon State University Radiation Safety Office. Responsibilities included managing a comprehensive radiation use authorization inspection program; managing internal dosimetry, dose investigation, sealed source leak test, and radioactive material shipment programs; performing functional supervision of Radiation Specialist 1 and student workers (2); assisting Radiation Safety Officer in analysis of applications for radiation use authorizations and amendments to ensure compliance with radioactive materials license, applicable regulations, radiation protection principles; providing pre-work orientations, refresher training to radiation users; performance, documentation of routine, special radiological surveys, including surface contamination, package receipt, dose rate, air monitoring surveys; calibration of radiation detection instruments; providing technical advice to radiation users; performance of other duties in support of the tasks listed above (writing reports, corresponding with licensees, procedure development, etc.)

Experience (cont.)

11/84 to 2/85 — **Radiation Specialist 1**, Oregon State University Radiation Safety Office. Collection and packaging of radioactive wastes; administration of the personnel dosimetry program for campus; routine, special radiological surveys, primarily contamination assessment, dose rate surveys; performance of bioassays; calibration of radiation detection instruments.

2/84 to 10/84 — **Research Assistant**, Oregon State University Radiation Safety Office. Duties were identical to those listed above for the Radiation Specialist 1 position.

1/80 to 6/83 — **Health Physics Assistant**, Oregon State University Radiation Center (part-time student worker, average 15 hours per week). Performance of routine radiological surveys; calibration of survey instruments; collection, analysis of environmental samples (soil, water, and vegetation); analysis of reactor water samples; training of new student workers; decontamination of facilities, equipment and personnel.

Education

Oregon State University, Corvallis, Oregon, M.S., Environmental Health Management, June, 1991; B.S., Management, June, 1983.

Honors: Phi Kappa Phi, Beta Gamma Sigma.

Professional

President, Cascade Chapter, Health Physics Society. (1998-99)
Secretary, Cascade Chapter, Health Physics Society. (1994-96)
Treasurer, Cascade Chapter, Health Physics Society. (1991-93)
Member-at-large, Cascade Chapter, Health Physics Society. (2001-03)
Plenary Member, Health Physics Society.

Service

Board of Directors, Cornerstone Associates Inc.
Chair, OSU Alternative Transportation Advisory Committee (current)

Steven Richard Reese
Director, Radiation Center
Oregon State University
100 Radiation Center
Corvallis, OR 97331
541-737-2341 (office)
541-737-0480 (fax)

DEGREES

Ph.D., Radiological Health Sciences, Colorado State University, Ft. Collins, CO 1997
B.S., General Science, Oregon State University, Corvallis, OR 1991

ACADEMIC POSITIONS

Director (0.9 FTE), Radiation Center, Oregon State University, Corvallis, OR
March 2005 - Present
Reactor Administrator (0.9 FTE), Radiation Center, Oregon State University, Corvallis, OR, October 1998 – February 2005
Instructor (0.1 FTE), Department of Nuclear Engineering and Radiation Health Physics, Oregon State University, Corvallis, OR, 1997 – Present

NON-ACADEMIC POSITIONS

University of California at Davis, MNRC Research Advisory Committee, 2006 – Present
National Science Foundation
    Domestic Nuclear Detection Research Initiative Review Panel - 2007
OSTR Reactor Operations Committee, 1998 – Present
Department of Homeland Security
    Nuclear Sector Coordinating Council 2006 – Present
        Co-Chair, Research and Test Reactor Sub Council, 2007 - Present
Reed College Nuclear Reactor Safety Committee, 2000 – Present
Corvallis Planning Commission, 2007 - Present
Radiation Specialist, Radiation Safety Office, Oregon State University, Corvallis, OR, 1997-1998
Research Scientist, Battelle Pacific Northwest National Laboratory, Richland, WA, 1991-1993
Radiation Technician, Oregon State University, Corvallis, OR, 1990-1991

PROFESSIONAL ACTIVITIES

Professional Societies

American Nuclear Society, 1988 – Present
    ASTM/ANS 15 Standards Committee, 2007 – Present
American Board of Health Physics, 2000 – Present

Health Physics Society, 1988 – Present
  National Program Committee, 2004 – 2007
  Continuing Education Committee, 2006 – 2008
    Chairperson, 2007-2008
  Administrative Dean, 2007 Professional Development School
  Cascade Chapter, 1997 – Present
    President, 1998-1999
National Organization of Training, Research and Test Reactors, 1998 – Present
  Chair, 2006-2007
  Executive Committee, 2005-2008

Licenses, Certifications, Courses

Certified, American Board of Health Physics, 2000 – Present
Senior Reactor Operator, 1999 – Present
1999 FEMA Radiological Emergency Response Operations Course
1995 IAEA Summer School on Environmental Sampling at the Chernobyl Nuclear Generation Plant, Kiev, Ukraine.
FAA Private Pilots License, 1987 – Present

Research Awards and Grants

"Global Nuclear Engineering Partnership Readiness Grant, U.S. Department of Energy, Steven R. Reese, September 2007 – September 2008, $100,000.

"Radiological Emergency Response Training and Support", Oregon Department of Energy, Steven R. Reese, January 2007 – December 2007, $24,350.

"University Reactor Sharing Grant", U.S. Department of Energy, Steven R. Reese, August 2006 - July 2008, $39,300.

"University Reactor Instrumentation Grant", U.S. Department of Energy, Steven R. Reese, June 2006 – June 2008, $65,522.

"Radiological Emergency Response Training and Support", Oregon Department of Energy, Steven R. Reese, January 2006 – December 2006, $22,915.

"University Reactor Instrumentation Grant", U.S. Department of Energy, Steven R. Reese, June 2005 – June 2006 $37,846.

"University Reactor Sharing Grant", U.S. Department of Energy, Steven R. Reese, August 2005 - July 2006, $37,800.

"Radiological Emergency Response Training and Support", Oregon Department of Energy, Steven R. Reese, January 2005 – December 2005, $20,832.

"University Reactor Instrumentation Grant", U.S. Department of Energy, Steven R. Reese, June 2004 – June 2005, $18,478.

"University Reactor Sharing Grant", U.S. Department of Energy, Steven R. Reese, August 2004 - July 2005, $25,000.

"Radiological Emergency Response Training and Support", Oregon Department of Energy, Steven R. Reese, January 2004 – December 2004, $21,183.

"University Reactor Instrumentation Grant", U.S. Department of Energy, Steven R. Reese, June 2003 – November 2004, $21,916.

"University Reactor Sharing Grant", U.S. Department of Energy, Steven R. Reese, August 2003 - July 2004, $30,000.

"Radiological Emergency Response Training and Support", Oregon Office of Energy, Steven R. Reese, January 2003 – December 2003, $20,837.

"Western Nuclear Science Alliance: Innovative Expansion of Nuclear Infrastructure and Education for the Western United States", U.S. Department of Energy, Stephen E. Binney, Andrew C. Klein, Steven R. Reese, Wade J. Richards, and Barry M. Klein, July 2002 – June 2007, $1,300,000 (annually).

"University Reactor Instrumentation Grant", U.S. Department of Energy, Steven R. Reese, August 2002 – January 2004, $48,396.

"Radiological Emergency Response Training and Support", Oregon Office of Energy, Steven R. Reese, February 2002 – September 2003, $20,837.

"Development of a Full-Scope NAA Program at the McClellan Nuclear Radiation Center", University of California, Davis, Stephen E. Binney, Steven R. Reese, and Erwin G. Schütfort, September 2000 – August 2003, $173,056.

"University Reactor Sharing Grant", U.S. Department of Energy, Steven R. Reese, August 2002 - September 2003, $34,000.

"Improvement in Neutron Radiography Capabilities", Oregon Metals Initiative and PCC-Structurals, Inc., Steven R. Reese, August 2001 – July 2002, $20,000.

"University Reactor Instrumentation Grant", U.S. Department of Energy, Steven R. Reese, April 2001 – October 2002, $21,284.

"Radiological Emergency Response Training and Support", Oregon Office of Energy, Steven R. Reese, October 2001 – June 2002, $20,888.

"Neutron Radiography Equipment Support", OSU Research Office, Steven R. Reese, March 1999 – June 2001, $7,914.

"University Reactor Instrumentation Grant", U.S. Department of Energy, Steven R. Reese, May 2000 – January 2001, $5,225.

"University Reactor Instrumentation Grant", U.S. Department of Energy, Steven R. Reese, April 1999 – May 2000, $27,000.

"University Reactor Sharing Grant", U.S. Department of Energy, Steven R. Reese, August 1998- September 1999, $58,000.

PUBLICATIONS/PRESENTATIONS

Michael R. Hartman, Steven R. Reese, and Stephen E. Binney, "Impact of INIE on the Oregon State TRIGA Reactor", Trans. American Nuclear Society 96, 401-402, 2007.

Steven R. Reese and Stephen E. Binney, "Impact of INIE on the Oregon State TRIGA Reactor and Other Radiation Center Facilities", Trans. American Nuclear Society 94, 531-532, 2006.

Eric Ashbacker, Steven R. Reese, and Larry Greenwood, "Characterization of the Neutron Spectra in Various Oregon State University TRIGA® Reactor Irradiation Facilities", Health Physics 89, 74-75, 2005.

Steven R. Reese, Stephen E. Binney, Todd S. Palmer, Todd S. Keller, Steven P. Smith, and Gary M. Wachs, "Development of a Neutron Radiography Facility at the Oregon State TRIGA® Reactor", Trans. American Nuclear Society 92, 159-152, 2005.

Steven R. Reese, "Oregon State TRIGA® Reactor Beam Port Modifications", U. S. Department of Energy Innovations in Nuclear Infrastructure and Education Consortium Meeting, Chicago, IL, April, 2004.

Stephen E. Binney, Wade J. Richards, and Steven R. Reese, "First Year Highlights of the Western Nuclear Science Alliance", Trans. American Nuclear Society 89, 849-850, 2003.

Stephen E. Binney, Andrew C. Klein, Steven R. Reese, and Jose N. Reyes, "Oregon State University's Role in Nuclear Energy Revitalization", Trans. American Nuclear Society 86, 377, 2002.

Stephen E. Binney and Steven R. Reese, "Recent Changes at the Oregon State TRIGA® Reactor," Trans. American Nuclear Society 83, 518-519, 2000.

Stephen E. Binney, Erwin G. Schütfort and Steven R. Reese, "Research Uses of the Oregon State University TRIGA® Reactor," Trans. American Nuclear Society 80, 89-90, 1999.

Poyarkov, V., Hordynsky, D., Kadenko, I., Nazarov, A., Arhipov, A., Stoliarevsky, I., Costa Ramos, A., Enyenze, K., Assadi, M., Hashary, M., Thompson, J., Reese, S., Watts, R. Post-accident Environmental Radiomonitoring in the Vicinity of Chernobyl NPP. IAEA-CN-63/293 One Decade After Chernobyl: Summing up the Consequences of the Accident. Vienna, Austria; 1996.

Reese, S.R. and Borak, T.B. Effects of Vegetation upon ^{222}Rn Transport, presented at the 1996 CRMCHPS technical meeting, Ft. Collins, CO.

Reese, S.R.. Locations of Criticality Alarms and Nuclear Accident Dosimeters at Hanford, PNL-MA-583, 1993, Pacific Northwest National Laboratory, Richland, WA, 1993.

Baumgartner, W.V. Endres, A.W., and Reese, S.R. Quality Control Program for the Hanford External Dosimetry Thermoluminescent Processing System, Thirty-seventh Annual Meeting of Health Physics Society, Supplement to Vol. 62, No. 6, June 1992.

Baumgartner, W.V. Endres, A.W., and Reese, S.R. Quality Control Program for the Hanford External Dosimetry Thermoluminescent Processing System, PNL-8299, Pacific Northwest National Laboratory, Richland WA 1992.

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Education

Doctor of Philosophy in Radiation Health Physics, Oregon State University, June 2001
Master of Science in Physics, Sam Houston State University, May 1995
Bachelor of Science in Physics, Sam Houston State University, May 1993

Employment

September 2002-present. Senior Health Physicist, Radiation Center, Oregon State University, Corvallis, OR. In charge of the radiation protection program for the OSU TRIGA reactor.

June 2000-September 2002 Assistant Health Physicist, Radiation Center, Oregon State University, Corvallis, OR. Perform daily health physics issues related to running the TRIGA reactor. Reactor water analysis, reactor maintenance, dosimeter evaluation and dose report reviews, environmental sample analysis, domestic and international shipping of radioactive material, radioactive waste, gamma spectroscopy, inventory and leak test of sources, student radiation monitor supervisor, worker orientations and reactor irradiations.

June 1997-June 2000 Radiation Safety Office, Oregon State University, Corvallis, OR. Independently inspect and survey radioisotope users laboratories. Inspect and deliver radioactive packages throughout the campus. DOT certified hazardous material shipper. Provide assistance with the dosimetry program and dose history reports. Write procedures relevant to the state license.

November 1998-June 1999 Rio Grande Physics, Albuquerque, NM. Field/Lab Operations – Took part in the radiological survey of Johnston Atoll using FIDLER detectors and NaI well detectors. Performed the QA/QC program for lab instruments, soil sampling, sample preparation and laboratory analysis of samples.

June 1991-May 1995 D. Hall Corporation, Huntsville, TX Research and development using lithium glass, ^{3}He, BF_3, foils, Ge-Li, high purity germanium and NaI detectors to measure the macroscopic thermal neutron absorption cross sections of solids and liquids, NORM measurements and nonreciprocity measurements. Sound transducers were used to measure ultrasonics in liquids

Sept 1992-May 1995 Sam Houston State University, Huntsville, TX Modern Physics I and II laboratory instructor (e.g., speed of light, Compton effect, nuclear reactions). Conducted surveys and leak test according to the guidelines of the Texas Department of Health, Bureau of Radiation Control, under the supervision of the Radiation Safety Officer.

Publications

Menn, S. and Higley, K.A., "Experimental Determination of Optimum Counting Geometry for a Low-Energy Gamma Emitter in Soil" Health Physics: Health Phy. 76:S117, 1999.

Higley, K.A., Arana, J.D., Boone, D.M., Chinudomsub, K., Crowley, P.R., Horn, C.M., Jue, T.M., Keillor, M.E., Marianno, C.M., Menn, S.A., Povetko, O.G;, Richardson, E.L., Schwab, K.E., Walker, M.J. Review and Testing of the Framework and EDUP Code. OSU-NE-9810, December, 1997, Oregon State University, Corvallis, OR

Menn, S., and Higley, K.A., "Construction and Usage of a Phantom for Classroom Purposes" Health Physics: Health Phys. 74:S37, 1998

Menn, S. and Hall, H. In-Situ Determination of Macroscopic Thermal Neutron Absorption Cross-section of Borehole Model Materials Using the Integrated Thermal Neutron Flux Method. Nuclear Geophysics. Vol. 9, No. 1, pp 45-54, 1995.

Menn, S.A., Macroscopic Thermal Neutron Absorption Cross Sections of Solids and Saline Solutions., Master's Thesis, Sam Houston State University, May, 1995.

Presentations

March 1995 - American Physical Society Meeting, Paper Title: Macroscopic Cross Sections of Saline Solutions.

October 1994 - American Physical Society Meeting, Paper Title: Macroscopic Cross Sections of Geologic Samples

March 1992 - American Physical Society Meeting, Paper Title: Nonreciprocity of Neutron.

Honors, Awards, Organizations

Health Physics Society (1995-present)
Sigma Pi Sigma (Physics Honor Society)
Society of Physics Students (1988-1995)
Outstanding Research Award (1991-1992, 1992-1993)

Appendix 8

Oregon State University

Special Nuclear Material License Application

Fuel Assembly Description and Safety Report

TABLE OF CONTENTS

1. INTRODUCTION 2

1.1 Synopsis of Test Setup 2

1.2 Description of Application 4

1.2.1 Advanced Test Reactor (ATR) 4

1.2.2 High Flux Isotope Reactor (HFIR) 4

1.2.3 Massachusetts Institute of Technology Reactor (MITR) 4

1.2.4 Missouri University Research Reactor (MURR) 5

1.2.5 National Bureau of Standards Reactor (NBSR) 5

2 SAFETY ANALYSIS 5

2.1 Description & Calculations 6

2.1.1 ATR Element (Case 1a & 1b) 6

2.1.2 HFIR Element (Case 2a & 2b) 8

2.1.3 MITR Element (Case 3a & 3b) 9

2.1.4 MURR Element (Case 4a & 4b) 10

2.1.5 NBSR Element (Case 5a & 5b) 12

2.1.6 Storage Configuration (Case 6a & 6b) 13

2.1.7 Lumped Configuration (Case 7a & 7b) 15

2.1.8 Materials 18

2.1.9 Eigenvalues 20

2. REFERENCES 21

1. INTRODUCTION

Oregon State University (OSU) has been tasked by the Reduced Enrichment for Research and Test Reactors (RERTR) Fuel Development Program to design construct, and utilize a Hydro-Mechanical Fuel Test Facility (HMFTF) with the primary objective of producing a database of information to support the qualification of a new prototypic uranium-molybdenum (U-Mo) alloy, low enrichment uranium (LEU) fuel to be inserted into the five U.S. High Performance Research Reactors (HPRRs). This database of information will include fuel plate and element, plastic and elastic deformation and vibration as a function of operating system pressure, temperature, and flow rate. The current design of the HMFTF permits for simulation of beyond design basis operating conditions of all the U.S. HPRRs including Lower Safety System Settings (LSSS) to Limiting Conditions of Operation (LCOs).

As a part of OSU's task it is required to procure a set of prototypic elemental specimens from each of the HPRRs. The objective of this document is to request permission to procure the specified materials. As a part of this request this document will provide sufficient evidence that while these elemental specimens are in the possession of OSU, safety will not be compromised at any time as a result of a criticality accident.

1.1 Synopsis of Test Setup

Figure 1.1 presents an isometric rendered sketch of the primary loop of the hydro-mechanical test facility. The test section is located approximately one third (1/3) the vertical distance of the front most vertical section. The test section length is approximately 6 feet which completely encompasses all HPRR fuel element geometry lengths. Figure 1.2 provides a closer rendering of the test section where the HPRR fuel elements will be placed during operation.

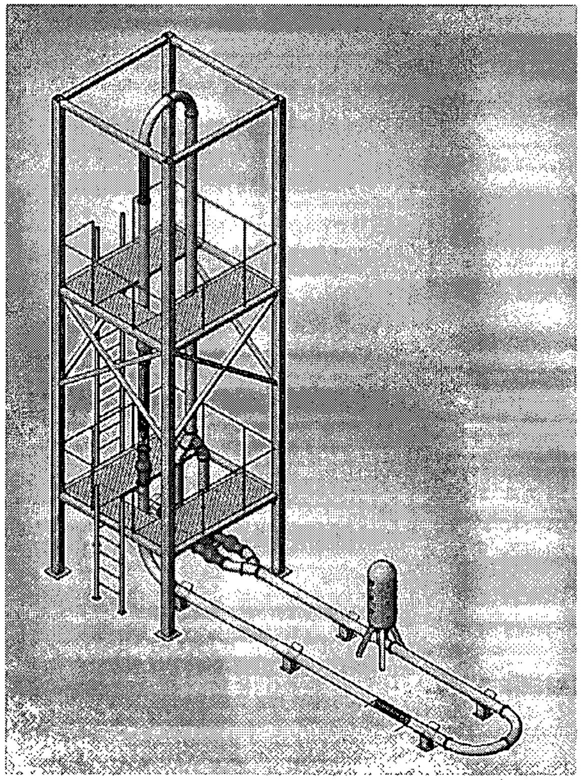

Figure 1.1: Isometric sketch of test loop

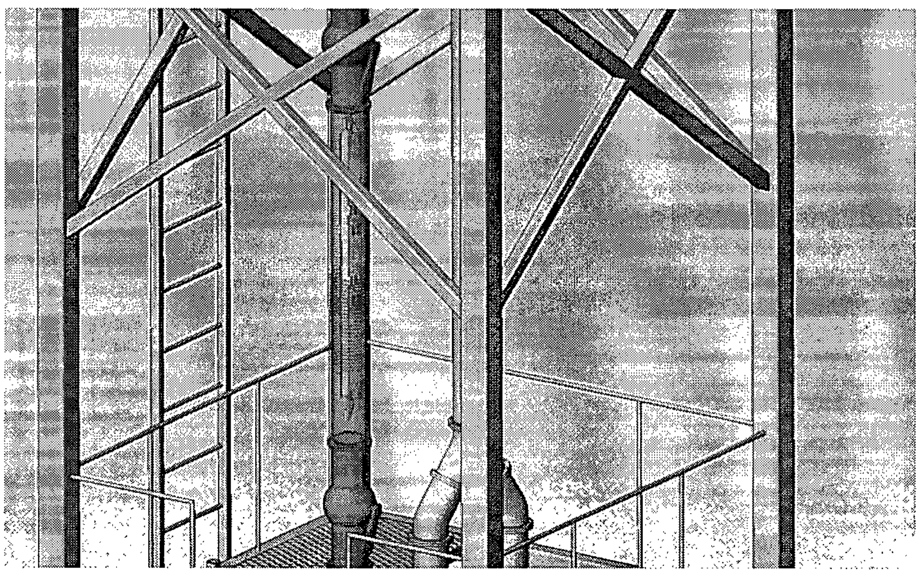

Figure 1.2: Graphical sketch of test loop test section including and Advanced Test Reactor element placed inside

1.2 Description of Application

This HMFTF study will utilize a single fuel element from each of the above listed five U.S. high performance research reactors to experimentally acquire hydro-mechanical fuel element properties

1.2.1 Advanced Test Reactor (ATR)

A prescribed number of hydro-mechanical fuel element characteristics for the Advanced Test Reactor shall be acquired as a result of this study in order to supplement the qualification of the U-Mo fuel currently under development at the Idaho National Laboratory for eventual insertion into the high performance research reactors.

The HMFTF will provide high fidelity plate vibration, local fluid pressure, local fluid temperature, and local plate deformation measurements within the test section of the test facility during operation [1].

1.2.2 High Flux Isotope Reactor (HFIR)

A prescribed number of hydro-mechanical fuel element characteristics for the High Flux Isotope Reactor shall be acquired as a result of this study in order to supplement the qualification of the U-Mo fuel currently under development at the Idaho National Laboratory for eventual insertion into the high performance research reactors.

The HMFTF will provide high fidelity plate vibration, local fluid pressure, local fluid temperature, and local plate deformation measurements within the test section of the test facility during operation [1].

1.2.3 Massachusetts Institute of Technology Reactor (MITR)

A prescribed number of hydro-mechanical fuel element characteristics for the Massachusetts Institute of Technology Reactor shall be acquired as a result of this study in order to supplement the qualification of the U-Mo fuel currently under development at the Idaho National Laboratory for eventual insertion into the high performance research reactors.

The HMFTF will provide high fidelity plate vibration, local fluid pressure, local fluid temperature, and local plate deformation measurements within the test section of the test facility during operation [1].

1.2.4 Missouri University Research Reactor (MURR)

A prescribed number of hydro-mechanical fuel element characteristics for the Missouri University Research Reactor shall be acquired as a result of this study in order to supplement the qualification of the U-Mo fuel currently under development at the Idaho National Laboratory for eventual insertion into the high performance research reactors.

The HMFTF will provide high fidelity plate vibration, local fluid pressure, local fluid temperature, and local plate deformation measurements within the test section of the test facility during operation [1].

1.2.5 National Bureau of Standards Reactor (NBSR)

A prescribed number of hydro-mechanical fuel element characteristics for the National Bureau of Standards Reactor shall be acquired as a result of this study in order to supplement the qualification of the U-Mo fuel currently under development at the Idaho National Laboratory for eventual insertion into the high performance research reactors.

The HMFTF will provide high fidelity plate vibration, local fluid pressure, local fluid temperature, and local plate deformation measurements within the test section of the test facility during operation [1].

2 SAFETY ANALYSIS

A criticality evaluation of all proposed HPRR fuel elements has been conducted. This analysis was performed using a conservative approach. As a result of this criticality analysis it has been demonstrated that during all experimental and storage configurations no critical event will occur. Similarly, it will produce subcritical eigenvalues under geometric configurations of the elements in a single MCNP5 model assuming that they are all arranged in a lumped configuration.

The largest effective multiplication factor (k_{eff}) was determined to be $k_{eff} = 0.86335 \pm 0.00126$ with the error presented in two standard deviations for a case where all fuel elements were grouped together as tightly as feasibly configurable and assumed to be in an infinite medium of water. All eigenvalues presented in this document include the multiplication factor $\pm$ two standard deviations.

2.1 Description & Calculations

To assess the impact of the LEU elements, detailed criticality calculations were undertaken. All eigenvalues were calculated using MCNP5 [2]. MCNP5 is a general purpose Monte Carlo transport code which permits detailed neutronics calculations of complex 3-dimensional systems, and it is well suited to explicitly handle the material and geometrical heterogeneities present in the HPRR fuel elements. In the models developed to describe the HPRR fuel elements, fabrication drawings, provided by the manufacturer, were used to define all fuel meat and clad geometries. In all models, the *maximum envelope* of fuel meat within the fuel plates was modeled in order to consider the maximum fissile material that these elements will plausibly contain.

Fourteen models were developed in total in order to completely describe all plausible fuel element configurations. Table 2.1 presents a summarized case numbering breakdown of all fourteen cases considered. All cases are described in the following sections.

Table 2.1: MCNP Model Case Number Breakdown

Model Description	Case Number	Water Moderator	Air Moderator
ATR Element	1	a	b
HFIR Element	2	a	b
MITR Element	3	a	b
MURR Element	4	a	b
NBSR Element	5	a	b
Storage Configuration	6	a	b
Lumped Configuration	7	a	b

2.1.1 ATR Element (Case 1a & 1b)

The ATR element has an overall envelope of 66.25 inches in length by 2.5 inches in depth and 4.25 inches in width. There are nineteen (19) fuel plates which make up an ATR element. Each plate is 49.5 inches in length by 0.05 inches in thickness and the width varies between 2.37 inches and 4.28 inches in arc length. The fuel meat within each fuel plate is centered in all directions of the plate with a length of

48 inches, a thickness of 0.03 inches, and varies in width between 1.88 inches and 3.95 inches in arc length [3].

A single ATR element was modeled and assumed to be surrounded in an infinite medium of water (case 1a) or air (case 1b). The infinite medium was modeled as a cubic volume of moderator material extending beyond the maximum fuel element dimensions by two (2) meters in all directions. This is assumed to be a sufficient attenuation distance such that it is analogous to that of an infinite medium. All elemental dimensions were taken from the most recent available manufacturer's drawings [4]. Figure 2.1 presents a horizontal cross sectional view of the ATR element modeled in MCNP5. All cladding and side plates were modeled, while the fuel element end fittings were neglected in the MCNP5 simulation as it was assumed that the moderator material fills its place. This is a good assumption as all cladding and end fittings are comprised of aluminum, therefore by replacing this volume with moderator causes a more thermal system. It is assumed that the moderator material creates a more critical condition than that which includes fuel element end fittings.

The MCNP5 k_{eff} value for case 1a was found to be 0.51766 $\pm$ 0.00116; for case 1b k_{eff} = 0.08875 $\pm$ 0.00030. A summary of all k_{eff} values are also presented in Table 2.6.

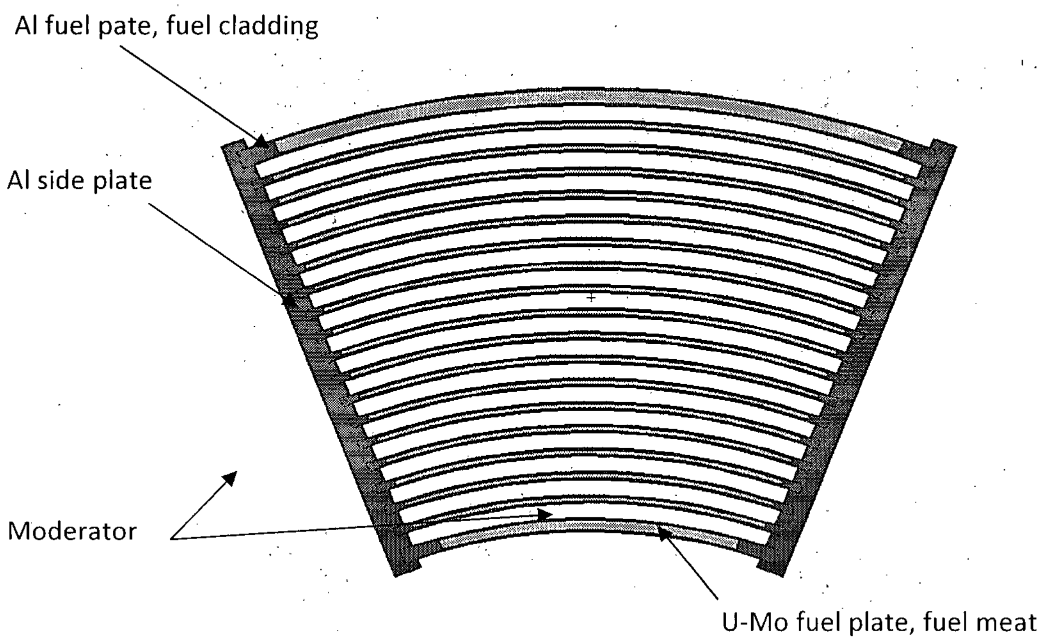

Figure 2.1: Horizontal cross-section of the MCNP5 ATR element

2.1.2 HFIR Element (Case 2a & 2b)

The fuel region of the HFIR is composed of two concentric, cylindrical fuel elements containing vertically oriented curved plates extending in the radial direction. The inner element contains 171 plates. The inner diameter is 5.067 inches, and the outer diameter is 10.590 inches. The outer element contains 369 plates. Its inner diameter is 11.250 inches, and the outer diameter is 17.134 inches [5]. The fuel element test section of the HFIR received for the purpose of this project will consist of a 45° or less azmuthal section of a complete HFIR fuel element.

A single 45° section of the HFIR element was modeled and assumed to be surrounded in an infinite medium of water (case 2a) or air (case 2b). The infinite medium was modeled as a cubic volume of moderator material extending beyond the maximum fuel element dimensions by two (2) meters in all directions. This is assumed to be a sufficient attenuation distance such that it is analogous to that of an infinite medium. All elemental dimensions were taken from the most recent available manufacturer's drawings [6-11]. Figure 2.2 presents a horizontal cross sectional view of the HFIR elemental test section modeled in MCNP5. The cladding was modeled, while the fuel element end fittings and side (ring) plates were neglected in the MCNP5 simulation as it was assumed that the moderator material fills its place. This is a good assumption as all cladding and end fittings are comprised of aluminum therefore by replacing this space with moderator causes a more thermal system. It is assumed that the moderator material creates a more critical condition than that which includes fuel element end fittings.

The MCNP5 k_{eff} value for case 2a was found to be 0.75160 ± 0.00150; for case 2b $k_{eff} = 0.17108 \pm 0.00042$. A summary of all k_{eff} values are also presented in Table 2.6.

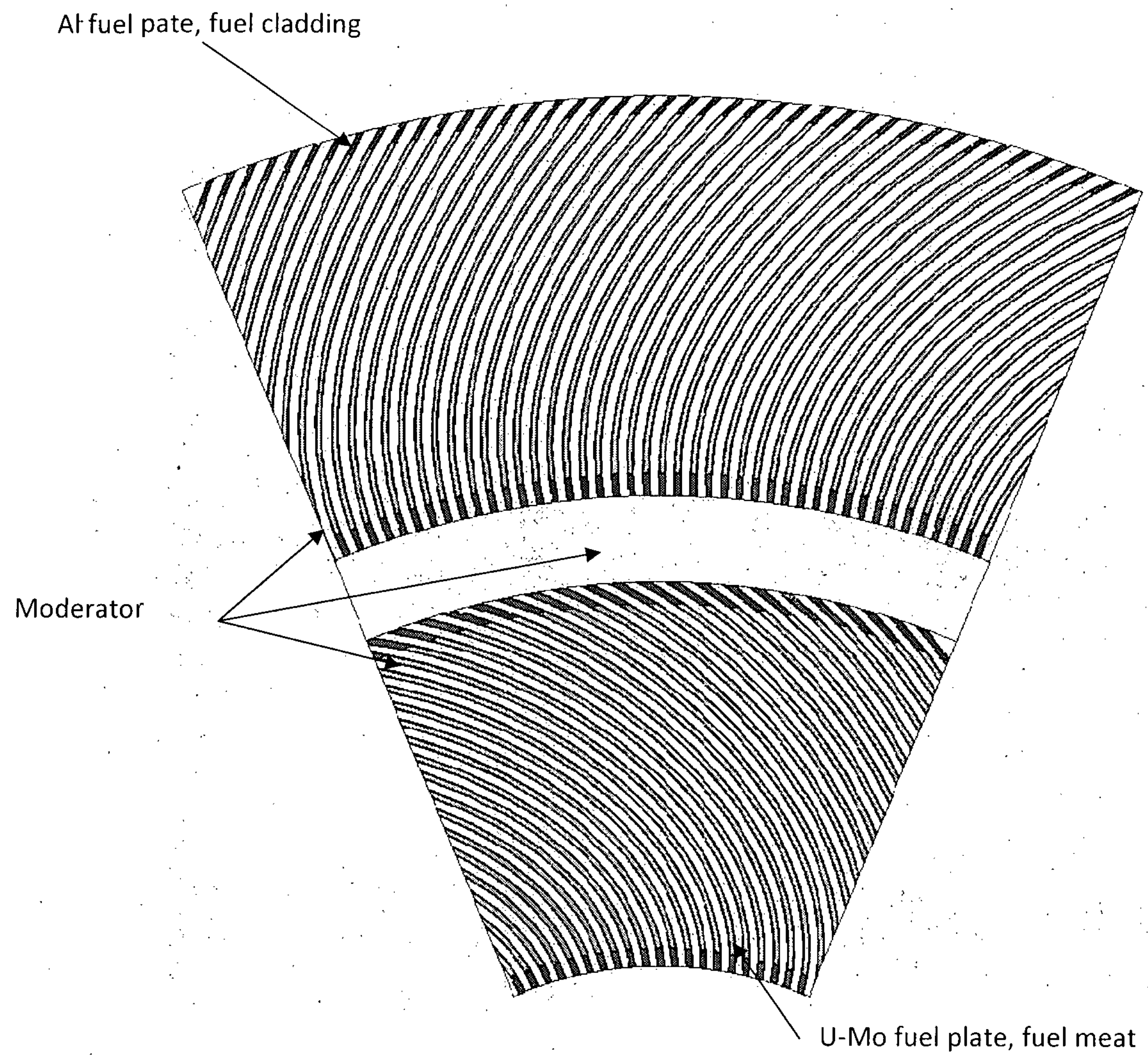

Figure 2.2: Horizontal cross-section of the MCNP5 HFIR elemental section

2.1.3 MITR Element (Case 3a & 3b)

The MITR element is rhomboid shaped element, with an overall length of 26.25 inches by 2.405 inches in depth and 2.405 inches in width. There are fifteen (15) fuel plates which make up and ATR element. Each plate is 23 inches in length by 0.08 inches in thickness and 2.525 inches in width. The fuel meat within each plate is centered in all directions of the plate with a length of 22.75 inches, a thickness of 0.03 inches and a width of 2.165 inches [12].

A single MITR element was modeled and assumed to be surrounded in an infinite medium of water (case 3a) or air (case 3b). The infinite medium was modeled as a cubic volume of moderator material extending beyond the maximum fuel element dimensions by two (2) meters in all directions. This is assumed to be a sufficient attenuation distance such that it is analogous to that of an infinite medium. All elemental dimensions were taken from the most recent available manufacturer's drawings [13-15]. Figure 2.3 presents a horizontal cross sectional view of the MITR elemental test section modeled in MCNP5. The cladding and side plates were modeled, while the fuel element end fittings plates were neglected in the MCNP5 simulation as it was assumed that the moderator material fills its place. This is a good assumption as all cladding and end fittings are comprised of aluminum therefore by replacing this space with moderator causes a more thermal system. It is assumed that the moderator material creates a more critical condition than that which includes fuel element end fittings.

The MCNP5 k_{eff} value for case 3a was found to be 0.46009 ± 0.00122; for case 3b $k_{eff} = 0.06862 \pm 0.00024$. A summary of all k_{eff} values are also presented in Table 2.6.

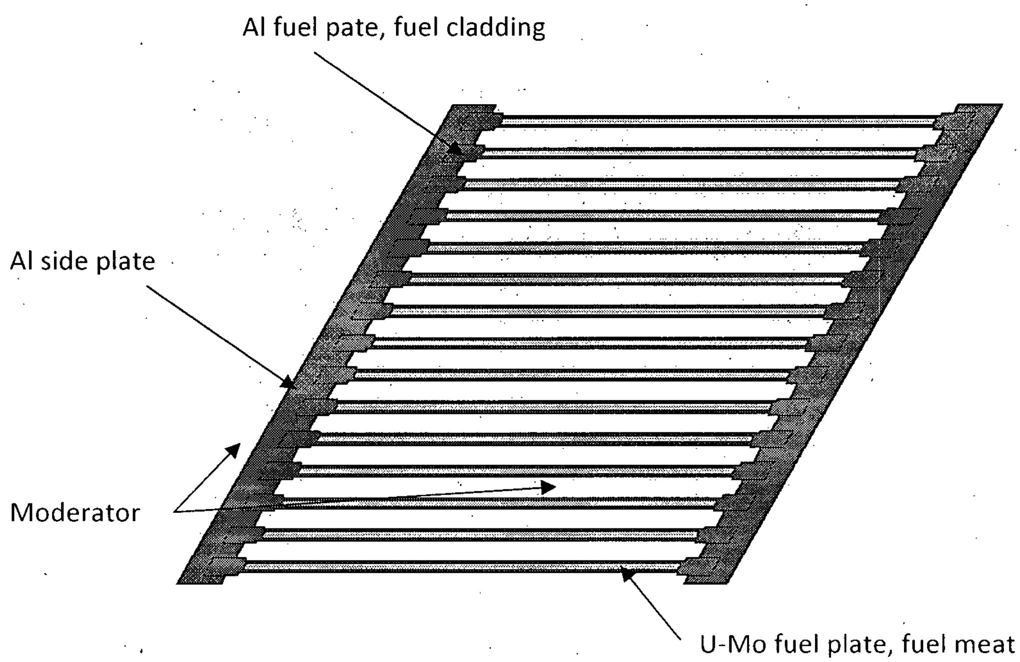

Figure 2.3: Horizontal cross-section of the MCNP5 MITR element

2.1.4 *MURR Element (Case 4a & 4b)*

The MURR element has an overall envelope of 32.5 inches in length by 4.5 inches in width and 3.16 inches in depth. There are twenty four (24) fuel plates which make up a MURR element. Each plate is 25.5 inches in length by 0.05 inches in thickness and the width varies between 1.993 inches and 4.342 inches in arc length. The fuel meat within each fuel plate is centered in all directions of the plate with a length of 24.75 inches, a thickness of 0.03 inches, and varies in width between 1.643 inches and 3.992 inches in arc length [16].

A single MURR element was modeled and assumed to be surrounded in an infinite medium of water (case 4a) or air (case 4b). The infinite medium was modeled as a cubic volume of moderator material extending beyond the maximum fuel element dimensions by two (2) meters in all directions. This is assumed to be a sufficient attenuation distance such that it is analogous to that of an infinite medium. All elemental dimensions were taken from the most recent available manufacturer's drawings [17, 18]. Figure 2.4 presents a horizontal cross sectional view of the MURR elemental test section modeled in MCNP5. The cladding and side plates were modeled, while the fuel element end fittings plates were neglected in the MCNP5 simulation as it was assumed that the moderator material fills its place. This is a good assumption as all cladding and end fittings are comprised of aluminum therefore by replacing this space with moderator causes a more thermal system. It is assumed that the moderator material creates a more critical condition than that which includes fuel element end fittings.

The MCNP5 k_{eff} value for case 4a was found to be 0.56449 ± 0.00126; for case 4b k_{eff} = 0.10923 ± 0.00030. A summary of all k_{eff} values are also presented in Table 2.6.

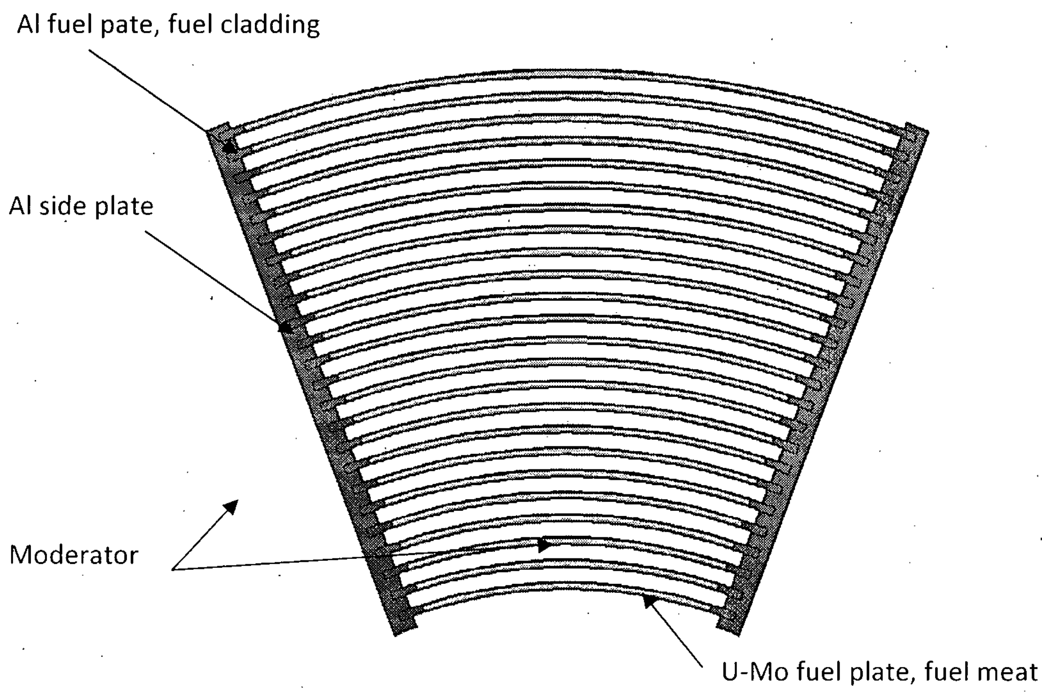

Figure 2.4: Horizontal cross-section of the MCNP5 MURR element

2.1.5 NBSR Element (Case 5a & 5b)

The NBSR element has a square cross sectional shape and is a Material Test Reactor (MTR) type fuel element. There are seventeen (17) fuel plates which make up an NBSR element. The fuel is contained in curved fuel plates approximately 13 inches in length by 2.793 inches in width by 0.050 inches in thickness. The dimensions of the fuel meat in each plate are 11 inches in length by 2.436 inches in width by 0.020 inch in thickness, and the cladding thickness is 0.015 inches. The radius of curvature is 5.5 inches. Figure 4.2.4 illustrates top and bottom flat short fuel plates. Each fuel element contains an upper and a lower fuel section separated by a 7 inch, non-fueled gap. Each plate has a 0.5 inch un-fueled region in this gap, and a 1.5 inch un-fueled region at its opposite end. The overall length of the fuel element assembly is approximately 68.8 inches [19].

A single NBSR element was modeled and assumed to be surrounded in an infinite medium of water (case 5a) or air (case 5b). The infinite medium was modeled as a cubic volume of moderator material extending beyond the maximum fuel element dimensions by two (2) meters in all directions. This is assumed to be a sufficient attenuation distance such that it is analogous to that of an infinite medium. All elemental dimensions were taken from the most recent available references [19, 20]. Figure 2.5 presents a horizontal cross sectional view of the NBSR elemental test section modeled in MCNP5. The cladding and side plates were modeled, while the fuel element end fittings plates were neglected in the MCNP5 simulation as it was assumed that the moderator material fills its place. This is a good assumption as all cladding and end fittings are comprised of aluminum therefore by replacing this space with moderator causes a more thermal system. It is assumed that the moderator material creates a more critical condition than that which includes fuel element end fittings.

The MCNP5 k_{eff} value for case 5a was found to be 0.47705 ± 0.00124; for case 5b $k_{eff} = 0.07023 \pm 0.00022$. A summary of all k_{eff} values are also presented in Table 2.6.

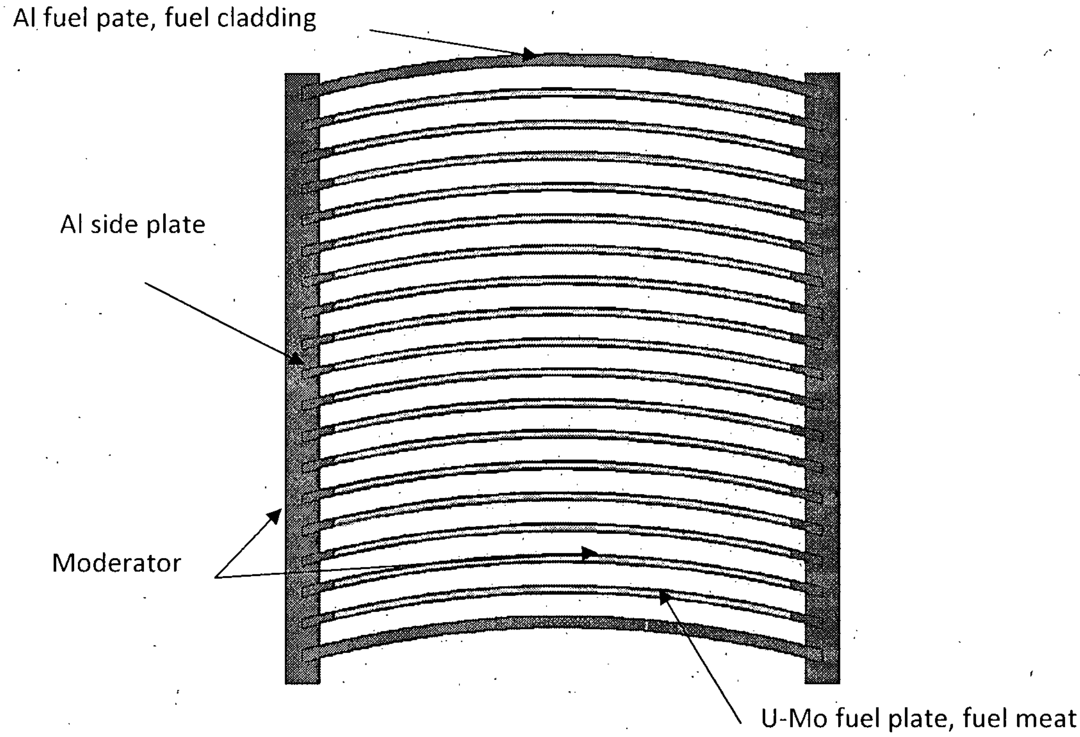

Figure 2.5: Horizontal cross-section of the MCNP5 NBSR element.

2.1.6 Storage Configuration (Case 6a & 6b)

All elements will be stored in their designated storage rack location at all times when not in use. The storage cabinet, presented in Figure 2.6, is constructed of aluminum. Each element is separated by a wall preventing any possible reconfiguration in spatial geometry as a result of any or all elements falling. The fuel elements are secured within each storage location such that they are horizontally separated at 8.00 inches on center with exception of the HFIR element which is separated via 12 inches on either side.

All five elements were modeled and assumed to be in an infinite medium of water (case 6a) or air (case 6b). The infinite medium was modeled as a cubic volume of moderator material extending beyond the maximum fuel element dimensions by two (2) meters in all directions. This is assumed to be a sufficient attenuation distance such that it is analogous to that of an infinite medium. All elements were modeled using the same methods as that for cases 1 through 5. The elements sit on the floor of the storage cabinet in their storage positions; only the fuel meat, cladding, and side plates were modeled in each of the elements. The vertical location in which the fuel elements are located relative to the floor of the cabinet including end fittings was taken into consideration in the model. A vertical cross section of the storage configuration MCNP5 model is presented in Figure 2.7. All elements were modeled such that their storage depth within the cabinet was similar to one another as will occur in the physical storage cabinet, this can be seen in Figure 2:8.

The MCNP5 k_{eff} value for case 6a was found to be 0.75161 $\pm$ 0.00136; for case 6b k_{eff} = 0.18868 $\pm$ 0.00048. A summary of all k_{eff} values are also presented in Table 2.6.

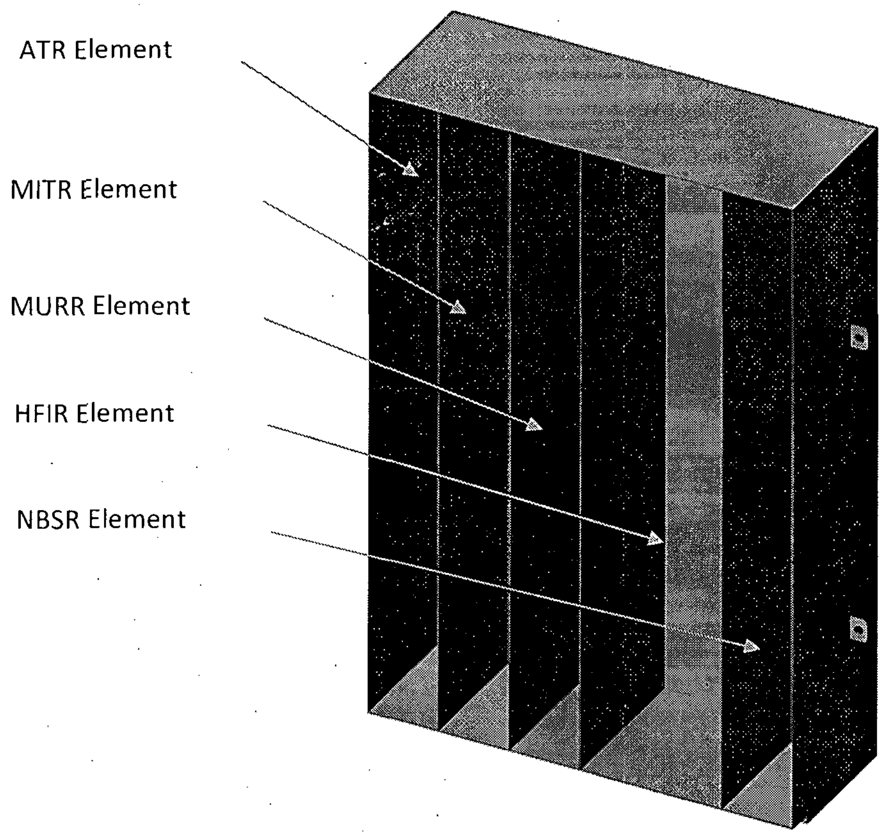

Figure 2.6: Element Storage Cabinet

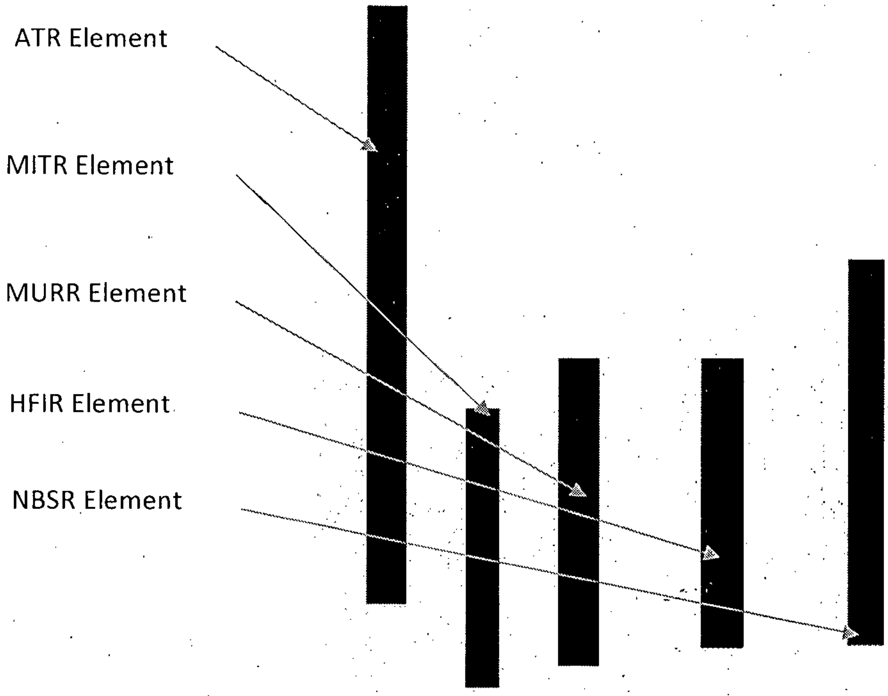

Figure 2.7: Vertical cross-section of the MCNP5 storage configuration model

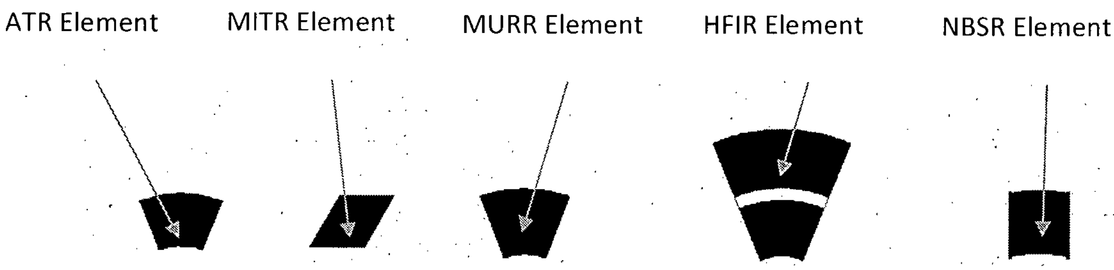

Figure 2.8: Horizontal cross-section of the MCNP5 storage configuration model

2.1.7 Lumped Configuration (Case 7a & 7b)

An eigenvalue calculation was conducted in a lumped configuration in order to provide sufficient evidence that under unrealistically 'packed' conditions all elements will remain subcritical. The lumped configuration considered as a part of this limiting eigenvalue simulation is unphysical as the elements are arranged closer than their end fittings will allow, it is the assumption that if the system of elements is subcritical in a more tightly packed configuration, it will remain subcritical in a configuration in which the elements are forced further apart as a result of their end fittings.

The MCNP5 model was developed using the same individual element angular orientation as that presented in all other models while translating their locations to produce a *lumped* configuration. Figure 2.9 presented a horizontal cross section of the MCNP5 model in the lumped configuration at a vertical position of +20 cm. Because the MURR element and MITR elements are shorter than the ATR, HFIR, and NBSR, they were stacked on top of each other. This can be seen by referring to Figure 2.10 where the horizontal cross section at a vertical position of –20 cm is presented. The vertical configuration of the MITR and MURR elements is further identified by referring to Figure 2.11 (a) which displays a *y,z* cut plane of the MCNP5 model. The vertical position of the HFIR and NBSR elements relative to the ATR element are presented in Figure 2.11(b) which displays an *x,z* cut plane of the MCNP5 model. The vertical locations of the NBSR and HFIR elements were chosen such that the center of the fuel portion of the fuel elements was lined vertically with that of the ATR element's fuel meat. It is assumed that by positioning the elements in this vertical configuration produces a more preferred critical geometry than if the elements were aligned at one end. All elements modeled in the lumped configuration were constructed just as done in their individual elemental models (cases 1 - 5).

The MCNP5 k_{eff} for case 7a was found to be 0.86335 $\pm$ 0.00126; for case 7b k_{eff} = 0.21454 $\pm$ 0.00054 A summary of all k_{eff} values are also presented in Table 2.6.

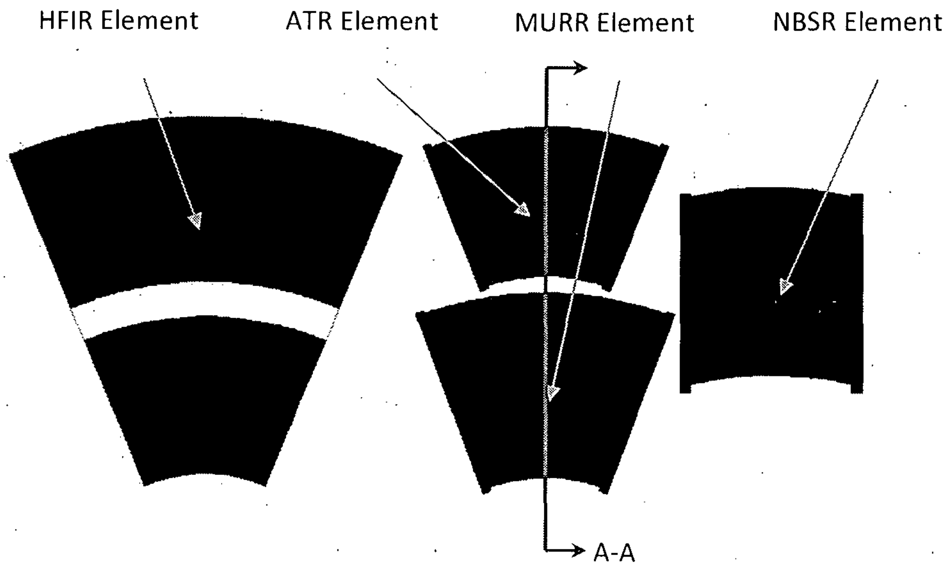

Figure 2.9: Horizontal cross-section (x,y plane) of the MCNP5 lumped configuration model at a vertical position of +20cm (z=+20)

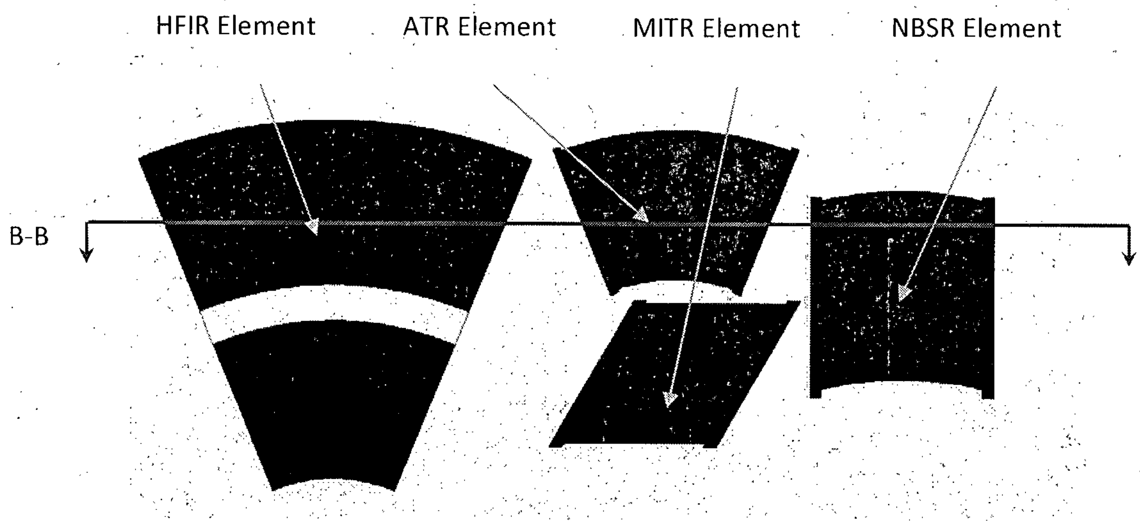

Figure 2.10: Horizontal cross-section (x,y plane) of the MCNP5 lumped configuration model at a vertical position of -20cm (z=-20)

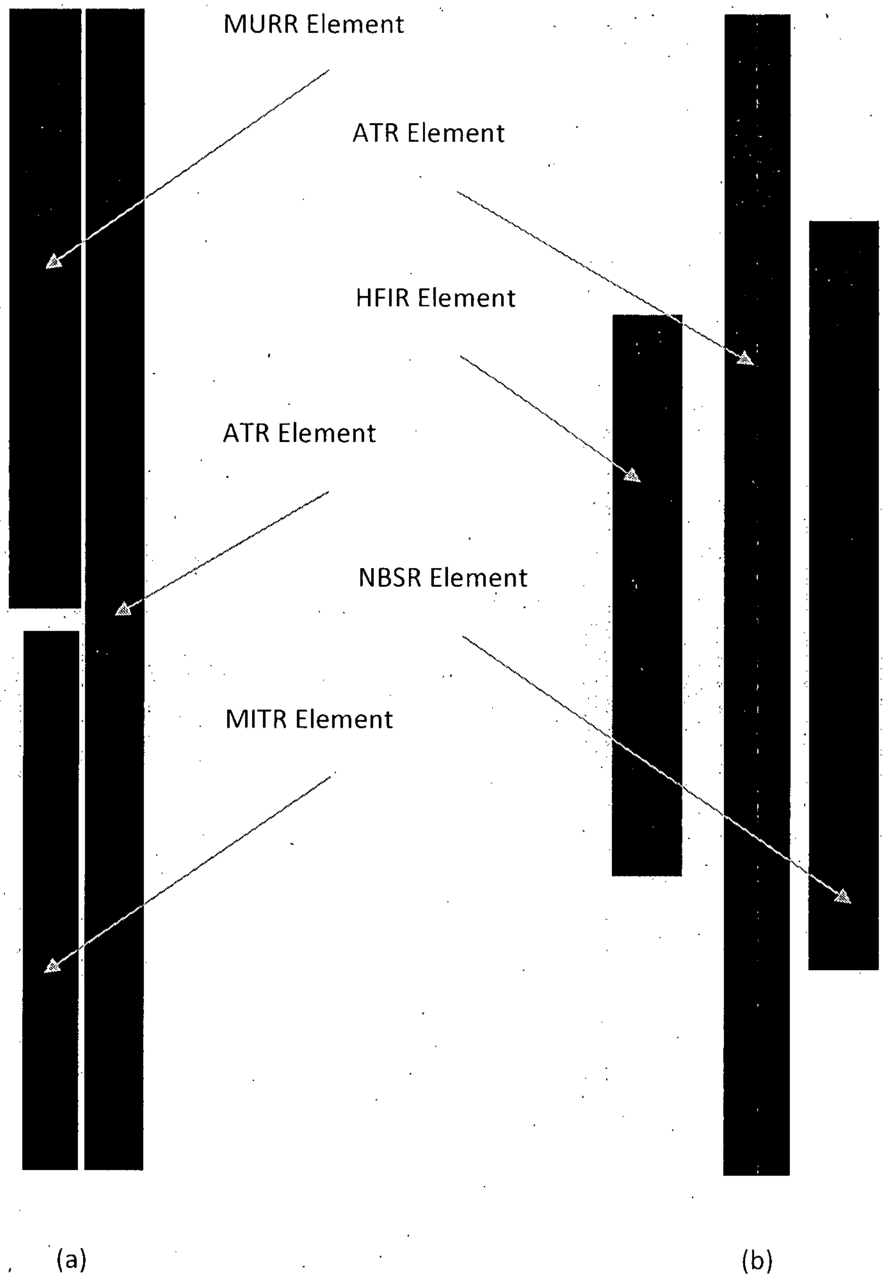

Figure 2.11: Vertical cross section in (a) y,z plane (refer to Figure 2.9 for A-A cut plane) and (b) x,z plane (refer to Figure 2.10 for B-B cut plane) showing vertical position of elements of the MCNP5 lumped configuration.

2.1.8 Materials

All fuel elements contain the same materials including the U-Mo fuel meat, Al clad and side plates (if applicable), and moderator (water or air). The fuel was assumed to be enriched to a value of 20.0 wt% U^{235} and includes trace elements of U^{234} which accompanies U^{235} at a ratio of 7.639E-3 atoms-

U^{234}/atoms-U^{235} [21]. It is conservatively assumed that U^{234} is contained in the fuel while enriching to 20.0wt%U^{235} to provide sufficient evidence that all models are subcritical with an abundance of fissile material.

A fabrication report produced by the Idaho National Laboratory reports that the loading density of Uranium in the U-Mo alloy is found to be 15.3 g/cc and the weight ratio of Uranium to Molybdenum is 90%U-10%Mo [22]. Table 2.2 provides a summary of the fuel meat material properties used to conduct the eigenvalue calculations.

The cladding material for all fuel elements considered as a part of this study, are made of aluminum. It was assumed that the cladding was fabricated from natural aluminum as presented in Table 2.3.

All water moderated calculation assumed to be of perfect H_2O content as seen in Table 2.4. Thermal neutron scattering in hydrogen was also incorporated into the models using the {lwtr} card in MCNP5.

All air moderated calculations assumed to only contain nitrogen, oxygen, and argon as presented in Table 2.5.

Table 2.2: MCNP5 material properties of U-Mo fuel meat

Density [g/cc] = 17.00		
Nuclide	ZAID	Weight Fraction
U^{234}	92234	[redacted]
U^{235}	92235	[redacted]
U^{238}	92238	[redacted]
$Mo^{Natural}$	42000	[redacted]

Table 2.3: MCNP5 material properties of Al fuel cladding

Density [g/cc] = 2.7		
Nuclide	ZAID	Weight Fraction
Al^{27}	13027	1.0000000

Table 2.4: MCNP5 material properties of water moderator

Density [g/cc] = 1.0		
Nuclide	ZAID	Weight Fraction
H^1	01001	0.1100000
O^{16}	08016	0.8900000

Table 2.5: MCNP5 material properties of air moderator

Density [g/cc] = 0.001204		
Nuclide	ZAID	Weight Fraction
N^{14}	07014	0.7550000
O^{16}	08016	0.2320000
$Ar^{Natural}$	18000	0.0130000

2.1.9 Eigenvalues

All eigenvalues (k_{eff}) presented in Table 2.6 include error values plus or minus two standard deviations. The summary of k_{eff} values presented below demonstrates that under the most conservative case considered (lumped configuration), all elements remain subcritical.

Table 2.6: Effective multiplication factor (k_{eff}) summary

Case	*a*	*b*
1	0.51766 ± 0.00116	0.08875 ± 0.00030
2	0.75160 ± 0.00150	0.17108 ± 0.00042
3	0.46009 ± 0.00122	0.06862 ± 0.00024
4	0.56449 ± 0.00126	0.10923 ± 0.00030
5	0.47705 ± 0.00124	0.07023 ± 0.00022
6	0.75161 ± 0.00136	0.18868 ± 0.00048
7	0.86335 ± 0.00126	0.21454 ± 0.00054

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Oregon State University

Addendum to the

Special Nuclear Material License Application

Fuel Assembly Description and Safety Report

1 INTRODUCTION

In further support of the safety analysis presented as part of the application, three further questions are proposed.

2 Purpose

This addendum identifies and addresses concerns regarding each of these three questions from a criticality safety perspective by conducting a criticality evaluation using MCNP5. These questions are:

1. What is the criticality level of the material in the event of a fire where all of the material melted together and water was added?
2. Is it possible for the elements to be closer than was shown for the lumped analysis? If so, the analysis should be redone with the closest possible configuration.
3. The analysis was done for only 0% and 100% water. The reviewer stated that higher k_{eff} values were possible with intermediate levels of water, and that the analysis should be performed for such configurations.

3 RESPONSE TO QUESTION 1

What is the criticality level of the material in the event of a fire where all of the material melted together and water was added?

3.1 Introduction

Let us assume that the most credible and limiting off-normal scenario associated with the oversight of the materials to be a fire which consumes the fuel elements such that the surrounding temperature exceeds their melting point temperature allowing for their reconfiguration.

A lock out/tag out procedure will allow for only single element to be removed from the storage cabinet at any given time. As a result of this administrative requirement, it is then assumed that the most limiting criticality scenario during a fire will occur if all elements are stored in their designated locations within the storage cabinet. The limited geometry of the cabinet will not allow for the fuel to melt and spread across the floor of the facility, but rather remain confined in their storage locations in a more favorable geometry. Figure A3.1 presents a rendering of the element storage cabinet.

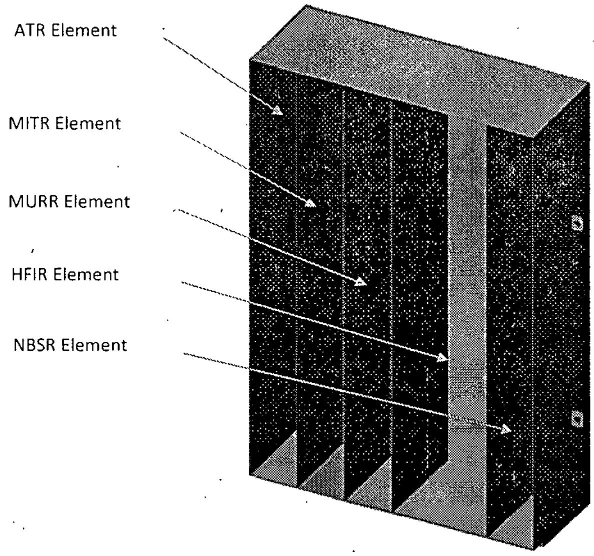

Figure A3.1: Element Storage Cabinet

3.2 Scenario Description and Configuration

A front and side view of the cabinet with its corresponding dimensions are presented in Figure A3.3.2. The cabinet will be made of stainless steel with wall of thickness 0.32cm or larger; it has a depth of 45.72cm and a height of 182.25cm.

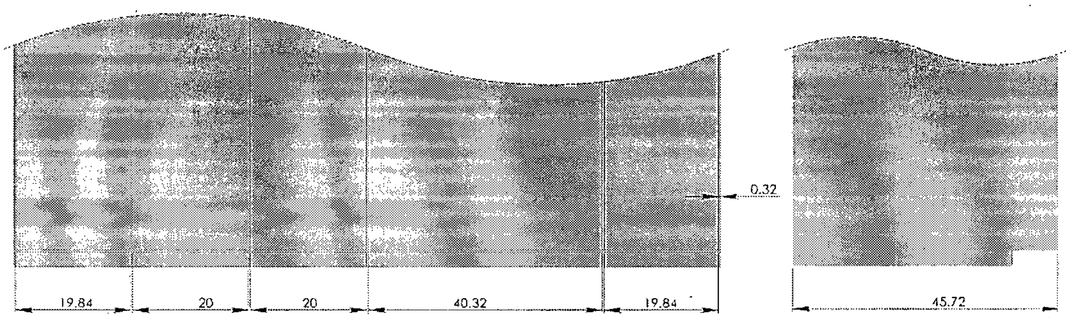

Figure A3.2: Broken out frontal and side dimensional view of cabinet (units in cm)

During this accident scenario it is assumed that fire takes place inside the cabinet melting all fuel meat, while burning all the aluminum cladding away. In order to stop the fire it is assumed that workers spray the cabinet with hoses. At the end of the scenario the cabinet is lined with the accumulative volume of fuel meat at its floor and a column of water above the fuel which fills the cabinet completely as a result of the fire hose. The integral fuel region volume of each element presented as a part of the safety analysis report is summarized in Table A3.1.

Table A3.1: Integral fuel volume of each element

Element	Fuel Volume [m^3]
ATR	[redacted]
MITR	[redacted]
MURR	[redacted]
HFIR	[redacted]
NBSR	[redacted]

An effective fuel meat height can be calculated for each fuel element in its designated location within the cabinet given the cabinet dimensions presented in Figure A3. and the integral volume of fuel meat presented in Table A3.1. These thicknesses are presented in Figure A3..

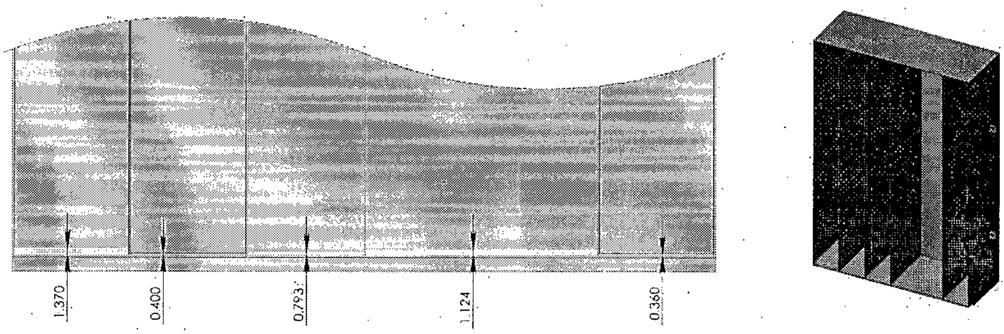

Figure A3.3: Broken frontal view of resulting thicknesses of melted fuel elements in their designated location and isometric view of cabinet. Cabinet door not shown (units in cm)

The cabinet is designed to have a one inch spacer between the floor of the cabinet and the ground to allow for the hinged cabinet door to open and shut properly. This spacer creates an air void between the fuel meat and the floor in the physical configuration, however, the MCNP5 model was developed such that it assumes the cabinet is directly on a concrete flow. This provides a more reflective base for the cabinet and is therefore more conservative than the physical configuration.

3.3 MCNP5 Model

An MCNP5 model was built to simulate the previously described scenario. Figure A3.4 and Figure A3.5 present a front and side view of the MCNP5 model. Concrete and air or water surround the farthest extents of the fuel storage cabinets by two meters in every direction. All stainless steel cabinet walls were included in the model as well.

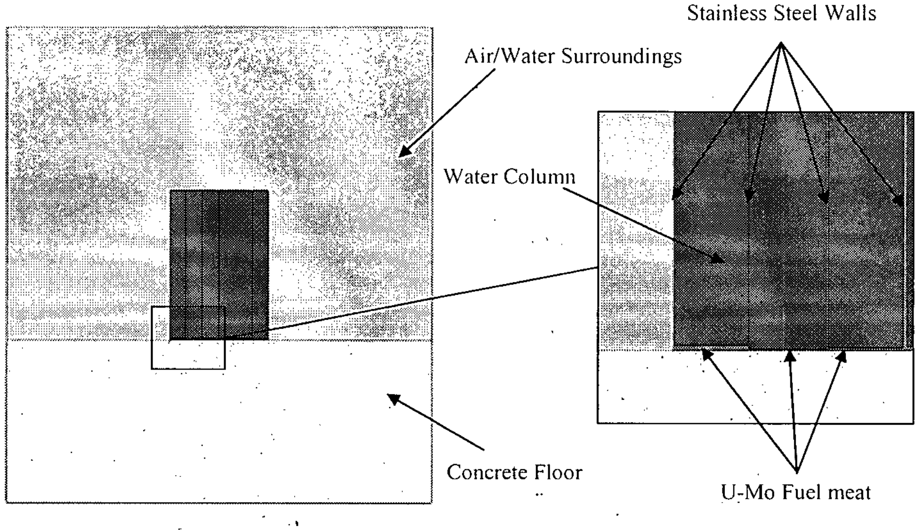

Figure A3.4: Front view and broken out view of MCNP model

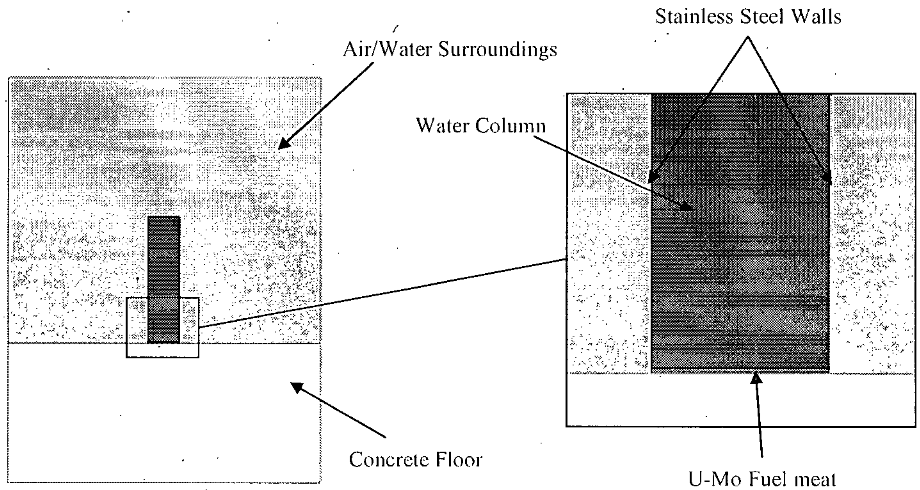

Figure A3.5: Side view and broken out view of MCNP model

Two simulations were run as a part of this scenario. The first considers the surroundings (Figure A3.4 and A3.5) to consist of air, and the second substitutes water as the atmospheric material surrounding the cabinet to simulate a flood circumstance after the fire. The eigenvalues of these two simulations are presented in Table A3.2 with 95% confidence levels (2 standard deviations).

Table A3.2: Summary of eigenvalues calculated by MCNP5 during the limiting scenario

Simulation	Eigenvalue
Air surroundings	0.64858 ± 0.00134
Water surroundings	0.65465 ± 0.00126

3.4 Materials

Shown in Table A3.3, all water moderated calculation assumed to be of perfect H_2O content. Thermal neutron scattering in hydrogen was also incorporated into the models using the {lwtr} card in MCNP5.

All fuel elements contain the same materials including the U-Mo fuel meat, Al clad and side plates (if applicable), and moderator (water or air). The fuel was assumed to be enriched to a value of [redacted] and includes trace elements of [redacted]. It is conservatively assumed that U^{234} is contained in the fuel while enriching to [redacted] to provide sufficient evidence that all models are subcritical with an abundance of fissile material.

A fabrication report produced by the Idaho National Laboratory reports that the loading density of Uranium in the U-Mo alloy is found to be [redacted] and the weight ratio of Uranium to Molybdenum is [redacted]. Table 2.2 provides a summary of the fuel meat material properties used to conduct the eigenvalue calculations.

The concrete material which supports the cabinets considered as a part of this study was taken from the MCNP criticality primer for standard concrete. Table 2.3 provides a summary of the concrete material properties used.

The cabinet material which holds all the elements is made of stainless steel. The material properties for stainless steel were assumed to be made of stainless steel 304. This material composition was taken from the MCNP criticality primer. Table A3.6 provides a summary of the stainless steel material properties used.

All air moderated calculations assumed to only contain nitrogen, oxygen, and argon as presented in Table 2.5.

Table A3.3: MCNP5 material properties of water moderator

Density [g/cc] = 1.0		
Nuclide	ZAID	Weight Fraction
H^1	01001	0.1100000
O^{16}	08016	0.8900000

Table A3.4: MCNP5 material properties of U-Mo fuel meat

Density [g/cc] = 17.00		
Nuclide	ZAID	Weight Fraction
U^{234}	92234	[redacted]
U^{235}	92235	[redacted]
U^{238}	92238	[redacted]
$Mo^{Natural}$	42000	[redacted]

Table A3.5: MCNP5 material properties for concrete

Density [g/cc] = 2.3		
Nuclide	ZAID	Weight Fraction
H^{1}	01001	0.1000000
O^{16}	08016	0.5320000
$Si^{Natural}$	14000	0.3370000
Al^{27}	13027	0.0340000
Na^{23}	11023	0.0290000
$Ca^{Natural}$	20000	0.0440000
$Fe^{Natural}$	26000	0.0140000

Table A3.6: MCNP5 material properties of stainless steel 304

Density [g/cc] = 7.92		
Nuclide	ZAID	Weight Fraction
$Fe^{Natural}$	26000	0.695000
$Cr^{Natural}$	24000	0.190000
$Ni^{Natural}$	28000	0.095000
Mn^{55}	25055	0.020000

Table A3.7: MCNP5 material properties of air moderator

Density [g/cc] = 0.001204		
Nuclide	ZAID	Weight Fraction
N^{14}	07014	0.7550000
O^{16}	08016	0.2320000
$Ar^{Natural}$	18000	0.0130000

4 RESPONSE TO QUESTION 2

Is it possible for the elements to be closer than was shown for the lumped analysis? If so, the analysis should be redone with the closest possible configuration.

4.1 Introduction

Administrative oversight allows for only a single element to be removed from the storage cabinet at any given time. If this is the case, then the storage configuration would be the closest possible configuration the elements will be oriented in at any given time.

5 RESPONSE TO QUESTION 3

An analysis was done for only 0% and 100% water. However, we are interest in determining the k_{eff} values with intermediate levels of water and air.

5.1 Introduction

It has been demonstrated that the storage configuration (Case 6a and 6b from the safety analysis report) is the closest the elements will be configured at any time. It is therefore determined that three additional eigenvalue calculations shall be run in the storage configuration to address the following moderator conditions:

- 25% air, 75% water
- 50% air, 50% water
- 75% air, 25% water

5.2 MCNP5 Model

All five elements were modeled and assumed to be in an infinite medium of moderator. The infinite medium was modeled as a cubic volume of moderator material extending beyond the maximum fuel element dimensions by two (2) meters in all directions. This is assumed to be a sufficient attenuation distance such that it is analogous to that of an infinite medium. All elements were modeled using their designated as-built drawings. The elements sit on the floor of the storage cabinet in their storage positions; only the fuel meat, cladding, and side plates were modeled in each of the elements. The vertical location in which the fuel elements are located relative to the floor of the cabinet including end fittings was taken into consideration in the model. A vertical cross section of the storage configuration MCNP5 model is presented in Figure 2.7. All elements were modeled such that their storage depth within the cabinet was similar to one another as will occur in the physical storage cabinet, this can be seen in Figure 2.8.

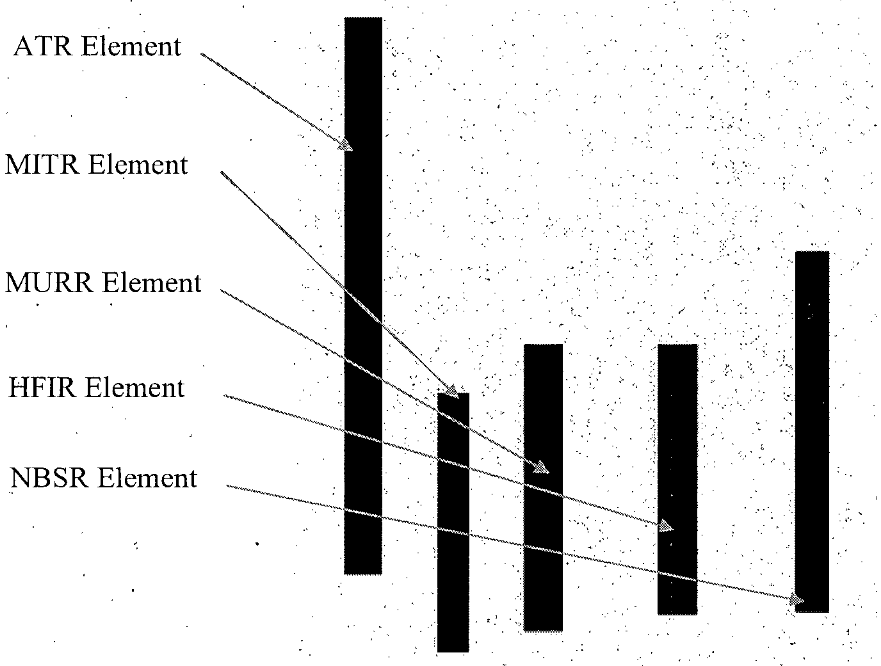

Figure A5.1: Vertical cross-section of the MCNP5 storage configuration model

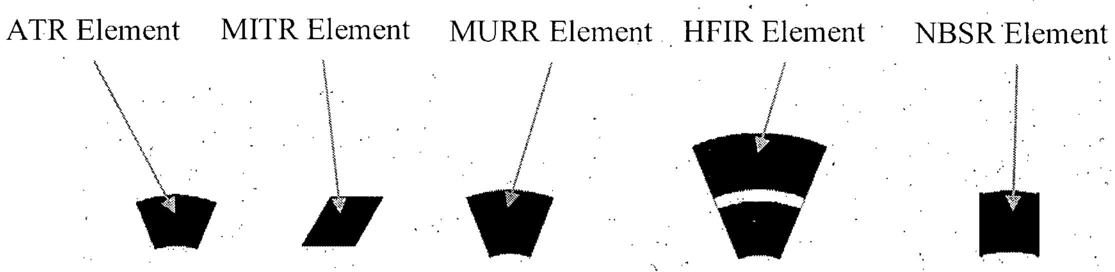

Figure A5.2: Horizontal cross-section of the MCNP5 storage configuration model

A summary of the effective multiplication factors calculated by MCNP5 is presented in Table A5.1. The largest eigenvalue was determined to be that of 100% water with a value of 0.75161 ± 0.00136.

Table A5.1: Effective multiplication factor (k_{eff}) summary

Moderator Composition	Eigenvalue
100% air, 0% water	0.18868 ± 0.00048[1]
25% air, 75% water	0.21664 ± 0.00078
50% air, 50% water	0.40289 ± 0.00102
75% air, 25% water	0.57544 ± 0.00122
0% air, 100% water	0.75161 ± 0.00136[1]

5.3 Materials

All materials used in the storage rack configuration are congruent with that found in section A3.4, with exception of the moderator, which was varied for three weight fraction ratios between air and water. The material properties used in the MCNP5 model for this sensitivity study are presented in Table A5.2.

Table A5.2: MCNP5 material properties for moderator

Density [g/cc]			0.250903	0.500602	0.750301
Material	Nuclide	ZAID	Weight Fraction		
			25%air,75%water	50%air,50%water	75%air,25%water
Air	N^{14}	07014	0.56625	0.37750	0.18875
	O^{16}	08016	0.17400	0.11600	0.05800
	$Ar^{Natural}$	18000	0.00975	0.00650	0.00325
Water	H^{1}	01001	0.02750	0.05500	0.08250
	O^{16}	08016	0.22250	0.44500	0.66750

6 SUPPLEMENT MATERIAL

Two MCNP5 eigenvalue calculations were run to benchmark the software against that of the *International Handbook of Evaluated Criticality Safety Benchmark Experiments.* These calculations were conducted using Reference [1] Model LEU-COMP-THERM-013 Case 1 and Case 4. These models consist of clusters of 36 inch long aluminum clad $U(4.31)O_2$ fuel rods in a larger water-filled tank. Where case 1 includes a stainless steel 304 separator plate between half of the fuel rods and case 4 substitutes the stainless steel plate for a Boroflex plate. Refer to [1] for a more thorough description.

Table A6.1 presents a summary of eigenvalues calculated at OSU to that provided by Reference [1]. In both cases the eigenvalues produced by the OSU calculation have a larger multiplication factor than that of the benchmark calculations. Employing a conservative perspective on the criticality safety analysis, it is therefore concluded that no bias value shall be added to the multiplication factor values provided by OSU.

Table A6.1: Benchmark eigenvalue calculations

Calculation Reference	Cross Section Identification	Eigenvalue $\pm$ 1sigma	
		LEU-COMP-THERM-013 Case 1	LEU-COMP-THERM-013 Case 4
Benchmark Calculations	MCNP (Continuous Energy ENDF/B-V)	0.9981 ± 0.0006	0.9972 ± 0.0007
OSU Calculation	MCNP (Continuous Energy ENDF/B-V)	0.99861 ± 0.00063	0.99864 ± 0.00066

7 REFERENCES

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