UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING CONVERSION ORDER TO CONVERT FROM

HIGH-ENRICHED TO LOW-ENRICHED URANIUM FUEL

FACILITY OPERATING LICENSE NO. R-75

OHIO STATE UNIVERSITY

DOCKET NO. 50-150

1.0 INTRODUCTION

In accordance with 10 CFR 50.64, which requires that non-power reactors convert to a low-enriched uranium (LEU) fuel, except under certain conditions, the Ohio State University (OSU) has proposed to convert the fuel in its pool-type nuclear reactor (the reactor) from high-enriched uranium (HEU) to LEU. OSU submitted a Safety Analysis Report (SAR) and revised Technical Specifications dealing with the changes needed to convert to LEU fuel and a power increase to 500 kW (thermal) by letter dated October 7, 1987. The staff's initial review led to the conclusion that the magnitude of the task of reviewing and approving both the conversion and the power increase (together) was such that the schedule for a timely issuance of the conversion order would be adversely affected. Accordingly, the staff requested that OSU separate the conversion and power increase issues and address the conversion first. OSU agreed and the staff's revised review led to additional questions pertinent to the HEU/LEU conversion to which OSU responded by letter dated May 6, 1988. This letter transmitted the answers to the questions and revisions to both the revised Technical Specifications and the new LEU Safety Analysis Report. The principal technical and safety analyses supporting the application are contained in the new LEU SAR (attached to the October 7 letter), which presents the assumptions, methods, and results of computations performed at OSU in support of the Ohio State University Research Reactor (OSURR) conversion. This material is available for review at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. This Safety Evaluation (SE) was prepared by T.S. Michaels, Project Manager, Division of Reactor Projects III, IV, V, and Special Projects, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. Major contributors to the technical review include W. R. Carpenter, R. E. Carter, and C. H. Cooper of EG&G, Idaho National Engineering Laboratory (INEL).

2.0 EVALUATION

2.1 General

The OSU reactor currently is licensed for operation at power levels not to exceed 10 kW thermal, using MTR-type flat-plate fuel, and cooled by natural thermal convection of the pool water. During the construction phase of the HEU/LEU conversion certain hardware and instrumentation changes will be made in anticipation of 500 kW operation, however, the Licensee has proposed no changes to any reactor system or operating characteristics which would adversely affect reactor operations at 10 kW (thermal). The following evaluations and conclusions are based on the assumption that the only pertinent facility modification necessitated by the conversion is the physical replacement of the HEU fuel elements by new LEU fuel elements.

2.2 Fuel Construction and Geometry

The HEU fuel elements currently installed at OSURR contain 10 plates each, in which the fuel meat is a 93.4% enriched uranium aluminum alloy. Each fueled plate contains approximately 14 grams of U-235 for a total U-235 loading of about 140 grams per fuel element. The new LEU standard fuel elements will have the same outer dimensions as the HEU fuel elements, but will contain 18 plates (16 fueled plates and two dummy plates) each, with the fuel meat in the form of uranium silicide (enriched to less than 20%) dispersed in an aluminum matrix. The LEU fueled plates will each contain approximately 12.5 grams of U-235 for a total U-235 loading of about 200 grams of U-235 per fuel element. The standardized LEU plates are thinner than the HEU plates, with thinner aluminum cladding, so, even though there are more LEU plates per fuel element, the metal-to-water ratio for the LEU and HEU fuel elements is very nearly equal, with the LEU fuel element being slightly lower i.e., slightly more water volume in the new LEU fuel elements. Fuel elements with plates and uranium composition essentially identical with the proposed OSSUR plates were developed especially for the U. S. Non-Power Reactor (NPR) conversion program by Argonne National Laboratory and reviewed and approved by the NRC. These fuel elements have been extensively tested under extremely adverse conditions in the Oak Ridge Research Reactor with no failures having a safety significance.[1]

In addition to the 200 gram U-235 standard fuel elements, OSU also proposes to acquire some fuel elements containing 10 plates (all fueled) to accommodate the three control rods and the single regulating rod and some partially loaded 18 plate fuel elements to accommodate precise excess reactivity adjustment. The 18 plate partial fuel elements are physically identical to the standard elements with the exception that they contain more aluminum dummy plates in order to make up loadings of 25%, 37.5%, 50% and 62.5% of the nominal uranium loading of a standard fuel element. Such use of the 10 plate control elements and the 18 plate partially loaded elements has been considered by the staff and judged acceptable.

2.3 Fuel Storage

A 10 x 3 fuel storage array is located in the fuel storage pit which is located at the east end of the Reactor Pool. The effective multiplication factor (K_{eff}) when this fuel rack is fully loaded with 30 LEU fuel elements is less than 0.9 and is, therefore, judged acceptable for storage of the new LEU fuel. This same fuel rack was previously found acceptable for the safe storage of HEU fuel. To accommodate the storage of both the HEU and LEU cores during the interim period while they reside at OSU, and, following removal of the HEU core, to accommodate the storage of some LEU fuel, two new fuel storage racks will be built and located in the bulk shielding pool.[2] Each of these racks will accommodate 16 fuel elements in a 2 x 8 array for a total fuel element loading of 32 in the bulk shielding pool. Criticality safety of these two racks was found acceptable by the staff[3] provided they are physically separated from each other on all sides by at least 24 inches. This separation has been required by the Technical Specifications.

2.4 Critical Operating Masses of U-235

The OSURR HEU core contains 24 fuel elements including four partially loaded control elements. Its critical mass is approximately 3.4 kg uranium containing 3.2 kg U-235. The critical mass of the LEU reactor is predicted to be approximately 19 kg uranium containing 3.7 kg U-235 in 20 fuel elements including the four partially loaded (10 plate) control elements. This increased LEU U-235 loading is necessitated by the large increase in concentration of U-238, which absorbs low energy (thermal) and epithermal (resonance) energy neutrons, and causes a hardening of the thermal neutron spectrum. The increase of uranium is achieved by increasing its concentration in the fuel matrix and increasing the number of fuel plates in the fuel elements. The proposed concentration is similar to that successfully achieved and tested in the Oak Ridge Research Reactor (ORR).

The OSU SAR and the OSU reply to the HEU/LEU conversion questions discusses sensitivity calculations of reactivity for different core configurations and a comparison of the calculated values of neutron lifetime and the effective delayed neutron fraction. In this study, fuel elements were shifted and/or removed to form various core configurations in the OSU LEU core in order to determine fuel element worth as a function of core position. These fuel element perturbations were compared, where possible between the HEU and LEU cores and the resultant reactivity changes were compared. In all cases the reactivity effects of core rearrangements between the HEU and LEU cores is as expected when the higher fuel element loading and lower total number of fuel elements in the LEU core is factored into the comparison. With regard to the comparison of the prompt neutron lifetime and effective delayed neutron fraction, the values for the HEU core and the LEU core are very similar with the lifetime being slightly shorter and the delay fraction being slightly larger for the LEU core as would be expected because of the increased neutron absorption in the U-238 and somewhat harder neutron spectrum of the LEU core. The results of this study demonstrate the basic neutronic similarity between the HEU and LEU cores at OSU.

2.5 Hydraulics and Thermal-hydraulics

At OSU, there are 18 plates in each new LEU fuel element as compared to 10 plates in each HEU fuel element. The LEU plates, however, are thinner and contain less U-235 (per plate). This results in less heat generation per plate, due to the decreased fuel loading, and improved heat transfer to the coolant, due to the thinner clad. Also, the increased number of LEU fuel plates per element is offset by the plates being thinner, which results in a very similar metal to water ratio for the two fuels with the cross section for coolant flow being slightly larger in each LEU fuel element. Accordingly, the OSU SAR calculations indicate that maximum fuel plate temperatures will not be significantly different between the HEU and LEU cores (the LEU plates being cooler), and at the 10 kw power level there is only a 0.2°F temperature difference between the LEU fuel plate surface and the coolant temperature. Since the OSURR Technical Specifications for 10 kilowatt operation apply a maximum core inlet water temperature of less than 95°F, the surface temperature of the fuel plates will be much less than that required to initiate the onset of nucleate boiling, and very much less than the melting point of aluminum.

2.6 Power Density and Power Peaking

Power densities and power density peaking, including both nuclear and engineering factors, were computed by OSU. Power densities in individual fuel plates were very similar between the HEU and LEU fuel with the LEU fuel being slightly higher. The power distribution among the fuel elements is similar in profile between the HEU and LEU cores but significantly flatter in the LEU core (a peak to average power density ratio of 1.64 versus 2.68). Therefore, the over-power condition at which onset of nucleate boiling occurs in the hottest channel is significantly higher in the LEU core than in the existing HEU core, primarily because of the the flatter power distribution and the lower initial heat generation per plate. Thus the safety margin between the licensed power and the power at which temperatures might lead to fuel damage at the OSURR is much higher in the LEU core than the currently licensed HEU core.

2.7 Control Rod Worths

The reactivity worths of the three control rods and single regulating rod were computed by acceptable methods for the OSURR LEU core. The calculated worths are 2.70% delta k/k, 2.47% delta k/k, 2.16% delta k/k, and 0.48% delta k/k respectively. These values are somewhat smaller than the HEU core, as expected, because of the increased neutron absorption in the U-238, but the OSURR LEU rod-worths are fully acceptable for safe reactor operation and control.

2.8 Shutdown Margin

On the basis of the computed control rod worths and the authorized excess reactivity, the OSU reactor would be subcritical by approximately 3.1% delta k/k with the control rod of maximum worth and the regulating rod fully withdrawn. This is substantially larger than the Technical Specification margin of at least 1% delta k/k, and is acceptable.

2.9 Excess Reactivity

Additional reactivity above cold, clean critical is required to allow a reactor to perform programmatic and academic functions. Ohio State's submittal discussed and presented calculated changes in reactivity caused by various LEU core configurations. There is reasonable assurance that the excess reactivity permitted by the Technical Specifications, which is the same for the OSURR LEU and HEU cores, can be achieved. Since the authorized maximum excess is 1.5% delta k/k, inadvertent step insertion of all of this excess would allow the reactor to become prompt critical. However, analysis presented in the original hazards evaluation and revisited in the new LEU SAR shows there is no credible means to add all of the authorized excess, and in fact, any reasonable estimate of a maximum inadvertent reactivity addition is well below the 0.7% delta k/k necessary for prompt critical. Therefore, any plausible transient power increase would be quickly terminated by a power level scram or operator intervention and would result in increased fuel temperature of only a few degrees C, which was previously accepted by the staff for the original HEU core and is currently accepted for the proposed LEU core.

2.10 Reactivity Feed-back Coefficients

The temperature coefficient of reactivity and the void coefficient of reactivity were computed for both the HEU and LEU cores. Both coefficients are more negative than required by the Technical Specifications. The void coefficient of the LEU core is somewhat more negative than for the HEU core because of the more under-moderate LEU core condition. The temperature coefficient of the LEU core is also more negative than in the HEU core because of the epithermal Doppler effect in the neutron capture resonances of the relatively much more abundant U-238 present in the LEU fuel. The Doppler feedback is prompt, and therefore more effective in countering a reactor transient in the LEU core than is the heat transfer dependent moderator temperature coefficient in the HEU core. Thus, these reactivity coefficients for the proposed LEU core are both larger and more effective in leading to reactor stability than the reactivity coefficients for the HEU core, and therefore are acceptable.

2.11 Fission Product Inventory and Containment

The total inventory of fission products from operation at 10 kW is low and will not be significantly different between the OSU HEU and LEU cores. Furthermore, because there is a predicted 291 fuel plates in the OSU LEU core versus an actual loading of 224 fuel plates in the OSU HEU core, the inventory per plate is less in the LEU core. The aluminum cladding however is thinner on the LEU plates which may tend to reduce the integrity of the fission product barrier. This cladding thickness, however has been successfully used for years on many NRC licensed HEU fueled research reactors and is currently in use on the University of Michigan LEU fueled reactor.[4] Because there have been no failures or significant releases of fission products, attributable to this cladding thickness, there is reasonable assurance that the new LEU fuel will perform satisfactorily in containing fission products in the OSU reactor.

2.12 Potential Accident Scenarios

Among the various potential accidents considered by the Licensee or the staff at the time of the original license issuance to OSU, only two could be affected by the conversion from HEU to LEU fuel. These two scenarios are addressed below.

2.12.1 Inadvertent Insertion of Excess Reactivity

In the approved Hazards Summary Report (HSR) for the original HEU-fueled OSURR, the maximum credible accident was defined as a 1.5% delta k/k reactivity insertion. The analysis shows, by comparison with experiments, that the OSURR achieves inherent shutdown through the negative void and temperature coefficients of reactivity feedback mechanisms, with peak fuel cladding temperatures less than those needed to damage the fuel plates to the point of causing release of fission products. For the proposed 10 kW OSURR LEU core an accident analysis was performed by the Licensee that compares the LEU core to the HEU core analyzed in the approved HSR for 10 kw operation. The conclusion reached was that the accident consequences for an LEU core are less severe than for HEU fuel. The technique used by OSU to reach this conclusion was to compare the nuclear parameters of prompt neutron lifetime and the effective delayed neutron fraction to compute an initial period for a 1.5% delta k/k step reactivity insertion in the LEU core then compare this period to the one calculated for the HEU core in the HSR. The values are nearly identical, 9.2 msec for the LEU and 9.3 msec for the HEU core. Following this, an evaluation of the inherent feedback was performed which shows the average fuel temperature necessary to mitigate the transient is less in the LEU core, and the maximum fuel plate temperature is significantly less due to the flatter peaking factors in the LEU core. The staff has reviewed this OSURR analysis and agrees with OSU's conclusion that the consequence parameters for a 1.5% delta k/k step reactivity insertion in the OSU LEU core are less severe than in the currently licensed OSU HEU core, and therefore, acceptable.

2.12.2 Fission Product Release Accident

A mechanistically undefined accident scenario leading to total release of all of the fission products present in one fuel plate located under water in the core under very conservative assumptions regarding pool conditions (dispersion and diffusion), and power operations conditions (maximum fission product inventory) was considered to be the maximum hypothetical accident (MHA) in the original hazards (HSR) evaluation of the OSURR facility for the granting of its original operating license for the HEU core. This same accident is also considered to be the MHA with LEU fuel at the OSURR. The only significant difference between the inventory of radioactivity in the LEU and in the HEU element is the plutonium-239 formed by neutron capture in the uranium-238, which is much more abundant in the LEU fuel. However, on the basis of the licensed power level and consequent burn-up of fuel at OSU, the additional build-up of the inventory of plutonium in the LEU fuel is radiologically insignificant, and release of the fission products including this plutonium from damaged LEU fuel would result in maximum potential radiation exposures in the unrestricted areas of only a small fraction of those allowed by 10 CFR Part 20 guidelines. Therefore, damage to LEU fuel in place of HEU fuel would cause no significant change in the risk to the health and safety of the public, which was already acceptably low for the current HEU core operated at 10 kw.

3.0 <u>ENVIRONMENTAL CONSIDERATION</u>

This amendment involves changes in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in inspection and surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously evaluated, or create the possibility of a new or different kind of accident from any accident previously evaluated, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

5.0 REFERENCES

1. Hobbins, R.R. et al. Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Nonpower Reactors, NUREG-1313, February 1988.

2. "Criticality Safety Analysis for Interim HEU Fuel Storage in the Bulk Shielding Facility Pool of the Ohio State University Research Reactor," by Joseph W. Talnagi, Senior Research Associate, Nuclear Reactor Laboratory, The Ohio State University, Columbus, Ohio, June 6, 1988.

3. Letter from T.S. Michaels (NRC) to R.F. Redmond (OSU), July 20, 1988, Subject: Criticality Safety Analysis for Interim HEU Fuel Storage.

4. Amendment No. 27 to the University of Michigan Ford Nuclear Reactor License, Dated February 10, 1981.

Dated: September 27, 1988

Enclosure 4

INFORMATION TO BE SUPPLIED BY OSU

1. In order to provide a comparison between measured values with computed predictions please provide a report six months after fuel loading that addresses the items in the outline of the attachment to this enclosure.

2. Please submit a Fuel Load and Reactor Start-Up Planning document, with specific procedures and instrumentation requirements. This document should also identify the personnel who will be involved with the start-up and who have previous experience with initial fuel-loading, power calibration and start-up of a non-power reactor. Please inform me of the expected date when fuel loading will begin and any changes that may occur to this date.

3. Your SAR submittals of October 7, 1987, and response to questions of May 6, 1988 reveal some inconsistencies with regard to the status of the Central Irradiation Facility (CIF). On page 103 of the October 7, 1987 SAR it is stated that the CIF is flooded. On page 12 of the May 6, 1988 submittal it is stated that the CIF is partially flooded. In response to question 12e (May 6, 1988), it is stated that the CIF is voided and the void coefficient is positive. Please advise us as to the true status of the CIF (% flooded) during normal operations and whether the CIF void coefficient was calculated for the LEU core. Also, provide the void status of the CIF for the flux profile calculations on pages 14 and 15 of Appendix B (May 6, 1988).

4. How does OSU intend to determine if the effective delayed neutron fraction is modified (see question 13, May 6, 1988).

5. Your analysis, on page 19 of Appendix B, second sentence (May 6, 1988), states that the heat capacity of the new LEU fuel plates is higher than the current HEU plates. This conclusion does not agree with NUREG-1313 and the IAEA Guide which you reference on page 19, Appendix B, because these references show the heat capacity for both HEU and LEU is virtually identical with the HEU being slightly higher. Although your conclusion does not alter the overall conclusion that LEU fuel stays cooler because of the flatter power distribution and higher feedback coefficients for LEU, please comment on your conclusion vis-a-vis the other references.

Attachment to Enclosure 4

OUTLINE OF REACTOR START-UP

REPORT AND COMPARISONS WITH CALCULATIONS

1. Critical Mass

   Measurement with HEU
   Measurement with LEU
   Comparisons with calculations for both LEU and HEU.

2. Excess (operational) reactivity

   Measurement with HEU
   Measurement with LEU
   Comparison with calculations for both LEU and HEU.

3. Control and regulating rod calibrations

   Measurements of differential and total rod worths, and comparisons with calculations for both HEU and LEU.

4. Reactor power calibration

   Methods and measurements that assure operation within the license limit. Comparison between HEU and LEU nuclear instrumentation setpoints, detector positions, and detector output.

5. Shutdown margin

   Measurement with HEU
   Measurement with LEU
   Comparisons between these, and with computations for both.

6. Partial fuel element worths for LEU

   Measurements of the worth of the 25%, 37.5%, 50% and 62.5% loaded fuel elements is the positions they are allowed to occupy.

7. Thermal neutron flux distributions.

   Measurements with HEU and LEU, and comparisons with each other and calculations.

8. Discussion of how compliance with void and temperature coefficient values in Technical Specifications is to be assured. Comparisons with any calculations for both HEU and LEU fuel.

9. Comparison of the various results, and discussion of the comparison, including an explanation of any significant differences which have an impact on both normal operation and potential accidents with the reactor.

10. Measurements made during initial loading of the LEU fuel, presenting subcritical multiplication measurements, predictions of multiplication for next fuel additions, and prediction and verification of final criticality conditions.