THE OHIO STATE UNIVERSITY

Nuclear Reactor Laboratory

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May 13, 1991

Ted Michaels
USNRC
PDNP
M.S. 11-B-20
Washington, D.C. 20555

Dear Ted:

As we discussed 5/10/91, I have enclosed a copy of the Power Increase Plan for the OSURR to go from 10 KW to 500 KW. We anticipate an outage of about one month to install the in-pool heat removal system, reload the core, and complete the necessary testing to return to 10 KW. This outage will begin about June 3, 1991.

All radiological monitoring as indicated in the plan will be completed starting at 10 KW and including 50 KW, 100 KW, 250 KW and 500 KW.

We do not anticipate operation above 100 KW until the latter part of the summer. It is most likely the results of our power increase will be included in this year's annual report.

Please call me if you have any questions.

Sincerely,

Rick

Richard D. Myser
Associate Director

RDM:sl

Enclosure

A020
1/1

Power Increase Plan

Date and Personnel

The expected date when the outage to begin the power increase is May 31, 1991. The following individuals shall be involved in the plan.

Richard D. Myser, Senior Reactor Operator, Associate Director of the Nuclear Reactor Laboratory. Supervised the unloading, reloading, and recalibration of the HEU core in 1981, 1984, and 1988, and the LEU core in 1988.

Joseph W. Talnagi, Senior Reactor Operator, Senior Research Associate, Engineering. Supervised the recalibration of the LEU and HEU cores in 1988 and participated in the unloading, reloading, and recalibration in 1984.

Joel M. Hatch, Senior Reactor Operator, Research Assistant, Engineering. Supervised control rod testing in 1984 and 1988 reloading operations and participated in the unloading, reloading, and recalibrations in 1984 and 1988.

Michael J. Davis, Senior Reactor Operator, Student Research Assistant. Participated in the unloading, reloading, and recalibration of the HEU core in 1988 and the current LEU core also in 1988.

Andrew Clark, P.E., is responsible for the design and installation of the heat removal system. The ex-pool part of the system is complete.

Instrumentation

Figure 3.12 of the SAR (attached) shows the location of the core grid plate, start-up source, and current neutron detection instrumentation. For the core loading the following instrumentation shall be utilized.

1. Logarithmic Power Monitoring Channel as described in Section 3.3.12 of the SAR (CIC #1).

2. Linear Power Monitoring Channel as described in Section 3.3.13 of the SAR (CIC #2).

3. Start-up Channel as described in Section 3.3.14 of the SAR (Fission Chamber).

4. Auxiliary Linear Power Level Monitoring Channel using a Compensated Ion Chamber (CIC) identical to those used in 1 and 2 above but placed in Beam Port #1.

5. Auxiliary (Fission Chamber) Pulse Channel to be placed in the Reactor Pool Thermal Column location G-7.

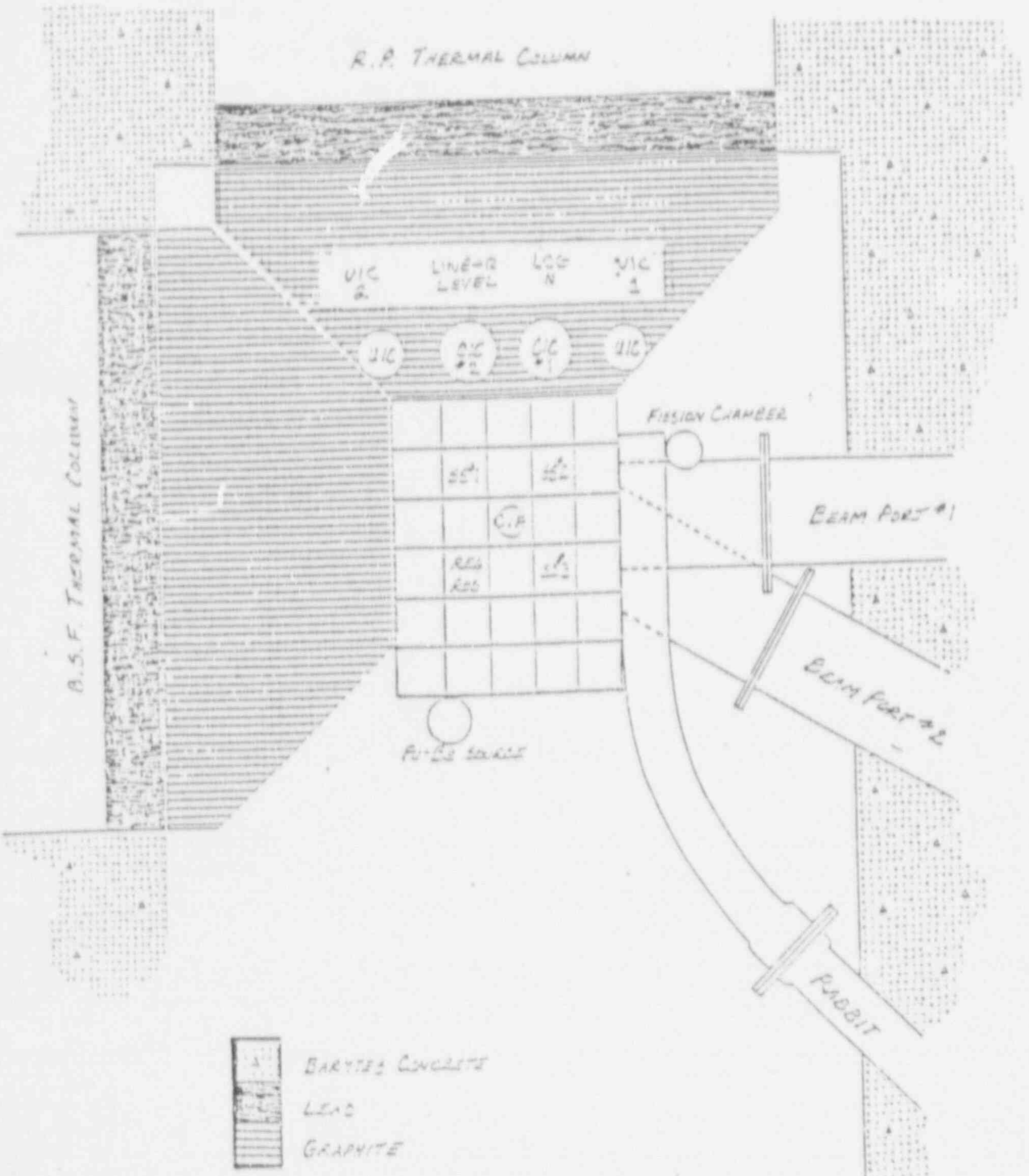

Figure 3.10

Locations of instruments and facilities in and around the OSURR core.

Pre-Installation Activities

Numerous items need to be completed before t's heat removal system can be installed in the pool. First the pool water will be analyzed for radioactivity to assure releases to the sanitary sewer system meet the requirements of 10CFR20. Then the core will be unloaded and the fuel placed in the storage pit. The control rods will be suspended from hangers along the N.E. corner of the pool. Lights and dry tubes will be moved to the East end. The CIF will be removed and stored in a shielded location (BSF Pool or BP#1 Area). Then the reactor pool will be drained by siphon. After draining, the pool will be carefully surveyed. Entry will be by extension ladder. Previous exposures from the 1988 LEU fuel conversion outage were below the minimum detectable limit for film badges. The floor will then be mopped dry and the storage pit sealed. The walls will be cleaned, rinsed and dried if necessary to help remove (control) bacterial growth. In addition, the NRL Staff committed to adding four additional film badges as environmental monitors about 100 feet away from the building in each direction. This was completed January 17, 1991.

Heat Removal System Installation and Other Dry Pool Activities

The installation of the heat removal system will be directed by Mr. Andy Clark, P.E. He designed the system that has been reviewed and approved by the ROC and NRC. The major components (decay tank, core shroud, and return line diffuser) have been fabricated by Allied Manufacturing. Installation should commence on or about June 3, 1991 and should require about two weeks.

During this same two week period the detector instrumentation rack and/or housings holding the detectors may be modified to allow for appropriate adjustment for the new power level. Prior to this we plan to test (move) at least UIC #1 and CIC #1 to help determine if modifications are necessary. Below are described the two proposed tests.

1. Level reactor power at 10 KW. Mark position of detector. Move UIC #1 away from the core until it reads a factor of 60 less or 166.66 watts. This position should correspond to a trip at 600 KW which is the new T.S. limit. Stop movement before the detector is removed from the lower support bracket. Return detector to original position.

2. Level reactor power at 10 KW. Mark position of detector. Move CIC #1 away from the core until it reads a factor of 100 less than 10 KW which is 100 watts. This position should correspond to full range on the Log N Channel. The collars on the detector housings will limit movement of CIC #1. UIC #1 may need to be moved away from the core to allow CIC #1 to be fully extended. We may also want to check period indications with this configuration since this comes from the CIC #1 signal.

Once the heat removal system is in place the control rods need to be reinstalled to check their alignment and the fit of the core shroud cover plates. We anticipate using the mock control rod fuel element and the spare

unirradiated control rod fuel element for the alignment checks. This will enable us to check two control rods at a time. The core shroud may also be heated to assure it does not twist or otherwise move out of tolerance.

Control Rods Tests

After loading the control rods in the critical experiment, but before continuing with loading, the following control rod tests shall be completed.

1. Magnet Release Times
2. Rod Insertion Times
3. Minimum magnet currents for holding, withdrawal, and pickup.
4. Rod drive times from lower limit to upper limit and upper to lower.

Critical Experiment

Utilizing where appropriate the five instrumentation channels a plot of 1/m versus the mass of the fuel in the core can be extrapolated to 1/m = 0 to give the critical mass of the core. The following is the <u>anticipated</u> loading sequence in an attempt to maintain a symmetrical core. Actual loading shall be based on results from 1/m plots and may be more conservative. Each fuel loading shall be less than or equal to 50% of the difference between the amount of fuel previously loaded into the core and the critical mass predicted by the 1/m curves with the possible exception of the last few loadings of single elements. Please refer to the attached Figure 1 and Table 1 from our LEU Fuel Loading . It indicates the sequence used to load LEU #1 core.

After the core is loaded, an approach to critical using 1/m vs rod height shall be completed. Following control rod calibrations, K_{xs} can be recalculated using a new rod worth information.

The above core was measured to have approximately 1.5% K_{xs}. Reactivity shimming can be accomplished by using fuel elements or inserting graphite elements if necessary. Data is available to estimate the worth of various grid locations. We anticipate loading additional fuel or graphite to obtain between 2.0 and 2.5% K_{xs}.

Control Rod Calibration

The following methods may be employed to complete control rod calibrations at low power estimated at a nominal 10w for rod drops; less than 10w for sub critical multiplication; and less than 1 kw for positive period measurements.

Figure 1. LEU CORE

TC

	A	B	C	D	E	
1	OHF 008 5a	OH-002 5	OH-003 3	OH-004 11	OHF 009 5a	
2	OH-006 9	OHC-001 SHIM SAFETY #1 1	OH-007 2	OHC-002 SHIM SAFETY #2 1	OH-008 7	Su
3	OH-009 4	OH-010 2	CENTRAL IRRADIATION FACILITY	OH-011 2	OH-012 4	BP
4	OH-013 8	OHC-004 REG. ROD 1	OH-014 3	OHC-003 SHIM SAFETY #3 1	OH-015 10	
5	OHF 006 5a	OH-017 12	OH-018 5	OH-019 6	OHF 007 5a	
6	OHF 001 5a	OHF 002 5a	OHF 003 5a	OHF 004 5a	OHF 005 5a	

PuBe

Proposed LEU Core for the OSURR
With An Estimated Excess Reactivity of 1.5% $\Delta K/K$

(X's Denote Filled Grid Plate Positions)

Loading Sequence	Grams	Loading Sequence	Grams
1	499.53	7	2697.64
2	1099.06	8	2897.49
3	1498.74	9	3097.32
4	1898.43	10	3297.15
5	2297.97	11	3496.91
5a	2297.97 + Fillers	12	3696.73
6	2497.81		

TABLE 1

OSURR Core Loading

Loading Number	Positions Loaded	Total U-235 Mass (grams)
0	source, CIF	0.00
1	2B, 2D, 4B, 4D	499.53
2	2C, 3B, 3D	1099.06
3	1C, 4C	1498.74
4	3A, 3E	1898.43
5	1B, 5C	2297.97
5a	filler plugs	2297.97
6	5D	2497.81
7	2E	2697.64
8	4A	2897.49
9	2A	3097.32
10	4E	3297.15
11	1D	3496.91
12	5B	3696.73

Rod Drop - Reactivity determination for full length of Shim Safety rods 1, 2, and 3 and for the minimum critical positions for the same rods.

Subcritical Multiplication - Reactivity determination for the lower portions of Shim Safety rods 1, 2, and 3.

Positive Period - Calibration of the entire length of the regulating rod and the upper portions of Shim Safety Rods 1, 2, and 3.

Power Calibration and Temperature Coefficient of Reactivity

A determination of reactor power is to be completed by measuring the increase in reactor pool water temperature due to the operation of the reactor. The reactor will be operated at 10 kw as indicated on the Linear Level Monitoring Channel and reactor pool heat-up rate shall be monitored. Initial pool temperature shall be ambient and reactor operation shall allow for a pool temperature rise of at least 5°F. From this heat-up rate the power is determined. A check of the power at 50 kw will also be made by measuring the heat up rate.

Moderator Temperature Coefficient of Reactivity

To determine the moderator temperature coefficient of reactivity the following measurements will be made:

1. Long (slow) period (100-300 sec.) starting at low power and continuing to ≈ 10KW, making the following measurements:

   a. Reactivity measurement at low power from doubling time measurement.

   b. Reactivity measurement at high power from doubling time measurement.

   c. Moderator temperature change above the core.

2. Short (fast) period (10-12 sec.) starting at low power and continuing to ≈ 10 KW, making the same measurements.

Below is a description of how this method was used for the current LEU core.

The reactivity change for the transient of the long, slow period was 0.046%. It was 0.074% at the start and 0.028% at the finish. The moderator temperature change was 4.2°C. It is assumed that the observed reactivity change of 0.046% was due to both moderator and fuel heating.

The reactivity change for the transient of the short, fast period was 0.02%. It was 0.30% at the start and 0.28% at the finish. There was no moderator temperature change measured during this short period. The reactivity change was therefore only due to fuel heating.

- 0.46% Moderator + Fuel Feedback
- - 0.20% Fuel Feedback only
- 0.26% Moderator Feedback only

$$- 0.26\%/4.2°C = - 6.19 \times 10^{-5}\ \Delta k/k/°C$$

This meets Technical Specification 3.1.1(5) which required a negative moderator temperature coefficient of reactivity with an absolute value of at least $2 \times 10^{-5}\ \Delta k/k/°C$.

Void Coefficient of Reactivity

The method used for measuring the moderator void coefficient of reactivity at the OSURR requires plastic "void boxes" fabricated to fit into the coolant flow channels in a reactor fuel element. One void box, with the internal volume air-filled, is required. An exact duplicate box, with water-filled internals, is also required.

The critical reg rod position with the air-filled box inserted into a particular fuel element and the position with the water-filled box inserted are compared. The difference in critical rod height is due to the reactivity worth of the air void inside the box. This reactivity worth can be determined from the rod worth data compiled in the rod calibration experiment. This, in turn, gives the moderator void coefficient (dk/k/1% void) when the air volume is ratioed to the total core floodable volume.

Radiation Monitoring for the Power Increase

1. Ar-41 - In previous correspondence with the NRC (attached) we have committed to extensive Ar-41 monitoring. This will begin during the power calibration at 10 KW after reloading the core. At the suggestion of ORS we will also monitor at 50 KW in addition to the already planned 10, 100, 250, and 500 KW levels.

2. Direct radiation monitoring will occur at each power level monitored for Ar-41 production. At least the following locations shall be monitored:

    1. Pool top including over CIF and Dry Tubes
    2. Beam Ports
    3. Thermal Column Area
    4. Rabbit Area
    5. Demineralizer Area

References

Initial Calibration of Standard Core No. 1 of The Ohio State University Reactor January 1966.

Recalibration of Standard Core No. 1 August 1975.

1981 Calibration Report for the OSURR.

Critical Experiment, Approved by the Advisory Committee on Reactor Operations April 13, 1965.

Neutronic Scoping Calculations for OSURR Core Design with Standardized U_3Si_2 Fuel Plates, 1986.

Standard Core Catalog, OSU Reactor Core Fuel Reactivity Worth, 1964.

Fuel Load and Start-up Planning Document for the LEU Core, 1988.

Initial Loading and Testing of Low-Enrichment Uranium Fuel in The Ohio State University Research Reactor, October 1989.

Testing to be completed per letter to NRC 2/28/90

Heating, Ventilation and Air Conditioning System (HVAC)

As a part of the overall startup testing program for operation at 500 KW, a plan to determine Ar-41 concentrations in the restricted area during various operating conditions shall be implemented. The measurements will be made using a shielded volume air sampling detection system calibrated for Ar-41. The system currently in use is calibrated annually and typically indicates MPC = 10 cps. Figure 6.1 on page 143 of the SAR indicates that the equilibrium concentration for Ar-41 inside the building is reached in about four hours. Measurements of air exiting the building at the exhaust fan will be made to verify when equilibrium has been reached. After this, other locations in the building shall be monitored. Furnace fans shall be running during all tests. The testing program to determine Ar-41 concentrations is described below.

Tests shall be completed at least for four power levels 10 KW, 100 KW, 250 KW and 500 KW. Three different operating conditions shall be investigated. These are: Ar-41 production from the reactor pool alone, Ar-41 production from the reactor pool with the rabbit running continuously, and Ar-41 production from the reactor pool including a puff release from the rabbit. A total of twelve locations may be sampled. These are:

1. Room 100, Reactor Bay Vent Fan
2. Room 100, Reactor Bay Reactor Pool Top Catwalk
3. Room 100, Reactor Bay BSF Pool Top Catwalk
4. Room 100, Reactor Bay Thermal Column Area
5. Room 100, Reactor Bay Beam Port Area
6. Room 103, Office
7. Room 103A, Office
8. Room 109, Counting Room
9. Room 104, Office
10. Room 201, Conference Room
11. Room 205, Reactor Control Room
12. Room 209, Office

It is anticipated that for the puff release, only locations 1, 7, and 11 will be sampled since the concentration of Ar-41 should peak and then decrease back to equilibrium not allowing time to sample all locations.

After measurements are completed for each operating condition, the reactor shall be shut down. Additional measurements at all locations shall be completed after shutdown to determine the purge rate. This will facilitate a determination of how long to monitor after shutdown during normal operations.

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From these measurements we shall determine: 1. if any areas need to be posted as airborne radioactivity areas per 20.203(d); 2. what areas reach the highest concentrations of Ar-41, and whether these are typically occupied; and 3. how long we can operate at various powers and conditions while maintaining doses ALARA per 20.103(b)(2).

Table 2 shows the proposed plan for determination of Ar-41 concentrations during start up testing. After this testing is completed, regular monitoring during operations will be done using the effluent monitor.

Table 2. Proposed Plan for Determination of Ar 41 Concentrations

Power Level	Operating Condition	Sample Hour	Sample Location
All (10,100,250,500 KW)	Pool Top	1,2,3,4	Vent
All	Pool Top	4-5	All
Shutdown @ 5 hours	Pool Top	6,8	All
All	Pool Top + Rabbit	1,2,3,4	Vent
All	Pool Top + Rabbit	4-5	All
Shutdown @ 5 hours	Pool Top + Rabbit	6,8	All
All	Pool Top + Puff	1,2,3,4	Vent
All	Pool Top + Puff	4-5	1,7,11
Shutdown @ 5 hours	Pool Top + Puff	6,8	All