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March 8, 1990

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Reference: Oregon State University TRIGA Reactor (OSTR); Technical Specification 6.7.d; License No. R-106; Docket No. 50-243

Subject: Startup Report in Response to Technical Specification 6.7.d and the Amendment of OSTR License No. R-106 Authorizing an Increase in the Maximum Steady-State Power From 1 MW to 1.1 MW

Gentlemen:

In accordance with the reporting provisions contained in OSTR Technical Specification 6.7.d, Oregon State University is submitting the attached startup report. The report deals with measurements of reactor operating parameters and reactor operating characteristics following an authorized increase in maximum steady-state power from 1 MW to 1.1 MW. The report has been reviewed and approved by the OSTR Reactor Operations Committee.

We trust that the report contains all necessary information. However, should there be questions, please let me know.

Yours sincerely,

A. G. Johnson
Director

AGJ/ef

cc: Non-Power Reactor, Decommissioning, and Environmental Projects Directorate, USNRC, Washington, DC. Attn: Mr. Al Adams, Project Manager
Regional Administrator, USNRC Region V
Oregon Department of Energy, Attn: Mr. Harry Moomey
T. V. Anderson, Reactor Supervisor
S. E. Binney, Chairman ROC
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D. S. Pratt, Health Physicist

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OREGON STATE UNIVERSITY TRIGA REACTOR (OSTR)

License No. R-106; Docket No. 50-243

STARTUP REPORT FOLLOWING A LICENSE AMENDMENT INCREASING THE OSTR MAXIMUM STEADY-STATE POWER FROM 1.0 MW TO 1.1 MW

Introduction

On January 1, 1990 the Oregon State University TRIGA Reactor (OSTR) received a license amendment increasing the authorized maximum steady-state power from 1000 kW to 1100 kW. As a result of this amendment, the reporting provisions of Technical Specification 6.7.d became effective and a report is required which describes certain measured values and other reactor characteristics outlined in the referenced specification. In order to comply with this requirement, a series of operational measurements were made on January 16 and February 5, 1990 with the OSTR operating at power levels up to 1060 kW. The measurements were designed to provide data which, through reasonable extrapolation, would allow a determination of important operating characteristics at the maximum licensed power level of 1100 kW.

The purpose of this report is to describe the major operational measurements which were made and to formally submit the measured values as well as the extrapolated data. The report is also intended to confirm that operation of the OSTR at steady-state power levels up to and including 1100 kW introduces no significant changes in key reactor operating characteristics and therefore is, as predicted, safe in every respect.

Startup Plan

The startup plan for determining key reactor operating characteristics at 1100 kW is described in detail in Attachment A. Part 1 of step A did not need to be completed because a reactor power calibration had recently been performed on December 18, 1989. Step B was performed on January 16, 1990 and Step C was performed on February 5, 1990. During Steps B and C, the maximum reactor power was intentionally limited to 1060 kW and the data were extrapolated up to 1100 kW. This was done to ensure that the new licensed power limit of 1100 kW was not exceeded, even by the small normal feedback of

the control mechanism. In keeping with this policy, the safety channel and the percent power channel each had their scram setpoint adjusted to 1090 kW, and the reactor power level was then limited to approximately 1060 kW.

Results

Results of Step A - Power Calibration and Scram Set Point Adjustment

Because a timely reactor power calibration was completed on December 18, 1989, it was not necessary to perform another such calibration prior to beginning the startup plan. However, scram set points for both the percent power and safety channels were adjusted to 1.09 MW (1090 kW) as part of the preoperational preparation.

Results of Step B - Short Duration Stepwise Power Increase to 1.06 MW

The purpose of this operation was to observe increases in fuel element temperature as power was gradually increased beyond 1000 kW. The ultimate objective was to establish that the maximum fuel temperature at 1100 kW would be well within the Limiting Safety System Setting (LSSS). The results, shown graphically in Fig. 1, indicated that the maximum fuel temperature at 1100 kW was 392 ± 2°C. This is well below the LSSS of 510 °C.

Results of Step C - Longer Term Operation at a Power Level of 1.06 MW

The purpose of this three hour operation at 1060 kW was to allow the performance of a comprehensive health physics radiation survey around the reactor and to provide adequate time to measure the reactor's normal operating parameters over a longer interval.

The results of the reactor radiation survey are included as Attachment B. A review of the data shows that most of the radiation dose rates did not significantly increase as the power went from 1000 kW to 1060 kW, and, therefore, a meaningful extrapolation to 1100 kW cannot be made. However, it seems clear that the dose rates at the new maximum steady state power level will be very similar to those at 1000 kW.

In reviewing the results of the radiation survey, a slight increase in the gamma dose rate will be noticed at the Beam Port #4 access door; however, it can also be seen that the increase occurs as a function of time. Typically, such an increase is not directly related to the reactor power

level, but instead is due to the buildup of ^{41}Ar in this beam tube. Even if the maximum 1060 kW dose rate is extrapolated to 1100 kW, the result is a dose rate of only 3.4 mrem/h at the 1100 kW power level.

The radiation survey results also show that the dose rate at the argon ventilation system manifold decreased as a function of time even when the reactor power increased to 1060 kW. The reason for this lack of correlation between reactor power and dose rate at the argon manifold, and the influence of time, is described in the remainder of this paragraph. The OSTR's rotating rack is purged with nitrogen gas to reduce argon-41 production in this irradiation facility. The nitrogen purge for the rotating rack is normally started each morning as part of the reactor startup process, and it takes a certain amount of time before the rotating rack housing is fully purged of air and filled with nitrogen. Therefore, argon-41 production in the rotating rack and the corresponding dose rate at the argon ventilation system manifold are initially higher, but decrease over time as the nitrogen replaces the air in the rotating rack.

Finally, the largest dose rate measured during the reactor survey continued to be directly over the top of the reactor tank at the level of the (metal grating) tank cover. During the current survey, the dose rate at the grating level initially increased from 123 mrem/h to 132 mrem/h, and then decreased to 125 mrem/h after the reactor had operated for a few hours. In this case, it is possible that the small fluctuation in dose rate was due to a modest reduction over time in the evolution rate of activated air components (e.g., ^{16}N and ^{41}Ar) from the reactor tank water. An extrapolation of the 132 mrem/h dose rate at the grating level on the reactor top indicates that the maximum dose rate at this position at 1100 kW would be approximately 138 mrem/h. At 1000 kW, the dose rate at this location is approximately 100 mrem/hr.

With respect to the measured dose rates over the top of the reactor, the subject area has always been treated as a high radiation area when the reactor is operating at 1 MW, even though the quoted dose rates exist <u>only</u> at the grating level and <u>do</u> <u>not</u> create a whole body dose rate to anyone standing on the grate. Nevertheless, the subject location has highly visible postings as a high radiation area; appropriate audible and visual alarms to alert personnel entering the area; similar audible and visual alarms for management notification of entry into the area; visual surveillance by the OSTR staff over personnel in the area; strict physical access control measures designed

to meet radiation protection and physical security requirements; and OSTR administrative procedures limiting occupational stay time on the reactor top. Therefore, we feel that no further restrictions are necessary to accommodate the slightly higher dose rates which will exist over the top of the reactor as the reactor power increases from 1000 kW to 1100 kW.

The results of other routine health physics measurements such as air monitoring and reactor water radioactivity measurements remained unchanged from values typically observed at 1000 kW.

With respect to reactor operational parameters measured over the three hour run, there is very little new information to report. The maximum fuel element temperature during the three-hour run was about 4°C lower than that measured during Step B, and only fluctuated by 1°C. Bulk water temperature in the reactor tank increased only by 1°C during the three hours, indicating that the reactor cooling system was more than adequate for the increased power. The extra reactivity required to increase reactor power from 1000 kW to 1060 kW was 6 cents. This indicates a power coefficient at this power level of 0.1 cent per kW. At low power levels this coefficient is about 0.4 cent per kW and an overall average is about 0.25 cent per kW. A total of 480 individual readings of reactor operating values were taken during the three hours at 1060 kW. None of the readings provided data which were considered unusual or unexpected, and apart from the minor differences mentioned above, no operational characteristics fell outside the range of values normally observed at 1000 kW.

Conclusions

1. An evaluation of OSTR performance at 1060 kW and subsequent extrapolations to 1100 kW shows good consistency with design predictions and specifications for operation.
2. A reassessment of the safety analysis submitted with the application for license amendment to increase the OSTR's maximum steady-state power to 1100 kW is not necessary because measurements of the reactor's operating characteristics while at power levels between 1000 kW and 1100 kW are consistent with data in the referenced safety analysis.

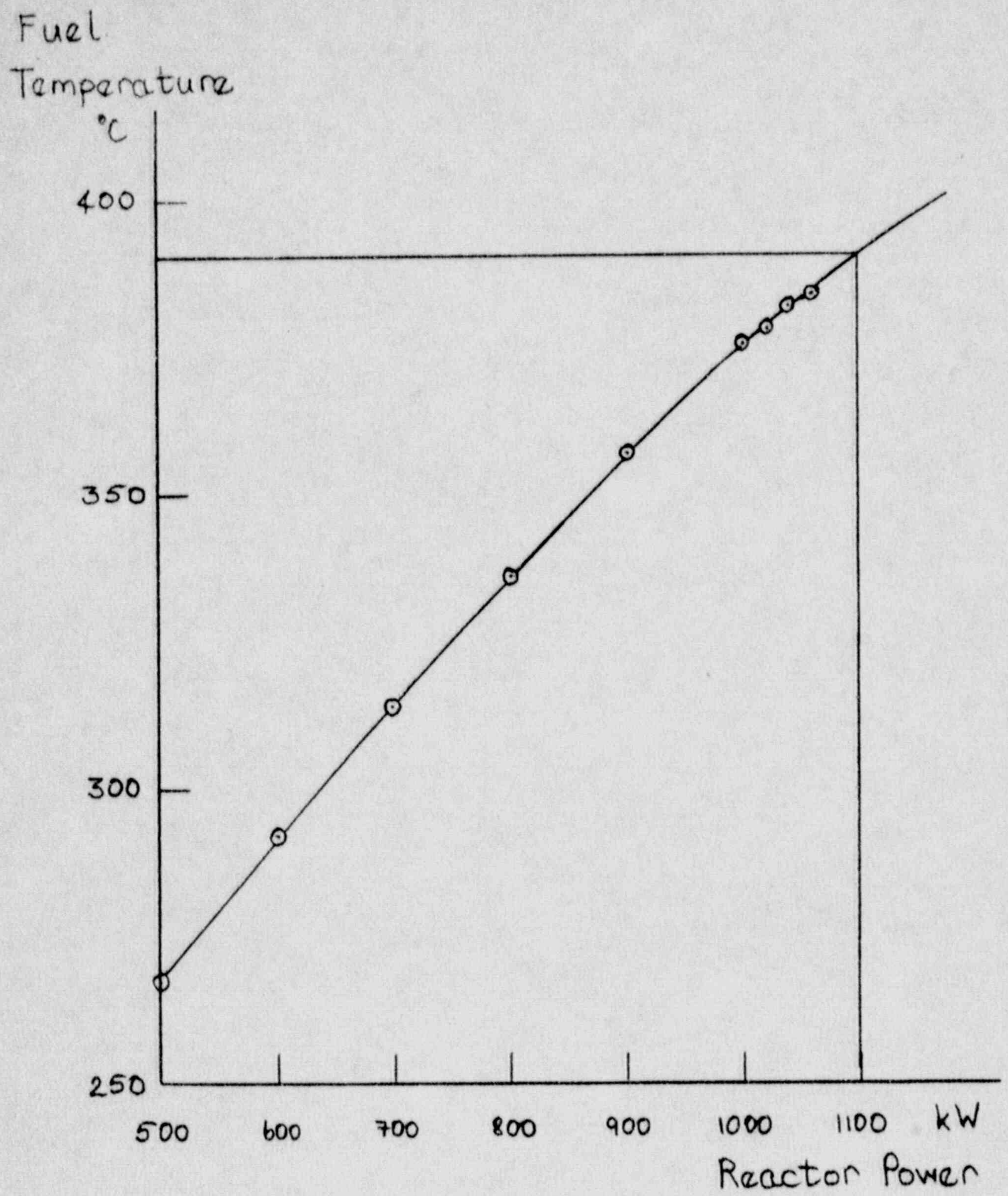

Figure 1. Maximum Fuel Element Temperature

Attachment A

STARTUP PLAN REQUIRED BY THE INCREASE IN THE OSTR LICENSED POWER TO 1.1 MW

The following procedure shall be followed as the startup plan to meet the requirements of Technical Specification 6.7.d.

A. Power Calibration and Scram Set Point Adjustment

1. If a recent (within 3 months) reactor power calibration has not been performed, calibrate the reactor power using the standard procedures in OSTROP 8. This will ensure that all of the reactor power channels are accurate within the limits given in OSTROP 8.0 (1.5%).

2. Adjust the reactor percent power and safety channel scram settings to 1.09 MW.

3. Proceed to Step B.

B. Short Duration Stepwise Power Increase to 1.06 MW

The purpose of this short duration power increase to 1.06 MW is to measure and record key reactor parameters (in particular fuel element temperature) prior to a longer run at 1.06 MW.

1. Operate the reactor at a power level of 1.0 MW and record operational reactor data as usual on an OSTR Control Room Log Sheet.

2. Increase the reactor power in 0.02 MW steps up to 1.06 MW, stopping at each step to record the usual reactor operational parameters on the OSTR Control Room Log Sheet. Stay at each power level only long enough to be able to record a stable fuel element temperature. If at any point during the stepwise increase in reactor power any of the reactor's operational parameters, or any of the area radiation monitors, fall outside the range of values to be expected based on past experience, shut down the reactor. In particular, if the fuel temperature exceeds 400°C shut down the reactor immediately. Although this temperature is well within the limiting safety system setting of 510°C it is higher than the temperature which should be observed at 1.06 MW.

3. If not proceeding to Step C, shut down the reactor and reset the percent power and safety channel scram settings to 1.06 MW.

C. Longer Term Operation at a Power Level of 1.06 MW

The purpose of the longer term operation at 1.06 MW is to measure key reactor parameters over at least three hours, as well as to enable a comprehensive health physics radiation survey to be conducted. This longer term operation can continue from Step B if operationally convenient, or it can be performed at a separate time.

1. If the longer term operation is not being performed as a continuation of

Step B, adjust the reactor percent power and safety channel scram settings to 1.09 MW and bring the reactor up to a power level of 1.06 MW.

2. About every 15 minutes, record all of the reactor's operational parameters called for on the OSTR Control Room Log Sheet. Again, if at any time abnormal readings are noticed, scram the reactor.

3. Health Physics staff will conduct a comprehensive radiological survey while the reactor is at 1.06 MW.

4. At the end of three hours, shut down the reactor and reset the percent power and safety channel scram settings to 1.06 MW.

D. Data Analysis and Report

1. Evaluate and analyze all of the data recorded. This will necesarily include an extrapolation of certain data (e.g., fuel temperature and dose rates) to determine expected values at a power level of 1.1 MW.

2. Write up the report required by Technical Specification 6.7.d and then submit this report to the USNRC after approval by the ROC. The report must be submitted to the USNRC within 90 days of the license amendment.

Attachment B

OREGON STATE UNIVERSITY - RADIATION CENTER
RECORD OF RADIATION SURVEY

Special Survey No. 1250

Building or Area: RADIATION CENTER   Room #: REACTOR BAY   Date: FEB. 5, 1990   Time: 945 - 1322

Description: RADIATION SURVEY OF REACTOR BAY @ 1 MW & @ 1.06 MW

Item or Area	Instrument Used and Serial No.	Distance	GAMMA @1 MW 945-1010 mrem/hr	NEUTRON @1 MW 945-1010 mrem/hr	TOTAL @1 MW 945-1010 mrem/hr	GAMMA @1.06 MW 1035-1050 mrem/hr	NEUTRON @1.06 MW 1035-1050 mrem/hr	TOTAL @1.06 MW 1035-1050 mrem/hr	GAMMA mrem/hr @1.06 MW 1310-1322	Neutrons mrem/hr @1.06 MW 1310-1322	Total mrem/hr @1.06 MW 1310-1322
THERMAL COLUMN ℄ @ FLOOR	γ: BICRON IC #A729R n: TN NEMO #199	CONTACT	8.0	0.05	8.05	8.0	0.05	8.05	8	0.06	8.06
BEAM PORT #4 ACCESS DOOR	↓	↓	0.8	0.5	~~0.12~~ 1.3	1.0	0.55	1.55	2.1	0.46	2.56
ARGON VENT MANIFOLD	↓	↓	3.6	0.08	3.68	1.0	0.06	1.06	0.5	0.1	0.6
BEAM PORT #1 SHUTTER AREA	↓	↓	0.7	0.2	0.9	0.7	0.2	0.9	0.7	0.1	0.8
2nd DECK NW BASE	↓	↓	50	13	63	50	13	63	50	14	64
2nd DECK ~~NE~~ SW BASE	↓	↓	45	15	60	45	15	60	45	15	60
2nd DECK SE BASE	↓	↓	40	10	50	38	14	52	42	12	54
2nd DECK @ DOOR TO 2nd FLR CORE	↓	↓	0.1	0.04	0.14	0.1	0.02	0.12	0.1	0.05	0.15
REACTOR TOP @ GRATE	↓	↓	95	28	~~133~~ 123	100	32	132	95	30	125
REACTOR TOP TOP OF CAM	↓	↓	3.6	0.4	4.0	3.1	0.32	3.42	3.1	0.2	3.3

NOTES:

(1) Make all radiation-field measurements involving dose rates >1 mrem/hr using an instrument which reads directly in mR/hr or mrem/hr. However, for local gamma fields <1 mR/hr, you may use a portable GM ratemeter (through the metal back, non-window side, of pancake probes, or through the closed window of other probes) and then convert net gamma cpm readings to mR/hr on the basis of 4000 net gamma cpm per mR/hr (or, if different, use the applicable conversion ratio marked on the instrument's calibration label).

(2) When using a portable survey instrument (e.g., a GM), direct readings from surfaces and smear results showing gross cpm values < twice background cpm (or net cpm values < background cpm) are considered to be below the lower limit of detection for such instrumentation, PROVIDED background is ≤ 100 cpm βγ.

(3) When using a portable survey instrument (e.g., a GM), if any direct reading from a surface or any smear shows a net cpm ≥ background cpm, enter the corresponding net dpm value in this column. For direct readings and smears showing a net cpm < background cpm, enter "<500 dpm" in this column, PROVIDED background is ≤ 100 cpm βγ, and the appropriate counting yield is approximately 20%. For yields significantly below 20% calculate the actual LLD.

(4) ____________(radioisotope) yield = ____________%; net dpm = net cpm/yield (expressed as a decimal).

(5) The LLD (at 95% confidence level) was ____________ net cpm = ____________ net dpm = ____________ net μCi.

(6) (All, Some) smears (did, did not) show removable radioactivity above the LLD. Exceptions: ____________

(7) Background rate was N/A cpm = N/A dpm; or 0.1 γ, 0.01 n mrem/hr. Direct radiation levels from the item(s) surveyed (were, (were not)) ≤ background radiation levels. Exceptions: 2nd DECK DOOR & EXPOSURE

Radiation Monitor: [signature]   Reviewed, Health Physicist: [signature]

USE BACK OF PAGE FOR NEEDED DIAGRAMS

Revised March 1986