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

OREGON STATE UNIVERSITY

DOCKET NO. 50-243

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 4
License No. R-106

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Oregon State University (the licensee) dated April 16, 1979, as supplemented July 11, 1979, August 17, 1979, and October 10, 1979 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

   F. Publication of notice of this amendment is not required since it does not involve a significant hazards consideration nor amendment of a license of the type described in 10 CFR Section 2.106 (a)(2).

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. R-106 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 4, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: December 18, 1979

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ATTACHMENT TO LICENSE AMENDMENT NO. 4

FACILITY OPERATING LICENSE NO. R-106

DOCKET NO. 50-243

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Pages

9 - 12

15

24

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3.3 PULSE MODE OPERATION

Applicability. This specification applies to the energy generated in the reactor as a result of a pulse insertion of reactivity.

Objective. The objective is to assure that the fuel temperature safety limit will not be exceeded.

Specification. The reactivity to be inserted for pulse operation shall be determined and limited by a mechanical block and electrical interlock on the pulse rod, such that the reactivity insertion will not exceed 2.55 dollars.

Basis. The fuel temperature rise during a pulse transient has been estimated conservatively by adiabatic models. This model accurately predicts pulse characteristics measured in an existing core of all Standard fuel. Pulse characteristics for operational mixed cores and FLIP cores thus may be estimated with confidence, relying also on information concerning prompt neutron life time and prompt temperature feedback of reactivity. These parameters have been established for mixed and full FLIP cores by calculations and have been confirmed in parts by measurement at existing facilities. In addition, the calculations rely on flux profiles and corresponding power densities which have been calculated for a variety of operational mixed and full FLIP cores in SAR I.

In this manner it is estimated conservatively that reactivity up to 2.55 dollars in operational cores will produce pulse transients with maximum fuel temperatures no greater than 950°C in FLIP fuel and 800°C in standard fuel; i.e., a safety margin of 200°C with respect to the safety limit of the fuel is maintained in either case, allowing for any uncertainties in measurements and/or calculations.

3.4 CORE CONFIGURATION LIMITATIONS

Applicability. This specification applies to mixed cores of FLIP and Standard types of fuel.

Objective. The objective is to assure that the fuel temperature safety limit will not be exceeded due to power peaking effects in a mixed core.

Specification. The FLIP fueled region in an operational core shall contain at least 80 FLIP fuel elements in a contiguous block of fuel in the central region of the reactor core. Single element positions may be left vacant or occupied by other items as specified in Section 5.2.c and 5.2.d of these Technical Specifications.

Basis. The limitation on allowable core configurations to those similar to the ones considered in Sections 3. and 5. of SAR I limits power

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peaking effects. The limitation of power peaking effects insures that the fuel temperature safety limit will not be exceeded in an operational core.

3.5 CONTROL AND SAFETY SYSTEM

3.5.1 Scram Time

Applicability. This specification applies to the time required for the scrammable control rods to be fully inserted from the instant that a safety channel variable reaches the Safety System setting.

Objective. The objective is to achieve prompt shutdown of the reactor to prevent fuel damage.

Specification. The scram time, measured from the instant the input signal reaches the value of the Safety System setting to the instant that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds.

Bases. This specification assures that the reactor will be promptly shutdown when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to assure the safety of the reactor.

3.5.2 Reactor Control System

Applicability. This specification applies to the information which must be available to the reactor operator during reactor operation.

Objective. The objective is to require that sufficient information is available to the operator to assure safe operation of the reactor.

Specification. The reactor shall not be operated in the specified mode unless the measuring channels listed in the following table are operable.

Measuring Channel	Effective Mode		
	S.S.	Pulse	S.W.
Fuel Element Temperature	X	X	X
Linear Power Level	X		X
Log Power Level	X		X
Percent Power Level	X		X
Nvt-Circuit		X	
Period-Circuit	X		
Safety Power Level	X		X

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Bases. Fuel temperature displayed at the control console gives continuous information on this paramater which has a specified safety limit. The power level monitors assure that the reactor power level is adequately monitored for both steady-state and pulsing modes of operation. The specifications on reactor power level indication are included in this section, since the power level is related to the fuel temperature. The specifications on reactor period are included to ensure that safety limits are not exceeded.

3.5.3 Reactor Safety System

Applicability. This specification applies to the reactor safety system channels.

Objective. The objective is to specify the minimum number of reactor safety system channels that must be operable for safe operation.

Specification. The reactor shall not be operated unless the safety channels described in Table I and interlocks described in Table II are operable.

Bases. The fuel temperature, power level, and period scrams provide protection to assure that the reactor can be shut down before the safety limit on the fuel element temperature will be exceeded. The manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs. The preset timer insures that the reactor power level will reduce to a low level after pulsing. The high voltage scram insures that the power measuring channels operate within their intended range as required for proper functioning of all power level scrams.

The interlock to prevent startup of the reactor at count rates less than 2 cps assures that start up is not initiated unless a reliable indication of the neutron flux level in the reactor core is available.

The interlock to prevent the initiation of a pulse above 1 kw is to assure that the magnitude of the pulse will not cause the fuel element temperature safety limits to be exceeded. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted is to prevent pulsing the reactor in the steady-state mode. The interlock to prevent withdrawal of the shim, safety or regulating rod in the pulse mode is to prevent the reactor from being pulsed while on a positive period. The interlock to prevent simultaneous withdrawal of two control rods is to limit reactivity insertion rate from the standard control rods. The interlock on the transient rod cylinder position prevents the pulse insertion of more than 2.55 dollars of reactivity during the pulse or square-wave mode.

TABLE I

Minimum Reactor Safety Channels

Safety Channel	Function	Effective Mode		
		S.S.	Pulse	S.W.
Fuel Element Temperature	SCRAM @ 510°C	X	X	X
Safety Power Level	SCRAM @ 110%*	X		X
Percent Power Level	SCRAM @ 110%*	X		X
Console Scram Button	SCRAM	X	X	X
Wide-Range Log Power Level	SCRAM @ period no less than 3 sec.	X		
Preset Timer	Transient rod SCRAM @ 15 sec or less after pulse		X	
High Voltage	SCRAM @ 25% of nominal operating voltage	X	X	X

*For the purpose of testing the full power safety channels, the reactor may be operated with the Linear Power Level and the Percent Power Level Scram setpoints set not greater than 120% of rated steady state power during the testing period.

TABLE II

Minimum Interlocks

Interlock	Function	Effective Mode		
		S.S.	Pulse	S.W.
Wide-Range Log Power Level Channel	Prevents control rod withdrawal @ less than 2 cps	X		
Transient Rod Cylinder	Prevents application of air unless fully inserted	X		
1 kw Pulse Interlock	Prevents pulsing above 1 kw		X	
Shim, Safety and Regulating Rod Drive Circuit	Prevents simultaneous withdrawal of two rods	X		X
Shim, Safety, and Regulating Rod Drive Circuit	Prevents movement of any rod except transient rod		X	
Transient Rod Cylinder Position	Prevents pulse insertion of reactivity greater than $2.55		X	X

Basis. The minimum height of 14 feet of water above the core guarantees that there is sufficient water for effective cooling of the fuel and that the radiation levels at the top of the reactor are within acceptable levels (SAR). The bulk water temperature limit is necessary, according to the reactor manufacturer, to ensure that the aluminum reactor tank maintains its integrity and is not degraded.

3.8 LIMITATIONS ON EXPERIMENTS

Applicability. This specification applies to experiments installed in the reactor and its experimental facilities.

Objective. The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications. The reactor shall not be operated unless the following conditions governing experiments exist:

a. Non-secured experiments shall have reactivity worths less than 1 dollar.

b. The reactivity worth of any single experiment will be less than 2.55 dollars.

c. Total experiment worth of all experiments will not exceed 3.00 dollars.

d. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 milligrams may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the design pressure of the container. EXCEPTION: Explosive materials not exceeding 0.014 lbs. equivalent of TNT may be irradiated in the laboratory area adjacent to the end of the OSTR tangential beamport for the purpose of neutron radiography.

e. Where the possibility exists that the failure of an experiment (except fueled experiments) under (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, or (3) possible accident conditions in the experiment, could release radioactive gases or aerosols to the reactor bay or the unrestricted area, the quantity and type of material shall be limited such that the airborne concentration of radioactivity averaged over a year will not exceed the limits of Appendix B of 10 CFR Part 20, assuming 100% of the gases or aerosols escape.

f. In calculations pursuant to d., above, the following assumptions shall be used:

increase in loading would result in an increase in power density of about 2%. Similarly, a minimum erbium content of 1.1% in an element is about 30% less than the design value. This variation would result in an increase in power density of only about 6%. An increase in local power density of 6% reduces the safety margin by, at most, 10%. The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than the value resulting from a hydrogen-to-zirconium ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad. When standard and FLIP fuel elements are used in mixed cores, visual identification of types of elements is necessary to verify correct fuel loadings.

b. A maximum uranium content of 9 wt-% in a standard TRIGA element is about 6% greater than the design value of 8.5 wt-%. Such an increase in loading would result in an increase in power density of less than 6%. An increase in local power density of 6% reduces the safety margin by, at most, 10%. The maximum hydrogen-to-zirconium ratio of 1.8 could result in a maximum stress under accident conditions in the fuel element clad about a factor of two greater than the value resulting from a hydrogen-to-zirconium ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad. When standard and FLIP fuel elements are used in mixed cores, visual identification of types of elements is necessary to verify correct fuel loadings.

5.2 REACTOR CORE

Applicability. This specification applies to the configuration of fuel and in-core experiments.

Objective. The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications.

a. The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator elements positioned in the reactor grid plate.

b. The TRIGA core assembly may consist of standard fuel elements, FLIP fuel elements, or a combination thereof (mixed core). Any operational core assembly involving FLIP fuel shall have no less than 80 FLIP fuel elements, located in a contiguous, central region.

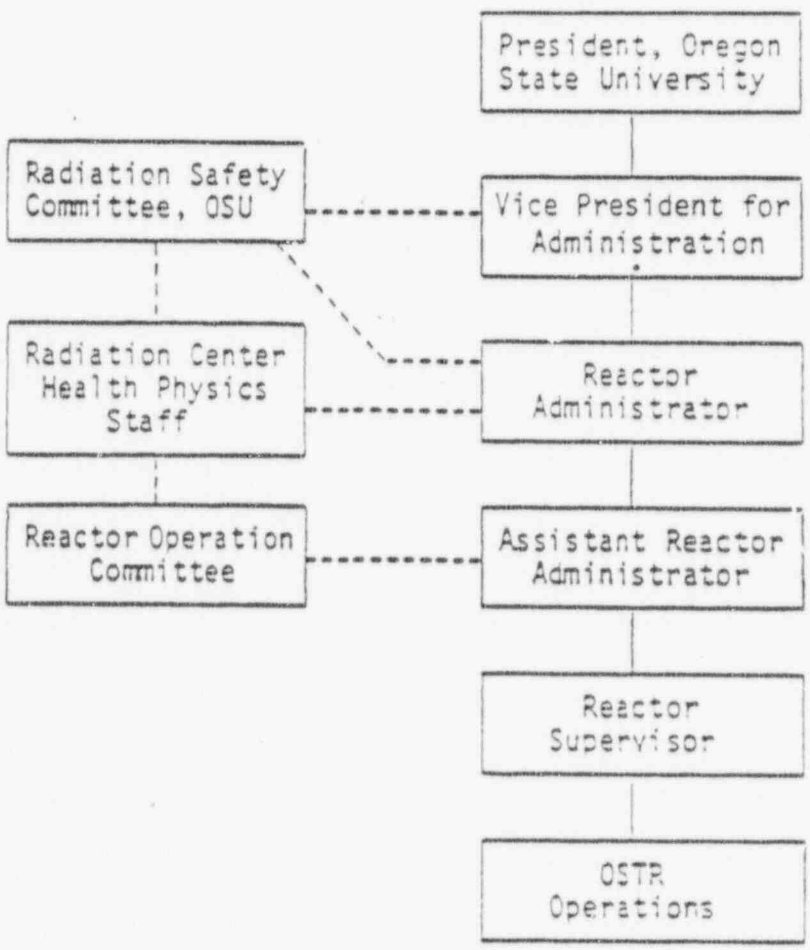
c. The fuel shall be arranged in a close-packed configuration except for single element positions occupied by in-core experiments, experimental facilities, graphite dummies, aluminum dummies, stainless steel dummies, control rods, and startup sources.

6. ADMINISTRATIVE CONTROLS

6.1 ORGANIZATION

a. The facility shall be under the direct control of the Reactor Administrator or a licensed senior operator designated by him to be in direct control. The Reactor Administrator shall be responsible to the Vice President for Administration of Oregon State University for safe operation and maintenance of the reactor and its associated equipment. The Reactor Administrator or his appointee shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and the requirements of the Reactor Operation Committee. He shall enforce rules for the protection of personnel against radiation.

b. The safety of operation of the OSTR shall be related to the University Administration as shown in the following chart:



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2. Those events reported as required by Sections 6.7.a.2 through 6.7.a.8.

c. A report within 30 days in writing to the NRC, Region V, Office of Inspection and Enforcement, with copies to the NRC, Director, Office of Inspection and Enforcement.

1. Any significant variation of measured values from a corresponding predicted or previously measured value of safety-connected operating characteristics occurring during operation of the reactor;

2. Any significant change in the transient or accident analyses as described in the Safety Analysis Report;

3. Any changes in facility organization or personnel; and

4. Any observed inadequacies in the implementation of administrative or procedural controls.

d. A report within 90 days after completion of starting testing of the reactor (in writing to the NRC, Region V, Office of Inspection and Enforcement and copies to NRC, Director, Office of Inspection and Enforcement) upon receipt of a new facility license, or an amendment to the license authorizing an increase in reactor power level, describing the measured values of the operating conditions or characteristics of the reactor under the new conditions including:

1. An evaluation of facility performance to date in com arison with design predictions and specifications.

2. A reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analysis.

e. An annual report within 75 days following the 30th of June of each year (in writing to the NRC, Region V, Office of Inspection and Enforcement, and copies to the NRC, Director, Office of Inspection and Enforcement).

1. A brief summary of operating experience including experiments performed and changes in facility design, performance characteristics and operating procedures related to reactor safety occurring during the reporting period, and results of surveillance test and inspections.

2. A tabulation showing the energy generated by the reactor (in megawatt-hours), hours reactor was critical, and the cumulative total energy output since initial criticality.

3. The number of emergency shutdowns and inadvertent scrams, including reasons therefore.

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4. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required.