June 18, 2008

Dr. William A. Baeslack III
Dean, College of Engineering
142A Hitchcock Hall
Ohio State University
2070 Neil Avenue
Columbus, OH 43210

SUBJECT: ISSUANCE OF RENEWED FACILITY LICENSE NO. R-75 FOR THE OHIO STATE UNIVERSITY RESEARCH REACTOR (TAC NO. MA7724)

Dear Dr. Baeslack:

The U.S. Nuclear Regulatory Commission (NRC) has issued renewed Facility License No. R-75 for the Ohio State University Research Reactor in response to the application for license renewal dated December 15, 1999, as supplemented on August 21, 2002; August 18, 2005; July 26, 2006; May 22, May 31, September 4, and September 28, 2007; and February 29, 2008. The renewed license is effective as of the date of this letter and shall expire at midnight, 20 years from the date of issuance, unless sooner terminated.

Enclosed with the renewed license is a copy of the Notice of Issuance of Renewed Facility License No. R-75 that is being sent to the Office of the Federal Register for publication and a copy of the safety evaluation report associated with the license renewal. The NRC sent the Notice of Issuance of Environmental Assessment and the environmental assessment associated with the license renewal to you under separate cover dated April 7, 2008. If you have any questions, please contact me at 301-415-1631.

Sincerely,

/RA/

Daniel E. Hughes, Project Manager
Research and Test Reactors Branch A
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-150

Enclosures:
1. Renewed Facility License No. R-75
2. Safety Evaluation Report
3. Notice of Issuance of Renewed Facility License No. R-75

cc w/enclosures: See next page

cc:

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Test, Research, and Training
  Reactor Newsletter
University of Florida
202 Nuclear Sciences Center
Gainesville, FL 32611

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cc w/enclosures: See next page

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Office	PRTA:PM	PRTA:LA	PFPB:BC	PRTA:PM	OGC	PRTA:BC	DPR:DD	DPR:D	NRR:D
Name	WKennedy wbk	EHylton egh	Input Provided	DHughes deh	SUttal NLO su	DCollins dsc	HNieh hn	MCase mc	ELeeds el
Date	4/10/2008	4/10/2008	4/02/2008	4/21/2008	5/28/08	5/30/08	6/5/08	6/18/08	6/18/08

<u>UNITED STATES NUCLEAR REGULATORY COMMISSION</u>
<u>DOCKET NO. 50-150</u>
<u>THE OHIO STATE UNIVERSITY</u>
<u>RENEWED FACILITY LICENSE</u>

License No. R-75

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

   A. The application for renewal of Facility License No. R-75 filed by the Ohio State University (the licensee) dated December 15, 1999, as supplemented on August 21, 2002; August 18, 2005; July 26, 2006; May 22, May 31, September 4, and September 28, 2007; and February 29, 2008 (the application), 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 Title 10, Chapter 1, of the Code of Federal Regulations;

   B. Construction of the Ohio State University Research Reactor (the facility) was completed in substantial conformity with Construction Permit No. CPRR-49 dated February 3, 1960, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance that (i) the activities authorized by this renewed license can be conducted at the designated location without endangering the health and safety of the public, and (ii) such activities will be conducted in compliance with the rules and regulations of the Commission;

   D. The licensee is technically and financially qualified to engage in the activities authorized by this license in accordance with the rules and regulations of the Commission;

   E. The licensee is a nonprofit educational institution and will use the reactor for the conduct of educational activities, and therefore the licensee is exempt from the financial protection requirement of subsection 170a of the Act;

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

   G. The issuance of this license is in accordance with 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," of the Commission's regulations and all applicable requirements; and

   H. The receipt, possession and use of byproduct and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30 and 10 CFR Part 70.

2. Facility License No. R-75 is hereby renewed in its entirety to read as follows:

   A. This license applies to the Ohio State University Research Reactor (the reactor) that is owned by the Ohio State University (the licensee), located on the Ohio State University's campus in Columbus, Ohio, and described in the licensee's application, as supplemented.

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Ohio State University:

1. Pursuant to subsection 104c of the Act, and Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the Code of Federal Regulations (10 CFR Part 50), to possess, use, and operate the reactor as a utilization facility at the designated location in Columbus, Ohio.

2. Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use in connection with operation of the facility:

    a. up to 5.2 kilograms of contained uranium-235 at enrichments less than 20 percent;

    b. up to 30 grams of highly enriched, contained uranium-235 in the form of fission chamber linings, foil targets, and other research applications;

    c. up to 80 grams of plutonium contained in encapsulated plutonium-beryllium sources; and

    d. to possess and use, but not to separate such special nuclear material as may be produced by operation of the reactor.

3. Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to possess and use, but not to separate, except for byproduct material produced in non-fueled experiments, such byproduct material as may be produced by operation of the reactor.

C. This license shall be deemed to contain and is subject to the conditions specified in Parts 20, 30, 50, 51, 55, 70, and 73 of the Commission's regulations; is subject to all applicable provisions of the Act and rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

1. The licensee is authorized to operate the reactor at steady-state power levels up to a maximum of 500 kilowatts (thermal).

2. The technical specifications contained in Appendix A are hereby incorporated in the license. The licensee shall operate the reactor in accordance with the technical specifications.

D. This license is effective as of the date of issuance and shall expire at midnight twenty years from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A, Technical Specifications

Date of Issuance: June 18, 2008

APPENDIX A

TO

RENEWED FACILITY OPERATING LICENSE NO. R-75

Technical Specifications

and Bases for

The Ohio State University

Pool-Type Nuclear Reactor

Columbus, Ohio

Docket No. 50-150

June 2008

CONTENTS

1.0 INTRODUCTION

1.1 Scope

This document constitutes the technical specifications for Facility License No. R-75 and supersedes all prior technical specifications. Included are the “specifications” and the “bases” for the technical specifications. These bases, which provide the technical support for the individual technical specifications, are included for information purposes only. They are not part of the technical specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

This document was written to be in conformance with American National Standards Institute/American Nuclear Society (ANSI/ANS)-15.1-1990, “The Development of Technical Specifications for Research Reactors.” The technical specifications include definitions, safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

1.2 Application

1.2.1 Purpose

These technical specifications have been written specifically for The Ohio State University Research Reactor (OSURR).

The technical specifications represent the agreement between the licensee and the U.S. Nuclear Regulatory Commission (NRC) on administrative controls, equipment availability, and operational parameters.

Specifications are limits, equipment requirements, and administrative requirements for safe reactor operation and for dealing with abnormal situations. They are typically derived from the safety analysis report (SAR) submitted with the application for renewal of Facility License No. R-75, as supplemented. These specifications represent a comprehensive envelope for safe operation. Only those operational parameters and equipment requirements directly related to preserving that safe envelope are listed.

1.2.2 Format

The format of this document is in general accordance with ANSI/ANS-15.1-1990.

1.3 Definitions

Administrative Controls—those organizational and procedural requirements established by the NRC and/or the facility management.

ALARA—as low as is reasonably achievable.

Channel—the combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

Channel Calibration—an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the measured parameter. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip settings, and shall be deemed to include a channel test.

Channel Check—a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

Channel Test—the introduction of a signal into the channel for verification that it is operable.

Cold Clean Core—when the core is at ambient temperature and the reactivity worth of xenon is negligible.

Confinement—a closure on the overall facility which controls the movement of air into it and out of it through a controlled path.

Control Rod—a device fabricated from neutron-absorbing material which is used to establish neutron flux changes.

Control Rod Scram Time—elapsed time from the receipt of a safety signal to when a shim/safety rod is fully inserted.

Controlled Area—an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason**.**

Controls—mechanisms used to regulate the operation of the reactor.

Core—the general arrangement of fuel elements and control rods.

Critical—when the effective multiplication factor (k_{eff}) of the reactor is equal to unity.

Excess Reactivity—the amount of reactivity that would exist if all control rods were removed from the core.

Experiment—any operation, or any apparatus, device, or material, installed in or near the core or which could conceivably have a reactivity effect on the core and which itself is not a core component or experimental facility, intended to investigate non-routine reactor parameters or radiation interaction parameters of materials.

Experimental Facility—any structure or device associated with the reactor that is intended to guide, orient, position, manipulate, or otherwise facilitate completion of experiments.

Explosive Material—any material that is given an Identification of Reactivity (Stability) index of 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, “Identification System for Fire Hazards of Materials,” or is enumerated in the *Handbook for Laboratory Safety* published by the Chemical Rubber Company in 1967.

Facility—the reactor building including offices and laboratories.

Fuel Element, Blank—A core element with no fuel plates. The bottom ends of these elements are closed to minimize coolant flow bypassing core elements with fuel. Also called "Filler Fuel Element."

Fuel Element, Control Rod—a fuel element with less than the full number of plates that is capable of holding a control rod.

Fuel Element, Partial—a fuel element with the full number of plates and less than 100 percent of the nominal fuel element loading.

Fuel Element, Standard—a fuel element with the full number of plates and 100 percent of the nominal fuel element loading.

Fueled Experiment—any experiment that contains uranium-235 or uranium-233 or plutonium-239, excluding the normal reactor fuel elements.

Indicated Value—see Measured Value.

Limiting Conditions for Operation—administratively established constraints on equipment and operational characteristics that shall be adhered to during operation of the facility. These constraints are the lowest functional capability or performance level required for safe operation of the facility.

Limiting Safety System Settings—settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting shall be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

Measured Value—the value of a parameter as it appears on the output of a channel.

Movable Experiment—one for which it is intended that all or part of the experiment may be moved in relation to the core while the reactor is operating.

NRC—U.S. Nuclear Regulatory Commission.

ONB—onset of nucleate boiling.

Operable—a component or system is capable of performing its intended functions in a normal manner.

Operating—a component or system is performing its intended function.

Protective Action—the initiation of a signal or the operation of equipment within the reactor safety system in response to a variable or condition of the reactor facility having reached a specified limit.

Reactivity Worth of an Experiment—value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

Reactor—the combination of core, permanently installed experimental facilities, control rods, and connected control instrumentation.

Reactor Operating—the reactor is operating whenever it is not secured or shut down.

Reactor Operator—an individual who is licensed to manipulate the controls of the reactor in accordance with Title 10, Part 55, "Operators' Licenses," of the *Code of Federal Regulations* (10 CFR Part 55).

Reactor Safety Systems—those systems, including their associated input channels, that are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

Reactor Secured—the reactor is secured when either of the following is true:

(1) Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection.

(2) The following conditions exist:

a. All shim/safety rods are fully inserted.

b. The console key switch is in the OFF position and the key is removed from the lock.

c. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods.

d. No experiments are being moved or serviced that that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment.

Reactor Shutdown—when the reactor is subcritical by at least 1% $\Delta k/k$ in the cold clean core condition.

Regulating Rod—a low reactivity-worth control rod used primarily to maintain an intended power level.

Restricted Area—area to which access is controlled for purposes of protection of individuals from exposure to radiation and radioactive materials.

ROC—Reactor Operations Committee.

Safety Channel—a measuring or protective channel in the reactor safety system.

Safety Limits—limits on important process variables that are found to be necessary to reasonably protect the integrity of the principal physical barriers that guard against the uncontrolled release of radioactivity. The principal physical barrier is the fuel cladding.

SAR—safety analysis report.

Scram—the rapid insertion of the shim/safety rods into the reactor for the purpose of quickly shutting down the reactor.

Secured Experiment—any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected from the normal environment of the experiment or by forces that can result from credible malfunctions.

Senior Reactor Operator—an individual who is licensed to direct the activities of reactor operators. Such an individual may also operate the controls of the reactor pursuant to 10 CFR Part 55.

Shall, Should, and May—the word “shall” is used to denote a requirement; the word “should” to denote a recommendation; and the word “may” to denote permission, which is neither a requirement nor a recommendation.

Shim/Safety Rods—high-reactivity worth control rods used primarily to provide coarse reactor control. They are connected electromagnetically to their drive mechanisms and have scram capabilities.

Shutdown Margin—the shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems with the most reactive shim/safety rod and the regulating rod in the most reactive position (fully withdrawn) and that the reactor will remain subcritical without further operator action.

Startup Source—a spontaneous source of neutrons which is used to provide a channel check of the startup (fission chamber) channel and to provide neutrons for subcritical multiplication during reactor startup.

Surveillance Time Intervals— maximum allowable intervals listed as follows are to provide operational flexibility only. Established frequencies shall be maintained over the long term.

5 Year	(interval not to exceed 6 years)
Biennial	(interval not to exceed 30 months)
Annual	(interval not to exceed 15 months)
Semiannual	(interval not to exceed 7-1/2 months)
Quarterly	(interval not to exceed 4 months)
Monthly	(interval not to exceed 6 weeks)
Weekly	(interval not to exceed 10 days)
Daily	(shall be done during the same working day)

Any extension of these intervals shall be occasional and for a valid reason.

True Value—the actual value of a parameter.

Unscheduled Shutdowns—any unplanned shutdown of the reactor caused by actuation of the reactor safety systems, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation. Excluded are those shutdowns resulting from expected testing operations or planned shutdowns, whether initiated by controlled insertion of control rods or planned manual scrams.

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit

Applicability: This specification applies to the melting temperature of the aluminum fuel cladding.

Objective: The objective is to ensure that the integrity of the fuel cladding is maintained.

Specification: The reactor fuel temperature shall be less than 550 °C.

Bases: The melting temperature of aluminum is 660 °C (1220 °F). The blister threshold temperature for uranium-silicide (U_3Si_2) dispersion fuel has been measured as approximately 550 °C (ANL/RERTR/TM-10, October 1987, NUREG-1313). Because the objective of this specification is to prevent release of fission products, any fuel whose maximum temperature reaches 550 °C is to be treated as though the safety limit has been reached until shown otherwise.

2.2 Limiting Safety System Settings

Applicability: This specification applies to the following two items associated with core thermodynamics:

(1) reactor thermal power level
(2) reactor coolant inlet temperature

Objective: To ensure that the fuel cladding integrity is maintained.

Specification:

(1) Reactor safety systems settings shall initiate automatic protective action at or below an indicated reactor power of 600 kilowatts (kW).

(2) Reactor safety systems settings shall initiate automatic protective action so that core inlet water temperature shall not exceed 35 °C.

Bases: The criterion for these safety system settings is established as the fuel integrity. If the temperature of the clad is maintained below that for blister threshold, then cladding integrity is maintained. This is the case for a power level of 600 kW and a core inlet temperature of 35 °C (normal inlet temperature is approximately 20–25 °C). Section 8.4.3.2 of the SAR provides the maximum credible accident analysis. The maximum credible accident assumes steady-state operation at 600 kW and initiation of a scram at 750 kW. The maximum temperature of the cladding reaches 91 °C (SAR Section 8.4.3.3).

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Parameters

3.1.1 Reactivity

Applicability: These specifications apply to the reactivity condition of the reactor and the reactivity worths of the shim/safety rods and regulating rod under any operating conditions.

Objective: To ensure the capability for safe shutdown of the reactor and that the safety limits are not exceeded.

Specification: With the exception of operations performed solely for determination of reactor reactivity worth values, the reactor shall be operated only if the following five conditions exist:

(1) The reactor core shall be loaded so that the excess reactivity, including the effects of installed experiments, does not exceed 2.6% Δk/k under any operating condition.

(2) The minimum shutdown margin under any operating condition with the maximum worth shim/safety rod and the regulating rod full out shall be no less than 1.0% Δk/k.

(3) All core grid positions internal to the active fuel boundary shall be occupied by a standard, partial, control rod, or blank fuel element or by an experimental facility.

(4) The moderator temperature coefficient shall be negative and shall have a minimum absolute reactivity value of at least 2×10^{-5} Δk/k/°C across the active core at all normal operating temperatures.

(5) The moderator void coefficient of reactivity shall be negative and shall have a minimum value of at least 2.8×10^{-3} Δk/k/1% void across the active core.

Bases:

(1) The maximum allowed excess reactivity of 2.6% Δk/k provides sufficient reactivity to accommodate fuel burnup, xenon buildup, experiments, control requirements, and fuel and moderator temperature feedback (Section 4.2 of the SAR). Also, calculations show that this excess reactivity ensures that the maximum temperature of the surface of the cladding will be well below the blister threshold of the U_3Si_2 fuel during a design-basis accident (SAR Section 8.4.3.2).

(2) The minimum shutdown margin ensures that the reactor can be shut down from any operating condition and remain shut down after cooling and xenon decay, even with the highest worth rod and the regulating rod fully withdrawn.

(3) The requirement that all grid positions be filled during reactor operation ensures that the volume flow rate of primary coolant which bypasses the heat producing elements will be within the range specified in Section 4.8 of the SAR. Furthermore, the possibility of accidentally dropping an object into a grid position and causing an increase of reactivity is precluded.

(4) A negative moderator temperature coefficient of reactivity ensures that any moderator temperature rise will cause a decrease in reactivity. The U_3Si_2 fuel also has a significant negative temperature coefficient of reactivity because of the Doppler broadening of neutron capture resonances in uranium-238, but no credit is taken for this effect in our safety analyses.

(5) A negative void coefficient of reactivity helps provide reactor stability in the event of moderator displacement by experimental devices or other means.

3.1.2 Maximum Power Level

Applicability: This specification applies to the reactor thermal power level.

Objective: To ensure that the fuel cladding integrity is maintained.

Specification: Steady state power level shall not exceed 500 kW thermal.

Basis: Thermal and hydraulic calculations presented in the SAR indicate that the fuel may be safely operated at power levels up to 500 kW.

3.2 Reactor Control and Safety System

3.2.1 Control Rod Scram Time

Applicability: This specification applies to the elapsed time from the receipt of a safety signal to when a shim/safety rod is fully inserted.

Objective: To ensure that the reactor can be shut down within a specified period of time.

Specification: The reactor will not be operated unless the control rod scram time for a fully withdrawn rod for each of the three shim/safety rods is less than 600 milliseconds.

Bases: Control rod scram times as specified ensure that the safety limit will not be exceeded in a short-period transient. Section 8.4.3.3 of the SAR provides the analysis for this.

3.2.2 Maximum Reactivity Insertion Rate

Applicability: This specification applies to the maximum positive reactivity insertion rate by the most reactive shim/safety rod and the regulating rod simultaneously.

Objective: To ensure that the reactor is operated safely and the safety limit is not exceeded as a result of a short period.

Specification: The reactor will not be operated unless the maximum reactivity insertion rate is less than 0.05% Δk/k per second.

Basis: This maximum reactivity insertion rate limit ensures that the safety limit will not be exceeded as a result of a short period generated by a continuous linear reactivity insertion. Section 8.4.3.5 of the SAR describes how the safety system will ensure that the reactivity inserted during a ramp reactivity insertion is bounded by the analysis performed for step reactivity insertions in SAR Sections 8.4.3.1–8.4.3.4, which shows that the safety limit will not be exceeded because of reactivity step insertions.

3.2.3 Minimum Number of Scram Channels

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

Objective: To stipulate the minimum number of reactor safety system channels that shall be operable to ensure that the safety limits are not exceeded by ensuring the reactor can be shut down at all times.

Specification: The reactor shall not be operated unless the safety system channels described in the following table are operable:

	Reactor Safety System Component	**Minimum Required**	**Function**
1.	Core H_20 Inlet Temperature	1	Scram if temperature $\geq$ 35 °C
2.	Reactor Thermal Power Level (Safety Channels)	2	Scram if thermal power $\geq$ 600 kW, as indicated on calibrated ionization chamber channels
3.	Reactor Period	1	Scram if period $\leq$ 1 second
4.	Reactor Thermal Power Level/Coolant System Pumps	1	Scram if primary and secondary coolant system pumps not on by $\geq$ 120 kW thermal power
5.	Coolant Flow Rate	1	Scram if coolant system has no flow (primary) by $\geq$ 120 kW thermal power
6.	Pool Water Level	1	Scram if pool level $\leq$ 20 feet (15 feet above core)

	Reactor Safety System Component	Minimum Required	Function
7.	Switches a. Magnet power key switch b. Effluent monitor counter switch c. Effluent monitor compressor power switch d. LOG-N amp calibrate or test mode switch e. Period amp calibrate or test mode switch f. Reactor power-level safety modules (2) calibrate or test mode switch g. Reactor period safety module calibrate or test mode switch	8	Scram if any listed switch is not set properly. Switches to select between operating mode and non-operating mode (e.g., on/off) must be set to operating mode. Switches to select between operating mode and a test or calibrate mode (e.g., norm/test) must be set to operating mode.
8.	Time-Trace Displays a. LOG-N b. Linear Level c. Startup d. Period e. Effluent Monitor	5	Scram if power is lost to any one of the listed time-trace displays
9.	Manual Scrams a. Control Room Console b. Pool Top Catwalk c. BSF Catwalk d. Rabbit/BP Area e. Thermal Column/BP Area	5	Scram upon activation of any one manual scram switch
10.	Neutron-Sensitive Ionization Chambers	4	Scram if bias voltage drops below operational specifications
11.	Safety Setpoints Associated with Time-Trace Display Signals a. Period b. Linear Level c. Startup	3	Scram if any condition listed below is met: $\leq$ 5 seconds $\geq$ 120% of licensed power $\leq$ 2 counts per second (may be bypassed if K_{eff} < 0.9)

	Reactor Safety System Component	Minimum Required	Function
12.	Safety System	2	Scram in case of a safety amp fault or if system is discontinuous
13.	Shim/Safety Rod Magnet Current	3	Rod drop will occur for any shim/safety rod that has magnet current $\geq$ 100 ma

Bases:

(1) Ensures safety limit is not exceeded.

(2) Ensures safety limit is not exceeded.

(3) Ensures safety limit is not exceeded.

(4) Ensures coolant system pumps are functional before raising power > 120 kW.

(5) Ensures there is always primary coolant flow when greater than 120 kW.

(6) Ensures there is enough primary coolant for natural convection cooling.

(7) Ensures nuclear instrumentation is in proper mode for operation.

(8) Ensures information is available for observation by the reactor operator during operation and is recorded, if required, as a record of reactor operations.

(9) Ensures that the reactor can be shut down by the reactor operator in the control room or at other locations near experimental facilities if deemed necessary by other reactor staff.

(10) Ensures shutdown if nuclear instrumentation fails.

(11) Ensures backup shutdown capability from short period or high power level. Ensures shutdown if count rate is too low to provide meaningful startup information. The startup interlock may be bypassed if K_{eff} is $\leq$ 0.9.

(12) Ensures all components of the safety system are installed and operational.

(13) Ensures that any control rod exhibiting excess magnet current will be released and fall to the bottom as a result of gravity.

3.3 Coolant System

3.3.1 Pump Requirements

Applicability: This specification applies to the operation of pumps for both the primary and secondary coolant loops.

Objective: To ensure that both pumps are functioning whenever the reactor is operated above 120 kW.

Specification: The reactor will not be operated above 120 kW unless both the primary and secondary coolant pumps are activated and there is flow in the primary coolant loop.

Bases: Having both pumps operating and flow in the primary loop will ensure there is adequate cooling of the primary coolant so the safety limit is not exceeded.

3.3.2 Coolant Level

Applicability: This specification applies to the height of the water in the reactor pool above the core.

Objective: To ensure adequate primary coolant in the reactor pool and sufficient biological shielding above the core.

Specification: The reactor shall not be operated unless there is 20 feet of water in the reactor pool and 15 feet of water above the core.

Bases: With the pool full of water to a level of 20 feet, there is adequate primary coolant for natural convection cooling. With 15 feet of water above the core, there is sufficient shielding at the licensed power level. Section 7.1.1.4 of the SAR discusses this shielding.

3.3.3 Water Chemistry Requirements

Applicability: This specification applies to the purity of the primary coolant water.

Objective: To minimize corrosion of the cladding on the fuel elements and to reduce the probability of neutron activation of ions in the water.

Specification:

(1) The conductivity of the pool water shall not exceed the limit of 2.0 μ mho/cm.
(2) The pH of the pool water shall not exceed 8.0.

Bases: Operation in accordance with these specifications ensures that aluminum corrosion is within acceptable limits and that the concentration of dissolved impurities that could be activated by neutron irradiation remains within acceptable limits.

3.3.4 Leak or Loss of Coolant Detection

Applicability: This specification applies to the capability of detecting and preventing the loss of primary coolant.

Objective: To ensure adequate primary coolant in the reactor pool and sufficient biological shielding above the core when the reactor is operating.

Specification: The pool water level shall be at least 15 feet above the top of the fuel in the core.

Bases: The same system that functions to scram the reactor upon low pool level will also be used as the detection system for this specification. Section 3.2.2.1 of the SAR discusses the design criteria of the cooling system to prevent large losses of pool water caused by siphoning.

3.3.5 Primary and Secondary Coolant Activity Limits

Applicability: This specification applies to the buildup of radioactive materials in the secondary coolant system.

Objective: To ensure a level of radioactive materials low enough so as not to exceed the limits of 10 CFR Part 20, "Standards for Protection against Radiation," if coolant is released to the sanitary sewer system.

Specification: The primary and secondary coolant system shall be monitored for the buildup of radioactivity and analyzed at least semiannually for increase in the concentration of radionuclides.

Basis: The basis for this specification is to ensure releases are legal and consistent with the as low as is reasonably achievable (ALARA) principal.

3.4 Confinement

Applicability: This specification applies to the capability to provide confinement for the reactor building.

Objective: To prevent the exposure of the public to airborne radioactivity exceeding the limits of 10 CFR Part 20 and the ALARA principle.

Specification: The reactor shall not be operated unless the following conditions are met:

(1) exhaust fan operating
(2) with exceptions for ingress and egress, all exterior doors and windows closed

Bases: By having the capability to provide confinement for the reactor building, exposure of the public to airborne radioactivity may be limited to the extent analyzed in the SAR.

3.5 Ventilation Systems

Applicability: This specification applies to all heating, ventilating, and air conditioning systems that exhaust building air to the outside environment.

Objective: To provide for normal ventilation and the reduction of airborne radioactivity within the reactor building during normal reactor operation and to provide a way to turn off all vent systems quickly in order to isolate the building for emergencies.

Specification:

(1) An exhaust fan with a capacity of at least 1000 cubic feet per minute shall be operable whenever the reactor is operating.

(2) This fan, as well as all other heating, ventilating, and air conditioning systems, shall have the capability to be shut off from a single switch in the control room.

Bases: In the unlikely event of a release of fission products or other airborne radioactivity, the ventilation system will reduce radioactivity inside the reactor building or be able to be isolated. Section 8.4.4 of the SAR includes an analysis of fission product release.

3.6 Radiation Monitoring Systems and Radioactive Effluents

3.6.1 Radiation Monitoring

Applicability: This specification applies to the availability of radiation monitoring equipment that shall be operable during reactor operation.

Objective: To ensure that monitoring equipment is available to evaluate radiation levels in restricted and unrestricted areas and to be consistent with the ALARA principle.

Specification:

(1) When the reactor is operating, the building gaseous effluent monitor shall be operating and have a readout and alarm in the control room. It may be used in either the “normal” mode or “sniffer” mode.

(2) When the reactor is operating, the following area radiation monitors (ARMs) shall be operating and have both local and control room readouts and alarms.

a. pool top
b. primary cooling system
c. beam port/rabbit area
d. thermal column area

(3) Portable survey instrumentation shall be available whenever the reactor is operating to measure beta-gamma exposure rates and neutron dose rates.

(4) When required monitors are inoperable, portable instruments, surveys, or analyses may be substituted for any of the normally installed monitors indicated in Section 3.6.1 above for periods of 1 week or for the duration of a reactor run in cases in which the reactor is continuously operated.

Bases:

(1) The gaseous effluent monitor will detect argon-41 levels in the reactor building. During “normal” mode operation, it will sample and monitor air just before it is released from the reactor building. (See SAR Section 6.3.1.) During “sniffer” mode of operation, it may be used for short periods to monitor in and around experimental facilities to determine local argon-41 levels.

(2) The ARMs provide a continuing evaluation of the radiation levels within the reactor building (see SAR Section 3.7) and provide a warning if levels are higher than anticipated.

(3) The availability of survey meters enables the reactor staff to independently confirm radiation levels throughout the building.

(4) In the event of instrument failure, short-term substitutions will enable the safe continued operation of the reactor.

3.6.2 Radioactive Effluents

Applicability: This specification applies to the monitoring of radioactive effluents from the facility.

Objectives:

(1) To ensure that liquid radioactive releases are safe and legal.

(2) To ensure that the release of argon-41 beyond the site boundary does not result in concentrations above the effluent concentration limit for unrestricted areas (see 10 CFR 20.1302, “Compliance with Dose Limits for Individual Members of the Public,” and Table 2 of Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” to 10 CFR Part 20).

(3) To ensure that the release of argon-41 in the restricted area does not result in concentrations above the derived air concentration (DACs).

Specification:

(1) The concentration of radioactive liquids released into the sanitary sewer shall not exceed the limits specified in 10 CFR 20.2003, “Disposal by Release into Sanitary Sewerage.”

(2) The concentration of argon-41 at ground level below the point of release into the unrestricted area shall not exceed the unrestricted area effluent concentration limit (10 CFR 20.1302 and Table 2 of Appendix B to 10 CFR Part 20) when averaged over 1 year or 10 times the effluent concentration limit when averaged over 1 day.

(3) The concentration of argon-41 in the restricted area shall not exceed the DAC when averaged over a 2000-hour work year.

Bases:

(1) Section 6.2 of the SAR includes the basis for this specification.

(2) Section 6.3 of the SAR includes the basis for this specification.

(3) Section 6.3 of the SAR and 10 CFR 20.1003 include the basis for this specification.

3.7 Experiments

3.7.1 Reactivity Limits

Applicability: This specification applies to experiments to be installed in or near the reactor and associated experimental facilities.

Objectives: To prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specification:

(1) The absolute value of the reactivity worth of any single secured experiment shall not exceed 0.7% Δk/k.

(2) The absolute value of the reactivity worth of any single movable experiment shall not exceed 0.4% Δk/k.

(3) The absolute value of the reactivity worth of all movable experiments shall not exceed 0.6% Δk/k.

(4) The absolute value of the reactivity worth of experiments having moving parts shall be designed to have an insertion rate less than 0.05% Δk/k/s.

(5) The absolute value of the reactivity worth of any movable experiment that may be oscillated shall have a reactivity change of less than 0.05% Δk/k.

(6) The total reactivity worth of all experiments shall not be greater than 0.7% Δk/k.

Bases:

(1) Section 8.4.3.2 of the SAR, which evaluates a step insertion of reactivity from an experiment, includes the bases for specifications 1, 2, 3, and 6.

(2) The bases for specifications 4 and 5 allow for certain reactor kinetics experiments to be performed but still limits the rate of change of reactivity insertions to levels that have been analyzed. Section 8.4.3.2 of the SAR evaluates a step insertion of reactivity from an experiment.

3.7.2 Design and Materials

Specification:

(1) No experiment shall be installed that could shadow the nuclear instrumentation, interfere with the insertion of a control rod, or credibly result in fuel element damage.

(2) All materials to be irradiated in the reactor shall be either corrosion resistant or doubly encapsulated within corrosion-resistant containers.

(3) Explosive materials shall not be allowed in experiments, except for neutron radiographic exposures of items performed outside of the core and experimental facilities. The amount of explosive material contained in capsules used for radiographic exposures shall not exceed 5 grains of gunpowder.

Bases:

(1) Specification 1 ensures that no physical interference with the operation of the reactor detectors, control rods, or physical damage to fuel elements will take place.

(2) Limiting corrosive materials in specification 2 and explosives in specification 3 reduces the likelihood of damage to reactor components and/or releases of radioactivity resulting from experiment failure.

(3) Limiting explosive materials to neutron radiographic exposures done outside of the core and experimental facilities reduces the likelihood of damage resulting from this experimental failure.

4.0 SURVEILLANCE REQUIREMENTS

4.1 Reactor Core Parameters

4.1.1 Excess Reactivity and Shutdown Margin

Applicability: This specification applies to surveillance requirements for determining the excess reactivity of the reactor core and its shutdown margin.

Objective: To ensure that the excess reactivity and shutdown margin limits of the reactor are not exceeded.

Specification:

(1) Whenever a net change in core configuration, for which the predicted change in reactivity is > 0.2% $\Delta k/k$, involving grid position is made, both excess reactivity and shutdown margin shall be determined.

(2) Both shutdown margin and excess reactivity shall be determined annually.

Bases: A determination of excess reactivity is needed to preclude operating without adequate shutdown margin. Moving a component out of the core and returning it to its same location is not a change in the core configuration and does not require a determination of excess reactivity.

4.1.2 Fuel Elements

Applicability: This specification applies to surveillance requirements for determining the physical condition of the reactor fuel.

Objective: To ensure that visible deterioration, corrosion, or other physical changes to the fuel elements are detected in a timely manner.

Specification: All fuel elements, both in-core and out, shall be visually inspected at least once every 5 years, by inspecting at least one-fifth of the elements annually.

Basis: If the water purity is continuously maintained within specified limits, it is projected that chemical corrosion of the fuel clad will proceed slowly. However, faults in the basic materials or fabrication could lead to loss of cladding integrity.

4.2 Reactor Control and Safety Systems

4.2.1 Control Rods

Applicability: This specification applies to the surveillance requirements for the shim safety rods and the regulating rod.

Objective: To ensure that all rods are operable.

Specification:

(1) The reactivity worth of the shim safety rods and regulating rod shall be determined annually and before the routine operation of any new core configuration.

(2) Shim safety control rod scram times and drive times and regulating rod drive time shall be determined annually or after maintenance or modification is completed on a mechanism.

(3) The shim safety rods and regulating rod shall be visually inspected annually for indication of corrosion and indication of excessive friction with guides.

Bases: The reactivity worth of the rods is measured to ensure that the required shutdown margin and reactivity insertion rates are maintained. It also provides a means for determining the reactivity of experiments. Measuring annually will provide corrections for burnup, and measuring after core changes ensures that altered rod worths will be known before continued operations.

The visual inspection of the rods and measurements of control rod scram times and drive times are made to ensure that the rods are capable of performing properly. Verification of operability after maintenance or modification of the control system will ensure proper reinstallation.

4.2.2 Reactor Safety System

Applicability: This specification applies to the surveillance requirements for the reactor safety system.

Objective: To ensure that the reactor safety system channels will remain operable and prevent safety limits from being exceeded.

Specification:

(1) A channel check of each measuring channel shall be performed daily when the reactor is operating.

(2) A channel test of each measuring channel shall be performed before each day's operation or before each operation extending more than 1 day.

(3) A channel calibration of the reactor power level measuring channels shall be made annually (linear level and LOG-N).

(4) A channel calibration of the level and period safety channels shall be made annually. Channel tests are done on these before each day's operation.

(5) A channel calibration of the following shall be made annually:

   a. core inlet temperature measuring system
   b. adequate pool water level indication
   c. indication of coolant system pumps operating
   d. indication that there is flow in the primary coolant loop

(6) The control room manual scram shall be verified to be operable before each day's operation. All other manual scram switches shall be tested annually.

(7) Other scram channels shall be tested/calibrated annually.

(8) Any instrument channel replacement shall be calibrated after installation and before utilization.

(9) Any instrument repair or replacement shall have a channel test before reactor operation.

Bases: The daily channel tests and checks will ensure that the scram channels are operable. Appropriate annual tests or calibrations will ensure that those long-term functions not tested before daily operation are operable.

4.3 Coolant System

4.3.1 Primary Coolant Water Purity

Applicability: This specification applies to the conductivity of the primary coolant water.

Objective: To ensure high-quality pool water.

Specification: The conductivity and pH of the pool water shall be measured weekly.

Bases: This specification ensures that changes that might increase the corrosion rate are detected in a timely manner and that the concentrations of impurities that might be made radioactive do not increase significantly.

4.3.2 Coolant System Radioactivity

Applicability: This specification applies to the radioactive material in the primary coolant or secondary coolant.

Objective: To identify radionuclides as potential sources of release to the sanitary sewer system.

Specification: Primary and secondary coolant shall be analyzed for radioactivity quarterly and before release.

Bases: Radionuclide analysis of the pool water or secondary coolant allows for the determination of any significant buildup of fission or activation products and helps ensure that radioactivity is not permitted to escape to the tertiary system in an uncontrolled manner.

4.4 Confinement

Applicability: This specification applies to the surveillance requirements for building confinement.

Objective: To ensure that building confinement capability exists.

Specification: A quarterly test shall be made to ensure that the building exhaust fan is operable and all exterior doors and windows have closure capability.

Bases: Quarterly surveillance of this equipment will verify that the confinement of the reactor bay can be maintained if needed.

4.5 Ventilation System

Applicability: This specification applies to the surveillance requirements for the building ventilation system.

Objective: To ensure that the ventilation shutoff functions satisfactorily.

Specification: The shutoff switch for all fans and air conditioning systems shall be tested on a quarterly basis.

Bases: This surveillance will ensure that the building can be isolated quickly if necessary to prevent uncontrolled escape of airborne radioactivity to the unrestricted environment.

4.6 Radiation Monitoring Systems and Radioactive Effluents

4.6.1 Effluent Monitor

Applicability: This specification applies to the surveillance requirement of the effluent monitor.

Objective: To ensure that the effluent monitor is operational and provides accurate effluent readings.

Specification: The effluent monitor shall have a channel calibration annually, and a channel test before each day’s operation.

Bases: The calibration will ensure that effluent release estimates are accurate, and the test will ensure that the monitor is operable whenever the reactor is operating.

4.6.2 Area Radiation Monitors

Applicability: This specification applies to the area radiation monitoring equipment.

Objective: To ensure that radiation monitoring equipment is operable whenever the reactor is operating.

Specification: A channel test of the ARMs shall be completed before each day's operation, and a channel calibration shall be completed annually.

Bases: Calibration annually will ensure the required reliability, and a check on days when the reactor is operated will detect obvious malfunctions in the system.

4.6.3 Portable Survey Instrumentation

Applicability: This specification applies to the portable survey instrumentation available to measure beta-gamma exposure rates and neutron dose rates.

Objective: To ensure that radiation survey instrumentation is operable whenever the reactor is operating.

Specification: Beta-gamma and neutron survey meters shall be checked with a source for operability quarterly and shall be calibrated annually.

Bases: Checks with a source will detect obvious detector deficiencies, and an annual calibration will ensure reliability.

5.0 DESIGN FEATURES

5.1 Site and Facility Description

5.1.1 Facility Location

The reactor and associated equipment is housed in a building at 1298 Kinnear Road, Columbus, Ohio, located on the West Campus of The Ohio State University. The minimum free air volume of the building housing the reactor will be greater than or equal to 70,000 cubic feet (ft^3). There is an exhaust fan with dampers providing control of the release of airborne radioactivity.

5.1.2 Controlled and Restricted Area

The fence surrounding the reactor building shall describe the controlled area. The restricted area, as defined in 10 CFR Part 20, shall consist of the reactor building.

5.2 Reactor Coolant System

5.2.1 Primary Coolant Loop

Natural convective cooling is the primary means of heat removal from the core. Water enters the core at the bottom and flows upward through the flow channels in the fuel elements.

5.2.2 Secondary and Tertiary Coolant Loops

The secondary coolant loop removes heat from the primary coolant. The secondary coolant (ethylene glycol and water) passes through two separate heat exchangers to remove heat if necessary. Heat is removed from the first by an outside fan-forced dry cooler. City water flow through the secondary side of an additional heat exchanger makes up the tertiary loop. It provides additional cooling for the secondary coolant.

5.3 Reactor Core and Fuel

Up to 30 positions on the core grid plate are available for use as fuel element positions. Control rod fuel elements occupy four of these positions and one is reserved for the central irradiation facility flux trap. Several arrangements for the cold, clean, critical core have been investigated. Approximately 18 standard fuel elements, in addition to the control rod fuel elements, are currently required. Partial elements, blank elements, and graphite elements may be utilized in various combinations to achieve the proper excess reactivity.

The reactor fuel is plate-type U_3Si_2, with a uranium-235 enrichment of less than 20 percent. Standard fuel elements have a total of 16 fueled plates and 2 outer aluminum plates. The control rod fuel elements have inner fuel plates removed to allow the control rods to enter. Aluminum guide plates are on the inside of this gap. The outer two plates for each control rod assembly are fueled. Partial elements are also available with 25, 40, 50, and 60 percent of the nominal loading of a standard element. These partial fuel elements are prefabricated by the vendor with fixed numbers of plates.

(1) References: NUREG-1313
ANL/RERTR/TM-10
ANL/RERTR/TM-11

5.4 Fuel Storage

The fuel storage pit, located below the floor of the reactor pool and at the end opposite from the core, shall be flooded with water whenever fuel is present and shall be capable of storing a complete core loading. When fully loaded with fuel and filled with water, K_{eff} shall not exceed 0.90, and natural convective cooling shall ensure that no fuel temperatures reach a point at which onset of nucleate boiling is possible.

5.5 Fuel-Handling Tools

All tools designed for or capable of removing fuel from core positions or storage rack positions shall be secured when not in use by a system controlled by the supervisor of reactor operations or the senior reactor operator on duty.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The Ohio State University Research Reactor is a part of the College of Engineering administered by the Engineering Experiment Station. Figure 6.1 illustrates the organizational structure.

6.1.2 Responsibility

The Director, Engineering Experiment Station (Level 1), is the contact person for communications between the NRC and The Ohio State University.

The Director, Nuclear Reactor Laboratory (Level 2), will have overall responsibility for the management of the facility.

The Associate Director (or Manager, Reactor Operations) (Level 3) shall be responsible for the day-to-day operation and for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and technical specifications. During periods when the associate director is absent, his/her responsibilities are delegated to a senior reactor operator (Level 4).

6.1.3 Staffing

(1) The following shall be the minimum staffing when the reactor is not secured:

a. A certified reactor operator shall be in the control room.

b. A second designated person shall be present at the facility complex able to carry out prescribed written instructions. Unexpected absence for as long as 2 hours to accommodate a personal emergency may be acceptable provided immediate action is taken to obtain a replacement.

c. A senior reactor operator shall be readily available on call. "Readily available on call" means an individual who (1) has been specifically designated and the designation known to the operator on duty, (2) keeps the operator on duty informed of where he/she may be rapidly contacted and the phone number, and (3) is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15-mile radius).

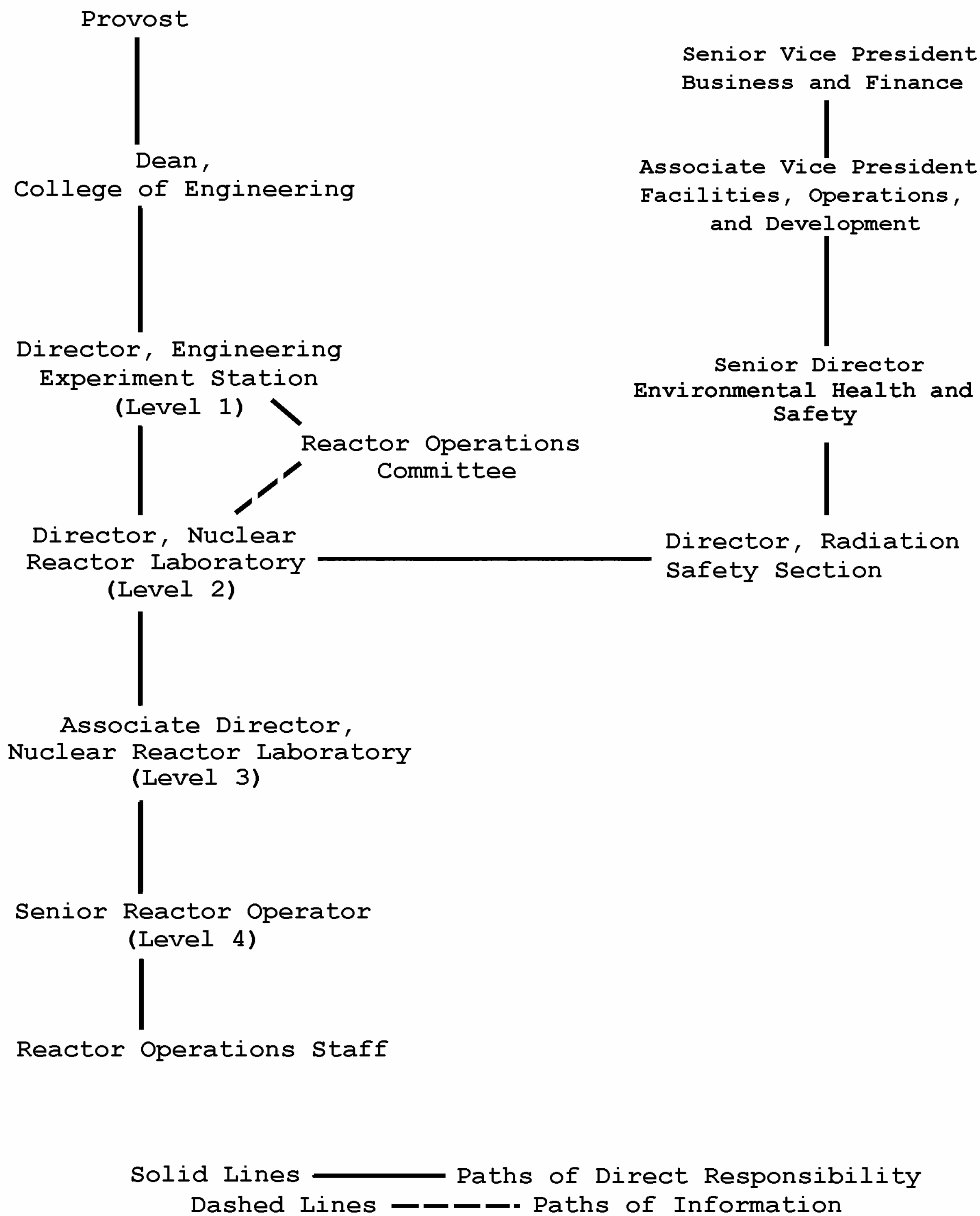

Figure 6.1 Administrative Organization

(2) A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the reactor operator. The list shall include the following:

a. management personnel
b. radiation safety personnel
c. other operations personnel

(3) The following events require the presence at the facility of a senior reactor operator:

a. initial startup and approach to power

b. all fuel or control-rod relocations within the reactor core region

c. recovery from an unplanned or unscheduled shutdown (these instances require documented verbal concurrence from a senior reactor operator)

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of American National Standard, ANSI/ANS-15.4-1988, "Selection and Training of Personnel for Research Reactors."

6.2 Review and Audit

There shall be a reactor operations committee (ROC) for independent review of the safety aspects of reactor operations to ensure that the facility is operating in a manner consistent with public safety and within the terms of the facility license.

A member or members of this committee or another qualified person or persons shall audit safety aspects of reactor operations as described in Section 6.2.4 of this document.

6.2.1 Composition and Qualifications of the Reactor Operations Committee

The ROC shall be composed of a minimum of three members who should collectively represent a broad spectrum of expertise in appropriate fields (i.e., having professional backgrounds in engineering, physical, biological, or medical sciences, as well as knowledge of and interest in applications of nuclear technology and ionizing radiation). Members and alternates shall be appointed by and report to Level 1 management. Individuals may be either from within or outside the operating organization. Qualified and approved alternates may serve in the absence of regular members.

6.2.2 Reactor Operations Committee Meetings

ROC functions shall be conducted in accordance with the following four items:

(1) Meetings shall be held at least once per calendar year and more frequently as circumstances warrant, consistent with effective monitoring of facility activities.

(2) A meeting quorum shall consist of at least one-half of the membership where the operating staff does not constitute a majority.

(3) The ROC may appoint a subcommittee from within its membership to act on behalf of the full committee on those matters that cannot await the next meeting. The ROC shall review the actions taken by the subcommittee at the next regular meeting.

(4) Meeting minutes shall be distributed to ROC members before the next meeting and shall be reviewed at the next meeting.

6.2.3 Review Function

The ROC shall review the following eight items:

(1) determination that proposed changes in equipment, systems, tests, experiments, or procedures do not require a license update, as described in 10 CFR 50.59, “Changes, Tests and Experiments”

(2) all new procedures and major revisions thereto having safety significance and proposed changes in reactor facility equipment or systems having safety significance

(3) all new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity

(4) proposed changes in technical specifications or license

(5) violations of technical specifications or license and violations of internal procedures having safety significance

(6) operating abnormalities having safety significance

(7) reportable occurrences listed in Section 6.6.2 of this document

(8) audit reports

A written report or minutes of the findings and recommendations of the review group shall be submitted to Level 1 management and ROC members in a timely manner after the review has been completed.

6.2.4 Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Discussions with cognizant personnel and observations of operations should be used as appropriate. In no case shall the individual immediately responsible for an area perform an audit in that area. The following four items shall be audited:

(1) facility operations for conformance to the technical specifications and license, at least once per calendar year (interval between audits not to exceed 15 months)

(2) the requalification program for the operating staff, at least once every other calendar year (interval between audits not to exceed 30 months)

(3) the results of action taken to correct those deficiencies that may occur in the reactor facility, equipment, systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval between audits not to exceed 15 months)

(4) the reactor facility emergency plan and implementing procedures at least once every other calendar year (interval between audits not to exceed 30 months)

Deficiencies found that affect reactor safety shall be reported immediately to Level 1 management. A written report of audit findings should be submitted to Level 1 management and the full ROC within 3 months of the audit's completion.

6.3 Procedures

6.3.1 Reactor Operating Procedures

Written procedures, reviewed and approved by the director, or his/her designee, and reviewed by the ROC, shall be in effect and followed. The procedures shall be adequate to ensure the safety of the reactor, but should not preclude the use of independent judgment and action, should the situation require such. All new procedures and changes to existing procedures shall be documented by the NRL staff and subsequently reviewed by the ROC. At least the following eight items shall be covered:

(1) startup, operation, and shutdown of the reactor

(2) installation, removal, or movement of fuel elements, control rods, experiments, and experimental facilities

(3) actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected cooling system leaks, and abnormal reactivity changes

(4) emergency conditions involving potential or actual release of radioactivity including provisions for evacuation, re-entry, recovery, and medical support

(5) preventive and corrective maintenance procedures for systems that could have an effect on reactor safety

(6) periodic surveillance of reactor instrumentation and safety systems, area monitors, and radiation safety equipment

(7) implementation of the emergency plan, reactor operator training and requalification requirements, and the security requirements of 10 CFR 73.67, "Licensee Fixed Site and In-Transit Requirements for the Physical Protection of Special Nuclear Material of Moderate and Low Strategic Advantage"

(8) personnel radiation protection

6.3.2 Administrative Procedures

Procedures shall also be written and maintained to ensure compliance with Federal regulations, the facility license, and commitments made to the ROC or other advisory or governing bodies. As a minimum, these procedures shall include the following:

(1) audits
(2) special nuclear material accounting
(3) operator requalification
(4) recordkeeping
(5) procedure writing and approval

6.4 Experiment Review and Approval

6.4.1 Definitions of Experiments

Approved experiments are those that have previously been reviewed and approved by the ROC. They shall be documented and may be included as part of the procedures manual. New experiments are those that have not previously been reviewed, approved, and performed. Routine tests and maintenance activities are not experiments.

6.4.2 Approved Experiments

All proposed experiments utilizing the reactor shall be evaluated by the experimenter and a licensed senior reactor operator to ensure compliance with the provisions of the utilization license, the technical specifications, 10 CFR Part 20, and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." If, in the judgment of the senior reactor operator, the experiment meets with the above provisions, is an approved experiment, and does not constitute a threat to the integrity of the reactor, it may be approved for performance. When pertinent, the evaluation shall include considerations of the following four items:

(1) the reactivity worth of the experiment

(2) the integrity of the experiment, including the effects of changes in temperature, pressure, or chemical composition

(3) any physical or chemical interaction that could occur with the reactor components

(4) any radiation hazard that may result from the activation of materials or from external beams

6.4.3 New Experiments

Before performing an experiment not previously approved for the reactor, the experiment shall be reviewed and approved by the ROC. Committee review shall consider the following three items:

(1) the purpose of the experiment
(2) the procedure for the performance of the experiment
(3) the safety evaluation previously reviewed by a licensed senior reactor operator

6.5 Required Actions

6.5.1 Action To Be Taken in the Event a Safety Limit Is Exceeded

(1) The reactor shall be shut down, and reactor operations shall not be resumed until authorized by the NRC.

(2) The safety limit violation shall be promptly reported to the director of the reactor laboratory.

(3) The safety limit violation shall be reported by telephone to the NRC within 24 hours.

(4) A safety limit violation report shall be prepared. The report shall describe the following:

   a. applicable circumstances leading to the violation including, when known, the cause and contributing factors

   b. effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public

   c. corrective action to be taken to prevent recurrence

(5) The report shall be reviewed by the ROC and shall be submitted to the NRC within 14 working days, and any follow-up report shall be submitted to the NRC when authorization is sought to resume operation of the reactor.

6.5.2 Action To Be Taken in the Event of a Reportable Occurrence

A reportable occurrence is any of the following seven conditions:

(1) operating with any safety system setting less conservative than stated in these specifications

(2) operating in violation of a limiting condition for operation established in Section 3.0 of these specifications

(3) safety system component malfunctions or other component or system malfunctions during reactor operation that could, or threaten to, render the safety system incapable of performing its intended function

(4) an uncontrolled or unanticipated increase in reactivity in excess of 0.4% $\Delta k/k$

(5) an observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor

(6) abnormal and significant degradation in reactor fuel and/or cladding, coolant boundary, or confinement boundary (excluding minor leaks), where applicable, that could result in exceeding prescribed radiation exposure limits of personnel and/or the environment

(7) any uncontrolled or unauthorized release of radioactivity to the unrestricted environment

In the event of a reportable occurrence, the following four actions shall be taken:

(1) The reactor conditions shall be returned to normal, or the reactor shall be shutdown, to correct the occurrence.

(2) The director of the reactor laboratory shall be notified as soon as possible and corrective action shall be taken before resuming the operation involved.

(3) A written report of the occurrence shall be made which shall include an analysis of the cause of the occurrence, the corrective action taken, and the recommendations for measures to preclude or reduce the probability of recurrence. This report shall be submitted to the director and the ROC for review and approval.

(4) A report shall be submitted to the NRC in accordance with Section 6.6.2 of these specifications.

6.6 Reports

Reports shall be made to the NRC as described in the following sections.

6.6.1 Operating Reports

An annual report shall be made by September 30 of each year to the NRC Document Control Desk (if on paper, the signed original), with a copy to the appropriate NRC Regional Office, in accordance with 10 CFR 50.4, "Written Communications," providing the following seven information items:

(1) a narrative summary of operating experience (including experiments performed) and of changes in facility design, performance characteristics, and operating procedures related to reactor safety occurring during the reporting period

(2) a tabulation showing the energy generated by the reactor (in kW hours) and the number of hours the reactor was in use

(3) the results of safety-related maintenance and inspections and the reasons for corrective maintenance of safety-related items

(4) a table of unscheduled shutdowns and inadvertent scrams, including their reasons and the corrective actions taken

(5) a summary of the safety analyses performed in connection with changes to the facility or procedures, which affect reactor safety, and performance of tests or experiments carried out under the conditions of 10 CFR 50.59

(6) a summary of the nature and amount of radioactive gaseous, liquid, and solid effluents released or discharged to the environs beyond the effective control of the licensee as measured or calculated at or before the point of such release or discharge

(7) a summary of radiation exposures received by facility personnel and visitors, including the dates and times of significant exposures

6.6.2 Special Reports

(1) A telephone or telegraph report of the following shall be submitted as soon as possible, but no later than the next working day, to the appropriate Regional Office:

   a. any accidental offsite release of radioactivity above authorized limits, whether or not the release resulted in property damage, personal injury, or known exposure

   b. any exceeding of the safety limit as defined in Section 2.1 of these specifications

   c. any reportable occurrences defined in Section 6.5.2 of these specifications

(2) A written report shall be submitted within 14 days to the NRC Document Control Desk (if on paper, the signed original), with a copy to the appropriate Regional Office, in accordance with 10 CFR 50.4, of the following:

a. any accidental offsite release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury, or known exposure

b. any exceeding of the safety limit as defined in Section 2.1 of these specifications

c. any reportable occurrence as defined in Section 6.5.2 of these specifications

(3) A written report shall be submitted within 30 days to the NRC Document Control Desk (if on paper, the signed original), with a copy to the appropriate Regional Office in accordance with 10 CFR 50.4, of the following:

a. any substantial variance from performance specifications contained in these specifications or in the SAR

b. any significant change in the transient or accident analyses as described in the SAR

c. changes in personnel serving as the Director, Engineering Experiment Station, Reactor Director, or Reactor Associate Director

(4) A report shall be submitted within 9 months after initial criticality of the reactor or within 90 days of completion of the startup test program, whichever is earlier, to the NRC Document Control Desk (if on paper, the signed original), with a copy to the appropriate Regional Office, upon receipt of a new facility license, an amendment to the license authorizing an increase in power level, or the installation of a new core of a different fuel element type or design than previously used.

The report shall include the measured values of the operating conditions or characteristics of the reactor under the new conditions and comparisons with predicted values, including the following:

a. total control rod reactivity worth
b. reactivity worth of the single control rod of highest reactivity worth
c. minimum shutdown margin both at ambient and operating temperatures
d. excess reactivity
e. calibration of operating power levels
f. radiation leakage outside the biological shielding
g. release of radioactive effluents to the unrestricted environment

6.7 Records

Records or logs of the items listed below shall be kept in a manner convenient for review and shall be retained for as long as indicated.

6.7.1 Records To Be Retained for a Period of at Least 5 Years

(1) normal plant operation
(2) principal maintenance activities
(3) experiments performed with the reactor
(4) reportable occurrences
(5) equipment and component surveillance activity
(6) facility radiation and contamination surveys
(7) transfer of radioactive material
(8) changes to operating procedures
(9) minutes of ROC meetings

6.7.2 Records To Be Retained for at Least One Requalification Cycle

Regarding retraining and requalification of licensed operations personnel, the records of the most recent complete requalification cycle shall be maintained at all times the individual is employed.

6.7.3 Records To Be Retained for the Life of the Facility

(1) gaseous and liquid radioactive effluents released to the environment

(2) fuel inventories and transfers

(3) radiation exposures for all personnel

(4) changes to reactor systems, components, or equipment that may affect reactor safety

(5) updated, corrected, and as-built drawings of the facility

(6) records of significant spills of radioactivity, and status

(7) annual operating reports provided to the NRC

(8) copies of NRC inspection reports, and related correspondence

Safety Evaluation Report Related to the Renewal of

Facility License No. R-75 for the

Ohio State University Research Reactor,

Ohio State University

June 2008
Office of Nuclear Reactor Regulation

ABSTRACT

This safety evaluation report summarizes the findings of a safety review conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation. The staff conducted this review in response to a timely application filed by Ohio State University (the licensee or OSU) for a 20-year renewal of Facility License No. R-75 to continue to operate the Ohio State University Research Reactor (OSURR or the facility). The facility is located at the Columbus campus of OSU in Columbus, Ohio. In its safety review, the staff considered information submitted by the licensee (including past operating history recorded in the licensee's annual reports to the NRC) as well as inspection reports prepared by NRC personnel and first-hand observations. On the basis of this review, the staff concludes that OSU can continue to operate the OSURR, in accordance with the renewed license, without posing a significant risk to the health and safety of the public, facility personnel, or the environment.

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LIST OF ABBREVIATIONS

LIST OF ABBREVIATIONS (cont.)

1 INTRODUCTION

1.1 Overview

By letter (and supporting documentation) dated December 15, 1999, as supplemented on August 21, 2002; August 18, 2005; July 26, 2006; May 22, May 31, September 4, and September 28, 2007; and February 29, 2008, Ohio State University (OSU or the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC or the Commission) a timely application for a 20-year renewal of the Class 104c Facility License No. R-75 (NRC Docket No. 50-150). The renewed license would authorize continued operation of the OSU Research Reactor (OSURR) located on the OSU campus in Columbus, Ohio. In accordance with Title 10, Section 2.109, “Effect of Timely Renewal Application,” of the Code of Federal Regulations (10 CFR 2.109), the current license will not be deemed to have expired until the Commission takes final action on the licensee’s application.

The staff conducted its review based on information contained in the renewal application, as supplemented. The renewal application includes the safety analysis report (SAR), proposed technical specifications (TSs), an environmental report, the operator requalification plan, financial qualifications, and responses to staff requests for additional information. The licensee also requested that the staff consider as part of the application the emergency plan (EP) previously filed with the NRC. The licensee has since updated the EP as part of the licensee’s routine maintenance of the EP under 10 CFR 50.54(q). The staff also based its review on annual reports of facility operation submitted by the licensee and inspection reports prepared by NRC staff. The review staff conducted site visits to observe facility conditions.

The licensee’s application and other materials reviewed by the staff may be examined, and/or copied for a fee, at the NRC’s Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC’s public documents. Documents related to this license renewal dated on or after November 24, 1999, may be accessed through the NRC’s Public Electronic Reading Room on the Internet at http://www.nrc.gov. If you do not have access to ADAMS or have problems accessing the documents located in ADAMS, or if you want access to documents dated before November 24, 1999, contact the NRC Public Document Room Reference staff at 1-800-397-4209 or 301-415-4737, or by email to pdr@nrc.gov.

This safety evaluation report (SER) summarizes the findings of the staff’s safety review of the licensee’s application. This SER and an environmental assessment will serve as the basis for issuance of a renewed license authorizing operation of the OSURR at power levels up to 500 kilowatts thermal (kW(t)). In conducting its safety review, the staff evaluated the facility against the requirements of 10 CFR Parts 19, 20, 30, 50, 51, 55, 70, and 73; applicable NRC regulatory guides; and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series. The staff also referred to the guidance contained in NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” issued 1996.

Mr. William B. Kennedy from the NRC’s Office of Nuclear Reactor Regulation, Division of Policy and Rulemaking, Research and Test Reactors Branch A prepared this SER. Other contributors to the safety review include Mr. Jessie F. Quichocho, Mr. Marvin M. Mendonca, Mr. Daniel E. Hughes, Mr. Paul V. Doyle, and Mr. Ronald B. Uleck of the NRC staff.

1.2 Summary and Conclusions on Principal Safety Considerations

On the basis of its safety evaluation, the staff reached the following findings:

- The design, testing, and performance of the OSURR structures, systems, and components (SSCs) important to safety during normal operation are acceptable. Safe operation of the facility can reasonably be expected to continue.

- The licensee's management organization is acceptable to maintain and safely operate the reactor. The licensee's management organization, training and research activities, and security measures continue to be acceptable to ensure safe operation of the facility and the protection of its special nuclear material (SNM).

- The licensee and NRC staff have considered the expected consequences of postulated accidents, including a bounding maximum hypothetical accident (MHA), using conservative initiating and mitigating assumptions. The calculated radiation doses resulting from the MHA meet the regulatory requirements of 10 CFR Part 20, "Standards for Protection Against Radiation," for facility personnel and members of the general public and are, therefore, acceptable.

- Exposures from and releases of radioactive effluents and wastes from the facility are not expected to result in doses or concentrations in excess of the limits specified by Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," to 10 CFR Part 20 and are consistent with as-low-as-reasonably-achievable (ALARA) principles.

- The TSs, which state limits controlling operation of the facility, provide reasonable assurance that the licensee will operate the facility in accordance with the assumptions and analyses in the SAR. No significant degradation of SSCs has occurred, and the TSs will continue to provide reasonable assurance that no significant degradation of SSCs will occur.

- The financial data submitted with the application demonstrate that the licensee has acceptable access to sufficient funds to cover operating costs and to eventually decommission the reactor facility.

- The licensee's procedures for training its reactor operators and the operator requalification plan give reasonable assurance that the licensee will continue to safely operate the reactor facility.

- The licensee's EP provides acceptable assurance that the licensee will continue to be prepared to assess and respond to emergency events.

- Continued operation of the OSURR poses no significant radiological risk to the health and safety of the public, facility personnel, or the environment.

On the basis of these findings, the NRC staff concludes that OSU can continue to operate the OSURR in accordance with its license and the regulations, without endangering the health and safety of the public.

1.3 General Facility Description

The OSURR site comprises the reactor building and a small area immediately surrounding it, bounded by a chain-link fence. The reactor building is a steel frame structure with insulated metal walls. The OSURR is a light-water-cooled-and-moderated, pool-type reactor licensed to operate at power levels up to 500 kW(t). The reactor pool is constructed of concrete, lined with fiberglass-reinforced epoxy paint, and has a capacity of 22,000 liters (L) (5800 gallons (gal)) and a depth of 6 meters (m) (20 feet (ft)). Natural-convection flow of pool water removes the heat generated in the fuel. The reactor core uses low-enriched uranium (LEU) uranium-silicide (U_3Si_2) fuel clad in aluminum. Control rods of boron-stainless steel composition (called shim safety rods) and an additional control rod composed only of stainless steel (called the regulating rod) provide reactivity control. The reactor safety system initiates gravity insertion of the shim safety rods to accomplish automatic reactor shutdown. The OSURR contains multiple experimental facilities, allowing simultaneous performance of a variety of experiments.

1.4 Shared Facilities and Equipment

In addition to the OSURR, the reactor building houses two gamma irradiators and a subcritical assembly that are currently licensed by the State of Ohio. The NRC originally licensed each of these devices when they were first housed in the reactor building. The NRC, the State of Ohio Bureau of Radiological Health, the Radiation Safety Section of the Ohio State University, and the staff of the OSURR have examined the installation and use of these devices for approximately 10 years. During this time, there has been no adverse impact on the operation and safety of the reactor. All utilities, ventilation equipment, and heating and air-conditioning equipment at the reactor building are dedicated to OSURR operations.

1.5 Comparison with Similar Facilities

The design of the OSURR is based on the Bulk Shielding Reactor, which was located at Oak Ridge National Laboratory. This reactor was of the type generally known as a materials testing reactor. Reactors of this type share various common features, such as light-water moderation and cooling, open pools, and plate-type fuel. Multiple reactors with similar design, construction, and operational characteristics have operated safely for more than four decades. The OSURR does not have any unique features that would preclude applying knowledge and experience gained in the operation of these other reactors to operation of the OSURR. In 1988, the OSURR converted its core to use LEU uranium-silicide fuel. Since then, the University of Missouri at Rolla (200 kW(t)), the Rhode Island Nuclear Science Center (2 megawatts thermal (MW(t))), and the University of Massachusetts at Lowell (1 MW(t)) converted their facilities to use LEU uranium-silicide fuel. Of these, the University of Missouri at Rolla is most similar to the OSURR in operating characteristics and facility features.

1.6 Summary of Operations

The licensee has operated the OSURR in accordance with Facility License No. R-75 and established procedures on an as-needed basis to provide teaching, research, and services for students, faculty, and the public. From 1991 to 2001, annual facility operation averaged approximately 2 megawatt-days (MWD) per year, with a maximum annual operation of approximately 7 MWD. These values represent expected annual facility operation during the period of the renewed license. However, the licensee’s accident analyses assume a much less restrictive operating schedule.

1.7 Compliance with the Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 provides that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant shall have entered into an agreement with the U.S. Department of Energy (DOE) for the disposal of high-level radioactive waste and spent nuclear fuel. DOE (represented by R.L. Morgan) has informed the NRC (represented by H. Denton) by letter dated May 3, 1983, that it has determined that universities and other government agencies operating non-power reactors have entered into contracts with DOE that provide that DOE retains title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage or reprocessing. Ohio State University has such a contract with DOE and is in conformance with the Nuclear Waste Policy Act of 1982.

1.8 Facility Modifications and History

On February 3, 1960, the Atomic Energy Commission (AEC) issued Construction Permit No. CPRR-49 authorizing construction of the OSURR. The original Facility License No. R-75, issued by AEC in 1961, authorized OSU to operate the OSURR at power levels up to 10 kW(t). The reactor achieved initial criticality that same year. The original core used high-enriched uranium (HEU) fuel. By order dated September 27, 1988, the NRC issued Amendment No. 12 requiring that the reactor be converted to use LEU fuel. On November 14, 1990, the NRC issued Amendment No. 13 authorizing an increase in the maximum licensed power level to 500 kW(t). As part of the power increase, the licensee upgraded the reactor coolant systems. The licensee has not made any significant modifications to the facility since the power increase in 1990. No significant facility modifications are requested as part of the license renewal.

2 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Geography

The OSURR is located in the City of Columbus in Franklin County, Ohio. The OSURR lies approximately 2 kilometers (km) (1.2 miles (mi)) west of the main campus on land owned by OSU and is a part of the Ohio State University Research Center. The site consists of the reactor building and a small area immediately surrounding it. The site is adjacent to parking lots to the north and east, fields to the west, and the main building of the OSU Research Center to the south.

Franklin County includes portions of two physiographic provinces called the Appalachian Plateau and the Central Lowlands. A series of north-south scarps and terraces form a gentle step-like ascent eastward to the Appalachian Plateau. The highest altitude of the county is 344 m (1130 ft) above mean sea level (MSL). The lowest elevation is 203 m (665 ft) above MSL at the efflux of the Scioto River from Franklin County. The City of Columbus is located in the center of the county at a ground elevation of approximately 247 m (812 ft) above MSL. The reactor site is approximately 240 m (780 ft) above MSL.

2.1.2 Demography

Students, faculty, and staff numbering approximately 65,000 comprise the population of OSU. Activities in the athletic and performing arts facilities periodically draw up to 100,000 additional people to the campus area. The nearest permanent residences are located approximately 0.5 km (0.3 mi) to the west and approximately 0.5 km (0.3 mi) to the south. Approximately 142,000 people reside within 5 km (3 mi) of the facility. This area encompasses residential communities including portions of Columbus, Upper Arlington, Grandview, and Bexley. Approximately 260,000 people reside within 8 km (5 mi) of the facility. This area also includes the fiscal and administrative center of the City of Columbus. The population of the City of Columbus decreased to 694,000 in 2005 from 711,000 in 2000.

2.2 Nearby Industrial, Transportation, and Military Facilities

Numerous manufacturing and processing facilities, research and development facilities, laboratories, machine shops, and educational and medical facilities are located within 8 km (5 mi) of the OSURR. The licensee held discussions with and reviewed information gathered by the Ohio Emergency Management Agency and the Ohio State University Office of Environmental Health and Safety regarding these facilities. As a result, the licensee concluded that accidents at nearby facilities are not expected to affect the safety of the OSURR.

The Marion line of the CSX Railway is located approximately 1.2 km (0.75 mi) to the east of the reactor facility, and the Norfolk and Southern line is about 2.4 km (1.5 mi) to the east. A major rail yard lies approximately 8 km (5 mi) to the west in Norwich Township. Carriers’ estimates, provided by the licensee, indicate that approximately 2 percent of the rail traffic traversing the county in a 1-year period may involve hazardous materials. According to the licensee, none of these materials are expected to pose a risk to safe operation of the OSURR.

Port Columbus International Airport lies approximately 12 km (7.5 mi) to the east of the OSURR. The west flight path for arrivals and departures is located approximately 0.5 km (0.3 mi) to the north of the OSURR at an altitude of approximately 900 m (3000 ft). According to data maintained by the National Transportation Safety Board (NTSB), the average accident rate for U.S. air carriers for 1987–2006 was 0.376 per 100,000 departures. An air carrier incident that the NTSB classifies as an accident is not necessarily the type of incident that could cause damage to the OSURR. In addition, the accident would have to occur at or near the OSURR site to cause damage. Consequently, the rate of air carrier accidents that could damage the OSURR is lower than the accident rate cited above. According to the Port Columbus International Airport website (http://www.port-columbus.com), the airport operates over 160 departures daily, or over 58,000 annual departures. The website estimates that airport activity will increase from 6.6 to 10 million annual enplaned passengers by 2018. Given this growth rate and the historic accident rate, approximately six accidents could occur over the next 20 years. During the period of the renewed license, there is reasonable assurance that air traffic will not affect safe operation of the OSURR.

Military facilities currently operating in the Columbus area include the Ohio Army National Guard Adjutant General's Office, the Ohio Air National Guard Headquarters, and the Defense Supply Center Columbus. Given their distance from the OSURR and the nature of the activities carried out at the facilities, none of these facilities will pose a significant risk to the OSURR.

Based on the above information, the staff concludes that industrial, transportation, and military facilities could pose no significant risk to the continued safe operation of the OSURR.

2.3 Meteorology

The National Oceanic and Atmospheric Administration (NOAA) maintains historical meteorological data sufficient to characterize the reactor site. Meteorological data describing current conditions in the City of Columbus can be readily obtained from the National Weather Service station at the Port Columbus International Airport. These meteorological data will enable the licensee to assess the dispersion of radiological effluents in the unlikely event of an accident-related gaseous release to the environment.

Temperatures in Franklin County over the past 60 years have ranged from a maximum of 39 degrees Celsius (°C) (102 degrees Fahrenheit (°F)) to a minimum of -30 °C (-22 °F). The mean annual temperature over the past 30 years was 11.6 °C (52.9 °F). Mean annual precipitation was 97.8 centimeters (cm) (38.5 inches (in)) from 1971 to 2000. The maximum 1-day snowfall during the past 59 years was 28 cm (11 in). Mean annual wind in the area of Columbus is from the south at approximately 4 meters per second (m/s) (8 miles per hour (mi/h)). The peak gust reported to NOAA from 1930 to 1996 was from the south at 26 m/s (59 mi/h). From 1950 to 2006, 24 tornadoes were observed in Franklin County, Ohio. Of these, seven were classified as F2 or F3 on the Fujita Scale. NUREG/CR-4461, Revision 2, "Tornado Climatology of the Contiguous United States," issued February 2007, gives a tornado strike probability of 3.58×10^{-4} per year for the Central Region of the United States. No instances of damaging high winds or tornadoes have occurred on the OSURR site since its construction. Section 3.2 of this SER discusses potential meteorological damage to the OSURR.

Based on meteorological information submitted by the licensee and staff independent review of meteorological data maintained by NOAA (http://www.ncdc.noaa.gov), the staff concludes that the reactor site meteorology does not pose any significant risk to continued safe operation of the OSURR.

2.4 Hydrology

Columbus is located in the center of the State of Ohio and in the drainage area of the Ohio River. The reactor site is located between the Scioto River, approximately 2.4 km (1.5 mi) to the west, and the Olentangy River, approximately 1.8 km (1.1 mi) to the east. Both rivers flow from north to south to their confluence located approximately 4.0 km (2.5 mi) south of the reactor site. The Scioto River flows south 50 km (30 mi) to Circleville and continues until emptying into the Ohio River at the Ohio-Kentucky border. Both rivers flow through gorge-like formations and have little flood plain. The OSURR site is above the 500-year flood level for both the Scioto and Olentangy Rivers.

The depth of the ground water underlying the OSURR site varies from 14 m to 15 m (45 ft to 50 ft). The ground water flows east toward the Olentangy River. The water for Columbus is taken from the Scioto River upstream of the confluence of the Scioto and Olentangy Rivers. Thus, significant contamination of the water supply for the City of Columbus by leaks or spills at the reactor site is not credible. Circleville, Ohio, is the next major town using water from the Scioto River. No privately or otherwise owned wells are located in the vicinity of the OSURR. The closest well is located approximately 1 km (0.6 mi) to the northeast. This well is not affected by reactor operations and is not monitored for radioactivity.

Based on the above information, the staff concludes that the reactor site hydrology does not pose a significant risk to the continued safe operation of the OSURR, nor does it provide a credible pathway for contamination of the local water supply.

2.5 Geology, Seismology, and Geotechnical Engineering

Columbus lies on the glaciated plains section at the eastern edge of the Central Lowlands physiographic province. The deposits from the most recent glacial advance, 3 m to 9 m (10 ft to 30 ft) of unsorted clay, silt, sand, pebbles, and boulders, lie directly on the limestone and shale bedrock underlying Columbus. Large out-wash deposits in the Scioto and the Olentangy Valleys resulted from the great volumes of melt water coming from the ice sheet. The deposits occurred above the present drainage level as gravel terraces and serve as water recharge areas north and south of downtown Columbus, principally on the west side of the Scioto River. Table 2.1 of the OSURR SAR shows the results of a ground boring analysis performed approximately 150 m (500 ft) southwest of the reactor site.

A belt of seismic activity (the St. Lawrence Seismic Belt) runs through northwest Ohio. Historically, the most active region in Ohio is approximately 100 km (60 mi) northwest of Columbus. A 2002 National Seismic Hazard Map produced by the U.S. Geological Survey shows only a 2-percent probability that in 50 years peak lateral ground acceleration will exceed 0.08 times the acceleration due to gravity.

Based on the above information, the staff concludes that the geology of the reactor site is suitable for supporting the reactor building, structure, and systems, and that potentially damaging seismic events are unlikely to occur during the period of the renewed license.

2.6 Conclusions

The staff concludes that the reactor site has experienced no significant geographical, meteorological, or geological change, and therefore the site remains suitable for continued operation of the OSURR. Infrequency of the occurrence of tornadoes and earthquakes and the robustness of the facility continue to make the site suitable for operation of the OSURR. Hazards related to industrial, transportation, and military facilities will not pose a significant risk to the continued safe operation of the OSURR. The demographics of the area surrounding the reactor have not changed in any way that significantly increases the risk to public health and safety from continued operation of the OSURR.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

The design criteria for structures, systems, and components (SSCs) are that the SSCs related to safe operation and shutdown of the reactor must be able to perform their intended functions as described in the OSURR SAR. The principal safety-related SSCs are the fuel, core support structure, reactor safety system, reactor pool, building exhaust fan, and reactor building. The staff evaluated the following specific design criteria for the above-mentioned SSCs during normal operation and credible accident scenarios:

- The fuel must prevent the release of fission products.
- The core support structure must maintain its orientation, geometry, and structural integrity.
- The reactor safety system must be able to shut down the reactor.
- The reactor pool must provide adequate shielding of radiation emitted from the reactor core and provide physical protection of the reactor core from external events.
- The building exhaust fan must provide an exhaust rate sufficient to control the buildup of argon-41 during normal operations.
- The reactor building must provide a controllable internal environment and protect the reactor from likely detrimental environmental conditions.

With the exception of the fuel, the SSCs mentioned above were designed and constructed in accordance with Construction Permit No. CPRR-49, issued February 3, 1960, on the basis of the Commission's related hazards analysis dated January 13, 1960. The fuel design and construction changed as a result of conversion, by NRC Order dated September 27, 1988, of the reactor core to use low-enriched uranium (LEU) fuel. These SSCs have since been maintained and/or changed using license amendments or licensee review processes, including 10 CFR 50.59, "Changes, Tests, and Experiments," maintenance, and special procedures, as appropriate, in accordance with the Commission's rules and regulations and Facility License No. R-75, as amended. The NRC previously evaluated all amendments to the facility license, and NRC inspections verified proper licensee reviews. In its safety evaluation supporting issuance of Amendment No. 13 dated November 14, 1990, the staff evaluated the licensee's request for an increase in the maximum reactor power level from 10 kW(t) to 500 kW(t). The staff concluded that the SSCs in place, with the exception of the reactor coolant systems, were adequate to support safe operation at the increased power level. The licensee increased the cooling capacity of those systems to support operation at the higher power level. Experience accumulated over the past 17 years of reactor operation supports the staff's conclusion. Chapter 16 of this SER discusses age-related issues. Based on the above, the staff concludes that the design and construction of safety-related SSCs provide reasonable assurance that SSCs will continue to meet the design criteria.

3.2 Meteorological Damage

Section 2.3 of this SER presents the meteorology of the reactor site. Severe storms or tornadoes are possible at the reactor site. However, the thick concrete walls of the reactor pool structure and the water within the pool provide considerable protection to the reactor core from direct wind damage or debris impact. Although very unlikely, damage to the pool wall could cause a loss-of-coolant accident. The licensee analyzed a loss-of-coolant accident and determined there would be no fuel failure and the consequences are bounded by the maximum hypothetical accident (MHA). (Chapter 13 of this SER presents the staff's evaluation of the loss-of-coolant accident and the MHA.) As discussed in Section 8.2 of this SER, there is reasonable assurance that failure of offsite power will not cause damage to the reactor. Local emergency sirens and a weather alert radio provide advanced warning of the approach of severe weather. According to the licensee, this should allow the operations staff to take appropriate actions, in accordance with established procedures, to protect personnel.

3.3 Water Damage

Section 2.4 of this SER presents the hydrology of the reactor site. The reactor site is not within the 500-year flood plain of either the Scioto or Olentangy Rivers, and, consequently, flooding should not pose a threat to the OSURR.

3.4 Seismic Damage

Section 2.5 of this SER presents the seismicity of the reactor site. An earthquake of sufficient magnitude could cause the reactor pool structure to vibrate as a whole. The vibration may lead to dislocations between the reactor pool structure and surrounding structures or damage to the reactor pool walls and pool liner. Any of these effects could lead to a loss of reactor pool water. As stated above, the licensee analyzed a loss-of-coolant accident and determined that there would be no fuel failure and the consequences are bounded by the MHA. If the vibration did not decouple the shim safety rods from their electromagnets and shut down the reactor, operations personnel would manually shut down the reactor. If the reactor did not immediately shut down and was not manually shut down, the loss of a significant quantity of pool water would cause an automatic reactor shutdown. As mentioned above, loss of offsite power also causes a safe reactor shutdown. Reactor operation would not resume until personnel verified the integrity of the reactor pool.

3.5 Systems and Components

The systems and components most important to safety are the reactor safety system and the fuel cladding. Section 4.2.1 of this SER discusses the fuel cladding design requirements, Chapter 13 examines accident scenarios, and Section 16.1 considers aging issues associated with the fuel cladding. These discussions show that the fuel cladding design basis and related TSs are adequate to ensure fuel cladding integrity under all credible circumstances.

The reactor safety system consists of the control rods, control rod electromagnets, and safety-related instrumentation and controls (I&C). Section 4.2.2 of this SER discusses the design requirements of the control rods. Sections 7.3 and 7.4 examine the design requirements of the safety-related I&C. Section 16.1 considers aging issues associated with the reactor safety system. These discussions show that the reactor safety system design bases and related TSs give reasonable assurance that the reactor safety system will function as designed to ensure safe operation and safe shutdown of the reactor.

3.6 Conclusions

On the basis of the above considerations, the staff concludes that the design and construction of the OSURR is adequate to withstand and/or ensure safe shutdown as a result of all credible and likely wind, water, and seismic events associated with the site. The design and performance of safety-related systems and components has been verified through safe operation during the period of the current license and routine NRC inspections. Accordingly, the staff concludes that the reactor systems and components are adequate to provide reasonable assurance that continued operation will not cause significant radiological risk to the health and safety of the public, personnel, and the environment.

4 REACTOR DESCRIPTION

4.1 Summary Description

The OSURR reactor core is composed of a variety of core components including fuel elements and control rods. The core is located near the bottom of the reactor pool, which holds the primary coolant and provides shielding from the radiation emitted by the core. The licensee maintains the configuration of the core to ensure that the reactor can be operated safely and the fuel temperature maintained below the safety limit (SL).

4.2 Reactor Core

4.2.1 Reactor Fuel

The OSURR uses low-enriched uranium (LEU) fuel developed as part of the reduced enrichment for research and test reactors (RERTR) fuel development program of the Argonne National Laboratory (ANL). The NRC previously evaluated and approved the results of the ANL fuel development program with respect to this fuel in NUREG-1313, “Evaluations of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors,” issued in July 1988. In addition to the OSURR, reactors at University of Missouri at Rolla, The Rhode Island Nuclear Science Center, and the University of Massachusetts at Lowell have accumulated considerable safe operating experience with similar fuel.

The OSURR fuel elements consist of fueled plates, unfueled plates, and aluminum side plates. The fueled plates consist of LEU fuel clad in aluminum. The unfueled plates consist of pure aluminum. Standard fuel elements contain fueled and unfueled plates joined at the edges between two aluminum side plates. The fueled plates are located in the central region of the element, while the outside two plates are unfueled. The element is fabricated to provide a space (flow channel) between adjacent plates. The flow channel allows coolant to pass between the plates and to remove heat generated in the fuel plates. Control rod fuel elements are similar in design to the standard fuel elements, except that several fueled plates are removed to provide a space in which the control rod moves. The plates bordering this space are unfueled, and the outer two plates in the element are fueled. Partial fuel elements are identical to the standard fuel elements, except that some of the fueled plates are replaced with unfueled plates. These elements allow the licensee to finely adjust the excess reactivity of the OSURR core.

All fuel elements have the same outer dimensions and have adapters that fit into the lower grid plate of the core support structure. The adapters minimize bypass flow between the fuel elements, and direct the coolant flow into the flow channels between the individual plates of the elements. The elements also have a mechanical feature that interfaces with the grid plates to fix their azimuthal orientation. Each fuel element is labeled with a unique alphanumeric label for positive identification during core loading and inspection of the fuel.

The NRC previously evaluated the OSURR fuel elements in the safety evaluation for Amendment No. 13 to the current license approving a power uprate to 500 kW(t). The staff found the fuel to be acceptable for operation at the increased power and in accordance with the OSURR TSs. No changes in the core operating parameters that would invalidate that safety evaluation are expected during the period of the renewed license. As discussed in Chapter 16 of this SER, the licensee has TSs and procedures to adequately address aging issues associated with the fuel. Additionally, as discussed in Chapter 13 of this SER, no credible accidents could cause fuel temperature to reach the SL specified by TS 2.1, “Safety Limit.” On these bases, the staff concludes that OSURR can safely use the current LEU fuel during the period of the renewed license.

4.2.2 Control Rods

The OSURR uses three shim safety rods to provide coarse control of reactor power. The active neutron absorber in the shim safety rods is boron stainless steel. The shim safety rods are connected to their respective drive mechanisms by electromagnets, and provide a means of rapid negative reactivity insertion for shutdown of the reactor. The regulating rod, which is composed of stainless steel, provides fine control of the reactor power. The control rods fit into the control rod fuel elements described above. The control rods move within housings attached to the top of the control rod fuel elements. Section 7.3 of this SER discusses the shim safety rod and regulating rod control systems.

The control rods and associated drive systems are designed to provide safe and reliable operation of the OSURR. Chapter 4 of the OSURR SAR presents the licensee’s analysis of the requirements for reactivity control systems. This analysis forms the bases for the designs of the control rod systems and TSs related to reactivity requirements for the control rods. TS 3.2.2, “Maximum Reactivity Insertion Rate,” requires that the maximum positive reactivity insertion rate be limited to 0.05% $\Delta k/k/s$. This TS is consistent with TS 3.7.1, “Reactivity Limits,” regarding experiment reactivity limits. Analysis and measurements performed by the licensee show that the control rod systems will not be able to generate a reactivity insertion rate greater than the limit of 0.05% $\Delta k/k/s$ with the control rods in their positions of greatest differential reactivity worth. The licensee analyzed a step reactivity insertion of 0.93% $\Delta k/k$, and found that the fuel temperature does not reach the SL (See Section 13.2 of this SER). TS 4.1, “Reactor Core Parameters,” and TS 4.2, “Reactor Control and Safety Systems,” contain surveillance requirements for the control rods that are consistent with ANSI/ANS-15.1, “The Development of Technical Specifications for Research Reactors,” issued 1990. Based on these considerations, the staff concludes that the control rod systems and related TSs provide reasonable assurance that the control rod systems will allow safe and reliable operation of the OSURR.

4.2.3 Neutron Moderation and Reflection

The coolant surrounding and in the reactor core primarily provides neutron moderation. Neutron reflection is provided by reactor-grade graphite, encased in watertight aluminum shells, and the coolant surrounding the reactor core. The graphite reflectors are located immediately to the south and west of the core. These reflectors extend to the thermal columns. Coolant provides neutron reflection on the north and east sides of the core. When installed in the core, the graphite isotope irradiation elements (GIIEs) provide additional reflection. The licensee’s analysis of nuclear operating characteristics of the OSURR core incorporates the moderator and reflector. The licensee analyzed the buildup of neutron-induced Wigner energy in the graphite in the reflectors. The licensee concluded that buildup of significant Wigner energy in the graphite will not occur during the period of the renewed license, assuming that reactor utilization is consistent with that during the current license. The staff finds the licensee’s analysis and conclusion acceptable. The licensee has conducted inspections of the watertight aluminum shells and found no evidence of degradation or leakage. The staff finds that the analysis and inspections provide reasonable assurance that the reflectors can continue to perform as designed and will not pose a significant risk to the OSURR during the period of the renewed license.

4.2.4 Neutron Startup Source

The OSURR uses a plutonium-beryllium source to provide an initial population of neutrons for reactor startup and sufficient to verify operation of the startup channel. A cadmium-lined housing provides storage and neutron shielding for the source during reactor operation at power. According to the licensee, the source has been successfully used during the period of the current license and is expected to continue to provide the required neutron emission rate during the period of the renewed license.

4.2.5 Core Support Structure

The OSURR core support structure consists of the grid plate and grid plate frame, which are constructed from aluminum. The grid plate frame is secured to pedestals mounted on the bottom of the reactor pool. The grid plate frame holds the grid plate in an elevated position to allow coolant to pass under the grid plate and flow up into the fuel elements. The grid plate is attached at the top of the grid plate frame. The grid plate contains a rectangular array of through-holes to accommodate the adapters located at the bottom of the fuel elements and other core components. Each grid plate position contains a dowel pin that interfaces with the fuel elements to fix their azimuthal orientation. TS 5.3, "Reactor Core and Fuel," specifies the design features of the grid plate and core components that may occupy the grid plate positions. The OSURR core support structure is similar in design to those found in other pool-type research reactors. The staff evaluated the design and features of the core support structure and found that it is constructed of appropriate material, provides for adequate support of the core components, provides features for reproducible positioning of core components, and provides negligible resistance to coolant flow to the fuel element adapters.

4.3 Reactor Pool

The reactor pool serves as the source of coolant for natural-convection cooling and as shielding from radiation emitted by the core, in-core experiments, and the nitrogen-16 control system decay tank. The pool holds approximately 22,000 L (5800 gal) and maintains roughly 4.5 m (15 ft) of water above the top of the reactor core for radiation shielding. The pool walls are constructed of barytes concrete to a height of 4.11 m (13.5 ft). The remaining 2.0 m (6.5 ft) of the pool walls are constructed of regular concrete. Experimental facilities penetrate the reactor pool walls in several locations. At these locations, welded steel plates or epoxy seals maintain the integrity of the reactor pool. The pool is completely lined with fiberglass-reinforced epoxy paint to prevent water leakage, or leaching of the materials in the concrete by the pool water, and to facilitate decontamination and repair of the walls. The licensee conducts inspections and repairs of the pool liner and experimental facility penetrations. Based on repairs performed during the period of the current license, the licensee predicts that pool liner repairs will be required approximately every 5 years. The licensee has not observed degradation of the welded steel plates. According to the licensee, reactor pool leaks would be detected by visual observation, increased makeup water usage, or the pool water level measuring channel of the reactor safety system. Chapter 13 of this SER discusses loss-of-coolant accidents. Chapter 5 of this SER discusses other considerations related to the reactor coolant systems.

4.4 Biological Shield

The OSURR biological shield comprises the reactor pool walls, the coolant in the reactor pool, and shielding for the experimental facilities that penetrate the reactor pool walls. Shielding for the pool wall penetrations includes lead, graphite, polyethylene, aluminum, and concrete. The reactor pool walls and experimental facility shielding provide the majority of shielding in the horizontal direction. The coolant in the reactor pool provides the majority of shielding above the reactor core. Chapter 11 of this SER discusses radiation protection and evaluates the effectiveness of the biological shield in reducing direct radiation exposure from the reactor core.

4.5 Nuclear Design

4.5.1 Normal Operating Characteristics

The OSURR core design objectives are to maintain no more than the core excess reactivity limit (TS 3.1.1, "Reactivity"), no less than the minimum shutdown margin (TS 3.1.1), minimize the power peaking in the fuel, and maximize the thermal neutron flux at the central irradiation facility. The licensee analyzed a variety of core configurations using different combinations and arrangements of core components to determine the effects of core configuration on neutron flux profiles, control rod worth, and excess reactivity. The OSURR core components include standard fuel elements, control rod fuel elements, partial fuel elements, GIIEs, blank fuel elements, shim safety rods, and the regulating rod. The licensee used the LEOPARD and 2DB computer codes, and benchmarked the results against several other codes and data from previous years of reactor operation. The analyses appropriately included the effects of the neutron moderator and reflector materials external to the core. Inputs to these analyses included relevant reactor parameters listed in Table 4.1 of the OSURR SAR. The licensee's analyses show that a variety of core configurations can satisfy the design objectives and the requirements of TS 3.1.1 and TS 3.2.2. The licensee also analyzed the reactivity effects of individual core components. The licensee found that removal of GIIEs would reduce the core excess reactivity, as would removal of fuel elements. As presented in Section 4.2.2 above, the licensee found that the dynamic effects of the control rods will not lead to any reactor transient greater than that analyzed in the reactivity insertion accident scenario.

The staff reviewed the licensee's analyses and found that the licensee considered an appropriate variety of core configurations and that these core configurations contained the components required for an operable reactor core. The staff found that the licensee used input parameters justified by analyses presented in the OSURR SAR. The staff found that the licensee adequately analyzed the reactivity effects of individual core components. TSs related to the normal operating conditions of the reactor core include reactivity limits on excess reactivity, minimum shutdown margin, and maximum reactivity insertion rate; allowable core configurations; and surveillance requirements for the core reactivity parameters and reactivity worth of the control rods. These TSs are consistent with ANSI/ANS-15.1. The staff found that the analyses presented in the OSURR SAR adequately justify these TSs and show that normal reactor operation will not lead to the release of fission products from the fuel. Based on these considerations, the staff concludes that the licensee has adequately analyzed expected normal reactor operation during the period of the renewed license. The staff further concludes that the TSs provide reasonable assurance that normal operation of the OSURR core will not pose a significant risk to the health and safety of the public or the environment.

4.5.2 Reactor Core Physics Parameters

The licensee performed analyses of the reactor core physics parameters using a "reference core." The reference core represents the core configuration that best satisfies the core design objectives stated in the previous section. The licensee analyzed the reactivity effects of void formation in the moderator. The analysis showed that the void coefficient of reactivity could range from -0.18% Δk/k/% void to -0.45% Δk/k/% void. The licensee analyzed a maximum reactivity insertion accident using the more conservative void coefficient of -0.18% Δk/k/% void and found that no fuel damage would result. TS 3.1.1 requires that the moderator void coefficient of reactivity have a maximum value of -0.28% Δk/k/% void, which is less than that used in the accident analysis, and thus is conservative. The licensee performed measurements on the current OSURR core that indicated a moderator void coefficient of reactivity of -0.79% Δk/k/% void, which satisfies the limit specified by TS 3.1.1.

The licensee analyzed the reactivity effects of an increase in the moderator temperature. The analysis yielded an estimated temperature coefficient of reactivity of $-6.3\times10^{-3}\%$ Δk/k/°C. The licensee performed measurements on the OSURR core that indicated a moderator temperature coefficient of reactivity of $-6.2\times10^{-3}\%$ Δk/k/°C. TS 3.1.1 requires that the moderator temperature coefficient of reactivity have a maximum value of $-2\times10^{-3}\%$ Δk/k/°C. The licensee's analysis of a reactivity insertion accident supports this TS limit.

Table 4.1 of the OSURR SAR presents a summary of reactor data. The data that were used in the licensee's analyses are evaluated elsewhere in this SER. These data include the result of analyses and measurements and the physical characteristics of the reactor. The staff evaluated these data and found that they were justified by analyses or descriptions contained in the SAR. The values of the core physics parameters are similar to those estimated and measured for similar research reactors.

TSs related to the reactor core physics parameters include limits on the moderator void coefficient of reactivity and moderator temperature coefficient of reactivity and the requirement that both coefficients be negative across the active core. These TSs are consistent with ANSI/ANS-15.1. The staff found that the analyses presented in the OSURR SAR adequately justify these TSs. The staff concluded that these TSs provide reasonable assurance that the void and temperature reactivity coefficients will offer additional safety during normal reactor operations and transients.

4.5.3 Operating Limits

The licensee analyzed the excess reactivity necessary to allow the reactor to operate at the licensed maximum power of 500 kW(t). The licensee's analysis considered xenon poisoning, temperature feedback, experiment requirements, fuel burnup, and control margin to generate reasonable reactor periods. The licensee did not consider the moderator void coefficient of reactivity, since no void formation is expected during normal reactor operation. This is a conservative approach because void formation would cause a negative reactivity insertion. The licensee's analysis showed that xenon poisoning and temperature feedback effects would require approximately 1.2% Δk/k in excess reactivity to maintain the reactor critical. Assuming a conservative operating history, burnup reactivity effects would require an additional 0.5% Δk/k to reasonably reduce the frequency of core configuration changes. The maximum reactivity associated with experiments, as specified in TS 3.7.1, requires 0.7% Δk/k. The licensee's analysis showed the need for 0.2% Δk/k in additional excess reactivity to generate a 30-second positive reactor period. These contributions amount to a maximum required excess reactivity of 2.6% Δk/k, which is the upper limit on excess reactivity specified by TS 3.1.1. The licensee measured the current excess reactivity and found it to be 1.35% Δk/k, which meets the TS requirement. The staff evaluated the sources of the individual contributions to excess reactivity and found them to be acceptable on the basis that they are valid sources of negative reactivity during reactor operation. The staff evaluated the magnitudes of the individual contributions to excess reactivity and found them to be adequately justified in the OSURR SAR.

The licensee analyzed the minimum shutdown margin required to assure adequate shutdown performance. The licensee conservatively assumed that the most reactive shim safety rod and the control rod are stuck in the full-out position. This assumption satisfies the "stuck rod" criterion found in the guidance in NUREG-1537 and ANSI/ANS-15.1. TS 3.1.1 requires a minimum shutdown margin of 1.0% $\Delta k/k$. The licensee's analysis showed a shutdown margin of 4.55% $\Delta k/k$ for the reference core. The licensee measured the current shutdown margin and found it to be 3.46% $\Delta k/k$, which satisfies the limit specified by TS 3.1.1.

TS 4.1.1, "Excess Reactivity and Shutdown Margin," requires the core excess reactivity and shutdown margin to be measured at least annually, and whenever there is a significant change in core configuration. TS 4.1.1 is consistent with ANSI/ANS-15.1, and the staff finds that it is acceptable.

4.6 Thermal-Hydraulic Design

Sections 4.8 and 4.9 of the OSURR SAR present the thermal-hydraulic characteristics of the OSURR. The OSURR thermal-hydraulic analysis is based on natural convection cooling of the fuel plates. Using conservative assumptions and the limiting conditions allowed by the TSs, the licensee calculated a maximum fuel plate surface temperature in the hot channel of 96 °C (205 °F), which is well below the SL specified in TS 2.1 of 550 °C (1020 °F). This surface temperature corresponds to a channel outlet temperature of 80 °C (180 °F). Operational data from the OSURR indicate a nominal core outlet temperature near 50 °C (120 °F). The staff evaluated the licensee's calculation methods and assumptions and found them to be conservative. The licensee appropriately used the limiting safety system setting value of 35 °C (95 °F) for the core inlet temperature specified by TS 2.2, "Limiting Safety System Settings." In the event that the fuel plate surface temperature was to lead to the formation of steam voids in the reactor coolant, the negative void reactivity coefficient would cause reactor power to decrease.

4.7 Conclusions

Based on the above considerations, the staff concludes that the licensee has presented adequate information and analyses to demonstrate the technical ability to configure and operate the OSURR core without undue risk to the health and safety of the public or the environment. The staff concludes that the OSURR TSs regarding the reactor design, reactor core components, reactivity limits, and related surveillance requirements provide reasonable assurance that the reactor will be operated safely in accordance with the TSs during the period of the renewed license.

5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The OSURR is a pool-type reactor that uses high-purity, demineralized light water as the primary coolant. Natural convection drives the coolant flow through the core. A forced-flow primary system transfers heat from the pool water to the secondary system. The secondary system uses a mixture of water and ethylene glycol to transport heat to a fan-forced dry cooler or a city-water-cooled heat exchanger. The total cooling capacity is adequate to remove the heat produced during extended periods of full-power operation under all meteorological conditions. Natural convection cooling is sufficient to remove decay heat following reactor shutdown. A primary coolant cleanup system and a primary coolant makeup water system, respectively, maintain the purity and inventory of the primary coolant.

5.2 Primary Coolant System

Natural-convection cooling is the primary means of removing heat generated in the reactor fuel during operation and after shutdown. The natural-convection flowpath comprises the reactor pool, the fuel element flow channels, and a plenum located above the core. Coolant contained in the reactor pool enters the bottom of the core, moves upward through the fuel element flow channels as it is heated by the fuel plates, and exits the top of the core into the plenum. A hole at the top of the plenum allows the heated coolant to mix with the bulk pool water if the forced-flow primary system is not operating. The reactor pool serves as the source of coolant for natural-convection cooling and as shielding from radiation emitted by the core, in-core experiments, and the nitrogen-16 control system decay tank.

The OSURR uses a forced-flow primary system to control the bulk temperature of the reactor pool and consequently the core inlet temperature. The system was a necessary facility modification for receipt of Amendment No. 13, which authorized operation at the current maximum steady-state reactor power. The system comprises the plenum located above the core, a nitrogen-16 control system decay tank, a centrifugal pump, a plate-and-frame-type heat exchanger, and associated piping and valves. Heated primary coolant exiting the core collects in the plenum above the core. The suction from the pump draws this coolant through the decay tank and forces it through the primary side of the heat exchanger. The cooled primary coolant is returned to the reactor pool in a manner that disperses any heated coolant that leaves the plenum through the hole in the top of the plenum.

The flow rate of the forced-flow system is controlled such that the natural-convection flow through the core is not significantly disturbed. The pump draws coolant from two sides of the plenum to minimize localized buildup of heated coolant at any point within the plenum. The outlet leg of the forced-flow primary system contains a siphon breaker to prevent siphoning of the reactor pool should a significant leak develop in the system. Primary system leaks would be detected by visual observation of coolant spills in the reactor building or instrument indication of an unusual decrease in the reactor pool coolant level. Cleanup and sampling of the primary coolant ensures that leaks to the reactor building or secondary coolant system will not result in undue radiological hazards to facility staff, the public, or the environment.

TSs related to the primary coolant system include a limiting safety system setting (LSSS) of 35 °C (95 °F) on the core inlet temperature (TS 2.2); a requirement that the “reactor thermal power level/coolant system pumps,” “core inlet temperature,” “coolant flow rate,” and “pool water level" safety system channels be operable during reactor operation (TS 3.2.3, "Minimum Number of Scram Channels"); requirements for operation of the system pump, reactor pool coolant level, coolant conductivity and pH, loss-of-coolant detection, and coolant radioactivity (TS 3.3, "Coolant System"); system and system instrumentation surveillance requirements (TS 4.2.2, "Reactor Safety System," and TS 4.3, "Coolant System"); and system design features (TS 5.2.1, "Primary Coolant Loop"). These TSs ensure that the primary system can perform all intended functions as described in the OSURR SAR. The licensee provided adequate analyses to support these TSs as they apply to the primary coolant system.

5.3 Secondary Coolant System

The secondary coolant system is a closed system that uses a water and ethylene glycol mixture to transfer heat from the primary system to an ultimate heat sink. The system comprises the secondary side of the primary heat exchanger, a variable speed positive-displacement pump, a surge tank, a fan-forced air cooling unit (dry cooler), a city-water-cooled secondary heat exchanger, and associated piping and valves. The pump allows adjustment of the flow rate to optimize the system efficiency for varying conditions. The system pump discharges coolant through the secondary side of the primary heat exchanger. The heated coolant flows through the surge tank, which accommodates changes in system pressure and serves as a point for addition of secondary coolant to the system. The coolant then passes through the dry cooler or a bypass leg, depending on the specific cooling requirements of reactor operation. These two flow paths merge downstream from the dry cooler before the coolant is drawn through the secondary heat exchanger by the pump suction. A system pump bypass leg contains a filter to limit the buildup of corrosion products in the secondary system.

The secondary coolant system has two methods for dissipating heat transferred from the primary system. Under most atmospheric conditions, the dry cooler is sufficient to maintain a steady reactor pool bulk temperature. The location of the dry cooler is outside the reactor building, near the east wall. This unit uses eight fans to transfer heat from the secondary coolant to the atmosphere. The fans do not have any cycling capability. Varying the total secondary coolant flow rate and the secondary coolant flow rate through the dry cooler provides fine control of the heat dissipated by the secondary coolant system. High outdoor air temperature or humidity may require additional cooling capacity. The city-water-cooled secondary heat exchanger provides that capacity. City water flows through a backflow preventer, through the heat exchanger, and into a floor drain on the main floor of the reactor building. The licensee refers to this flow path as the "tertiary loop." The heat exchanger design provides the necessary additional cooling capacity, while preventing city water discharge to the floor drain that exceeds local temperature limits.

The secondary coolant mixture uses lithium tetraborate as a buffer to reduce corrosion in the secondary system. The boron contained in this buffer ensures that any leak of secondary coolant into the primary system will result in the addition of negative reactivity to the reactor core. Sampling for radioactivity ensures that leaks from the secondary system to the reactor building or dry cooler area will not result in radiological hazards to facility staff, the public, or the environment.

TSs related to the secondary cooling system include a requirement that the "reactor thermal power level/coolant system pumps" safety channel be operable during reactor operation (TS 3.2.3), requirements for operation of the system pump and coolant radioactivity (TS 3.3), system and system instrumentation surveillance requirements (TS 4.2.2 and TS 4.3), and system design features (TS 5.2.2, “Secondary and Tertiary Coolant Loops”). These TSs ensure that the secondary system can perform all intended functions, as described in the OSURR SAR.

5.4 Primary Coolant Cleanup System

The primary coolant cleanup system maintains the purity of the primary coolant to reduce corrosion of reactor components. The system also filters out activation products to protect OSURR staff from undue radiation hazards and minimize the consequences of a primary coolant spill or leak. The system comprises particulate filters, ion-exchange cartridges, a pump, and associated piping and valves. The pump suction draws coolant from the reactor pool through the particulate filters and discharges it through the ion-exchange filters and into the reactor pool. An identical system maintains the purity of the bulk shielding facility pool.

TSs related to the primary coolant cleanup system include limits on the coolant conductivity and pH (TS 3.3) and surveillance requirements (TS 4.3). These TSs ensure that the functionality of materials in contact with the primary coolant will not be significantly degraded by less than optimal coolant chemistry, and that any deficiency in the coolant chemistry will be identified and corrected in a timely manner.

5.5 Primary Coolant Makeup Water System

The primary coolant makeup water system comprises city water supply lines, a series of filters, a carbon filter, two mixed-bed ion exchange cartridges, and associated piping and valves. City water passes through the filters, carbon filter, and mixed-bed ion exchange cartridges before being manually added to the reactor pool using a fill valve. The system also contains outlets for use elsewhere in the reactor building. This system is not intended to provide core cooling in the case of an emergency.

5.6 Nitrogen-16 Control System

The OSURR nitrogen-16 control system is of a type that is commonly and effectively used at TRIGA reactors. The system is necessary to allow the nitrogen-16 in the primary coolant to decay before the coolant reaches the surface of the reactor pool or enters unshielded primary system piping. The OSURR system comprises the plenum above the core, a decay tank, tank inlet and outlet piping, and the forced-convection primary system coolant return outlet. The nitrogen-16 control system is an integral part of the forced-convection primary system.

The tank inlet piping routes nitrogen-16-rich coolant from the plenum to the decay tank. This tank delays the coolant for a time equal to approximately 11 half-lives of nitrogen-16 (approximately 80 seconds), which is adequate to sufficiently reduce radiation levels at the reactor pool surface. The decay tank is vertically mounted in the reactor pool near the southeast corner and rests on a type 304 stainless steel base plate on the bottom of the pool. The decay tank is made of type 304 stainless steel and is insulated to minimize heat transfer between the heated coolant inside the tank and the cooler water in the reactor pool. The tank outlet piping routes the coolant to the forced-convection primary system pump. The forced-convection primary system coolant return outlet returns primary coolant to the reactor pool in a manner that disperses any heated coolant that is not drawn into the tank inlet piping. This prevents a thermal plume of nitrogen-16-rich coolant from reaching the reactor pool surface.

TSs related to the nitrogen-16 control system include requirements for operation of the forced-flow primary system pump when reactor power is greater than 120 kW(t) (TS 3.3), operation of radiation detection equipment at the reactor pool top (TS 3.6.1, “Radiation Monitoring”), and surveillance requirements for the radiation detection equipment (TS 4.6.2, "Area Radiation Monitors (ARMs)"). These TSs provide reasonable assurance that no undue radiation hazard, resulting from the nitrogen-16 produced during reactor operation, is present at the reactor pool top.

5.7 Conclusions

The design and construction of the OSURR cooling systems provide adequate cooling capacity to remove the heat generated during extended periods of full-power reactor operation under all meteorological conditions. The systems contain sufficient features to protect personnel from excessive radiation hazards, minimize corrosion of core components and fuel, prevent or detect losses of coolant, and provide for reactor operation within the related limits of the facility license and TSs.

6 ENGINEERED SAFETY FEATURES

6.1 Summary Descriptions

6.1.1 Reactor Building Ventilation

Section 9.1 of this SER describes the reactor building ventilation. The licensee analyzed the radiological consequences of a maximum hypothetical accident (MHA) (SAR Section 8.4.4.4). The analysis shows that, with the building ventilation secured, doses to personnel and the public will be below the limits specified in 10 CFR Part 20. Because this system will not be relied upon to mitigate the consequences of such an accident, the staff does not consider the reactor building ventilation to be an engineered safety feature.

6.1.2 Emergency Core Cooling System

The OSURR does not have an emergency core cooling system. The licensee analyzed a hypothetical instantaneous loss-of-coolant accident (SAR Section 8.4.2.1). The analysis shows that this accident will not lead to fuel damage and that the consequences of this accident are bounded by the MHA.

6.1.3 Building Evacuation System

The reactor operator can activate the building evacuation klaxon to alert any persons in the reactor building of the need to evacuate. In the case of the MHA, prompt evacuation would minimize doses to personnel, but would not be necessary to keep doses to personnel below the limits specified in 10 CFR Part 20. If the evacuation klaxon failed to perform its function, the reactor operator could sweep the reactor building and alert personnel of the need to evacuate. Therefore, the staff finds that the evacuation klaxon is not required as an engineered safety feature.

6.2 Conclusions

Section 13.1 of this SER contains the staff's evaluation of the MHA. The staff found that the radiological consequences of the MHA would be below regulatory limits, given that the reactor building ventilation system was secured. Accordingly, the staff concludes that the OSURR does not require any engineered safety features.

7 INSTRUMENTATION AND CONTROL

7.1 Summary Description

The OSURR instrumentation and control (I&C) systems provide functions for monitoring and controlling the reactor during normal operation, shutdown, and abnormal conditions. The I&C systems include the control console, reactor control system (RCS), reactor protection system (RPS), coolant system I&C, auxiliary controls, and area radiation monitoring system. The instrumentation provides indication of process variables, reactor core nuclear parameters, radiation levels at various locations in the reactor building, effluent activity levels, alarms, and other parameters necessary to allow safe operation and shutdown of the reactor and protection of personnel. According to the licensee, the control systems provide flexible and reliable control of the reactor during all regimes of operation and shutdown. The OSURR I&C systems are functionally similar to those that provided reliable operation for many years at similar research reactors at Oak Ridge National Laboratory and various universities.

7.2 Design of Instrumentation and Control Systems

The OSURR I&C systems are designed to allow reliable and safe operation and shutdown of the reactor during normal reactor operation and abnormal conditions. The I&C systems consist of nuclear and nonnuclear instrumentation; signal processing equipment; displays and recorders of instrument output signals; controls and electric drive motors for positioning the shim safety rods, regulating rod, startup channel fission chamber, and startup neutron source; local and remote controls for pumps, valves, fans, and other support equipment; power supplies; and cabling. The I&C systems contain appropriate interlocks, redundancy, and common-mode failure protection, and they are failsafe.

The I&C systems design bases are derived from the analysis presented in the OSURR SAR and are specified in TS 3.1, "Reactor Core Parameters," TS 3.2, "Reactor Control and Safety System," and TS 3.6, "Radiation Monitoring Systems and Radioactive Effluents." The design bases are consistent with the licensee's analyses of core operating parameters and postulated accidents and are therefore acceptable. The TSs contain surveillance intervals for components and functions of the I&C systems. These intervals are consistent with the guidance found in ANSI/ANS-15.1 and the intervals used at similar research reactors. The staff finds that the specified intervals provide reasonable assurance that I&C component failure and degradation will be detected in a timely manner, and that specified calibration frequencies are adequate to prevent significant drift in instrument setpoints and detection ranges.

The current OSURR I&C systems designs are not being changed as part of the license renewal process. Based on the above considerations and years of safe operation with the current systems, the staff finds the I&C designs to be acceptable for operation of the OSURR during the period of the renewed license. Chapter 16 of this SER discusses aging issues related to the I&C systems.

7.3 Reactor Control System

Sections 3.3, 3.4, and 3.5 of the OSURR SAR describe the RCS. The primary functions of the RCS are to provide the reactor operator the information and control capability necessary to safely operate and shut down the reactor. The RCS includes functions for controlling and monitoring reactor power and period, cooling systems, the ventilation system, experimental facilities, and radiation detection systems. I&C for these systems are provided in the control room and in some cases locally.

Sections 3.3.12 and 3.3.13 of the OSURR describe the logarithmic power monitoring channel (Log-N) and linear power monitoring channel. Each channel consists of a detector, high-voltage and compensating-voltage power supply, signal amplifier, local and remote meters, and a recorder. These monitoring channels use gamma-compensated ionization chambers to detect reactor neutron power. This type of detector is appropriate for neutron power monitoring and is widely used in similar research reactors. The remote meters and the recorder are located in the control room. The Log-N channel also provides indication of the reactor period and a reactor period recorder. The Log-N and linear power monitoring channels are independent and redundant. The combination of a logarithmic power monitoring channel and a linear power monitoring channel is common for research reactors. The staff evaluated these channels and found that they have sufficient range and sensitivity to detect reactor power and period over all regimes of operation analyzed in the OSURR SAR. During a site visit, the staff observed the meters and recorders and found that they provide the types of information necessary to allow the reactor operator to safely and reliably control the reactor power.

Coarse reactivity control of the reactor is achieved by changing the height of the shim safety rods relative to the bottom of the core. Fine reactivity control is accomplished by changing the height of the regulating rod relative to the bottom of the core. Section 4.2.2 of this SER discusses the design and reactivity requirements of the control elements. Manual switches on the reactor control console allow the reactor operator to manipulate the shim safety rods and regulating rod. The switches activate the associated shim safety rod or regulating rod drive motor. A mechanical interlock ensures that only one shim safety rod can be moved in the upward direction at any time. This provides reasonable assurance that the total reactivity insertion rate will be consistent with that analyzed in the OSURR SAR and within the limit specified in TS 3.2.2. Dial position indicators and positioning system indicator lamps are located on the control console. These displays provide the reactor operator with information regarding the position and status of each shim safety rod and the regulating rod. The staff evaluated the reactivity control systems against the design bases specified in the OSURR TSs. The staff found that these systems are capable of providing reactivity control within the limits specified in the TSs, including the reactivity insertion rate.

Additional I&C are provided for startup of the reactor. A moveable startup neutron source, discussed in Section 4.2.4 of this SER, provides an initial population of neutrons in the reactor core. The startup source is positioned using controls similar to those described for the regulating rod. The startup source controls and position indicator are located on the control console. A startup channel provides instrumentation for startup and low-power operation of the reactor. The channel consists of a moveable neutron detector, high-voltage power supply, signal amplifier, discriminator, digital timer/counter, linear-log rate meter, and a recorder. The neutron detector is a fission chamber that is sensitive to low neutron flux. The detector is positioned using controls similar to those described for the regulating rod. The detector controls and position indicator are located on the control console. An interlock prevents simultaneous upward motion of the detector and the shim safety rods. This ensures that the detector will not be moved into a region of lower neutron flux during the addition of positive reactivity to the core. The staff evaluated the startup I&C against the requirement of TS 3.2.3. This TS requires the reactor to automatically shut down, or remain shut down, if the startup channel count rate is less than or equal to 2 counts per second (cps). Core loading experiments may require the withdrawal of control rods when the startup channel count rate is less than 2 cps. In these cases, the reactor operator may bypass the automatic reactor shutdown if the core cannot achieve criticality with the control rods withdrawn (core keff < 0.9). Strict administrative controls provide reasonable assurance that the reactor cannot achieve criticality with the bypass in place. The staff found that the startup instrumentation is of the type capable of detecting low neutron count rates and commonly used in similar research reactors for startup channels.

Chapter 5 of this SER discusses the reactor cooling system and system requirements. Sections 3.2.2.3 and 3.4 of the OSURR SAR provide a description of the cooling system I&C. Cooling system instrumentation displays and system controls are provided in the control room. Some system instrumentation also provides local displays. Instrumentation is provided for coolant system process variables, including coolant temperatures, pressures, flow rates, valve positions, and component actuation and operation. Controls are provided for system components such as pumps, fans, and valves. The staff evaluated the cooling system I&C and found them to be adequate to allow the reactor operator to assess current system conditions and make appropriate adjustments to system operation. The staff also found that the instrumentation provides process variable information for the locations of greatest interest in the system.

Sections 9.1 and 6.1.1 of this SER discuss the reactor building ventilation system. Ventilation system controls are provided in the control room. These controls allow the reactor operator to initiate and secure building ventilation.

Chapter 10 of this SER discusses the experimental facilities, including the controls for the pneumatically-operated irradiation facility or "rabbit system." Experimental facilities are generally controlled at or near the access point to the experimental facility. Indicators located on the control console provide the reactor operator with information about the status of the facilities or components of the facilities. Examples include the position of the thermal column door and the status of the beam port shutters. These indicators provide reasonable assurance that the reactor operator will be cognizant of the use of the experimental facilities and can take appropriate action if the status of a facility changes unexpectedly. A permit switch located in the control room provides the reactor operator with ultimate control of the rabbit facility.

Section 7.7 of this SER discusses the radiation monitoring systems.

7.4 Reactor Protection System

The primary function of the RPS is to automatically shut down the reactor, or "scram" the reactor, in the case of abnormal conditions or unanticipated transients. The RPS functions by interrupting current flow to the shim safety rod electromagnets that connect the shim safety rods to their associated drives. With the electromagnets de-energized, the shim safety rods drop into the core due to gravity, thereby shutting down the nuclear reaction. Interruption in the current is caused by opening a relay or biasing a current-controlling element in the circuit connecting the electromagnet power supplies to the electromagnets. The RPS receives input signals from many of the RCS systems and components. Two safety channels also provide inputs to the RPS. Both of these channels monitor reactor power and one monitors reactor period as well. The channels are similar in design to the RCS power-level monitoring channels described in the previous section, except that the detectors are uncompensated ionization chambers. The staff has found this type of detector acceptable for use in safety channels at other research reactors. Because there are no unique conditions at the OSURR that would preclude their use, the staff concludes that these detectors are appropriate. Additionally, the two channels provide simple redundancy, as described in ANSI/ANS-15.15, "Criteria for the Reactor Safety Systems of Research Reactors," issued 1978. The RPS is designed to fail safe in the case of a malfunction or failure of individual components or the loss of power to individual components or the system. The RPS also includes manual scram switches located in the control room and at other appropriate locations within the reactor building.

The secondary function of the RPS is to provide alarms that alert the reactor operator to abnormal conditions and the reason for RPS activation. The alarms are located on the reactor console. The alarms annunciate audibly and visually and will activate the building evacuation klaxon if unacknowledged within a set time to alert personnel to the alarm condition.

TSs related to the RPS include the limiting safety system settings (LSSSs) for the core inlet water temperature and reactor thermal power (TS 2.2), a limit on the maximum scram time (TS 3.2.1, “Control Rod Scram Time”), minimum number of scram channels (TS 3.2.3), and system and component surveillance requirements (TS 4.2.1, “Control Rods,” and TS 4.2.2). The staff evaluated the RPS against the requirements contained in these TSs and the associated bases analyzed in the OSURR SAR. The staff also evaluated the RPS surveillance requirements using ANSI/ANS-15.1. The staff found that the LSSS is consistent with the licensee’s analyses of core operating parameters and postulated accidents and will preclude reaching the safety limit (SL) on fuel temperature specified by TS 2.1. This satisfies the requirement specified in 10 CFR 50.36(c)(1)(ii)(A) for the LSSS. As stated in Section 13.2 of this SER, the SL will not be reached even if the RPS fails to automatically shut down the reactor. The staff also found that the types of RCS components that provide input to the RPS are consistent with those at similar research reactors and are diverse and redundant. Based on consistency with ANSI/ANS-15.1, the staff found the RPS surveillance intervals to be acceptable.

7.5 Engineered Safety Features Actuation Systems

As described in Chapter 6 of this SER, the OSURR does not have any required engineered safety features. In the event of an accident with radiological consequences similar to those of the maximum hypothetical accident (MHA), the reactor operator would secure the building ventilation system. A master switch in the control room can be used to secure all building ventilation.

7.6 Control Console and Display Instruments

The OSURR control console comprises three main panels. The center panel contains the controls and position indicators for the control elements, startup source, and startup channel. Analog meters located above this panel indicate the logarithmic and linear power monitoring channels and the reactor period. The left panel contains the RPS indicators and controls and the alarm annunciators. The right panel contains the linear power monitoring channel indication and auxiliary systems controls and indicators. The staff compared the general arrangement and types of controls and displays provided by the control console to those at similar research reactors and found that the designs are similar. The staff observed the control console during a site visit and found that the control console provides the reactor operator with the types of information and controls necessary to facilitate reliable and safe operation of the reactor.

7.7 Radiation Monitoring Systems

Area radiation monitors (ARMs) provide local and remote alarms and indications of radiation levels at four locations within the reactor building. The remote alarms and indicators are located in the control room. Alarm trip points are nominally set at 0.1 millisievert per hour (mSv/h) (10 millirem per hour (mrem/h)) to 0.3 mSv/h (30 mrem/h) and are adjusted using controls located in the control room. The ARM system uses Geiger-Mueller detectors. These detectors are of the type usually used for area radiation monitoring and the staff finds them appropriate for use at the OSURR. The individual ARM detectors are located above the reactor pool, opposite the thermal column and beam ports, near the primary coolant loop heat exchanger, and next to the water processing system. The staff evaluated these locations and found that they are the locations where radiation levels would be expected to be highest during normal reactor operation. The staff also found that these locations have the greatest chance of increased radiation levels from experiments and abnormal conditions such as a loss of coolant or an increase in activation products in the primary coolant. Based on the above findings, the staff concludes that the ARM system is adequately designed to provide information about the magnitude of the radiation fields of greatest interest in the reactor building and to alert personnel to the existence of any abnormally elevated radiation fields. TS 3.6.1 contains a requirement that the ARM system be operating and have a read-out and alarm in the control room whenever the reactor is operating. TS 4.6.2 gives surveillance requirements and intervals for the ARM system. These TSs provide reasonable assurance that the ARM system will be capable of performing its intended function, as described in the OSURR SAR.

A gaseous effluent monitoring system samples the air exhausted by the building exhaust fan. The detector system consists of a shielded volume equipped with a pancake-type Geiger-Mueller detector and a compressor to draw air into the shielded volume. Detector response information is transmitted to a rate meter and time trace display located in the control room. These provide the reactor operator with the current effluent activity and maintain a record of facility gaseous effluents for comparison with regulatory limits. Chapter 6 of the OSURR provides analysis of the expected gaseous effluents for routine operation of the OSURR. The licensee identified argon-41 as the only significant radionuclide expected in the effluent stream. Accordingly, the gaseous effluent monitoring system response is calibrated for argon-41. Chapter 11 of this SER presents the staff's evaluation of the licensee's analysis of expected gaseous effluents. The staff evaluated the location and design of the gaseous effluent monitoring system and observed its operation during a site visit. The staff finds that the system uses an appropriate type of radiation detector for monitoring argon-41. Additionally, the staff finds that shielding the detector is appropriate for eliminating background radiation that could artificially increase the amount of effluent detected by the system. The system samples air from the effluent stream as the effluent is ejected from the reactor building, thus providing an appropriate radiological characterization of the effluent. Based on its observations at the facility, the staff concluded that the control room rate meter and time trace display provide adequate information to keep the reactor operator cognizant of the effluent being released and to provide an adequate record of the effluent being released. TS 3.6.1 contains a requirement that the gaseous effluent monitoring system be operating and have a read-out and alarm in the control room whenever the reactor is operating. TS 4.6.1, "Effluent Monitor," gives surveillance requirements and intervals for the system. TS 3.2.3 requires the RPS to initiate an automatic reactor shutdown if the system compressor is secured or power is lost to the time trace display. These TSs provide reasonable assurance that the gaseous effluent monitoring system will be capable of performing its intended function, as described in the OSURR SAR.

Section 11.1.4 of this SER discusses personnel monitoring systems and radiation survey equipment.

7.8 Conclusions

Based on the above discussion, the staff concludes that the nuclear and non-nuclear I&C systems are adequately designed and implemented to provide for safe and reliable startup, operation, and shutdown of the reactor during normal facility operation. The staff concludes that the RPS is adequate to protect the SL on fuel temperature and maintain the reactor in a state analyzed in the OSURR SAR. The staff also concludes that the I&C systems used for radiation monitoring are positioned at appropriate locations within the facility, use appropriate detectors and displays, and provide reasonable assurance that facility personnel will be aware of area radiation levels.

8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

The Columbus and Southern Ohio Electric Company supplies normal electrical power to the reactor building area. Historically, this service has had an annual failure probability of 0.2. The reactor building electrical service consists of 120/240 volts, three wire, single phase; 240 volts, three phase; and 120/208 volts, four wire, solid neutral for lighting and power. Outlets provide 120-volt and 240-volt electrical connections at various locations throughout the reactor building. Breaker panels located in the machine shop area and adjacent to the east service room allow for remote isolation of electrical power.

Building electrical service provides the initial source of electrical power for the nuclear instrumentation and reactor control system (RCS). Separate constant voltage isolation transformers provide electrical service to the control console and instrument systems. The console transformer is located on the main floor of the reactor building, while the instrument power transformer is housed in the control room.

8.2 Emergency Electrical Power Systems

The OSURR does not have any requirements for emergency electrical power. In the event of a loss of normal electrical power, the control rod magnets de-energize, dropping the shim safety control rods into the core and shutting down the reactor. Natural convection provides sufficient cooling during shutdown. Several wall-mounted lamps and handheld flashlights provide lighting in the reactor building in the case of an interruption of normal electrical power. These units function using battery power, and the wall-mounted lamps energize automatically.

8.3 Conclusions

On the basis of its review, the staff concludes that the normal electrical power system at the OSURR facility provides reasonable assurance of adequate operation. In addition, the staff concludes that loss of normal electrical power will lead to safe shutdown of the facility and that emergency power is not required to maintain safe shutdown.

9 AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air-Conditioning Systems

A combination of heating units, overhead gas heaters, air-conditioning units (both central and window mounted), blowers, and ducts provide heating, ventilation, and air-conditioning (HVAC) throughout the reactor building. Two rooms house the heating units and blowers that heat the reactor building. Two overhead gas heaters, one mounted above the front service door and one mounted above the rear entrance, provide additional reactor bay heating capability. An air-conditioning unit located outside the reactor building provides cooling for the reactor bay. Window-mounted air conditioners provide additional cooling for the control room and other individual rooms. An exhaust fan located at the top of the north wall of the reactor building provides general ventilation for the reactor bay. The exhaust rate is approximately 0.5 cubic meters per second (m^3/s) (1000 cubic feet per minute (ft^3/min)). This flow rate exhausts a volume equal to that of the reactor building (2000 cubic meters (m^3) (70,000 cubic feet (ft^3))), approximately once every 70 minutes. A master switch located in the control room allows the operator to secure all ventilation equipment and close the reactor bay exhaust fan dampers.

Chapter 11 of this SER discusses the role of the HVAC system in controlling the release of gaseous effluents and minimizing the concentration of airborne radioactivity within the reactor building. TSs related to the HVAC system include requirements that the exhaust fan be operating during reactor operation (TS 3.4, "Confinement," and TS 3.5, "Ventilation Systems"), a requirement that the ventilation system can be secured with a single switch located in the control room (TS 3.5), system surveillance requirements (TS 4.4, "Confinement," and TS 4.5, "Ventilation System"), and system design features (TS 5.1, "Site and Facility Description"). These TSs provide reasonable assurance that the HVAC can perform all intended functions, as described in the OSURR SAR.

Based on the above discussion and operating experience at the OSURR and similar facilities, the staff concludes that the HVAC systems are adequate to maintain conditions conducive to reliable reactor operation, including instrumentation and equipment temperature control and operator comfort. Additionally, the staff concludes that the ventilation system design and controls are adequate to control the release of radioactive materials during normal reactor operation and abnormal facility conditions.

9.2 Handling and Storage of Reactor Fuel

Two unique fuel-handling tools allow personnel to move reactor fuel within the core and between the core and fuel storage pit in accordance with established procedures. The fuel-handling tools are kept secured when not in use. According to the licensee, the design of the tools helps to minimize radiation exposure during fuel handling and prevent mechanical damage to the fuel elements.

The fuel storage pit is located at the bottom of the reactor pool. The design of the pit includes a fuel storage rack that provides storage space for an entire core loading of fuel elements. Boral spacers span the width of the rack and separate the individual rows of storage spaces. Given a full loading of fuel elements, the spacers limit the maximum effective multiplication factor (k_{eff}) to a value of 0.68. Natural-convection cooling removes decay heat from the fuel stored in the pit. Two lead-filled, aluminum plugs cover the fuel storage pit when no fuel is being moved between the reactor core and the pit.

TSs related to the handling and storage of reactor fuel include a surveillance requirement for fuel elements located out of core (TS 4.1.2, “Fuel Elements”), fuel storage facility and fuel-handling tool design features (TS 5.4, “Fuel Storage,” and TS 5 5, “Fuel Handling Tools”), a requirement on the minimum number and qualification of personnel necessary for fuel movement (TS 6.1.3, “Staffing”), and a requirement for fuel-handling procedures (TS 6.3.1, “Reactor Operating Procedures”). These TSs provide reasonable assurance that fuel handling and storage at the OSURR will be conducted in a manner that does not pose undue risk to public health and safety or the environment.

Based on the above discussion, the staff concludes the fuel storage facility design, fuel-handling tool design, and fuel-handling and storage procedures provide adequate measures to preclude inadvertent criticality and unauthorized fuel movement, and to minimize the risk of mechanical or chemical damage to the fuel during movement and storage.

9.3 Fire Protection Systems and Programs

The reactor building, reactor pool, and reactor core and support structures are all constructed of materials highly resistant to fire, such as concrete, steel, and aluminum. The reactor building contains a limited quantity of combustible material, such as furniture, paper, and superficial building materials. Hard-wired smoke detection equipment or manual pull-boxes alert the Clinton Township Fire Department and facility personnel to the existence of a fire. The facility personnel train in the use of portable firefighting equipment that is stored at various locations throughout the reactor building. This equipment is adequate to extinguish small fires. An organization independent of the facility conducts regular inspections to ensure that the portable firefighting equipment is functional. The Clinton Township Fire Department would be summoned in accordance with the OSURR emergency plan (EP) to assist with larger, prolonged fires.

Fire within the confinement building poses no threat to the safe shutdown of the reactor. Any interruption of the electrical supply to safety-related systems or signals from safety-related instrumentation as a result of destruction of cables or circuits by fire would cause the control rods to drop into the core and shut down the reactor. Following shutdown, the reactor pool walls and primary coolant would protect the core from a fire until it could be extinguished.

During a site visit, the NRC staff observed the fire detection system and portable firefighting equipment at the facility. Based on those observations and the above discussion, the staff concludes that adequate measures are in place to prevent and mitigate fire, and fire damage does not pose a significant threat to the safe operation or shutdown of the reactor.

9.4 Communication Systems

A two-way intercom system allows the control room operator to listen to and communicate with various stations throughout the reactor building. The intercom stations feature a local “push-to-talk” button that allows personnel and experimenters to talk to the control room operator. A dedicated telephone line services the control room. All telephones located in the reactor building are connected to the University for both internal and external communication. The building evacuation klaxon allows the control room operator to alert occupants of the reactor building to an abnormal reactor condition that necessitates building evacuation. The building evacuation klaxon automatically annunciates if an alarm is not acknowledged by the control room operator within a set time period. Before sounding the klaxon, the control room operator would use the two-way intercom system to inform building occupants of the situation.

The OSURR EP and physical security procedures (PSP) specify additional requirements for communication systems. The staff reviewed those requirements and systems as part of its review of the EP and PSP. The staff determined that the systems are adequate to allow proper communication among the various organizations and persons potentially involved in a response to an emergency or security event.

Based on the above, the staff concludes that the OSURR communication systems provide adequate means of communication for reactor personnel, experimenters, and offsite organizations during normal operation and abnormal facility conditions.

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

The OSURR generates byproduct material during normal reactor operation and use of the experimental facilities. Byproduct materials produced at the OSURR are typical of those associated with the operation of similar facilities. The byproduct materials include gaseous and liquid effluents, activated core and facility components, isotopes produced in experimental facilities, and fission products, for example. The OSURR uses a designated area for handling and storage of byproduct material that is not fixed within the reactor or discharged as an effluent. Chapter 11 of this SER discusses byproduct materials and the associated radiation protection requirements and TSs.

The OSURR license does not contain any provisions for the possession of source material. As mentioned in Section 1.4 of this SER, a State of Ohio license covers possession and use of the source material used in the subcritical assembly.

The OSURR facility license permits the receipt, possession, and use of limited quantities of special nuclear material (SNM) in the form of 5.2 kilograms (kg) (11 pounds (lb)) of contained uranium-235 at enrichments less than 20 percent for reactor fuel; up to 30.0 grams (g) (1.1 ounce (oz)) of high-enriched uranium (HEU) for fission chamber linings, foil targets, and other research applications; and up to 80.0 g (2.8 oz) of plutonium contained in encapsulated plutonium-beryllium neutron sources. Section 4.2.1 of this SER discusses reactor fuel. Section 4.2.4 of this SER discusses the plutonium-beryllium neutron source. TSs related to SNM include fuel element surveillance requirements (TS 4.1.2), fuel design features (TS 5.3, TS 5.4, and TS 5.5), staffing requirements during fuel handling (TS 6.1.3), procedural requirements (TS 6.3.1 and TS 6.3.2, “Administrative Procedures”), and requirements on records (TS 6.7.3, “Records to be Retained for the Life of the Facility”).

The NRC inspection program verifies that the licensee properly uses and maintains procedures related to SNM. The staff most recently reviewed selected procedures related to SNM during an inspection in May 2007 (NRC Inspection Report No. 50-150/2007-201, ADAMS Accession No. ML071500486). The inspection concluded that the licensee satisfied the procedural requirements.

9.6 Conclusions

Based on the above discussions, the staff concludes that the auxiliary systems in place at the OSURR generally enhance safe and reliable operation of the reactor. Additionally, the staff concludes that the related TSs and procedures provide reasonable assurance that possession and use of byproduct and SNM at the OSURR will not pose a significant risk to the health and safety of the public, OSURR personnel, or the environment.

10 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

The OSURR experiment program allows the conduct of a wide range of experiments utilizing a variety of experimental facilities. Experiments include, but are not limited to, neutron activation experiments, isotope production, materials damage studies, and medical experiments. Experimental facilities include the central irradiation facility (CIF), two beam ports, the rabbit system, the main thermal column, the bulk shielding facility (BSF) thermal column, graphite isotope irradiation elements (GIIEs), and movable dry tubes. In accordance with Subsection 104c of the Atomic Energy Act, as amended, the facility license allows the licensee to conduct widespread research through the development, review, and approval of new experiments and experimental facilities, provided that those experiments and experimental facilities do not require prior Commission approval per 10 CFR 50.59. The facility license contains specifications on experiment maximum reactivity worth, design, and materials that aid the licensee in determining if an experiment or experimental facility satisfies the requirements of 10 CFR 50.59.

10.2 Experimental Facilities

The OSURR experimental facilities allow gamma and neutron irradiation of materials. The design and location of the different facilities provide a spectrum of neutron energies and fluxes. Experimenters can extract radiation beams from the reactor core and perform irradiations in the active region of the core or in the moderator near the core. The experimental facilities are comparable in design, construction, utilization, and purpose to experimental facilities at other similar research reactors. The experimental facilities have been successfully and safely utilized during the period of the current facility license.

Accidents involving the experimental facilities include rupture leading to coolant loss, reactivity insertion caused by flooding of the experimental facility, and reactivity insertion caused by rapid movement of the experimental facility. Chapter 8 of the OSURR SAR and Chapter 13 of this SER show that the maximum hypothetical accident (MHA) bounds these accidents. Additionally, the design, construction, and utilization of the experimental facilities are such that these accidents are extremely unlikely. Chapter 11 of this SER discusses radiation hazards associated with the experimental facilities and methods for protecting experimenters and personnel from these hazards. Access to experimental facilities is controlled by the use of operating and radiation protection procedures. Additionally, the control room contains a permit switch for use of the rabbit facility and indicators of the status of access points to the thermal column and beam ports. Use of appropriate radiation detection equipment, radiation protection practices (including the as-low-as-reasonably-achievable (ALARA) program), and established experiment review procedures provide reasonable assurance that doses from experimental facilities will meet the requirements of 10 CFR Part 20 for personnel and members of the general public.

TSs related to the experimental facilities that constitute the OSURR experiment program include limits on the reactivity worth of individual experiments and all experiments (TS 3.7.1), design and construction criteria for experiments (TS 3.7.2, “Design and Materials”), procedural requirements (TS 6.3.1), requirements for experiment review and approval (TS 6.4, “Experiment Review and Approval”), a reporting requirement (TS 6.6.1, “Operating Reports”), and a requirement on record keeping (TS 6.7.1, “Records to be Retained for a Period of at Least Five

Years”). These TSs provide reasonable assurance that the experimental program and facilities will not cause an accident with consequences not bounded by the consequences of the MHA. Additionally, these TSs adequately assure that radiation hazards associated with the experimental facilities will not pose an undue risk to the health and safety of the public, facility personnel, experimenters, or the environment.

10.2.1 Central Irradiation Facility

Section 3.8.1 of the OSURR SAR provides a detailed description of the design and construction of the CIF. The CIF is an aluminum tube with an outer diameter of 3.8 cm (1.5 in) that extends from above the pool surface down into the central core array position. The CIF allows sample irradiation in the active core region where thermal neutron flux is highest.

10.2.2 Beam Ports

Section 3.8.2 of the OSURR SAR provides a detailed description of the design and construction of the two beam ports. The beam ports are aluminum tubes with an inner diameter of 15.56 cm (6.125 in) that extend horizontally from the outside face of the reactor pool wall into the reactor pool at the level of the centerline of the core. The beam ports allow insertion of samples to a position adjacent to the core or extraction of radiation beams to experiments external to the reactor pool wall.

10.2.3 Rabbit Facility

Section 3.8.3 of the OSURR SAR provides a detailed description of the design and construction of the rabbit system. The rabbit system allows rapid insertion of small samples using a pneumatic tube from a control station next to the outside of the reactor pool wall. The irradiation position is adjacent to the reactor core and above the beam ports. A permit switch located in the control room provides the control room operator with overall control of the system.

10.2.4 Main Graphite Thermal Column

Section 3.8.4 of the OSURR SAR provides a detailed description of the design and construction of the main thermal column. The thermal column consists of a cavity in the reactor pool wall filled with graphite bars. Removal of up to 13 bars creates space for sample irradiation in a fairly uniform field of thermal neutrons. Boral plates and a 79-centimeter-thick (31-inch-thick) concrete door provide shielding for personnel in the reactor bay.

10.2.5 Bulk Shielding Facility Thermal Column

Section 3.8.5 of the OSURR SAR provides a detailed description of the design and construction of the BSF thermal column. The BSF thermal column is similar to the main thermal column except that it is located in a cavity in the reactor pool wall between the reactor pool and the BSF.

10.2.6 Graphite Isotope Irradiation Elements

Section 3.8.6 of the OSURR SAR provides a detailed description of the design and construction of the GIIEs. The GIIEs are graphite-filled boxes that fit into the reactor core grid positions. The GIIEs have the same outer dimensions as a standard fuel element. Each GIIE contains a vertical hole with a diameter of 2.5 cm (1.0 in) that allows sample irradiation in the active core region.

10.2.7 Movable Dry Tubes

Section 3.8.7 of the OSURR SAR provides a detailed description of the design and construction of the movable dry tubes. The movable dry tubes are metal tubes that extend from above the reactor pool surface down into the reactor pool. The movable dry tubes allow sample irradiation at various locations within the reactor pool, including positions adjacent to the reactor core.

10.3 Experiment Review

The Reactor Operations Committee (ROC), discussed further in Section 12.2 of this SER, reviews all new experiments. The ROC has the responsibility to determine whether a proposed experiment requires a change in accordance with 10 CFR 50.59. The ROC review considers at a minimum (1) the purpose of the experiment, (2) the procedure for performance of the experiment, and (3) the safety evaluation previously approved by a licensed senior reactor operator (SRO). Based on the review, the ROC may approve the experiment.

A licensed SRO and the experimenter evaluate all experiments utilizing the reactor to ensure compliance with the provisions of the utilization license, the TSs, and 10 CFR Part 20. The experiment may be performed if, in the judgment of the SRO, the experiment meets with the above provisions, received previous approval by the ROC, and does not constitute a threat to the integrity of the reactor. When pertinent, the evaluation includes considerations of (1) the reactivity worth of the experiment, (2) the integrity of the experiment, including the effects of changes in temperature, pressure, or chemical composition, (3) any physical or chemical interaction that could occur with the reactor components, and (4) any radiation hazard that may result from the activation of materials or from external beams.

TSs related to experiment review include responsibilities of the ROC (TS 6.2.3, “Review Function”) and requirements for experiment review and approval (TS 6.4). These TSs adequately describe the processes used to evaluate previously approved experiments and review and approve new experiments, including determining the need for a change to the TSs pursuant to 10 CFR 50.59. Additionally, these TSs provide reasonable assurance that administrative oversight will preclude the experiment program from posing a significant risk to the health and safety of the public, facility personnel, experimenters, and the environment.

10.4 Conclusions

The staff concludes that the administrative controls governing the OSURR experimental program will minimize the risk of unintended radiation exposure of experimenters and facility personnel. Based on the preceding discussions and safe utilization during the period of the current facility license, the staff concludes that the OSURR experiment program will not pose a significant risk to the health and safety of the public, facility personnel, experimenters, or the environment during routine use or from accidents.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

11.1.1 Radiation Sources

Sources of radiation related to operation of the OSURR include the reactor core, primary coolant cleanup system demineralizer cartridge, liquid-borne nitrogen-16, and gaseous argon-41. Other sources of radiation at the OSURR may include calibration and check sources, activation products produced by the experimental programs, and radioactive contamination.

The reactor core is shielded by the concrete reactor pool walls and the primary coolant in the reactor pool. The licensee analyzed the expected radiation levels outside the reactor pool walls and above the pool surface resulting from direct radiation from the reactor core. Radiation surveys confirmed acceptable radiation levels outside the reactor pool walls and above the pool surface. The staff evaluated the licensee's analysis and assumptions and found them to be acceptable based on the use of appropriate conservatism and calculation methods. The licensee analyzed the production and dose consequences of nitrogen-16. As described in Section 5.6 of this SER, the licensee has equipment to control nitrogen-16 in the reactor pool. As discussed in Section 11.2.2 below, the licensee analyzed the production of argon-41 in the reactor pool and experimental facilities and the expected releases to the reactor building. The licensee conducts periodic radiation surveys and continuous monitoring of radiation levels in the reactor building. The licensee also conducts an appropriate personnel dosimetry program. These activities have shown that personnel exposure from operation of the reactor can be kept well below the occupational limits specified in 10 CFR 20.1201, "Occupational Dose Limits for Adults." The staff concludes that the licensee's analyses and monitoring programs provide reasonable assurance that the licensee will maintain radiation exposure from the OSURR well below the applicable regulatory limits during the period of the renewed license.

The use of administrative controls, procedures, and appropriate storage facilities limits the exposure to other sources of radiation at the OSURR. A designated area of the reactor building is used for the storage of check and calibration sources and activation products from the experimental program. Sources are generally stored in lead-shielded containers. Administrative controls prohibit certain activities in this area that could increase the likelihood of personnel contamination or increased radiation exposure. The radiation hazards associated with the experimental programs are controlled by Reactor Operations Committee (ROC) review of new types of experiments and senior reactor operator (SRO) review of individual approved experiments. The SRO on duty has the authority to terminate any experiment. TS 6.3.1 requires procedures for installation, removal, and movement of experimental facilities. The ROC must review those procedures. TS 6.4.2, "Approved Experiments," requires an evaluation of individual approved experiments that may include an assessment of radiation hazards from the activation of materials or from external radiation beams. The staff found that these TSs provide reasonable assurance that radiation hazards associated with the experimental programs will be identified and controlled.

11.1.2 Radiation Protection Program

Section 7.2 and Section 7.3 of the OSURR SAR describe the radiation protection strategy and methodology used at the OSURR. Guidelines issued by the university's Radiation Safety Section (RSS) of the Office of Environmental Health and Safety, which is under the authority of the Vice President for Health Services, govern the strategy and methodology. According to the licensee, the university's guidelines are designed to, at the very least, meet the requirements of applicable NRC and State of Ohio regulations. The OSURR radiation protection strategy and methodology are based on a university-wide administrative commitment to keep radiation doses as-low-as-reasonably-achievable (ALARA). Section 11.1.3 below discusses the ALARA commitment and practices. The elements of the OSURR radiation protection strategy and methodology include training of personnel, administrative controls, radiation monitoring and surveying, personnel dosimetry, and access control. These elements are implemented by TS-required procedures developed and maintained by the OSURR administration and personnel.

The staff evaluated the radiation protection strategy and methodology against ANSI/ANS-15.1; ANSI/ANS-15.11, "Radiation Protection at Research Reactor Facilities," issued 1993; and the OSURR TSs. The staff found that the radiation protection strategy and methodology incorporate elements that are consistent with the standards. These include training personnel in accordance with 10 CFR 19.12, "Instruction to Workers"; administrative controls related to experiments, records, reporting, and procedures; radiation monitoring and surveying programs that meet the requirements of 10 CFR 20.1501, "General"; access control to radioactive materials; and posting of radiation and high-radiation areas. The staff found that the OSURR radiation protection procedures are required by TS 6.3.1 and are maintained in accordance with TS 6.2, "Review and Audit." The NRC inspection program checks for compliance with these TSs on a routine basis. Based on the findings discussed above, the staff concludes that the OSURR radiation protection program is adequate to provide reasonable assurance that personnel will be protected from undue radiation hazards.

11.1.3 ALARA Program

Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable," Revision 1-R, issued May 1977, is the basis of the ALARA philosophy used by the RSS and OSURR. The OSURR SAR indicates that the facility will maintain doses ALARA by using the practices of minimizing exposure time to radiation, using appropriate shielding of radiation sources, and increasing distance between radiation sources and individuals. These are the standard practices for minimizing radiation exposure, and the staff finds them to be acceptable. The OSURR implements these practices using established procedures and careful, advanced planning for work in an area where a radiation field might exist. The staff finds that this provides reasonable assurance that the OSURR will implement the ALARA philosophy. Section 11.2.3 discusses the ALARA requirement for doses to members of the general public.

11.1.4 Radiation Monitoring and Surveying

Section 7.7 of this SER discusses the area radiation monitor (ARM) system used to monitor radiation levels in the reactor building. The staff concluded that the ARM system is adequate to detect radiation levels at appropriate locations within the reactor building and provides reasonable assurance that personnel will be cognizant of radiation levels within the building.

The facility personnel complete a radiation survey of the reactor building each day that the reactor is operated and at least once per week, in accordance with established procedures. The OSU RSS conducts an independent monthly survey. Portable survey meters used by the OSURR personnel are capable of detecting gamma, beta, and neutron radiation. These are the types of radiation expected during normal reactor operation and facility utilization. The RSS can provide supplemental detection capability, if required. The staff evaluated the monitoring and surveying equipment and found that the licensee has the appropriate equipment to detect expected types radiation with adequate sensitivity. This equipment is required by TS 3.6.1 and tested and calibrated as required by TS 4.6.3, "Portable Survey Instrumentation." The staff evaluated the test and calibration requirements and found them to be consistent with ANSI/ANS-15.1. The staff concludes that the radiation monitoring and surveying equipment and programs satisfy the requirements of 10 CFR 20.1501(a) and (b) and provide reasonable assurance that doses to personnel will be kept below the limits specified in 10 CFR 20.1201.

11.1.5 Radiation Exposure Control and Dosimetry

Personnel and visitor exposures at the OSURR are detected using film badges, pocket ionization chambers, and ring badges (for measuring extremity dose). RSS processes the film badges and ring badges on a monthly basis. The pocket ionization chambers are used to provide dosimetry for individuals visiting the reactor for short periods of time (on the order of hours) and to provide near real-time dose information for personnel. These types of dosimetry are consistent with those widely used at research reactors and the types listed in ANSI/ANS-15.11.

The staff evaluated the last five annual reports submitted to the NRC to gauge the historical effectiveness of the exposure control and dosimetry programs related to visitor and personnel exposures. The licensee did not report any measurable exposures of visitors to the facility. During years of normal reactor operation, the average and maximum occupational total effective dose equivalent (TEDE) were 0.55 millisievert (mSv) (55 millirem (mrem)) and 1.3 mSv (130 mrem), respectively. During years of increased maintenance activity, the average and maximum occupational TEDE were 8.15 mSv (815 mrem) and 24.12 mSv (2412 mrem), respectively. These doses are below the occupational dose limit of 50 mSv (5000 mrem) specified in 10 CFR 20.1201. The staff compared the reported doses against those reported by other research reactors and found them to be similar. No changes in reactor operation that would lead to increased occupational dose are expected during the period of the renewed license. TS 6.2.3 requires administrative review of any proposed changes to the facility, experiments, or the operating procedures to assess potential increases in radiation exposure. Accordingly, the staff concludes that exposure control and dosimetry at the OSURR are adequate to maintain occupational doses below the regulatory limit during the period of the renewed license.

11.1.6 Contamination Control

The RSS conducts periodic radiation and contamination surveys. According to the licensee, these surveys should detect loose contamination within the reactor building. A Geiger-Mueller radiation detector with detachable probe is provided at the main entrance to the reactor building to allow for detection of personal contamination. This detector is of the type commonly used for personal contamination monitoring. The licensee uses administrative controls to reduce the likelihood of contamination events at the OSURR. These include storage of radioactive materials in secured locations, the use of protective clothing, and review of experiments for radiation hazards. The facility maintains records of contamination surveys for a 5-year period as required by TS 6.7.1. Based on the measures in place at the OSURR to control, detect, and record contamination, the staff finds that contamination does not pose an undue risk to personnel.

11.1.7 Environmental Monitoring

The licensee conducts a voluntary environmental monitoring dosimetry program and an unrestricted area radiation survey program to record and track radiological effects of facility operation on the surrounding unrestricted area. The environmental monitoring dosimetry program consists of quarterly exposure measurements at five stations surrounding the OSURR restricted area. Exposures are recorded using film badges. Film badges are widely used for these types of monitoring programs at research reactors, and the staff finds them appropriate for use at the OSURR. The RSS processes the film badges and retains records of the measurements. The unrestricted area radiation survey program consists of dose rate measurements at eight locations surrounding the restricted area during full-power reactor operation. The staff evaluated the environmental monitoring program and found that it provides radiation measurements at sufficiently diverse and appropriate locations. The staff also found that the radiation survey program is conducted frequently enough to detect any day-to-day changes in radiation levels. The monitoring and survey programs indicate that, given very conservative assumptions, the maximally exposed member of the public could receive an annual dose of approximately 0.61 mSv (61 mrem). This value is below the annual limit of 1.0 mSv (100 mrem) specified in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public."

11.2 Radioactive Waste Management

11.2.1 Radioactive Waste Management Program

Chapter 6 of the OSURR SAR describes the licensee's radioactive waste management program. The program addresses gaseous, liquid, and solid radioactive wastes. The waste management program includes characterization of waste streams, procedures for monitoring wastes, procedures for the release of wastes, and requirements for records. Sections 11.2.2 and 11.2.3 of this SER address characterization, monitoring, and release of wastes. TS 6.7.3 requires that records of gaseous and liquid effluents released to the environment be retained for the life of the facility. TS 6.7.1 requires that records of the transfer of radioactive material be retained for a 5-year period.

11.2.2 Radioactive Waste Controls

The licensee controls gaseous radioactive wastes (gaseous effluents) by minimizing production of the waste products and discharging the effluents to the unrestricted area. The only gaseous waste of concern is argon-41. Chapter 6 of the OSURR SAR provides analysis of the production of argon-41 in the reactor pool water and experimental facilities. This analysis gives maximum permissible annual operating times for the reactor and the experimental facilities based on the concentration of argon-41 within the reactor building and the calculated radiological impact of releasing the effluent. Section 11.2.3 below evaluates the radiological impact of releasing the effluent. In the analysis of production of argon-41, the licensee did not take credit for administrative controls in place at the OSURR to minimize the production of argon-41. The staff evaluated the analysis in the SAR and found it to be conservative and acceptable. The staff found that the scope of the licensee's effluent monitoring program and the licensee's characterization of argon-41 sources provide reasonable assurance that gaseous waste will not pose an undue risk to personnel or result in releases above the limits specified in Appendix B to 10 CFR Part 20.

The licensee controls liquid radioactive waste by using the primary coolant cleanup system, described in Section 5.4 of this SER, and sampling the primary coolant for radioactivity. The cleanup system removes activation products, primarily sodium-24, from the primary coolant by passing the coolant through a demineralizer cartridge. The licensee performs radiation surveys of the demineralizer cartridge following each shutdown of the reactor and posts the area as a radiation area, if necessary. Primary coolant sampling, which is required by TS 3.3.5, "Primary and Secondary Coolant Activity Limits," provides reasonable assurance that the buildup over time of radioactivity in the primary coolant will be detected.

The spent demineralizer cartridge constitutes the majority of the solid waste generated at the OSURR. The licensee maintains control of the demineralizer cartridge to allow the radioactivity to decay to negligible levels before returning the cartridge to the manufacturer for regeneration. If longer-lived isotopes are present in the demineralizer cartridge, the licensee may transfer the cartridge to the RSS for disposal. Other low-level solid waste consists of gloves, pads, and various activation products from the experimental facilities. The staff compared the waste to that of similar facilities and found that the types of waste are common for research reactors and can be safely stored and disposed of in accordance with applicable regulations.

11.2.3 Release of Radioactive Waste

Gaseous effluents are discharged via the ventilation fan, located 9 m (30 ft) up the north wall of the reactor building, at a volumetric flow rate of 0.5 m^3/s (1000 ft^3/min). The licensee analyzes and records the effluent stream activity daily using the gaseous effluent monitor described in Section 7.7 of this SER. According to the licensee, other release pathways exist, but they are normally secured during reactor operation and have insignificant volumetric flow rates compared to the ventilation fan. Gaseous radioactive releases reported in annual reports to the NRC were within the limits set by Appendix B to 10 CFR Part 20. Annual doses to members of the general public, calculated using the COMPLY code of the U.S. Environmental Protection Agency, were typically on the order of 0.001 mSv (0.1 mrem). These dose rates are within the limits set by 10 CFR 20.1301 and demonstrate compliance with 10 CFR 20.1101(d). TS 3.6.2, "Radioactive Effluents," places limits on the concentration of argon-41 that may be released to the unrestricted area. This TS conforms with ANSI/ANS-15.1, and the staff found it to be acceptable to limit the release of argon-41 from the facility. The staff evaluated the licensee's assumptions regarding argon-41 releases and found them to be conservative.

Liquid effluents are discharged to the sanitary sewer, and releases are reported to the RSS. The RSS maintains records of releases made by OSU to the sanitary sewer and ensures that total liquid effluents from the campus do not exceed the regulatory limits. Measurable releases occur every few years as a result of maintenance work that requires the reactor pool to be drained. The volume, activity, and inventory of the effluent stream are recorded to ensure compliance with applicable regulations. The only nuclide of significant concentration found in the liquid effluent stream is tritium. Tritium releases reported in annual reports to the NRC were well below the limits set by Appendix B to 10 CFR Part 20. TS 3.6.2 requires all liquid releases to be within the requirements of 10 CFR Part 20.

Un-compacted solid low-level radioactive waste is transferred to the RSS for ultimate disposal. The RSS maintains a State license for receipt of transferred material. Solid radioactive releases for the period between July 1999 and June 2004 totaled 0.34 m^3 (12 ft^3) and had a total activity of 275 megabecquerels (MBq) (7.44 millicuries (mCi)). The OSURR has not shipped any high-level radioactive waste since the high-enriched uranium (HEU) fuel was removed in 1995 following the conversion to low-enriched uranium (LEU) fuel. The licensee does not anticipate the need to ship any high-level radioactive waste during the 20-year period of the renewed license. If spent fuel needs to be removed from the site, the facility will return it to the U.S. Department of Energy (DOE) in accordance with an established contract.

Based on the findings discussed above, the staff concludes that administrative controls, radioactive release methods, and TSs are adequate to provide reasonable assurance that the release of radioactive waste from the OSURR will meet the requirements of 10 CFR Part 20 and therefore will not pose an undue risk to the health and safety of the public or the environment.

11.3 Conclusions

The staff concludes that the OSURR radiation protection and ALARA programs, radiation monitoring and surveying, and exposure control and dosimetry are adequate to provide reasonable assurance that doses to facility personnel will be maintained below the regulatory limit and ALARA. The staff concludes that the licensee's environmental monitoring program and radioactive waste disposal methods provide reasonable assurance that doses to members of the public will be kept below the regulatory limit and ALARA. Additionally, the staff concludes that the licensee's radioactive waste management program is adequate to provide reasonable assurance that radioactive wastes will be handled and disposed of in accordance with applicable regulations and should not have a significant impact on the environment.

12 CONDUCT OF OPERATIONS

12.1 Organization

The OSURR is a part of the College of Engineering administered by the Engineering Experiment Station of OSU. Responsibility for the safe operation of the OSURR is vested in the administrative organization shown in Figure 6.1 of the TSs. TS 6.1, “Organization,” provides requirements for the structure and responsibility of the administrative organization. The Director of the Engineering Experiment Station, designated as Level 1, is the primary contact person for communications between the NRC and OSU. The Director of the Engineering Experiment Station reports to the Dean of the College of Engineering, who reports to the Provost. The Director of the OSURR, designated as Level 2, is responsible for overall management of the facility. The Director of the OSURR is directly responsible to both the Level 1 manager and the Director of the university’s Radiation Safety Section (RSS) of the Office of Environmental Health and Safety. This provides reasonable assurance that radiation protection practices and reactor operations issues will be considered equally. The Associate Director, designated as Level 3, is responsible for ensuring safe day-to-day operation of the OSURR and reports to the Director of the OSURR. Senior reactor operators (SROs), designated as Level 4, and reactor operators report to the Associate Director. The staff finds that TS 6.1 is consistent with the guidance of ANSI/ANS-15.1 and specifies an adequate level of independence for the administration of the radiation protection program and facility operations.

TS 6.1.4, “Selection and Training of Personnel,” states that selection and training of operations personnel shall meet or exceed the requirements of ANSI/ANS-15.4, “Selection and Training of Personnel for Research Reactors,” issued 1988. The staff finds that the requirements in ANSI/ANS-15.4 are appropriate for the selection and training of personnel at the OSURR. Accordingly, the staff finds that TS 6.1.4 provides reasonable assurance that operations personnel will be able to operate the OSURR in a safe manner in accordance with the TSs. TS 6.1.3 provides requirements for minimum staffing when the reactor is not secured. The requirements include a reactor operator in the control room, a second person at the facility able to carry out written instructions, and a SRO readily available on call. TS 6.1.3 also identifies events that require the presence of a SRO at the facility. These events include initial startup and approach to power, fuel movements, and recovery from unplanned or unscheduled shutdowns. The staff evaluated the requirements of TS 6.1.3 and found that they are consistent with ANSI/ANS-15.1 and satisfy the requirements of 10 CFR 50.54(m)(1).

12.2 Review and Audit Activities

The Reactor Operations Committee (ROC) is responsible for the review and audit of reactor operations to assure the OSURR is operating in a manner consistent with the TSs and other conditions of the license. TS 6.2 specifies the ROC composition and qualifications, meetings, subcommittees, review and approval function, and audit function. The staff evaluated that TS against ANSI/ANS-15.1. The staff found that the requirements of ROC composition and qualifications, meetings, and subcommittees are consistent with the standard. The staff evaluated the review and approval function and the audit function of the ROC and found that they are consistent with ANSI/ANS-15.1 and that the appropriate levels of the organization required by Figure 6.1 of the TSs will be notified of the activities and findings of the ROC.

12.3 Procedures

The OSURR maintains written procedures regarding reactor operations and administrative functions. The Director of the OSURR reviews and approves procedures, and the ROC reviews them. TS 6.3, "Procedures," specifies the required procedures and specific areas to be covered. The staff evaluated the requirements of TS 6.3 and found that they are consistent with ANSI/ANS-15.1. The staff found that the types of procedures required cover the full range of reactor operations and administrative functions and that appropriate levels of the OSURR organization are responsible for review, approval, and change of the procedures. The staff concludes that the procedural requirements at the OSURR provide reasonable assurance of safe operation of the reactor and proper administration of the facility.

12.4 Required Actions

TS 6.5, "Required Actions," specifies actions to be performed by the licensee if there is a reportable occurrence or a safety limit (SL) is exceeded. As discussed in other chapters of this SER, the licensee provided information and analyses demonstrating that no credible failure could lead to a fuel temperature equal to the SL specified by TS 2.1. The required actions specified by TS 6.5 include actions by the reactor operator to ensure that the reactor is in a safe condition, reporting and notification, and review of the occurrence by the ROC. The staff evaluated the requirements of TS 6.5 and found that they are consistent with ANSI/ANS-15.1; satisfy the requirements of 10 CFR 50.36, "Technical Specifications"; and provide reasonable assurance that the facility will respond to unanticipated occurrences in a manner that emphasizes reactor safety and protection of public health and safety.

12.5 Reports

TS 6.6, "Reports," specifies reports that the facility is required to make to the NRC. These reports include annual operating reports and special reports. Annual operating reports provide information regarding reactor utilization, maintenance and inspections, changes to the facility or procedures, radioactive effluents, and personnel and visitor exposures to radiation. Special reports provide information regarding non-routine occurrences, changes in organization personnel, and significant changes in the analyses presented in the OSURR SAR. The staff evaluated the requirements of TS 6.6 and found that they are consistent with ANSI/ANS-15.1 and provide reasonable assurance that the facility will report appropriate information regarding routine operation, non-routine occurrences, and changes to the facility and personnel to the NRC in a timely manner.

12.6 Records

TS 6.7, "Records," specifies records that the OSURR is required to maintain in a manner that facilitates convenient review. The licensee specified the types of records that the facility will retain and the period of retention to ensure that important records will be retained for an appropriate time. The staff evaluated the requirements of TS 6.7 and found that they are consistent with ANSI/ANS-15.1 and give reasonable assurance that the facility will maintain appropriate records to facilitate NRC inspection and provide adequate history of the facility.

12.7 Emergency Planning

The licensee requested that the current emergency plan (EP) be considered as part of the license renewal application. The staff reviewed the EP against NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test

Reactors," issued October 1983; Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors," Revision 1, issued March 1983; ANSI/ANS-15.16, "Emergency Planning for Research Reactors," issued 1982; and NRC Information Notice 97-34, "Deficiencies in Licensee Submittals Regarding Terminology for Radiological Emergency Action Levels in Accordance with the New Part 20," issued June 1997. The staff concluded that the OSURR EP is in accordance with the guidance and regulations. A letter dated October 5, 2005, from the NRC Office of Nuclear Reactor Regulation to the Ohio State University (ADAMS Accession No. ML052560604) documented this review. The licensee has demonstrated the ability to make changes to the EP in accordance with 10 CFR 50.54(q). Accordingly, the staff concludes that the OSURR EP provides reasonable assurance that the licensee can respond appropriately to a variety of emergency situations and that the OSURR EP will be adequately maintained during the period of the renewed license.

12.8 Security Planning

Because the facility license authorizes possession of special nuclear material (SNM) of low strategic significance, the licensee must maintain security measures that satisfy the requirements of 10 CFR 73.67(f), "Fixed Site Requirements for Special Nuclear Material of Low Strategic Significance." The licensee has met these requirements since the license condition to have an approved Security Plan was removed from the license by Amendment No. 16, issued December 4, 1995. The NRC inspects the licensee's measures for physical security and protection of SNM on a routine basis. A recent inspection verified that the licensee's security measures satisfy all applicable regulations and are acceptable.

12.9 Quality Assurance

The OSURR SAR states that the licensee maintains a quality assurance (QA) program that is consistent with the guidance found in ANSI/ANS-15.8, "Quality Assurance Program Requirements for Research Reactors," issued 1995. The Director of the OSURR has responsibility for the QA program. Normally, the Associate Director is responsible for the daily implementation of the program. The ROC has responsibility for independent review and audit functions associated with the program. QA records include inspection and test results, reviews by the ROC, and analyses of modifications and design changes.

12.10 Operator Training and Requalification

The Manager of Reactor Operations (or Associate Director) serves as the Training Coordinator and is responsible for the implementation, coordination, and operation of the Requalification Program including the training of new operators. The licensee's Requalification Program and training program provide reasonable assurance that the licensee will have technically qualified reactor operators. The licensee submitted a revised Operator Requalification Program with the application for license renewal. The staff reviewed the program and found that it meets all applicable regulations (10 CFR 50.54(i-l) and 10 CFR Part 55, "Operators' Licenses") and is consistent with guidance contained in ANSI/ANS-15.4.

12.11 Conclusions

Based on the preceding considerations, the staff concludes that the licensee has sufficient experience, an appropriate organization, and adequate procedures to provide reasonable assurance that the OSURR will continue to be managed in a way that will not cause any significant radiological risk to the health and safety of the public or the environment.

13 ACCIDENT ANALYSES

13.1 Maximum Hypothetical Accident

As a bounding analysis of the radiological consequences of an accident at the OSURR, the licensee analyzed a maximum hypothetical accident (MHA) that is not a credible accident at the OSURR. The calculated hypothetical doses presented in this section of the SER are not representative of doses, if any, resulting from the credible accidents presented in subsequent sections of this chapter.

Section 8.4.4 of the OSURR SAR presents the complete scenario and consequences of the MHA. The MHA considered was the complete removal of the cladding from one side of one fuel plate. The licensee did not postulate the event effecting removal of the cladding. This scenario is not considered credible for the following reasons:

(1) Melting of the cladding cannot result from any credible accident scenario (see subsequent sections of this chapter).

(2) Stripping of the cladding from a fuel plate by chemical processes (oxidation, dissolving, etc.) is not credible because the fuel plate is immersed in a large volume of low-conductivity, mean neutral pH water at near-ambient temperature of 20 °C (68 °F).

(3) Stripping of the cladding from a fuel plate by mechanical processes is not credible because fuel plates are located only at positions internal to the fuel element. Thus, an element would need to be intentionally removed from the core and disassembled to expose a fuel plate.

Removal of an element from the core requires generation of a large upward force on an element. Other than intentional manipulation, which is only performed under the supervision of a senior reactor operator (SRO), there is no credible mechanism for generation of such a force in the OSURR systems. Appropriate procedures strictly control intentional manipulation of a fuel element.

13.1.1 Accident Source Term

In analyzing the MHA, the licensee made the following assumptions regarding the fission product source term. The staff's evaluation follows each assumption.

(1) The reactor is operated at 500 kW(t) for an infinite irradiation time with the least number of fuel plates necessary.

This assumption implies that the concentrations of the fission products of interest have reached equilibrium at the maximum licensed power of 500 kW(t). This assumption ensures that the consequences of the MHA represent the least restrictive (i.e., most conservative) operating schedule. The reactor typically operates at a power levels less 500 kW(t) for several hours at a time. Over the course of an average year, the reactor operates at an average power of 64 kW(t) for 440 hours. Radionuclides with longer half-lives will not reach the equilibrium concentrations used in the analysis of the MHA during typical operations. The fission product inventory of an individual fuel plate will be greater when there are fewer total fuel plates in the core.

(2) The fuel plate that fails does so at the end of this operation.

The series of events that effects removal of the cladding would take some amount of time. The licensee's assumption does not account for radioactive decay during that time and is therefore conservative.

(3) The failed fuel plate is located at the peak radial flux position in the core. The licensee used a power peaking factor of 1.8.

The fuel plate at the peak flux position in the core will have the greatest fission product inventory. A power peaking factor of 1.8 is conservative in comparison to the value of 1.64 presented in the staff's safety evaluation supporting the order for conversion to low-enriched uranium (LEU) fuel dated September 27, 1988. Use of the radial peaking factor is in accordance with Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," issued March 1972.

(4) Of the iodine, krypton, and xenon isotopes within recoil range 1.37×10^{-3} cm (5.39×10^{-4} in) from the surface of the fuel, 100 percent are released to the reactor pool water.

This assumption is consistent with accident analyses performed for similar research reactors, including analysis presented in NUREG/CR-2079, "Analysis of Credible Accidents for Argonaut Reactors," issued April 1981, of fission product releases from plate-type fuel similar to that used at the OSURR.

Using the above assumptions and appropriate methods, the licensee calculated a gaseous fission product (iodine, xenon, and krypton) source term of 1.4×10^3 gigabequerels (GBq) (37 curies (Ci)). The staff verified the licensee's calculation and performed an independent calculation using the assumptions and method described in Regulatory Guide 1.25. The staff's calculation yielded a gaseous fission product source term of 8.9×10^2 GBq (24 Ci). The staff also performed a sensitivity study to determine the conservatism of assumption (1) of infinite operating time. The staff found that, given an operating time of 4 hours, the average fission product concentration would be half of that for an infinite operating time. Section 13.1.4 below discusses the impact of this assumption on the total radiation dose.

13.1.2 Thyroid Dose

In analyzing the thyroid dose resulting from inhalation of iodine, the licensee made the following additional assumptions. The staff's evaluation follows each assumption.

(1) At the time of the release, the iodine instantaneously mixes with the coolant in the reactor pool.

This assumption ignores radioactive decay during the finite mixing time.

(2) The released iodine is transferred to the reactor bay air by diffusion.

Diffusion because of the partial pressure of iodine in air is an appropriate method for calculating the concentration of iodine in the reactor bay air. The licensee used a reactor pool water temperature of 20 °C (68 °F), which represents a realistic value. In evaluating the licensee's assessment of the thyroid dose, the staff used a more conservative value of 35 °C (95 °F).

(3) The concentration of iodine in the reactor bay is reduced only by radioactive decay.

This assumption ignores reduction of the iodine concentration by plate-out and is conservative because it leads to higher iodine concentrations in the building air and the effluent stream.

Using the above assumptions and appropriate methods, the licensee calculated an occupational thyroid dose of 1.2 mSv (120 mrem) for a stay-time of 2 hours. This thyroid dose corresponds to a committed effective dose equivalent (CEDE) of 0.036 mSv (3.6 mrem) given the tissue weighting factor of 0.03 for the thyroid specified in 10 CFR 20.1003, "Definitions." A more realistic stay-time of 5 minutes decreases the CEDE to 0.002 mSv (0.2 mrem). The staff independently verified the doses calculated by the licensee by using the licensee's method and methods described in Regulatory Guide 1.25 and NUREG/CR-2079. As mentioned above, the staff used a reactor pool water temperature of 35 °C (95 °F), the limit given in TS 2.2, to calculate the concentration of iodine in the air in the reactor building. Using the licensee's method, the staff calculated a CEDE of 0.007 mSv (0.7 mrem) for a stay-time of 5 minutes. The other two methods of calculating occupational dose contained additional conservatism and, accordingly, yielded higher occupational thyroid doses. All calculated doses were significantly below the applicable regulatory limit of 50 mSv (5000 mrem) specified in 10 CFR 20.1201.

The licensee calculated the thyroid dose to members of the general public located immediately outside the restricted area. The licensee's calculation yielded a thyroid dose of less than 0.01 mSv (1 mrem). The staff independently calculated the dose to the maximally exposed member of the public using the methods described in Regulatory Guide 1.25 and NUREG/CR-2079. The staff used an atmospheric dispersion factor of 2.6×10^{-3} seconds per cubic meter (s/m^3) (7.4×10^{-5} seconds per cubic foot (s/ft^3)), which was calculated using Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, issued November 1982, and conservatively assuming a distance of 0.5 km (0.3 mi) (closest permanent residence), a wind speed of 1 m/s (2 mi/h), and a Pasquill stability category of G (extremely stable). The calculated CEDE was well below 0.01 mSv (1 mrem) at the nearest permanent residence. The staff also calculated a CEDE of 0.017 mSv (1.7 mrem) at the site boundary using an atmospheric dilution factor of 0.1. These doses satisfy the regulatory limit of 1 mSv (100 mrem) specified in 10 CFR 20.1301 for members of the general public.

13.1.3 Whole-Body Gamma Dose

In analyzing the whole-body gamma dose resulting from submersion in a cloud of gaseous fission products (iodine, krypton, and xenon), the licensee made the following additional assumptions. The staff's evaluation follows each assumption.

(1) At the time of the release, the gaseous fission products instantaneously mix with the air in the reactor bay.

This assumption ignores radioactive decay during the finite mixing time.

(2) For the case of dose to personnel in the reactor building, the dose results from a finite, hemispherical cloud of uniformly distributed gaseous fission products. The finite hemisphere has a volume equal to that of the reactor building (2000 m^3 (70,000 ft^3)).

This assumption leads to a more realistic dose estimate than an infinite cloud model. Given that the dimensions of the reactor building are small compared to the mean free path of gamma rays in air, this assumption is appropriate.

(3) For the case of dose outside the reactor building resulting from the gaseous fission products trapped within the reactor building ("direct radiation" dose), the reactor building is modeled as sections of two concentric spherical shells of uniformly distributed gaseous fission products. The two sections have a combined volume equal to that of the reactor building. Additionally, the reactor structural concrete and reactor building walls are assumed to provide negligible shielding.

As in the case of assumption (2), this assumption leads to a more realistic dose estimate. Ignoring the shielding effects of the reactor structural concrete and reactor building walls adds conservatism to the dose estimate.

(4) For the case of dose outside the reactor building that results from gaseous fission products vented from the reactor building ("plume" dose), the dose is due to a plume of gaseous fission products dispersed by atmospheric transport.

This assumption is consistent with Regulatory Guide 1.145.

In addition to these assumptions, the licensee calculated the occupational dose, the direct radiation dose, and the plume dose assuming (1) that the reactor building exhaust fan was secured at the time of the fission product release (ventilation off) and (2) that the reactor building exhaust fan continued to operate indefinitely (ventilation on). With the exhaust fan secured, the licensee measured the reactor building air leakage rate to be 2.3×10^{-3} m^3/s (4.9 ft^3/min). With the exhaust fan operating, the ventilation rate is 0.5 m^3/s (1000 ft^3/min).

The licensee calculated occupational doses for a worker located near the center of the reactor bay and ignored shielding by the concrete reactor pool structure. This yielded a conservative dose estimate because the reactor pool structure would provide shielding from a significant portion of the fission products uniformly distributed throughout the air in the reactor building. With the building ventilation off, the licensee calculated an occupational dose of 1.8 mSv (180 mrem) for a stay-time of 2 hours and a dose of 0.1 mSv (10 mrem) for a more realistic stay-time of 5 minutes. With the building ventilation on, doses were 0.92 mSv (92 mrem) and 0.1 mSv (10 mrem), for the respective stay-times. Based on firsthand observations of the reactor building and the licensee's emergency procedures, the staff finds that an evacuation time of 5 minutes is realistic. The staff verified the licensee's dose calculations and found them to be correct. The staff also independently calculated doses for the same stay-times and obtained similar results. All calculated doses to a worker were significantly below the applicable regulatory limit of 50 mSv (5000 mrem) specified in 10 CFR 20.1201.

The licensee calculated direct radiation doses for an individual standing at the site boundary. With the building ventilation off, the licensee calculated a direct radiation dose of 0.98 mSv (98 mrem) for a stay-time of 30 days. With the building ventilation on, the dose decreased to 0.25 mSv (25 mrem) for the same stay-time. The staff verified the licensee's calculations and found them to be acceptable. The staff performed independent calculations of the direct radiation dose and obtained similar results.

The licensee calculated plume doses for a member of the public located at the site boundary. The licensee used an atmospheric dispersion factor of 9×10^{-3} s/m^3 (3×10^{-4} s/ft^3). This dispersion factor was based on a member of the public standing at the site boundary, stable atmospheric conditions, wind velocity of 1 m/s (2 mi/hr), and mixing resulting from building wake effects. With the building ventilation off, the licensee calculated a plume dose of 0.008 mSv (0.8 mrem) for a stay-time of 30 days. With the building ventilation on, the licensee calculated a plume dose of 0.16 mSv (16 mrem) for the same stay-time. The staff considers a stay-time of 30 days unrealistic but useful for estimating an upper bound on the plume dose. Any variation in meteorological conditions (wind speed, wind direction, stability class) over the release period would increase atmospheric dispersion, and consequently reduce the dose estimate. The staff verified the licensee's calculations and found them to be conservative and appropriate. The staff also independently calculated doses for an infinite stay-time for the nearest permanent residences. With the building ventilation off, the staff calculated a plume dose of less than 0.001 mSv (0.1 mrem). With the building ventilation on, the staff calculated a plume dose of 0.017 mSv (1.7 mrem). The staff used an atmospheric dispersion factor of 2.6×10^{-3} s/m^3 (7.4×10^{-5} s/ft^3), as described in Section 13.1.2 of this SER. All calculated doses to a member of the general public were below the applicable regulatory limit of 1 mSv (100 mrem) specified in 10 CFR 20.1301.

13.1.4 Total Dose Consequences of the Maximum Hypothetical Accident

To quantify the total occupational dose, the staff summed the occupational thyroid CEDE and the occupational whole-body dose for a stay-time of 2 hours. This resulted in a total effective dose equivalent (TEDE) below the applicable regulatory limit of 50 mSv (5000 mrem) specified in 10 CFR 20.1201. To quantify the dose to a member of the general public, the staff summed the thyroid CEDE at the site boundary, the direct radiation dose at the site boundary, and the plume dose at the site boundary. The staff used the doses calculated for the cases in which the building ventilation is secured. This is consistent with the licensee's procedures and emergency plan. The summation resulted in a TEDE below the applicable regulatory limit of 1 mSv (100 mrem) specified in 10 CFR 20.1301. As mentioned in Section 13.1.1 of this SER, the staff performed a sensitivity study of the reactor operating time on the gaseous fission product source term. The staff found that an operating time of 4 hours would reduce total dose by approximately 40 percent as compared to an infinite operating time before the release.

13.1.5 Conclusions

The staff evaluated the licensee's assumptions and calculation methods and found them to be consistent with guidance contained in regulatory guides, NUREG series documents, and previous NRC licensing actions. The staff verified the licensee's calculations and found them to be correct. The staff performed independent calculations to verify that the licensee's results were correct and to gauge the conservatism of the dose estimates. Based on these evaluations, the staff concludes that the licensee is capable of calculating conservative doses for the MHA, and all doses resulting from the MHA would be well below the applicable regulatory limits.

13.2 Reactivity Insertion

The licensee analyzed a reactivity insertion accident for a variety of initial conditions and reactor safety system responses. Section 8.4.3 of the OSURR SAR provides discussions and analyses of these scenarios. The staff focused its evaluation on the most limiting case, a step reactivity insertion initiated at high power with no response from the reactor safety system. After considering various mechanisms for reactivity insertion and the magnitude of the insertion associated with each mechanism, the licensee appropriately assumed the reactivity insertion was the result of rapid movement of an experiment. The licensee used a conservative value of 0.93% Δk/k for the reactivity insertion based on the limit of 0.7% Δk/k for the total reactivity worth of all experiments as specified in TS 3.7.1. Additionally, the licensee used conservative values for the reactor power (600 kW(t)), the void reactivity coefficient (−0.18% Δk/k/% void), and the reactor coolant temperature (35 °C (95 °F)). As mentioned above, the licensee assumed the reactor safety system did not function and the transient terminated because of steam void formation.

The licensee used the PARET code to analyze the reactivity insertion accident. The PARET code uses a coupled neutronics-hydraulics-heat transfer model with inputs for reactor-specific parameters. Table 8.2 and Table 8.3 of the OSURR SAR list the complete set of parameters used in the analysis. The results of the transient analysis show that reactor power would increase to 49 megawatts (MW) at 0.11 seconds after the reactivity insertion. At a time 0.14 seconds after the power peaked, the maximum cladding temperature would reach 146 °C (295 °F). This temperature is well below the safety limit (SL) of 550 °C (1020 °F) specified by TS 2.1 for the aluminum cladding of the fuel. Accordingly, the staff concludes that a reactivity insertion accident, given very conservative assumptions, would not cause fuel damage or the release of fission products.

13.3 Loss of Primary Coolant

The licensee analyzed two loss-of-coolant accidents. Section 8.4.2.1 of the OSURR SAR provides a description and analysis of a hypothetical instantaneous loss of all reactor pool water. Section 8.4.2.2 of the OSURR SAR provides a description and analysis of a non-instantaneous loss of reactor pool water to the mid-plane of the reactor core. The accident scenarios assumed operation at 500 kW(t) for an infinite time. Both loss-of-coolant accidents would result in increased fuel temperature and increased radiation levels in the reactor building. The staff considers the complete, instantaneous loss of coolant to be bounding in terms of the consequences of a loss-of-coolant accident.

To demonstrate that a complete, instantaneous loss of coolant would not lead to fuel damage, the licensee cited studies of the surface temperature of plate-type fuel cooled by natural convection in air. These studies were conducted at the Oak Ridge Research Reactor (ORR). In a 1967 study, "Water-Loss Tests in Water-Cooled and Moderated Research Reactors," C.C. Webster examined data from the ORR, as well as data from similar experiments conducted at the low intensity testing reactor (LITR) and the Livermore pool-type reactor (LPTR). These reactors were light-water-moderated research reactors that utilized flat-plate fuel and were similar in design to the OSURR. Webster's conclusions indicate that plate-type fuel can withstand a loss-of-coolant accident after infinite operation at power levels up to 3 MW. Given the similarity of the OSURR design to the LITR and LPTR, the staff concludes that the results of these studies are applicable to the OSURR. Accordingly, the staff finds that, given that the OSURR licensed maximum power is 0.5 MW, a complete, instantaneous loss of coolant will not lead to fuel damage or the release of fission products.

Credible failures that could lead to a non-instantaneous loss of coolant sufficient to uncover a portion of the core include rupture of experimental facilities. If the reactor did not shut down immediately, the reactor safety system would automatically shut down the reactor because of a low pool water level of less than 4.6 m (15 ft) above the core (TS 3.2.3). The licensee calculated the draining time associated with a rupture of beam port 2. Conservatively assuming that the beam port plugs are removed and the beam port shutter is in the open position, the reactor pool would drain to the level of beam port 2 in approximately 200 seconds. During this time, radioactive decay would reduce the core fission product inventory, and the reactor operator could commence evacuation of the building. The licensee analyzed the maximum fuel plate temperature for a partially-submerged core using conservative assumptions. The licensee's analysis resulted in a maximum fuel plate temperature of approximately 259 °C (499 °F) which is well below the SL specified by TS 2.1.

A loss of coolant to the level of beam port 2 would allow gamma radiation from the decay of fission products to stream out of the reactor pool and scatter off the reactor building roof. The licensee calculated direct radiation dose rates at the reactor pool top of 170 sievert per hour (Sv/hr) (17,000 rem per hour (rem/hr)) at 200 seconds after the initiating event (time when the core is uncovered) and 5.3 Sv/hr (530 rem/hr) at 24 hours after the initiating event. The licensee calculated scattered radiation dose rates at 1.5 m (5 ft) above the reactor bay floor of 0.031 Sv/h (3.1 rem/h) and 0.0014 Sv/h (0.14 rem/h) at the same times as above. The licensee made conservative assumptions regarding power history and geometric factors when calculating the dose rates. The staff found the licensee's method for calculating dose rates to be acceptable.

Given the initial dose rates for this accident, evacuation of the reactor building would be an appropriate response. Because it was shown that no fuel damage would result, facility personnel would not be required to remain in the reactor building to perform mitigating actions. Personnel could return to the reactor building to recover from the accident after sufficient decay time and in accordance with the radiation protection principles of as-low-as-reasonably-achievable (ALARA). Assuming an evacuation time of 5 minutes, a pool draining time of 3 minutes, and a dose rate of 0.031 Sv/h (3.1 rem/h) from scattered radiation, the maximally exposed worker on the reactor bay floor would receive a whole-body gamma dose of approximately 0.86 mSv (86 mrem). This value is below the total dose resulting from the MHA. No significant risk to public health and safety would result from this accident because the reactor pool walls provide adequate shielding in all directions other than upward from the core, and the reactor building roof is uninhabited. Additionally, periodic monitoring of the activity of the reactor pool water provides reasonable assurance that the lost coolant, if it were to leak out of the reactor building, would not exceed the limits specified in Appendix B to 10 CFR Part 20 for release to the unrestricted area.

13.4 Loss of Heat Sink

The licensee analyzed a postulated loss-of-heat-sink accident. Section 8.4.1 of the OSURR SAR presents the complete scenario and consequences of this accident. The licensee assumed that the reactor cooling systems stopped removing heat from the reactor pool. The reactor was assumed to have shut down as a result of the loss of the heat sink. TS 3.2.3 requires the reactor safety system to automatically shut down the reactor if either the primary or secondary coolant pumps are off when reactor power is greater than 120 kW(t) or there is no primary coolant flow and reactor power is greater than 120 kW(t). TS 2.2 requires the reactor safety system to automatically shut down the reactor if the core inlet temperature is greater than 35 °C (95 °F) (also stated as a limiting safety system setting (LSSS) in TS 2.2). The licensee made conservative assumptions regarding heat transfer mechanisms and pathways and the power history of the reactor (5 years of continuous operation at 500 kW(t)). Conservatively assuming a reactor shutdown because of high core inlet temperature and no mitigating actions by the reactor operator, the maximum pool temperature will be 48 °C (120 °F) approximately 10 weeks after shutdown. The long delay before the pool reaches the maximum temperature is the result of conservative assumptions made by the licensee regarding heat transfer mechanisms and pathways. No fuel damage or significant radiation hazard results from this accident, and accordingly, the staff concludes that the accident would not have any significant consequences.

13.5 Loss of Normal Electrical Power

Loss of normal electrical power causes the control rod magnets to de-energize, dropping the control rods into the core and shutting down the reactor. Because the coolant pumps would shut off upon loss of electrical power, this accident is similar to the loss-of-heat-sink accident analyzed above. As stated in Section 8.2 of this SER, the OSURR does not have a requirement for emergency power. The staff concludes that loss of normal electric power would not pose a significant risk to the reactor, facility personnel, or the public.

13.6 Experiment Malfunction

The licensee analyzed two types of experiment malfunctions leading to reactivity insertions and one malfunction leading to a loss-of-coolant accident. The first reactivity insertion accident analyzed was the flooding of an in-pool experimental facility such as a beam port tube or the rabbit system tube. The licensee calculated a maximum positive reactivity insertion of 0.5% $\Delta k/k$ for flooding of one of these facilities. The analysis in Section 13.2 of this SER would bound the effects of this reactivity insertion. Section 13.2 also evaluated the second reactivity insertion accident, caused by rapid movement of an experiment. Section 13.3 evaluated the rupture of an experimental facility leading to a loss-of-coolant accident. Based on the findings discussed above and the conservative assumptions made by the licensee, the staff concludes that the licensee properly addressed experiment malfunctions and that experiment malfunctions will not lead to fuel damage or the release of fission products.

13.7 External Events

Chapter 2 and Chapter 3 of this SER discuss the meteorological and seismic hazards of the reactor site and the design criteria of the reactor structures, systems, and components (SSCs). These discussions show that the consequences of meteorologically or seismically induced reactor damage are bounded by the analyses presented in this chapter for a loss-of-coolant accident, a loss of normal electrical power, and consequently the MHA. Chapter 9 of this SER discusses fire protection. This discussion shows that onsite or offsite responders can acceptably mitigate a fire at the OSURR. As stated in Section 12.7 and Section 12.8 of this SER, the licensee maintains adequate plans and procedures to respond to emergency and security events. Based on these considerations, the staff concludes that external events at the OSURR would not pose a significant risk to the health and safety of the public.

13.8 Conclusions

The licensee analyzed an MHA and found the radiological consequences to be below the applicable regulatory limits for occupational doses and doses to members of the general public. The staff evaluated the licensee’s assumptions and methods of calculating doses and found them to be conservative and appropriate. The licensee analyzed a variety of credible, although unlikely, accident scenarios and found the consequences to be bounded by the MHA. The staff evaluated the accident scenarios and assumptions and concludes that the licensee has analyzed an appropriate spectrum of credible accidents for the OSURR and that the MHA bounds the consequences of the credible accidents. The licensee has shown that credible accidents at the OSURR do not have any offsite radiological consequences. Accordingly, the staff concludes that accidents at the OSURR will not pose a significant risk to the health and safety of the public, facility personnel, or the environment.

14 TECHNICAL SPECIFICATIONS

The staff has evaluated the TSs as part of its review of the application for renewal of Facility License No. R-75. The TSs define certain features, characteristics, and conditions governing the operation of the OSURR. The TSs are explicitly included in the renewed license as Appendix A. The staff reviewed the format and content of the TSs for consistency with the guidance found in ANSI/ANS-15.1 and NUREG-1537. Other chapters of this SER discuss the evaluations of individual TSs. The staff specifically evaluated the content of the TSs to determine if the TSs meet the requirements in 10 CFR 50.36. The staff concluded that the OSURR TSs do meet the requirements of the regulations. The staff based this conclusion on the following findings:

- To satisfy the requirements of 10 CFR 50.36(a), the licensee provided proposed TSs with the application for license renewal. As required by the regulation, the proposed TSs included appropriate summary bases for the TSs. Those summary bases are not part of the TSs

- The OSURR is a facility of the type described in 10 CFR 50.21(c), and therefore, as required by 10 CFR 50.36(b), the facility license will include the TSs. To satisfy the requirements of 10 CFR 50.36(b), the licensee provided TSs derived from analyses in the OSURR SAR.

- To satisfy the requirements of 10 CFR 50.36(c)(1), the licensee provided TSs specifying a safety limit (SL) on the fuel temperature and limiting safety system settings for the reactor protection system to preclude reaching the SL.

- The TSs contain limiting conditions for operation on each item that meets one or more of the criteria specified in 10 CFR 50.36(c)(2)(ii).

- The TSs contain surveillance requirements that satisfy the requirements of 10 CFR 50.36(c)(3).

- The TSs contain design features that satisfy the requirements of 10 CFR 50.36(c)(4).

- The TSs contain administrative controls that satisfy the requirements of 10 CFR 50.36(c)(5). The licensee's administrative controls contain requirements for initial notification, written reports, and records that meet the requirements specified in 10 CFR 50.36(c)(1,2,7,8).

The staff finds the TSs to be acceptable and concludes that normal operation of the OSURR within the limits of the TSs will not result in radiation exposures in excess of the limits specified in 10 CFR Part 20 for members of the general public or occupational exposures. The staff also finds that the TSs provide reasonable assurance that the facility will be operated as analyzed in the OSURR SAR, and adherence to the TSs will limit the likelihood of malfunctions and the potential accident scenarios discussed in Chapter 13 of this SER.

15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Operate the Reactor

As stated in 10 CFR 50.33(f), "Except for an electric utility applicant for a license to operate a utilization facility of the type described in 10 CFR 50.21(b) or 10 CFR 50.22, [an application shall state] information sufficient to demonstrate to the Commission the financial qualifications of the applicant to carry out, in accordance with the regulations of this chapter, the activities for which the permit or license is sought."

OSU does not qualify as an "electric utility," as defined in 10 CFR 50.2. Further, pursuant to 10 CFR 50.33(f)(2), the application to renew or extend the term of any operating license for a non-power reactor shall include financial information that is required in an application for an initial license. Therefore, the staff has determined that OSU must meet the financial qualifications requirements pursuant to 10 CFR 50.33(f), and is therefore subject to a full financial qualifications review by the NRC. OSU must provide information to demonstrate that it possesses or has reasonable assurance of obtaining the necessary funds to cover estimated operating costs for the period of the license. OSU must submit estimates for the total annual operating costs for each of the first 5 years of facility operations from the time of expected license renewal, and indicate the source(s) of funds to cover those costs.

OSU submitted an estimated expense and income table for the reactor facility for the 5-year period 2008 to 2012. The table shows that projected expenses range from approximately $366,000 in FY2008 to $428,000 in FY2012, and that they would be covered by income from a University allocation, research grants, commercial services, and instrumentation grants. The staff reviewed the applicant's estimated operating costs and projected sources of funds, and found them to be reasonable.

Based on its review, the staff concludes that OSU has demonstrated reasonable assurance of obtaining necessary funds to cover the estimated facility operations costs for the period of the license. Accordingly, the NRC staff finds that OSU has met the financial qualifications requirements pursuant to 10 CFR 50.33(f), and is financially qualified to hold the renewed license for the OSURR.

15.2 Financial Ability to Decommission the Facility

The NRC has determined that the requirements to provide reasonable assurance of decommissioning funding are necessary to ensure the adequate protection of public health and safety. The regulation at 10 CFR 50.33(k) requires that an application for an operating license for a utilization facility contain information to demonstrate how reasonable assurance will be provided and that funds will be available to decommission the facility. The regulation at 10 CFR 50.75(d) requires that each non-power reactor applicant for or holder of an operating license shall submit a decommissioning report which contains a cost estimate for decommissioning the facility, an indication of the funding method(s) to be used to provide funding for decommissioning, and a description of the means of adjusting the cost estimate and associated funding level periodically over the life of the facility. The acceptable methods for providing financial assurance for decommissioning are specified in 10 CFR 50.75(e)(1).

OSU has elected to use a statement of intent to provide financial assurance, as allowed by 10 CFR 50.75(e)(1)(iv) for a Federal, State, or local government licensee. The statement of intent must contain a cost estimate for decommissioning and indicate that funds for decommissioning will be obtained when necessary.

The licensee's statement of intent, dated May 14, 2007, contains a decommissioning cost estimate of approximately $3.4 million for the DECON option, and states that ". . . funds for decommissioning of the OSURR will be obtained when necessary in the future and that they will be provided sufficiently in advance of such decommissioning to prevent delay of required activities." The cost estimate includes the costs for labor, disposal of radioactive materials, and other costs (e.g., tools and supplies), and a 60% contingency. OSU assumed that labor and disposal of radioactive materials costs would comprise 90% of the total cost to decommission the facility. In reviewing the cost estimate, the staff took into consideration experience at other facilities with similar construction and operational history. The applicant stated that it will update its decommissioning cost estimate using the following methodology: current staff labor rates will be used for labor and waste disposal costs will be obtained from a waste broker. The estimate for other costs (10% of the total) will be updated based on the increased costs for labor and disposal of radioactive materials.

To support the statement of intent and the licensee's qualifications to use a statement of intent, the OSU stated that it is an instrumentality of the state of Ohio and provided references (including portions of the Ohio Revised Code and selected legal cases) which corroborate this statement. The licensee also provided information supporting OSU's representation that the decommissioning funding obligations of OSU are backed by the full faith and credit of the state of Ohio. Further, the Ohio State University Board of Trustees bylaws authorize the OSU Executive Vice President for Academic Affairs (the signator of the statement of intent) to execute contracts on behalf of the educational institution (OSU).

The staff reviewed the OSU's information on decommissioning funding assurance as described above and finds that OSU is a state government licensee under 10 CFR 50.75(e)(1)(iv), the statement of intent is acceptable, the decommissioning cost estimate for the DECON option is reasonable, and OSU's means of adjusting the cost estimate and associated funding level periodically over the life of the facility is reasonable.

15.3 <u>Foreign Ownership, Control, or Domination</u>

Section 104d of the Atomic Energy Act, as amended (AEA), prohibits the NRC from issuing a license under Section 104 of the AEA to "any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government." The NRC regulation at 10 CFR 50.38, "Ineligibility of Certain Applicants," contains language to implement this prohibition. According to the application, OSU is an agency of the State of Ohio and is not a corporation, and there is no foreign control of the University. The NRC staff does not know or have reason to believe that OSU will be owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government during the period of the renewed license.

15.4 <u>Nuclear Indemnity</u>

The staff notes that the licensee currently has an indemnity agreement with the Commission, and said agreement does not have a termination date. Therefore, OSU will continue to be a party to the present indemnity agreement following issuance of the renewed license. Under 10 CFR 140.71, OSU, as an educational institution licensee, is not required to provide nuclear liability insurance. The Commission will indemnify OSU for any claims arising out of a nuclear incident under the Price-Anderson Act (Section 170 of the Atomic Energy Act, as amended) and in accordance with the provisions under its indemnity agreement pursuant to 10 CFR 140.95, for up to $500 million and above $250,000. Also, OSU is not required to purchase property insurance under 10 CFR 50.54(w).

15.5 Conclusions

The staff reviewed the financial status of the licensee and finds that there is reasonable assurance that the necessary funds will be available to support the continued safe operation of the OSURR and, when necessary, to shut down the facility and carry out decommissioning activities.

16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

As detailed in previous sections of this SER, the NRC staff concludes that continued operation of the OSURR will not pose a significant radiological risk. The bases for these conclusions include the assumption that the facility systems and components are in good working condition. However, reactor systems and components may experience chemical, mechanical, and radiation-induced degradation, especially over years of reactor operation. Systems and components that perform safety-related functions must be maintained or replaced to ensure that they continue to protect adequately against accidents. Such systems and components found at the OSURR include the fuel cladding and the reactor safety system.

Section 4.2.1 of this document describes the reactor fuel. Possible mechanisms of degradation of the fuel cladding over time include (1) thermal cycling and high fuel temperature, (2) radiation damage, (3) erosion, (4) mechanical impact or fuel handling, and (5) corrosion.

(1) Because the OSURR does not have any pulse capability, thermal cycling occurs only because of startup and shutdown of the reactor. During a cycle, the maximum cladding surface temperature change is approximately 75 °C (170 °F). The temperature change at the cladding-fuel interface is not expected to be significantly greater. This temperature change does not have the potential to cause degradation of the fuel cladding. The licensee calculated a maximum cladding temperature of 95 °C (200 °F). This temperature is too low to cause degradation of the cladding.

(2) Aluminum-clad materials-testing-reactor-type fuel does not have a history of failure resulting from radiation damage. The Reduced Enrichment for Research and Test Reactors (RERTR) fuel development program tested the fuel type used at the OSURR at high burnup and observed no fuel failures. The staff evaluated those results in NUREG-1313. Exposure to radiation doses greater than those expected at the OSURR caused no significant degradation in similar fuel plates.

(3) Natural convection cooling does not generate the coolant velocities or pressures necessary to erode the cladding.

(4) The design of in-pool structures and components minimizes the chance for mechanical impact. The design of the standard fuel element places aluminum plates at the outside of the element, effectively shielding the cladding of the fueled plates. The design of the control rod fuel element places aluminum plates on either side of the center gap, thus effectively shielding the cladding of the fueled plates from impact with the control rod. The control rod elements do have fueled outer plates. These elements are centrally located in the core and thus protected from external impacts. Fuel handling requires specially designed tools that do not come in contact with the cladding. The core plenum shields the fuel from tools and small objects, should they fall into the reactor pool.

(5) TS 3.3 places requirements on the conductivity and pH of the primary coolant. TS 4.3 specifies surveillance intervals for the chemical properties of the coolant. These TSs adequately ensure that no significant corrosion of the cladding has occurred or will occur.

The electrical design of the reactor safety system (safety channel circuitry, control rod magnets, etc.) precludes accidents as a result of failure of system components. As discussed in Chapter 7 of this SER, failure or removal for maintenance of required safety-related instrumentation and control (I&C) components causes a safe reactor shutdown. TS 4.2 specifies surveillance requirements for the reactor safety system. These requirements ensure that gradual degradation of system components will be detected. Additionally, the OSURR staff performs regular preventive and corrective maintenance and replaces system components as necessary. Nevertheless, some equipment malfunctions have occurred. The staff's review indicates that most of these malfunctions were one-of-a-kind and typical of even industrial-quality electrical and mechanical instrumentation and components. There is no indication of significant degradation of the instrumentation and components, and there is strong evidence that the OSURR staff will remedy any future degradation with prompt corrective action.

The staff did not consider prior utilization of other systems and components because degradation would occur gradually, be readily detectable, or not affect the likelihood of accidents. Some examples include degradation of the reactor pool liner, secondary coolant pump, or chart recorders. Section 4.3 of this SER discussed pool liner degradation and repair frequency.

16.2 Conclusions

In addition to the considerations discussed above, the NRC staff reviewed licensee event reports and inspection reports. On the basis of this review and the preceding considerations, the NRC staff concludes that there has been no significant degradation of facility systems or components. The NRC staff further concludes that the surveillance requirements in the TSs provide reasonable assurance that the facility will continue to be adequately monitored for degradation of systems and components.

17 CONCLUSIONS

On the basis of its evaluation of the application as discussed in the previous chapters of this SER, the staff concludes the following:

- The application for license renewal dated December 15, 1999, as supplemented on August 21, 2002; August 18, 2005; July 26, 2006; May 22, May 31, September 4, and September 28, 2007; and February 29, 2008, 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 Chapter I of Title 10 of the Code of Federal Regulations.

- The facility will operate in conformity with the application, as well as the provisions of the Act and the rules and regulations of the Commission.

- There is reasonable assurance that (1) the activities authorized by the renewed license can be conducted at the designated location without endangering the health and safety of the public and (2) such activities will be conducted in compliance with the rules and regulations of the Commission.

- As discussed in Chapters 4, 12, and 15 of this SER, the licensee is technically and financially qualified to engage in the activities authorized by the renewed license in accordance with the rules and regulations of the Commission.

- The issuance of the renewed license will not be inimical to the common defense and security or to the health and safety of the public.

18 REFERENCES

ANSI/ANS-15.1, "The Development of Technical Specifications for Research Reactors," 1990

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UNITED STATES NUCLEAR REGULATORY COMMISSION

OHIO STATE UNIVERSITY RESEARCH REACTOR

DOCKET NO. 50-150

NOTICE OF ISSUANCE OF RENEWED FACILITY LICENSE NO. R-75

The U.S. Nuclear Regulatory Commission (NRC) has issued renewed Facility License No. R-75, held by Ohio State University (the licensee), which authorizes continued operation of the Ohio State University Research Reactor (OSURR), located in Columbus, Franklin County, Ohio. The OSURR is a pool-type, light-water-moderated-and-cooled research reactor licensed to operate at a steady-state power level of 500 kilowatts thermal power. Renewed Facility License No. R-75 will expire at midnight 20 years from its date of issuance.

The renewed license complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in Title 10, Chapter 1, "Nuclear Regulatory Commission," of the *Code of Federal Regulations* (10 CFR), and sets forth those findings in the renewed license. The agency afforded an opportunity for hearing in the Notice of Opportunity for Hearing published in the *Federal Register* on August 2, 2006, at 71 FR 43818, and September 1, 2006, at 71 FR 52173. The NRC received no request for a hearing or petition for leave to intervene following those notices.

The NRC staff prepared a safety evaluation report for the renewal of Facility License No. R-75 and concluded, based on that evaluation, that the licensee can continue to operate the facility without endangering the health and safety of the public. The NRC staff also prepared an environmental assessment for license renewal, noticed in the *Federal Register* on April 14, 2008, at 73 FR 20072, and concluded, based on that assessment, that renewal of the license will not have a significant impact on the quality of the human environment.

For details with respect to the application for renewal, see the licensee's letter dated December 15, 1999 (ADAMS Accession No. ML993610185), as supplemented by letters dated August 21, 2002 (ADAMS Accession No. ML022380431); August 18, 2005 (ADAMS Accession No. ML052350564); July 26, 2006 (ADAMS Accession No. ML062090072); May 22, 2007 (ADAMS Accession No. ML071430417); May 31, 2007 (ADAMS Accession No. ML071550098); September 4, 2007 (ADAMS Accession No. ML072490367); September 28, 2007 (ADAMS Accession No. ML072750038); and February 29, 2008 (ADAMS Accession No. ML 080650352). For details with respect to the issuance of the renewed facility license, see renewed Facility License No. R-75 (ADAMS Accession No. ML081000618), the related safety evaluation report (ADAMS Accession No. ML081000618), and the related environmental assessment dated April 7, 2008 (ADAMS Accession No. ML070230004). Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room (PDR), located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room on the NRC Web site, http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC PDR Reference staff at 1-800-397-4209 or 301-415-4737, or send an email to pdr@nrc.gov.

Dated at Rockville, Maryland, this 18th day of June, 2008.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Daniel S. Collins, Chief
Research and Test Reactors Branch A
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation