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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING POWER INCREASE TO 500 Kw

FACILITY OPERATING LICENSE NO. R-75

OHIO STATE UNIVERSITY

DOCKET NO. 50-150

1. INTRODUCTION

The Ohio State University (OSU) submitted an application to the U.S. Nuclear Regulatory Commission (NRC) for authorization to convert the Ohio State University Research Reactor (OSURR) from high enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel and to increase the maximum licensed thermal power level from 10 kW to 500 kW (0.5 MW). Attachments to this letter were the revised safety analysis report (SAR) of September 1987 supporting the conversion and the power increase, the revised technical specifications, and the environmental impact appraisal.[1] Because of time constraints on the conversion and the complexity of the power increase, the NRC decided to address these issues separately. Regarding the power increase, NRC forwarded questions to OSU seeking clarification of some issues by letters of March 30, 1988, and December 22, 1989, to which OSU responded by letters, with enclosures, of May 26, 1989,[2] February 28, 1990, and June 12, 1990.[3]

This safety evaluation report (SER) summarizes the results of the safety review of the OSURR as the safety review relates to the power increase. In addition, the SER delineates the scope of the technical details considered in evaluating the radiological safety aspects at the increased power level. The hardware modifications necessitated by the power increase are only those modifications needed to increase the capacity of the heat removal system and to increase the radioactivity monitoring system along with their associated instrumentation and controls. The modifications were reviewed as were analyses regarding thermal hydraulics, radiation protection, and accidents.

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2. REACTOR

The OSURR is designed by Lockheed Nuclear Products, and is currently licensed to operate at 10 kW. The maximum allowed excess reactivity is currently 1.5 percent $\Delta k/k$; however, primarily to accommodate burnup and the increased temperature and the xenon poisoning effect of the requested power increase to 500 kW (thermal), the licensee has requested that the maximum allowed excess reactivity be increased to 2.6 percent $\Delta k/k$.

The OSURR is a pool-type reactor that uses light water as a moderator and coolant. The fuel shaped as solid flat plates, commonly called MTR-type (Materials Testing Reactor) fuel. These fuel plates are mechanically joined into fuel assemblies that are positioned in a grid plate in a 5-by-6 rectangular matrix. This matrix has a total of 30 available fuel assembly positions. The grid plate is bolted to the floor of the reactor pool. A plutonium-beryllium (Pu-Be) neutron source whose strength at the end of 1986 was 10^7 neutrons/second provides an initial population of neutrons to the core for controlled reactor startup. Reactor power is regulated by inserting or withdrawing neutron absorbing control rods.

The OSURR achieved initial criticality in 1961. The current operating license extends through February 2000. Principally, OSU uses the OSURR for a variety of instructional, research, irradiation, and service activities. The licensee has projected, following the granting of the power increase, a conservative but realistic estimate for reactor use of 400 MWh/y. However, because the operating license contains no such limitation the NRC staff (staff) evaluations will be based on the assumption that the reactor will produce at least 1000 MWh during a normal year of operation. The principal design parameters for the OSURR are listed in Table 1.

The OSURR is located on property owned by OSU, west of the main campus. The reactor building is a multipurpose steel-framed stand-alone structure with insulated metal wall panels. The exterior ground floor dimensions are 62 feet (length) by 48 feet (width). The 62 foot length is divided into three 48 foot wide sections extending from west to east. The western section is 16 feet long by 11 feet high and is used for office space and the materials testing laboratories. The center section is 30 feet long by 35 feet high and containes the reactor pools and the reactor with its associated support facilities. The eastern section is 16 feet long by 23 feet high and contains the control room, class rooms, and offices on two levels. The entire building is heated and air conditioned by its own self-contained heating and ventilation system, which is discussed in Section 4. The interior of the entire reactor building is designated the restricted area, pursuant to 10 CFR Part 20. The exterior is designated the unrestricted area, per 10 CFR Part 20, but access to the immediate area surrounding the building is limited by fences and gates.

TABLE 1. REACTOR PARAMETERS FOR THE OHIO STATE UNIVERSITY RESEARCH REACTOR

GENERAL

Maximum Licensed Thermal Power (kW)	500
Critical mass (g U-235)	3480
Operational Mass (g U-235)	4100*
Excess Reactivity (% $\Delta k/k$)	2.6*
β_{eff}	0.00766*
Neutron lifetime, ℓ, (μsec)	66*

TABLE 1. REACTOR PARAMETERS FOR THE OHIO STATE UNIVERSITY RESEARCH REACTOR (continued)

FUEL	
Number of assemblies (maximum)	30 (5x6 array)
Number of plates per fuel assembly:	
Standard Assembly	16 Fueled 2 Dummy
Control Rod Assembly	10 Fueled 2 Guide Plates (Dummy)
Partial Assembly (Ratio of Fueled Plates to Dummy Plates)	4 6 8 10 Fueled 14 12 10 8 Dummy
Loading (g U-235/plate)	12.5
Thickness (in.)	0.050
Thickness (cm)	0.127
Clad Material	6061 Aluminum
Clad Thickness (in.)	0.015
Clad Thickness (cm)	0.038
Fuel Meat Thickness (in.)	0.020
Fuel Meat Thickness (cm)	0.051
Fuel Alloy	Uranium-Silicide (U_3Si_2)
Uranium Enrichment (%)	19.5
Active Fuel Length (in.)	24
Active Fuel Length (cm)	61
CONTROL SYSTEMS	
Shim Safety Rods:	
Number	Three
Composition	Boron-Stainless Steel
Poison Section Length (in.)	26
WORTHS (% $\Delta k/k$)	2.54, 2.22, 1.91*
Regulating Rod:	
Number	One
Composition	Stainless Steel
Control Rod Stroke (in.)	24
Control Rod Stroke (cm)	61
Nominal Drive Speed (in./s)	0.072
Nominal Drive Speed (cm/s)	0.183

* Typical values for a core having 22 fuel assemblies, 18 standard and 4 control rod assemblies, reflected on two sides by graphite and the remaining sides by pool water.

TABLE 1. REACTOR PARAMETERS FOR THE OHIO STATE UNIVERSITY RESEARCH REACTOR (continued)

REACTIVITY COEFFICIENTS

Moderator Temperature Coefficient (% $\Delta k/k/^{o}C$)	- 0.0062*
Fuel Temperature Coefficient (% $\Delta k/k/^{o}C$)	- 0.0012*
Void Coefficient (% $\Delta k/k$ per 1% moderator void)	- 0.79*

* Typical values for a core having 22 fuel assemblies, 18 standard and 4 control rod assemblies, reflected on two sides by graphite and the remaining sides by pool water.

2.1 Reactor Core

The reactor core consists of a rectangular array of fuel elements, control rods, and irradiation facilities, all of which are located on a 5-by-6 position grid plate that is positioned by a support structure at the west end of the reactor pool. To obtain the desired coolant flow, aluminum dummy elements occupy grid positions not loaded with core components. The cross section of the reactor assembly is 15 by 18 inches, has a 24 inch high active fuel length and contains 22 fully or partially loaded fuel elements for a total U-235 loading of about 4.1 kg. The reactor is controlled by means of four control rods that run vertically in four partially loaded control rod fuel elements.

2.1.1 Fuel Elements and Reflector

The fuel in the OSURR contains uranium that is enriched to less than 20 percent U-235 in the form of uranium-silicide (U_3Si_2) dispersed in aluminum. This type of fuel was developed within a program directed by the Argonne National Laboratory (ANL) for the U.S. Department of Energy (DOE). The NRC has evaluated and approved the results.[4] The active portion of the fuel plate is 0.020 inches thick by 24 inches long and contains 12.5 g of U-235. This fuel is contained in a sandwich arrangement within flat aluminum plates 0.015 inch thick for a total fuel plate thickness of 0.050 inches. The fuel plates are joined to aluminum side plates to form either standard, partial, or control rod fuel elements. All fuel elements have cross-sectional dimensions of 3-by-3 inches. Each fuel element contains both fueled and unfueled ("dummy") plates. The unfueled plates are made of pure aluminum with no U_3Si_2 content, but are the same size as the fueled plates.

The new low energy uranium (LEU) fuel elements installed in the OSURR in 1988 are of the type standardized by previous agreement among NRC, DOE, and the licensees to simplify analyses and effect reasonable economic benefit. These OSURR standard fuel elements have a total of 18 plates per fuel element. Of these 18 plates, the inner 16 contain uranium for a total U-235 loading of 200g, and the outside 2 are unfueled aluminum dummy plates. Use of the two dummy plates allows a lower loading of uranium in each standard fuel assembly,

which allows the LEU core of the OSURR to be of the proper dimensions to use the existing core hardware as designed for the original HEU reactor.

The control rod fuel elements are similar to the standard elements in that they use the standard fuel plates and a similar side plate. However, several of the inner fuel plates have been completely removed to allow a water-filled gap for the control rod to pass through the element. On either side of this gap for the control rod, there are pure aluminum guide plates. The outer two plates in each control rod fuel assembly contain uranium fuel. Each control rod element contains 10 fuel plates.

Additional partial fuel elements have the same dimensions as the standard elements except that some of the fuel plates have been replaced with pure aluminum dummy plates. Partial elements are available with 25, 37.5, 50, and 62.5 percent of the nominal uranium loading of a standard element. Use of partial fuel elements allows for precise adjustment of the excess reactivity of the OSURR core.

In the OSURR core, neutrons are reflected by reactor-grade graphite and high-purity water. Solid-graphite reflectors, about 11 inches thick, each encased in its own watertight aluminum shell, are secured to the reactor pool walls and reflect the west and south sides of the core. Where graphite is not present, most neutrons are reflected by the high-purity, light-water filling in the reactor pool. Water is the primary means for neutron reflection at the top, bottom, north, and east faces of the core.

2.1.2 Control Rods

Four control rods (three scrammable shim safety rods and one nonscrammable regulating rod) are used to control reactivity and regulate the power level at the OSURR. The measured worths of these control rods are 2.54 percent $\Delta k/k$, 2.22 percent $\Delta k/k$, 1.91 percent $\Delta k/k$, and 0.52 percent $\Delta k/k$, respectively, for a total worth of 7.19 percent $\Delta k/k$.

The active neutron absorber in the three shim safety rods is a 26-inch piece of boron stainless steel. The absorber in the regulating rod is stainless steel containing no boron. All rods have a 24-inch stroke with a total time for gravity-driven automatic reactor shutdown of less than 1 second.

The nominal drive speed for the control rods is 11 cm per minute (0.07 in./s), which corresponds to an average 0.008 percent $\Delta k/k/s$ for the rod of most worth. Conservatively doubling this to correspond to a ratio of peak-to-average axial power of about 2, results in a peak reactivity addition rate of 0.016 percent $\Delta k/k/s$ or about 0.021 dollars/s.

2.1.3 Reactor Pool

The reactor pool is built on a 15-inch thick rectangular concrete foundation. The pool has inner dimensions of 3 feet 8.5 inches by 10 feet 7 inches by 20 feet (depth), with a capacity of 5800 gallons of water. The top of the reactor core is located 15 feet below the surface of the water. The walls are constructed of barytes concrete to an elevation of 13 feet 6 inches above the foundation. The remaining 6 feet 6 inches of the walls are made with regular concrete. The pool is completely lined with a layer of fiberglass-reinforced epoxy paint to prevent water leakge, or leaching of the materials in the

concrete by the pool water, and to facilitate decontamination and repair of the walls. The reactor core is surrounded on the south and west sides by thermal column extensions. These extensions are boxes made of 0.25 inch thick 3003-H112 aluminum plates welded at the joints and filled with reactor-grade graphite. In addition, these extensions are the neutron reflectors noted in Section 2.1.1. To attenuate core gamma rays, 3 inch thick lead shields separate these thermal column extensions and the principal thermal columns. These lead shields are less than 11 inches from the reactor core faces, and have no provision for heat removal. Therefore, the temperatures in these lead shields might increase during extended operation at full power at 0.5 MW. The licensee and the staff have made independent estimates of the heating rate. Although these estimates do not totally agree, the staff analysis confirms that there is reasonable assurance that any degradation of the integrity of these lead shields will not lead to reactor core damage.

Furthermore, the unusually narrow pool dimensions for a 0.5 MW reactor might lead to accelerated radiation deterioration of the epoxy coating. However, there is no indication that abrupt failure of this coating will occur.

2.1.4 Assessment

Sections 2 and 2.1 presented an overview of the OSURR physical plant and reactor facility hardware including discussions regarding the building structure, reactor core and fuel elements, control rods, and reactor pool. None of these structural or hardware components have been modified because of the power increase. The staff has reviewed the licensee's submittal and agrees that the increased power level would not cause these physical components to perform less reliably at 500 kW than they did at 10 kW. Specifically, the staff and the licensee have investigated the increased heating of the lead shielding, possible dimensional changes of the graphite in the thermal columns, and the deterioration of the beam port gaskets caused by the increased neutron flux and concluded the increased power level does not adversely affect these effects unacceptably. A separate assessment of the reactor building heating and ventilation system is presented in Section 4, and an assessment of the water height over the core as a radiation shield during normal operations is presented in Section 6.3.1.

2.2 Dynamic Design Evaluation

The OSURR is operated by manipulating control rods in response to changes in parameters such as temperature and neutron flux (power) as measured by the instrument channels. There are interlocks to prevent excessive reactivity additions and an automatic reactor shutdown system to initiate a rapid shutdown (reactor scram) if a preset power limit has been reached. In addition, if a significant increase in fuel or moderator temperature occurs, a negative temperature coefficient throughout the operating range that results from both moderator feedback and doppler broadening of the U-238 neutron absorption resonances reduces the reactivity. This inherent limitation of reactivity provides additional operating stability and safety during inadvertent power increases.

2.2.1 Excess Reactivity and Shutdown Margin

The technical specifications for the 500 kW OSURR limit the maximum excess reactivity to 2.6 percent Δk/k in the cold, xenon-free condition, with all experiments in their most positive reactivity state. The technical specifications also require a minimum shutdown margin under any operating condition of 1.0 percent Δk/k when the shim safety rod of highest worth and the nonscramming regulating rod are fully withdrawn.

The reactivity worth of any single movable experiment in the reactor is limited by the technical specifications to less than 0.4 percent Δk/k, all movable experiments are limited to less than 0.6 percent Δk/k, any single secured experiment is limited to less than 0.7 percent Δk/k, and the maximum total value of all experiments in the reactor is limited to less than 0.7 percent Δk/k. The control rod worths for a typical core measured during the LEU initial loading and testing program at the OSURR are 2.54 percent Δk/k, 2.22 percent Δk/k, 1.91 percent Δk/k for the shim safety rods, and 0.52 percent Δk/k for the regulating rod, for a total rod worth of 7.19 percent Δk/k. Assuming the OSURR is configured with its maximum allowed excess reactivity of 2.6 percent Δk/k, the shutdown reactivity is -1.53 percent Δk/k (2.6- 2.22-1.91), which adequately satisfies the requirements in the technical specification for shutdown margins. With all rods fully inserted (the normal shutdown condition), the LEU reactor is predicted to be subcritical -4.59 percent Δk/k.

2.2.2 Normal Operating Conditions

The revised OSURR technical specifications impose a safety limit (SL) of 550°C (1022°F) on the peak fuel temperature. To protect the reactor from reaching this safety limit, the OSURR technical specifications impose a limiting safety system setting (LSSS) set point for automatic reactor shutdown at 600 kW (120 percent of full licensed power), which corresponds to a peak cladding temperature of less than 91°C (196°F). The fuel temperature safety limit for the aluminum-clad OSURR fuel is based on the blistering temperature of the clad alloy of 550°C (1022°F).[4] Calculations performed specifically for the power increase safety analysis at OSU for an enveloping overpower condition, where the initial steady state power was assumed to be 600 kW, showed the maximum cladding temperature to be less than 91°C (196°F). Normal operations at 500 kW with the overpower set points for automatic reactor shutdown i.e., scram at 600 kW (120 percent of full power) will ensure that the safety limit of 550°C (1022°F) fuel temperature safety limit is not exceeded anywhere in the core.

2.2.3 Functional Design of Reactivity Control System

The four standard OSURR control rods are positioned in the core by electromechanical devices controlled from the main console in the control room. The three shim safety rods are scrammable; the regulating rod is not. The technical specifications require the three scrammable rods to enter the core within 600 milliseconds of a scram signal. All of the control rods are actuated by rod-drive mechanisms and motors that are supported at the top of the reactor with brackets that are attached to the walls of the pool. Details concerning the protective functions of the control rod scram systems are provided in Section 5.

2.2.4 Assessment

The control rods, control rod drives, and control rod-drive mechanisms have performed reliably and satisfactorily in the OSURR for many years and provided the primary basis for safe operation of the OSURR at 10 kW. Therefore, the staff concludes that these systems should continue to operate satisfactorily and provide the basis for safe operation at 500 kW. The power increase should not affect the ability of these protective systems to function properly. In addition, throughout the full range of reactor operations, the negative temperature coefficient of reactivity should contribute significantly to safe and stable operation.

Furthermore, the technical specifications require that the core excess reactivity and experiment reactivity worths be limited so that the reactor can always be brought to a safe subcritical condition, even if the scrammable control rod of the highest worth and the regulating rod were totally removed from the reactor. The authorized 500 kW core configuration will meet all of these limitations.

The fuel element temperature safety limits at the OSURR are based on theoretical and experimental investigations and are consistent with those limits used at other similar reactors using the uranium-silicide-aluminum-clad fuel. Adherence to these limits ensures that the integrity of fuel elements will be maintained. According to safety analysis calculations performed at OSURR for power levels exceeding 500 kW, the maximum fuel element temperature at the OSURR will remain below 100°C, which is well below the prescribed safety limit, and also well within the temperature range at which the uranium-silicide fuel has been tested and evaluated. The staff has reviewed these calculations and agrees with the licensee's conclusions.

2.3 Operational Procedures

The Ohio State University has implemented administrative controls that are stated in the technical specifications and that require review, audit, and written procedures for all reactor safety-related activities. The Reactor Operations Committee (ROC) reviews all aspects of current reactor operation to ensure that the reactor facility is operated and used within the terms of the facility license and consistent with the safety of the public and the operating personnel. The responsibilities of this committee include the review of operating procedures, experiments, and proposed changes to the facility or the facility operating license, including its technical specifications. Specifically, for escalation of the power to 500 kW, the licensee has committed that the ROC will review and approve all operational steps and procedures including physics measurements and radiation monitoring at specific hold points in the planned power ascension.

2.4 Conclusion

In its review of the OSURR power increase application, the staff studied the physical plant; the reactor core; the fuel design; the control rods; the technical specification requirements on reactivity and safety limits; and the

inherent safety features. Based on this review, the staff concludes that the power increase to 500 kW will not cause the safety-related system hardware discussed in this section to perform less reliably than it did at 10 kW. The staff further concludes that there is reasonable assurance that the OSURR can safely operate at 500 kW as limited by its technical specification reactivity requirements regarding shutdown margin, excess reactivity, experiment worth, and automatic shutdown times. Additionally, the negative reactivity response associated with increased moderator temperature and moderator void formation contributes to inherent safety and reactor operating stability.

3. REACTOR COOLANT AND ASSOCIATED SYSTEMS

Natural circulation of light water in the reactor pool is the primary coolant. This water passes through both a heat removal system and a water processing system. These systems are controlled and operated independently. The heat removal system was required for the increase in operating power from 10 kW to 500 kW. The system removes reactor-generated heat from the pool water and transfers this water to a secondary heat sink (either the outside atmosphere or the city water, which is then dumped into the sanitary sewer system). The water processing system removes impurities from the pool water to meet required limits on water purity. A schematic diagram of the cooling system is shown in Figure 1.

3.1 Cooling System

Natural convective cooling enhanced by a mechanical pump is the primary means of heat removal from the OSURR core. Water enters the core at the bottom, flows upward between the fuel plates through the flow channels in the fuel elements, and is heated by the warm surfaces of the fuel plates. The heated water rises and enters a plenum at the top of the core, which controls the coolant flow. Upon exiting the plenum, the primary coolant passes into a 20 foot [illegible] inch long decay, or holdup tank, vertically mounted in one corner of the reactor pool. The delay time for the water in this tank is equal to about 11 half-lives of N-16. The outlet from the decay tank passes to the primary coolant pump and then into the primary heat exchanger. This primary heat exchanger transfers heat contained in the primary coolant (pool water) to a secondary cooling fluid. Upon exiting the heat exchanger, the primary water is returned to the reactor pool, completing the primary loop. A siphon breaker protects against accidental siphoning of the pool water through the inlet leg of the primary coolant pump if the integrity of the primary system piping is lost.

The secondary coolant loop removes heat that was transferred from the primary coolant to the secondary coolant. This secondary coolant, a mixture of ethylene glycol and water, passes through two separate heat exchangers, which removes heat to the outside atmosphere, or, under certain environmental conditions, to the city sanitary waste water system. The secondary coolant can be directed into either the outdoor cooling unit or the bypass leg. If the atmospheric air temperature is at or below about 26°C (78°F), the outdoor fan-forced drycooler, is of sufficient size to remove all heat that the core will generate when operating at 500 kW. Under normal operating conditions, this drycooler is the

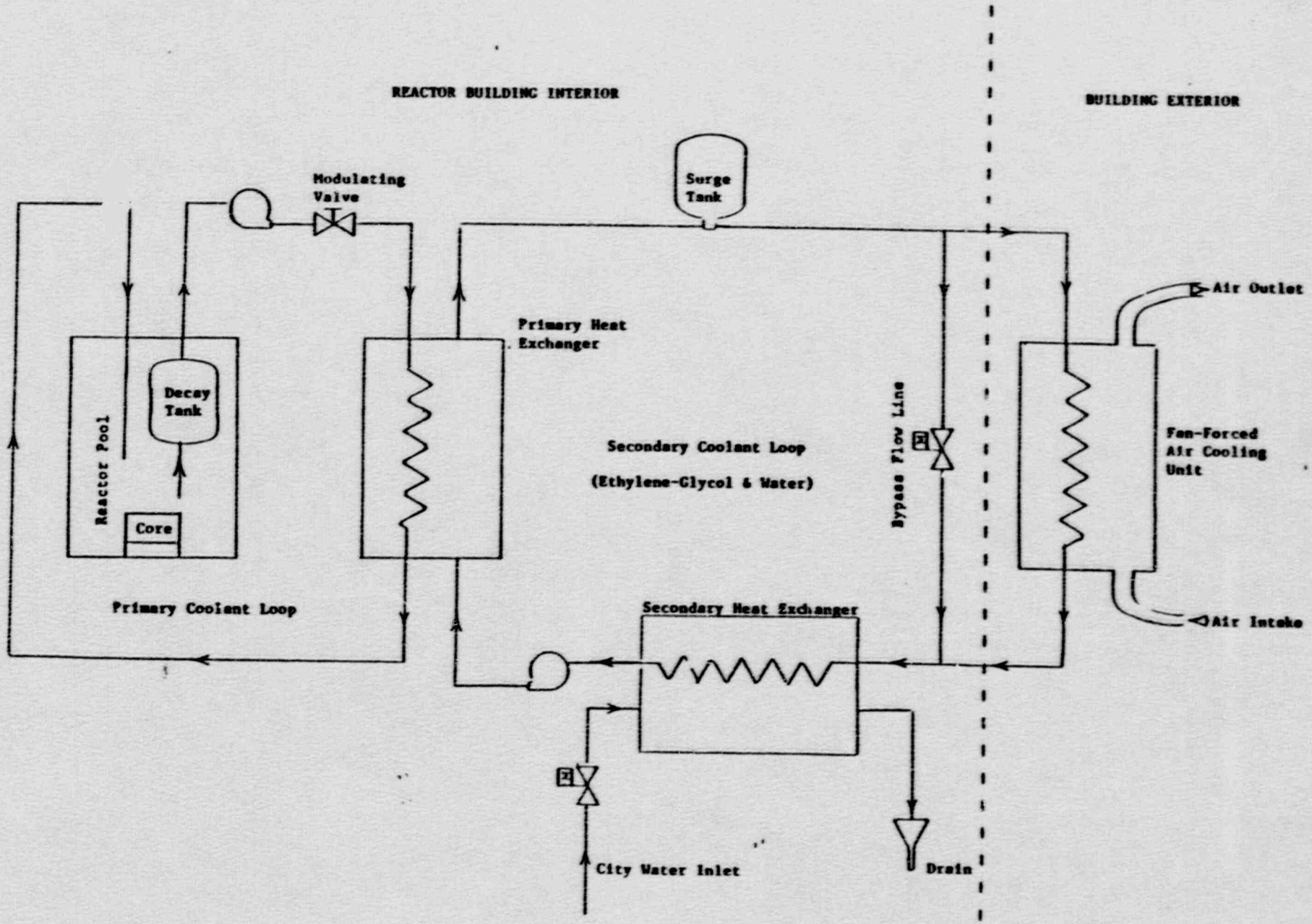

Figure 1: Schematic Diagram of the OSURR Cooling System

sole means of heat rejection. Upon exiting the drycooler, the coolant enters a heat exchanger that uses city water as its heat sink. Although normally deactivated, this unit provides additional cooling capacity for the secondary coolant if the outside fan-forced air cooling unit does not provide a sufficiently low secondary coolant temperature. The outlet from this heat exchanger connects to the secondary side of the primary heat exchanger through the secondary pump that completes the secondary coolant system loop.

3.2 Water Processing System

A water processing system purifies the water in the reactor pool. The system includes particulate and ion-exchange filters that retard fuel cladding corrosion and associated corrosion products. The water processing system uses pumps rated at 3 HP, with a capacity of 50 gal/min. The pumps enable this system to completely recirculate the water in the reactor pool every 116 minutes. Each pump is activated by either a timer or by an operator in the control room. Water from the reactor and bulk shielding facilities recirculates through particulate and ion-exchange filters of the process assembly on a 15 hour/day schedule. Monitoring instruments ensure that the water conductivity is maintained below 10 μmhos/cm.

3.3 Water Makeup System

A permutite MBD-20A mixed-bed demineralizer provides high-purity demineralized makeup water for the process system. Valves direct various fluid flows, depending on the operational mode of the regenerable demineralizer. A water-use meter records the amount of makeup water passed by the system each week. During service operations, water enters the top of the demineralizer tank, flows downward through the mixed resin bed, and exits the bottom of the unit. Check valves prevent water from flowing back up from the reactor through the makeup system and into the city water supply. Additionally, manually operated valves allowing makeup water into the system are normally closed except for actual additions, which are relatively infrequent. These valves are an additional barrier to inadvertent backflow.

3.4 Assessment

The potential risk to the public health and safety, is due to the increase in the radioactivity of the coolant. This will be the largest effect on the cooling and purification system that results from the power increase to 500 kW. An analysis of the system shows that there is very little possibility that reactor coolant will enter the city water supply because of the design features that use pressure differential, check valves, and manual shutoff valves. However, reactor coolant could enter the city sewer system through the floor drain from a loss from the primary coolant loop or a loss from the purification system, caused by pipe failure. The licensee and the staff have examined possible failure scenarios and have concluded the most adverse radiological consequences would result from the inadvertent loss of a maximum of 870 gallons of water because of a break in the primary piping upstream of the check valve, which would lower the reactor pool water to the syphon break. Assuming this

water was at the highest credible activation level, OSURR personnel have estimated a maximum radioactivity level in the city sewer system of 5.3×10^{-6} µCi/cc for sodium-24, and a maximum tritium concentration after operating for 25 years of less than 10^{-4} µCi/cc, both of which are well below the limits in 10 CFR Part 20 for water discharge to an unrestricted area, even if no further dilution were to occur.

3.5 Conclusion

The staff concludes the following:

1. The new OSURR cooling system components are of appropriate size, design, and condition to ensure adequate operation of the reactor under routine operating conditions specified in the OSURR operating license at power levels up to 500 kW.

2. The OSURR primary coolant purification system is of proper size, design, and condition to ensure that the quality of the coolant water remains sufficiently high to preclude significant water corrosion damage to the core or structure under all operating conditions permitted in the operating license.

3. The OSURR cooling system is designed such that an inadvertent injection of primary water into the city water supply system is not credible. An inadvertent release of up to 870 gallons of primary water to the city sewer system is credible; however, the associated radioactivity release is well below NRC regulatory limits, and is acceptable.

4. VENTILATION SYSTEM

Fresh air for the reactor bay and the adjoining offices, class rooms, and storage areas is supplied by the forced air handlers of the heating and air condition system. During normal operations air is evacuated through the top of the building by a 1000 cubic feet per minute exhaust fan that places a small partial vacuum on the building. Air may then enter the building through various openings (mail slot, cracks under doors). The building internal air circulation system takes a suction on the reactor bay area through return air grills and this air is heated or cooled and distributed to all rooms.

A single switch in the control room enables operators to turn off all fans, air conditioners, and furnace blowers in the building, effectively confining the air within the building. This action will quickly isolate any accidental release of radioactivity from the reactor building to the unrestricted area upon warning from the radiation alarms associated with the building gaseous effluent monitor and the rabbit vent system. Once the building has been isolated, airborne contamination will be mitigated through radioactive decay. The OSURR has no air filtration system that could control and filter a release of airborne radioactive particulates to the unrestricted area.

4.1 Assessment

The staff concludes that the ventilation system at the OSURR facility can control the release of airborne radioactive effluents to the unrestricted area during normal operations in compliance with regulations and to limit credible potential releases of airborne radioactivity to the unrestricted area during abnormal conditions. However, the staff is concerned about the ability of the facility to limit airborne radioactivity in the restricted area during normal operations. In the reactor bay, the airborne radioactivity levels will increase with the increase in power. In addition, this air will be pumped throughout the reactor building through the ventilation system. Such a monolithic ventilation system is unusual for a licensed research reactor. Most facilities either separate the heating, ventilating, and air conditioning (HVAC) systems of the reactor bay from the systems for the remainder of the building, or maintain a negative pressure in the reactor bay with respect to the remainder of the building. Each of these alternatives provides as low as reasonably achievable (ALARA) irradiation exposures to all occupants of the building and limits background radiation in laboratory spaces as much as possible.

In response to these concerns, the licensee has, in its February 28, 1990, response, committed to an extensive Ar-41 monitoring program as part of its overall startup testing program. This monitoring program will determine Ar-41 levels at 12 locations in the reactor building at power levels of 10, 100, 250, and 500 kW for various operating conditions and time periods up to times sufficient to ensure that the Ar-41 concentration is at equilibrium. When the NRC receives this information, with evaluations, the staff will have more information on which to determine the need for possible facility changes, additional administrative controls, or license changes.

5. CONTROL AND INSTRUMENTATION SYSTEM

The increased power level at the OSURR will affect the instrumentation that monitors the increased capacity coolant system and the instrumentation that monitors area radiation. In addition, the power increase may affect some safety and process-related set points on other instrumentation that monitors startup, power level, control rods, and other reactor facility and process conditions. However, the actual instrumentation and controls, that the operators at OSURR have implemented and are using for 10 kW operation, will also be used for 500 kW operation. Those instruments and controls have performed accurately and reliably at the OSURR, and increased power should not affect their performance. However, ion chambers for the nuclear instrumentation may need to be relocated to positions further from the reactor core center.

Other reactors that have similar designs and operating characteristics with the OSURR have operated safely and reliably for many years. The basic design of the OSURR instrumentation is similar to that of the Bulk Shielding Reactor and the Tower Shielding Reactor, both of which are located at Oak Ridge National Laboratory, and the Breazeale Reactor at Pennsylvania State University. Similarly, the essential features of the OSURR control system resemble those

of control systems used in operating reactors at the University of Virginia and Purdue University.

This section includes discussion of new instrumentation and controls for the coolant system and all set points for automatic reactor shutdown during operation of 500 kW. Section 6 includes discussion of new instrumentation for area radiation monitoring.

5.1 Cooling System Controls and Instrumentation

The operation of the heat removal system is controlled from the cooling system panel rack that is located above the control console. Indicators can be viewed from a seated or standing position. Various push-button and toggle switches allow operators to actuate the various system components such as pumps, valves, and fans. Various subsystems (e.g., primary water circulation loop) can be controlled independently or in conjunction with associated systems.

Cooling system indicators include display of temperature, conductivity, pressure, flow rates, valve positions, component actuation and operation, and yes/no indications of flow. Either digital or analog meters display numerical data, while simple LED displays indicate the on/off status.

Temperature of the pool water at various locations throughout the volume of the reactor pool is measured by both resistance-temperature devices (RTD) and thermocouples. Bulk pool temperature is indicated by RTD sensors located near the surface, midplane, and bottom of the reactor pool, while thermocouples indicate water temperatures at the inlet (bottom) and outlet (top) of the core. Additionally, thermocouples monitor the temperature at the inlet to and outlet from the cooling system. A thermocouple is also used to monitor the temperature of the secondary coolant. The water conductivity monitor system consists of three conductivity probes in the purification system with the control switch and readout located on the control panel.

The cooling system instrumentation and controls interface with the reactor safety system. The cooling system must operate effectively under specified reactor power conditions. To avoid automatically shutting down the reactor, both the primary and secondary cooling loop pumps must be operating, and the positive flow indications received from both primary and secondary loops, if the reactor is operated above a specified power level. Additionally, a loss of cooling system capacity (as indicated by elevated outlet water temperatures) will cause the reactor to shut down automatically. If the outlet temperature changes suddenly during a short period of time, a warning annunciator will alert the operator to anticipate a reactivity change.

5.2 Automatic Reactor Shutdown System (Scram)

The safety system protects the reactor if an event occurs that could cause the reactor to operate outside allowable system parameters. The reactor protection system (RPS) shuts the reactor down by allowing the shim safety rods to be inserted into the reactor core under the force of gravity. The system is designed to release the shim safety rods within 50 milliseconds of receipt of a shutdown signal and insert these rods into the reactor core within 600 milliseconds.

The OSURR has two types of reactor scrams, the slow scram and the fast scram. The slow scram system shuts the reactor down by turning off the current to the electromagnets holding the shim safety control rods to the rod-drive assemblies by opening a set of relay contacts. With the electromagnet current turned off, the control rods drop into the reactor core under gravity. The fast scram system turns off shim safety rod electromagnet current by biasing a current-controlling element in the magnet control modules that reduces the current flow through the electromagnets. When current is reduced below the minimum holding current, the control rod drops into the reactor core. The fast scram system is somewhat quicker than the slow scram system because the lag time associated with the opening of a relay is eliminated. The OSURR scram functions are shown in Table 2.

TABLE 2 OSURR SCRAM FUNCTIONS

Reactor Safety System Component	Minimum Required	Function
1. Core H_2O Inlet Temp.	1	Slow scram if temp. ≧35°C
2. Reactor Thermal power level (safety channels)	2	Fast scram if thermal power ≧600 kW, as indicated on calibrated ionization chamber channels.
3. Reactor Period	1	Fast scram if period ≦1 s
4. Reactor Thermal power level/coolant system pumps	1	Slow scram if coolant system pumps not on by ≧120 kW thermal power
5. Coolant Flow Rate	1	Slow scram if coolant system has no flow (primary) by ≧120 kW thermal power
6. Pool Water Level	1	Slow scram if pool level ≦20 ft (15 ft above core)
7. Switches a. Magnet Power Key "On" b. Startup Cal-Use in "Use" c. Period Generator Switch "Off" d. LOG-N Amp Calibrate Switch "Norm" e. LOG-Period Amp Calibrate Switch "Norm" f. Effluent Monitor Compressor "On"	6	Slow scram if any one switch is not properly set at the position indicated in quotes. (Also prohibits startup)
8. Recorders a. LOG-N b. Linear Level c. Startup Channel	5	Slow scram if power is lost to any one of the listed recorders

TABLE 2. OSURR SCRAM FUNCTIONS (Continued)

Reactor Safety System Component	Minimum Required	Function
d. Period e. Effluent Monitor		
9. Manual Scrams a. Control Room Console b. Pool Top Catwalk c. BSF Catwalk d. Rabbit BP Area e. Thermal Column/BP Area	5	Slow scram upon activation of any one manual scram switch
10. Compensated Ion Chambers	2	Slow scram if voltage drops below operational specifications
11. Safety Set Points On Recorders a. Period b. Linear Level c. Linear Level d. Startup Channel	4	Slow scram if associated recorder values are exceeded ≦5 s ≧120% of licensed power Servo deviation ≧Set point (nominal 10%) ≦2 cts/s (may be bypassed if K_{eff} <0.9)
12. Safety System	2	Slow scram in case of a safety amp fault or if system is discontinuous
13. Backup Shutdown Mechanisms	3	Rod drop will occur for any control rod which has excess magnet current ≧60 ma

5.3 Assessment

The control and instrumentation systems at the OSURR, which are similar to those in other operating MTR-type reactors operating at 500 kW or higher, are adequately designed and provide for reliability and flexibility. The nuclear power monitoring circuits are redundant and diverse and an increase in reactor power to 500 kW will not reduce the accuracy or reliability of the nuclear instrumentation at the OSURR. New instrumentation and controls associated with the increased capacity cooling system are adequately designed and engineered to perform their intended functions. The reactor protective system and the scram function are adequately designed to protect the reactor from exceeding any safety limits. The control system will shut down the reactor automatically if electrical power is lost or interrupted.

5.4 Conclusion

From this analysis and the formal administrative controls required in the operation of the OSURR, the staff concludes that the control and instrumentation systems at the OSURR comply with the requirements and performance objectives of the technical specifications, and that they are acceptable to adequately ensure the continued safe operation of the reactor at the increased power level of 500 kW.

6. RADIOACTIVE WASTE MANAGEMENT

The major radioactive waste generated by operation of the Ohio State University reactor consists of activated gases, principally argon-41 (Ar-41) and nitrogen-16 (N-16). The reactor also generates small volumes of liquid and solid radioactive waste, principally spent ion-exchange resins primarily in connection with the water processing system.

6.1 ALARA

The following Ohio State as-low-as-reasonably-achievable (ALARA) policy statement is quoted from its application for renewal of its materials operating license as submitted to NRC in July 1986.[5] It is signed by the vice president for health services and is deemed to be currently applicable to the power increase request.

> The administration of the Ohio State University remains committed to the ALARA philosophy, a philosophy which has always been a major part of the University's Radiation Safety Programs. The Vice President for Health Services maintains an Office of Radiation Safety, whose responsibility it is to develop and monitor ALARA concepts, and to which the authority to enforce such concepts throughout the University has been delegated. The Office of Radiation Safety is supported in these efforts by the Vice President's Office, and by the University Radiation Safety Committee and the Human Use Radionuclides Committee.
>
> Implementation of the ALARA philosophy at the Ohio State University completely incorporates all elements of the model ALARA program published in USNRC Regulatory Guide 10.8, Proposed Revision 2, August 1985, Appendix G.

6.2 Radiation Protection Program

The Ohio State University has a structured radiation safety program with personnel and equipment to detect, measure, and control area and personnel radiation exposures. Use of radioactive material and radiation sources is controlled, so that releases of radioactive material to the environment are kept to a minimum.

6.2.1 Health Physics

The Ohio State Office of Radiation Safety (ORS) provides health physics services to the OSURR. Professional and technical health physics staff oversee the radiation protection program and provide support and information for all the users of radiation and radioactive materials on campus. Reactor personnel perform the normal radiation safety function at the reactor facility. One

of the health physicists from the ORS oversees the radiation protection program at the reactor facility and participates in experiment review and approval through his position on the Reactor Operations Committee (ROC). The director of the ORS is notified of any emergency within the OSURR and when requested, he or his designated alternate assist in conducting emergency control measures. Detailed written procedures address the radiation safety support that is provided to the routine operation of the OSURR facility. Reactor personnel use a variety of detecting and measuring instruments for monitoring potentially hazardous ionizing radiation at the OSURR facility. The instrument calibration procedures and techniques ensure that any credible type of radiation and any significant intensities can be detected promptly and measured correctly.

6.2.2 Training

Persons working daily in the reactor building are trained in accordance with Part 19 of Title 10 of the Code of Federal Regulations in the appropriate radiation protection concepts. In addition, these persons are trained to assist in responding to abnormal radiological conditions in the reactor building. Individuals not working daily in the reactor building but routinely involved in continuing research and instructional activities at the laboratory, are provided instruction on self-protection against radiation exposure, according to the requirements stated in 10 CFR Part 19. Occasional visitors to the OSURR, such as commercial vendors, one-time visitors for tours and demonstrations, and nonroutine experiment personnel, are given instruction in basic health physics procedures and are escorted at all times.

6.2.3 Access Control

The entire reactor building is designated as the restricted area. The licensee states that outside doors are locked to prevent access from the outside. Persons entering the building must either be admitted by an authorized individual or be specifically authorized to have a key. Only persons trained in radiation protection and security procedures are issued building keys. Once admitted, an individual must go through the sign-in procedure and wear a dose monitor at all times.

6.3 Radiation Sources

Sources of radiation directly related to reactor operations include the reactor core, ion-exchange columns, cooling water cleanup system, and radioactive gases, primarily Ar-41, and to a lesser extent, N-16 that is generated from activation of oxygen in the pool water. The reactor core is the largest single source of direct radiation. At the increased power level, Ar-41 is the most significant radiological concern.

6.3.1 Direct Radiation

The 15 feet of water over the OSURR core provides shielding for the gamma rays produced during reactor operations. However, OSURR personnel have calculated the direct gamma radiation dose at the surface of the pool with the reactor at 500 kW to be 140 mr/h. According to 10 CFR Part 20, any area where the dose rate is high enough to provide 100 mr/h to the major portion of the human body is designated as a "high radiation area." Accordingly, if during the power ascension radiation surveillance tests, any area near the edge of the pool is shown to be a high radiation area, the licensee will implement procedures and controls to limit dose to persons required to work in this area.[1,3]

6.3.2 Airborne Radiation

Argon-41 is produced by neutron activation of the Ar-40, which is a natural component of ordinary air. Some of this air is contained in all the reactor experiment irradiation facilities, and is dissolved in the light water reactor coolant. An increase in the power of the reactor from 10 kW to 500 kW could multiply the Ar-41 source term by 50 in the reactor pool water and the irradiation facilities.

The OSURR staff assessed the Ar-41 released during normal and abnormal operating conditions both inside the reactor building (restricted area) and outside the reactor building, which is part of the campus-wide unrestricted area. The staff has reviewed the OSURR assumptions and calculations, which are presented herein, and determines their calculated values regarding sources of Ar-41 and their source term calculations, that is, curies generated and curie concentration, to be very conservative. The calculations indicate that almost 40 percent of the Ar-41 produced 15 feet below the surface of the pool is released from the surface. This is higher than typical Ar-41 release values reported for similar NPRs and is judged by the staff to represent an upper bound release of Ar-41 from the pool to the reactor room air at the OSURR. In converting the OSURR source terms to dose rate, the licensee used standard and industry-wide accepted techniques and concludes that the OSURR can be operated safely at 500 kW, and that the Ar-41 does not pose an unacceptable radiological risk to either the reactor building occupants or the neighboring student and general public populations.

The OSURR has two significant sources of Ar-41 at the OSURR, the continuous operation of the vacuum operated experiment transfer device (RABBIT) and the pool water. All other devices that contain air are sealed and do not contribute to airborne Ar-41 in the reactor building as a result of normal operations. At 500 kW and equilibrium Ar-41 production, the licensee calculated that the RABBIT generates 1.71 μCi/s and the pool releases 1.94 μCi/s for a combined release of 3.65 μCi/s.

To assess the radiological consequences in the unrestricted area, the licensee assumed that the entire 3.65 μCi/s Ar-41 is pumped outside through the 1000 cfm building fan in the reactor bay, which results in a maximum concentration in the unrestricted area of about 5×10^{-8} μCi/cm^3. This value is somewhat higher than the 10 CFR Part 20 guidelines of 4×10^{-8} μCi/cm^3 year-long average for a semi-infinite cloud. However, this calculation includes several mitigating

factors. Aside from those conservatisms previously mentioned in the source-term evaluations, the largest conservatism is that the 10 CFR Part 20 guideline of 4×10^{-8} μCi/cm^3 is based on continuous reactor operation, that is, 168 h/wk, 52 wk/y for 8736 h/y. However, the licensee currently uses the reactor for less than 200 effective full power hours (EFPH) per year, and if a conservative estimate of future operation at about 600 EFPH per year is assumed, with RABBIT operation at about 5 percent of the time, or about 30 h/y, the assessment of the maximum probable, dose in the unrestricted area is less than 40 mr/y, which is well within the guidelines of 10 CFR Part 20. If the reactor were to be operated for 2000 EFPH (1000 MWh) per year, the resultant 130 mr/y would still be within 10 CFR Part 20 upper limits. These doses were calculated by the licensee assuming the receptor was located in an infinite cloud of Ar-41. Use of this infinite cloud provides for about twice the estimated dose resulting from the more traditional semi-infinite cloud approximation, the use of which is widely accepted to provide an overestimate of likely dose rate. Also, these dose assessments represent maximum credible levels anywhere in the unrestricted area, which, for the OSURR, is near the reactor building in an area that is fairly inaccessible and normally not occupied. To provide a more reasonable assessment of upper limit dose in areas that would be considered to be normally occupied by members of the general public, a yearly dose at the nearest permanent residence (1200 feet), assuming constant occupancy, was estimated. Assuming conservative meteorological and wind conditions (Class F at 1 m/s), and using standard dispersion techniques, the source strength at 1200 feet would decrease by a factor of about 1000. This calculation yields a maximum credible dose assessment of about 0.2 mr/y to members of the general public in areas that would normally be occupied.

To assess the radiological consequences of the Ar-41 production in the restricted area (reactor building) the licensee assumed the reactor has operated at 500 kW long enough (approximately 9 h) to establish an equilibrium concentration of Ar-41. The Ar-41 source into the reactor building is from simultaneous operation of the reactor and continuous operation of the RABBIT blower system. Argon-41 is removed from the reactor building by means of radioactive decay and purging through the 1000 cfm building fan. Consistent with these assumptions, the equilibrium concentration of Ar-41 is calculated to be 6.83×16^{-6} μCi/cm^3. Exposure to the Ar-41 concentration for 600 h/y results in a maximum dose of about 280 mr/y, which is about 6 percent of the 5 rem limit in 10 CFR Part 20 for a restricted area. This calculation assumes that the receptor was located in the middle of a sphere with the same volume as the OSURR reactor building. This calculated annual dose is conservative because the contribution from the RABBIT, which is about one half the total of the calculated 280 mr/y, is overstated in that the RABBIT only operates approximately 5 percent of the time that the reactor is operating. An additional conservatism, by nearly a factor of 2, is gained because the licensee modeled the reactor building sphere of equal volume rather than the more usual practice of using a hemisphere. Furthermore, if an occupant of the reactor building were exposed at this rate for 2000 h/y, the accumulative dose would not exceed 1 r/y.

Although the continuous release of Ar-41 from the pool and RABBIT is the dominant radiological concern, the licensee evaluated two puff release

scenarios for completeness. The first scenario assumes that the RABBIT carrier tube containing an equilibrium amount of Ar-41 of 1.11×10^3 μCi is manually opened by an operator, and releases a short-term cloud of Ar-41 near the operator. Assuming the cloud lasts about 1 minute before dispersing the operator could receive about a whole body dose of about 0.04 mr. The Ohio State University has established administrative procedures based on ALARA to prevent such an event. The second puff release assumes the experiment facility with the highest equilibrium Ar-41 level instantly releases all this Ar-41 into the reactor room. This release (from beam port 2) is estimated to result in a whole body dose of about 3.4 mr to any personnel who remain in the reactor bay long enough for the Ar-41 to decay (about 2 hours). This type of event is precluded because the experiment ports are normally capped, which eliminates Ar-41 release.

Radioactive N-16 is formed in the core of the OSURR. However, because of the 15 foot depth of the core and the placement of the decay tank, the licensee has conservatively estimated the transport time from the core to the surface of the pool to be at least 11 times the 7.4 second half-life of N-16. Therefore, the release of radioactive N-16 to both the restricted and unrestricted environment is considered not to be significant.

6.3.3 Solid Radioactive Waste

Operation of the OSURR will generate very little solid low-level radioactive waste. The primary source of low-level solid waste will be the demineralizer cartridge in the reactor pool water processing system. Current procedures at the 10kW power level require the cartridges to be kept onsite for a decay period sufficient to reduce the activity of short-lived radionuclides to negligible levels, then to be shipped to their manufacturer for regeneration. Because operation of the OSURR at 500 kW will generate additional radioisotopes in the resins of the demineralizer cartridge, the licensee will perform a bulk radioassay to determine specific and total activity in the cartridges before releasing them from the restricted area.

If the manufacturer does not continue to regenerate the cartridges, the resins may have to be removed by the OSURR staff and replaced with fresh resins. This would generate a quantity of radioactive material to be transported offsite as low-level waste. Experience with similar reactor facilities indicate that about 13 ft^3 of resin might be generated once or twice a year. OSU already has established arrangements for transporting low-level waste in accordance with all applicable regulations.

Spent fuel assemblies may also be classified as solid radioactive waste. These assemblies are stored in the fuel storage pit at the east end of the reactor pool, unless otherwise approved by the ROC and the NRC. After suitable decay times, spent fuel assemblies are returned to the DOE under an existing agreement for ultimate disposal. Because of their isotopic content and activity inventory, used fuel assemblies are not considered low-level radioactive waste. Therefore, the assemblies are handled separately from other waste forms generated by the laboratory. Spent fuel shipment would be performed in accordance with approved procedures that meet appropriate federal, state, and local requirements.

6.3.4 Liquid Radioactive Waste

No liquid-borne radioactive materials are discharged from the OSURR during normal operation. However, certain maintenance and repair activities may result in accumulation of liquid wastes.

As an alternative to the regeneration of demineralizer cartridges by the manufacturer or replacement of the resins by OSURR personnel, the OSURR staff would regenerate the cartridges onsite. This alternative would release some liquid-borne radionuclides. These radionuclides eventually might be released from the reactor building as liquid radioactive waste. Before the release of liquid radioactive waste, OSURR staff would determine the isotopic content of the material and specific activities of radioisotopes present in the liquid. The liquids may then be kept in a holding tank to allow decay to reduce the total activity inventory, or for later dilution or other treatment before release. If these procedures are required, OSU would need to acquire approved storage facilities, which it does not currently possess. In any case, liquid radioactive waste will be disposed of in accordance with all applicable regulations.

6.4 Area Radiation, Effluent, and Environmental Monitors

The area radiation monitors (ARMs) provide information on radiation levels around the reactor building. These monitors alert the reactor operator and people in the surrounding areas to the existence of radiation dose rates above a specified set point. The ARM system uses four model GA-2TMO plastic scintillator detectors manufactured by the Nuclear Measurement Corporation. These monitors 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 ARM system includes duplicate display instrumentation at both local (near the detector position) and remote (control room) locations. Analog meters indicate gamma radiation levels from 0.1 to 1000 mr/h. Amber and red lamps indicate high background and alarm conditions, respectively. An alarm bell is located atop each of the local ARM meters, and provides a local warning to personnel in the area.

An effluent monitoring system detects gaseous radionuclides being released to the unrestricted area. This system extracts a sidestream of air ejected from the reactor building by the building ventilation fan. The sampled air is introduced to a shielded volume containing a double-sided pancake-type Geiger-Mueller detector. Detector output is counted on a rate meter in the control room, and recorded on a panel-mounted stripchart recorder. System response is calibrated for Ar-41 activity. Detector count rate is noted in the control room. The count rate corresponding to the 10 CFR Part 20 guideline of 4×10^{-8} $\mu Ci/cm^3$ for Ar-41 in an unrestricted area is posted at the recorder. With 500 kW operation, a check will be made each day the reactor is operated to compare the concentration of Ar-41 at the point of release to this unrestricted area guideline.

For current environmental monitoring during operation at 10 kW, film badges are mounted at four different locations about 20 feet outside the reactor building,

but within the security fence. For 500 kW operations, the licensee has committed to add additional monitors outside of the restricted area.[3] At least four of these monitors will be positioned in the unrestricted areas about 100 feet from the reactor building. These areas are usually unoccupied in all directions, except at the Van de Graaff Laboratory, which is located about 150 feet to the south. The licensee should implement the additional monitoring well in advance of actually increasing the reactor power level, so that personnel can accurately compare radiation levels.

6.5 Conclusions

The staff considers that the radiation protection program at the OSURR has received acceptable support from Ohio State University for 10 kW operation and that adequate additional support, as needed for 500 kW operation, will be provided. At the OSURR facility the OSURR staff has conducted and are expected to conduct the waste management activities when OSURR is operating at 500 kW power level in a manner consistent with 10 CFR Part 20 and the ALARA principles. Among other guidance, the staff review has followed the methods of ANSI/ANS 15.11, "Radiological Control at Research Reactor Facilities."[6]

Because Ar-41 is the only significant radionuclide released from the reactor to the environment during normal operations, the staff has reviewed the history, current practices, and future expectations of reactor operations at 500 kW with respect to this radionuclide. Additionally, the staff has reviewed the licensee's assessment of the increased Ar-41 production at 500 kW and concurs that the calculated source terms and dose rates in the reactor building (restricted area) and immediately outside the reactor building (unrestricted area) provide a high estimate of the possible dose to those people associated with operations of the OSURR, the OSU student body, and the general public, as a result of normal reactor operation. Furthermore, the staff has reviewed and accepted the licensee's commitment to make and report to the NRC the extensive radiological measurements at the reactor facility during incremental increases of power. This review gives reasonable assurance that exposures to Ar-41 produced by the reactor can be maintained well within the guidelines of 10 CFR Part 20 at 500 kW.

7. ACCIDENT ANALYSIS

The staff has evaluated the documentation submitted by the staff of the Ohio State University and its analysis of possible site-specific accidents. These analyses included the various types of possible accidents and the possible consequences to the public because of operation of the OSURR. Of the various accident scenarios analyzed in the 1987 safety analysis report (SAR) for the power increase and fuel conversion from high-enriched uranium (HEU) to low-enriched uranium, the three scenarios directly affected by the power increase are the loss-of-coolant accident (LOCA), the reactivity insertion accident, and the fuel plate failure accident.

Of these possible events, the fuel plate failure accident with the postulated complete loss of cladding integrity of one irradiated fuel plate in the reactor pool could release the highest level of radioactivity to the environment, both inside and outside the OSURR facility. Thus, the fuel plate failure accident will be designated as the maximum hypothetical accident (MHA). The results of the analyses of the other two credible accidents with less severe consequences than the MHA are also addressed in this section.

7.1 Fuel Plate Failure Accident

This accident scenario, designated the MHA, addresses various challenges to at least one or more irradiated fuel elements that could cause a breach or rupture in the fuel cladding. To remain conservative, the staff and OSURR did not try to develop a detailed failure scenario, but the staff concurred with OSURR that the upper limit scenario is the complete failure of the cladding of one fuel plate underwater in the reactor core, and the release of fission products to the water.

Because the fission product noble gases do not condense or combine chemically, any noble gases released are assumed to diffuse until radioactive decay has reduced the concentration to an insignificant value. Conversely, the iodines are chemically active but are not volatile at temperatures below approximately 180°C. Some of these radionuclides will be trapped by the materials they come in contact with, such as water and structures. Evidence indicates that most of the iodines either will not become or will not remain airborne under many accident scenarios that are applicable to nonpower reactors.[7,8,9] However, to be certain that the fuel-cladding-failure scenario leads to upper-limit dose estimates for all possible events, the licensee assumed that all the cladding was stripped off both sides of a fuel plate, totally exposing the meat, and that 100 percent of the iodine in the fuel is released into the pool and a calculated fraction of this iodine then percolates through the water into the air in the reactor bay. These assumptions will lead to computed thyroid doses that most certainly are high for most credible scenarios. Additionally, OSURR assumed that 100 percent of the noble gases contained in the fuel plate are released into the pool and instantly reach the air in the reactor bay. These assumptions will lead to unrealistically high noble gas immersion doses for most reasonable scenarios.

7.1.1 Scenario

OSURR personnel evaluated an accident scenario that assumed all of the cladding was stripped from both sides of one fuel plate under water in the reactor causing the release of 100 percent of its volatile fission products. The analysis assumed that the cladding failure occurred in the highest power fuel plate following an extended run at the maximum licensed power level of 500 kW so that all fission products had reached their saturated activity levels. This assumption is conservative considering the typical operating history at the OSURR. The analysis assumed that all fission-product radionuclides were still at saturated activity levels at the time of release from the cladding. All the noble gases and halogens in the fuel plate are assumed to be released into the pool water instantaneously. The licensee calculated the release fraction of iodine that reaches the reactor bay air to be 4×10^{-4}. The analysis assumed that 100 percent of the noble gases reach the reactor bay air. Using this scenario as a basis, the whole-body immersion dose (gamma-ray) and the possible thyroid dose from iodine inhalation were calculated for an individual in the reactor building (occupational) and in the unrestricted area immediately outside the reactor building (maximum exposed member of the public).

For the occupational exposure, the analysis assumed that all of the fission products released from the pool are uniformly mixed with the air in the 1982 m^3 reactor building. The analysis also assumed that the 1000 cfm reactor bay unfiltered air exhaust fan was shut down at the time of the accident, the core contained 312 fuel plates, and the failed fuel plate developed a power level approximately 2.75 times as high as an average fuel plate. The postulated MHA could not credibly occur without the operating personnel being immediately alerted by radiation monitors, the analysis assumed that the reactor building would be evacuated in an orderly manner within five minutes. For a one hour public exposure, the analysis assumed that the unfiltered air exhaust fan was shut down and the fission products present in the reactor building leaked to the outside at the rate of 10 percent of the building volume per day, but that source strength was not decreased by the decay of the radioactivity. The whole body dose calculations assumed immersion in a uniform finite cloud in the reactor building and in a uniform semi-infinite cloud in the unrestricted area. Table 3 lists the doses calculated by the licensee for these assumptions and locations.

TABLE 3. OHIO STATE CALCULATED DOSES RESULTING FROM THE POSTULATED FUEL PLATE FAILURE ACCIDENT (MHA) AT THE OSURR.

Exposure and Location	Whole-Body Immersion Dose (mr)	Thyroid Committed Dose (mr)
5 min (occupational) exposure in reactor room	1200	472
1 h (public) exposure immediately outside the restricted area	18	0.2

7.1.2 Assessment

In the assessment of the licensee's results presented in Table 3, the staff concurred that the scenario involving the total loss of all cladding as postulated by the OSURR staff is an upper-limit accident for fission product release. Because the immersion doses listed in Table 3 result almost entirely from the 100 percent noble gas release, these values of a 1200 mr occupational dose and a 18 mr public dose, are upper-limit estimates and, as such, are determined to be acceptable by the staff. A supplementary calculation that assumed a one hour public exposure was performed where it was assumed the unfiltered air exhaust fan continued running. Even for this relatively unlikely condition, the one hour immersion dose in the unrestricted area was less than 1 rem.

To assess the licensee's calculated thyroid committed dose, the staff verified the total iodine inventory of the highest power OSURR fuel plate, and used methods contained in NUREG/CR-2079[7] and Regulatory Guide 1.25[10] to estimate the fraction of this iodine inventory that reaches the reactor bay environment. Thyroid dose estimates obtained in this manner were consistent with the licensee-calculated results. This assessment demonstrates the radiological

consequences to the thyroid for both the occupational and public exposure, caused by the MHA, do not pose an unacceptable safety risk. A supplementary calculation for a one hour public exposure for the thyroid dose, assuming the building exhaust fan continued running, produced a dose of 38 mr, which still does not pose a public safety risk.

Because of the conservative nature of the assumptions regarding the maximum hypothetical fuel plate failure accident, and because of the conservative approach taken to analyze the resultant doses, the staff concludes that there is reasonable assurance that the calculated consequences of the postulated fuel plate failure accident do not pose an undue health risk to OSURR reactor operations personnel, students, or the general public.

7.2 Reactivity Insertion Accident

The power increase to 500 kw will increase the core excess reactivity from 1.5 percent $\Delta k/k$ to 2.6 percent $\Delta k/k$. Therefore, the power increase will affect the assessment of the reactivity insertion accident. OSU demonstrated that this increased excess reactivity results from the increased power defect, the fission-product poisoning, and the U-235 burnup associated with the higher power operation. The staff has reviewed OSU's reactivity requirements and agrees that the increase of excess reactivity from 1.5 percent $\Delta k/k$ to 2.6 percent $\Delta k/k$ is warranted. In Section 2, the staff assessed and found acceptable the safety implications of 2.6 percent $\Delta k/k$ excess with regard to rod worth, shutdown margin, and experiment requirements. In this section, the staff assesses safety implications regarding an inadvertent reactivity addition because of a 2.6 percent $\Delta k/k$ core excess.

7.2.1 Scenario

The staff has reviewed the licensee's postulated scenario of OSURR operation at zero power and full power of 500 kW, at which time some failure mechanism incrementally inserts added reactivity. An analysis of the OSURR reactor and experiment facilities shows that inadvertent incremental additions of reactivity can credibly be caused by two mechanisms: experiment movement, or an accident leading to flooding of the experimental facility. The maximum worth of any movable experiment is 0.4 percent $\Delta k/k$, with the maximum of all experiments combined of 0.7 percent $\Delta k/k$. The sudden inadvertent flooding of an experiment facility could add a maximum of 0.5 percent $\Delta k/k$. However, the staff does not consider as credible the inadvertent flooding of the experimental facility coincident with an experiment movement that would add the maximum authorized possible reactivity. Therefore, the licensee has postulated that a single failure involving experiment movement or the flooding of the experimental facility can produce a maximum inadvertent addition of 0.7 percent $\Delta k/k$ of reactivity.

The staff does not consider as credible an inadvertent addition of more than 0.7 percent $\Delta k/k$ of reactivity from a control rod fault, even though three of the four control rods are worth more than 0.7 percent $\Delta k/k$. Such an addition is not credible because of the standard basic design of the OSURR control rod system. In this system the control rods are magnetically connected to their motor-driven drives and pulled upward out of the core to add reactivity. Upon scram, the magnets are de-energized and the rods fall by gravity into the core. The OSURR system contains no physical forces that would result in the upward movement of a control rod, such as coolant flow or pressure.

For the zero power case, the staff performed an analysis that is similar to the analysis presented in HEU/LEU conversion documentation,[11] which assumed any credible reactivity accident would be enveloped by a 1.5 percent $\Delta k/k$ incremental increase in reactivity added to the cold, clean reactor core. This step reactivity insertion resulted in a reactor period of 9.2 ms, and a maximum fuel plate temperature below the blistering temperature of the aluminum-clad alloy. Temperature and void coefficient measurements[12] for the LEU core confirmed that these parameters are more negative (temperature: -0.0062 percent $\Delta k/k/°C$, LEU, versus -0.0021 percent $\Delta k/k/°C$, HEU; void: -0.79 percent $\Delta k/k$ 1% void, LEU, versus -0.28 percent $\Delta k/k$ 1 percent void, HEU) than the parameters used for the HEU analysis herein previously. Because these parameters are more negative and because the consequence parameters for a 1.5 percent $\Delta k/k$ increase are more severe than a 0.7 percent $\Delta k/k$ increase the staff has assurance that the aluminum cladding of the OSURR fuel plates will remain significantly below the safety limit of the aluminum cladding during the maximum credible reactivity accident at zero power.

For the the maximum postulated incremental reactivity insertion at full power, the OSURR staff postulated the following conditions. Initial power level is 600 kW, reactivity insertion is a 0.84 percent $\Delta k/k$ increment, and the 120 percent overpower is assumed to fail. The initial power of 600 kw is conservative because the steady state fuel temperature is higher than the fuel temperature would be at 500 kW. Therefore, a smaller temperature increase would be necessary to induce fuel or clad failure, as caused by the power transient. The 0.84 percent $\Delta k/k$ increment is conservative because it is 20 percent higher than the 0.7 percent $\Delta k/k$ increment that is the postulated maximum reactivity addition at full power operation, and the failure of the system to scram provides the opportunity for the analysis to demonstrate the effects of the negative temperature and void coefficients in the termination of this transient. In this analysis, the power peak is 49 MW at 0.11 seconds after the reactivity insertion, and the cladding reaches a maximum temperature of 146°C, 0.03 seconds later. The power transient, and hence the temperature rise, is arrested by the negative reactivity feedback caused by the increased temperature and the steam void formation in the coolant channels. Void formation causes a somewhat slower decrease in reactor power than would be experienced if the reactor scrammed (i.e., control rods were released to drop into the core). However, the power does significantly decrease in several seconds. In addition, the temperature of the cladding stops rising and begins to drop, but tends to approach an equilibrium level as the reactor power approaches decay power. However, the maximum cladding temperature never exceeds 150°C, and remains well below the blistering temperature of aluminum.

7.2.2 Assessment

The staff agrees with the licensee that the inadvertent maximum credible incremental reactivity addition at the OSURR is 0.7 percent $\Delta k/k$, and that the maximum cladding temperature calculated for this transient at zero power or full power conditions at the OSURR will not result in fuel or aluminum cladding failure. Therefore, the staff concludes that there is a reasonable assurance that the fission products contained in the fuel will not be released to the environment as a result of the postulated rapid insertion of reactivity accident.

7.3 Loss-of-Coolant Accident

The rapid loss of pool water, and its inherent shielding effects, immediately following reactor operation is a possible accident that results in increased fuel and cladding temperatures and increased radiation levels in the reactor room. Because water is required to moderate the neutrons, the loss of coolant in the reactor terminates the neutron chain reaction and, thus, terminates the fission power production. However, the residual radioactivity resulting from fission-product decay would continue to deposit heat energy in the fuel, and would become an unshielded radiation source in the bottom of the tank.

Both OSU and the staff consider the complete loss of cooling water at the OSURR to be a very unlikely event. Because all primary water lines enter the reactor tank from the top of the vessel, draining through an improperly aligned valve is not possible. Water can only be pumped from the vessel down to the suction break that still provides for about 12 feet of water above the core. The steel tank liner of the vessel itself and the associated concrete structure were initially designed and thoroughly tested to ensure tightness against leakage. Since that time, no leaks have been identified or suspected. However, the OSURR loss-of-coolant accident assumes that some unidentified failure mechanism leads to a complete and rapid loss of water from the pool. The staff concurs that this accident would constitute a worse-case scenario for loss of coolant.

To show that a sudden loss of coolant at the OSURR would result in no fission-product release through melted fuel, the licensee cites studies[13] of the surface temperature of the fuel elements at the Oak Ridge Research Reactor (ORR) with natural air convective cooling. Webster,[14] correlated this work, along with experimental results from the Low Intensity Testing Reactor (LITR) and the Livermore Pool-Type Reactor (LPTR). Webster's conclusions are pertinent because the ORR, LITR, and LPTR are all light-water moderated research reactors with solid plate-type fuel (commonly called MTR- or BSR-type elements), which are very similar in design to the OSURR. These studies indicate that this type reactor could sustain a loss of pool water without causing fuel melting following infinite operation at 3 MW. Therefore, these studies indicate that the OSURR, which is operated at 500 kW or less for times of 30 hour or less (postulated maximum operating time), will not experience fuel melting following a rapid loss of all pool water.

Dose levels were calculated at three positions in the reactor area: the pool top, the second floor, and the ground floor. Time is measured from the cessation of infinite operation at 500 kW. At the pool top, the direct dose and scattered dose were calculated. At the second and ground floors, only the scattered dose was calculated, because there is no line-of-sight exposure to the reactor core at these locations. The results are presented in Table 4.

TABLE 4. DOSE LEVEL CALCULATIONS

Location	Dose Rate R/h		
	199 s	1 h	24 h
Pool Top (Direct)	3451.00	1931.00	1019.00
Pool Top (Scattered)	2.24	1.25	0.66
Second Floor (Scattered)	1.14	0.64	0.34
Ground Floor (Scattered)	0.41	0.23	0.12

Clearly, dose rates of this magnitude would require the reactor building to be evacuated. Following this evacuation, the hazard to the public would not increase further because the radiation from the unshielded core would be collimated upward by the tank and shield structure toward the uninhabited ceiling area of the reactor building.

7.3.1 Assessment

The staff has reviewed OSU's LOCA analysis, and concurs with the assumptions and general methods. Because of the design of the reactor pool and because the pool has no penetrations below the normal water level, the staff agrees with the licensee that loss of coolant at the OSURR is a very improbable event. However, should a LOCA occur at the OSURR, the staff agrees that the analysis presented by the licensee provides reasonable assurance that the maximum core temperature will remain below that necessary to induce any cladding failure.

7.4 Conclusion

The staff has evaluated the credible SAR accidents for the OSURR that would be affected by an increase in the maximum authorized power level from 10 kW to 500 kW. From this evaluation, the staff concurs with the licensee that the postulated accident with the greatest possible effect on the environment is the total loss of cladding integrity of one irradiated fuel plate in the highest power region of the core, 15 feet below the surface of the pool. However, the analysis of this accident, shows that even under conservative assumptions, the expected dose equivalent in restricted and unrestricted areas still would be significantly below the guideline values of 10 CFR Part 20.

Of the other SAR accidents analyzed, the only two accidents that would be altered by the increased licensed power are the loss-of-coolant accident and the reactivity insertion accident. As shown in Section 7.3, the likelihood of a LOCA is very small at the OSURR. Even if the accident were to occur under the most pessimistic assumptions, the staff concludes that the licensee has provided reasonable assurance that a LOCA would not result in fuel damage or cladding failure with a consequent release of radioactivity to the environment. Also, as shown in Section 7.2, the reactivity insertion accident initiated at

both zero power and full power would incude the maximum credible reactivity addition to the OSURR would not result in core power or temperatures that would lead to cladding failure.

Therefore, the staff concludes that the design of the facility and the technical specifications provide reasonable assurance that the OSURR can be operated with a low chance of accidents and that even the maximum hypothetical accident will pose no significant risk to the health and safety of the OSURR reactor operations personnel, the OSURR student population, or the general public.

8.0 ENVIRONMENTAL CONSIDERATION

The Commission has prepared an environmental assessment (EA), which was published in the Federal Register on November 13, 1990 (55FR47407). On the basis of the EA and the safety evaluation referenced herein, the Commission has determined that no environmental impact statement is required and that issuance of this amendment will have no significant adverse effect on the quality of the human environment.

9.0 CONCLUSION

Based on the considerations discussed herein, the staff has concluded that (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed activities, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

10. REFERENCES

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(14) C. C. Webster, "Water-Loss Tests in Water-Cooled and Moderated Research Reactors," Nuclear Safety, Vol. 8, No. 6, November-December 1967.