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August 21, 2002

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U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: REQUESTED ADDITIONAL INFORMATION

Gentlemen:

Enclosed are responses to questions seeking additional information and clarification regarding the application for renewal of Facility Operating License No. R-75 for the Ohio State University Research Reactor (OSURR) submitted December 15, 1999. These responses have been reviewed and approved by the The Ohio State University Reactor Operations Committee. Questions regarding these responses should be directed to Mr. Richard Myser, Associate Director of the OSURR at 614-688-8220.

Sincerely,

James C Williams

James C. Williams
Dean, College of Engineering

c: D. Hughes, USNRC; D. Miller, OSURR; R. Myser, OSURR; U.S. NRC Region III

STATE OF OHIO,
COUNTY OF FRANKLIN, ss:

Before me, a Notary Public, in and for said county, personally appeared JAMES C. WILLIAMS, who acknowledged that he did sign the foregoing letter and that the same is his free and voluntary act and deed for the uses and purposes therein set forth.

IN TESTIMONY WHEREOF, I have hereunto set my hand and affixed by official seal this 21st day of August, 2002.

NOTARIAL SEAL STATE OF OHIO

STACY WEISLOGEL, Attorney-At-Law
NOTARY PUBLIC - STATE OF OHIO
My commission has no expiration date.
Section 147.03 R.C.

A020

Response to NRC Request for Additional Information Dated February 5, 2002
The Ohio State University Research Reactor
Docket No. 50-150

1. *Please list the shared facilities or equipment, if any, and discuss the impact on the operation and safety of the facility. (Reference: NUREG 1537, Part I, Section 1.4)*

There are two gamma irradiators and a subcritical assembly that are housed in the Reactor Building.

A. A Tech/Ops Model 547 Specimen Irradiator is currently located on concrete blocks and is positioned six feet east of the biological shield for the reactor. It has been housed in the Reactor Building since 1989 and is currently licensed by the State of Ohio # 0211025037. It is a self-shielded unit and contains about 135 curies of Cs-137.
B. A self-contained wet storage irradiator for in water irradiations is currently located in the Bulk Shielding Facility pool south of the reactor pool. It has been housed in this pool since 1992 and is currently licensed by the State of Ohio #0211025037 and contains about 3190 Curies of Co-60.
C. A graphite moderated subcritical assembly contains 1298 Kg of natural uranium in 720 cylindrical slugs canned in aluminum tubes. It is located about five feet north of the biological shield for the reactor. It has been in this location since 1991 and is currently licensed by the state of Ohio # 01129250037.

The Nuclear Regulatory Commission (NRC) originally licensed each of these when they were first housed in the Reactor Building. Their installation and utilization has been examined by the NRC, the State of Ohio Bureau of Radiological Health, the Radiation Safety Section of the Ohio State University and the staff of the OSU Nuclear Reactor Laboratory for about ten years. During this time there has been no adverse impact on the operation and safety of the OSU Research Reactor and none is anticipated in the future.

2. *Provide a list of or confirm that there are no other reactors with principal similarities with OSURR now that it has low-enriched uranium silicide-aluminum dispersion fuel. (Reference: NUREG 1537, Part I, Section 1.5)*

The Ohio State University Research Reactor (OSURR) began operations with LEU fuel in 1988. At that time, it was the only reactor in this country or worldwide that used this fuel form. Since then, others have adopted its use. In this country, these include the University of Missouri at Rolla, The Rhode Island Nuclear Science Center, and the University of Massachusetts at Lowell. Reactor facilities in this country using LEU prior to being permanently shutdown include The University of Virginia Reactor, The Manhattan College Zero Power Reactor, and the University of Iowa. Of these, the University of Missouri at Rolla is most similar to the OSURR in operating characteristics and facility features.

3. *Briefly discuss reactor operations, experimental programs, and mission of the reactor in reference to the license renewal period. Refer to current and proposed operational plans. (Reference: NUREG 1537, Part I, Section 1.6)*

The Ohio State University is a land grant institution that provides teaching, research, and service. The OSU Nuclear Reactor Laboratory's mission is to provide nuclear science teaching, research, and service opportunities for students, faculty, and the public. Based on the period 1991-2001 the average thermal energy production was 56,359 KW-hrs/yr. We anticipate this average level of utilization to continue for the license renewal period. The highest annual utilization from 1991 to 2001 was 160,000 KW-hrs. Again, we do not expect to exceed this utilization in any one year of the license renewal. Operations are scheduled on an as needed basis with a balance between teaching, research, and service. Most teaching utilization involves operating the reactor at 20KW or less, most service operations are at 5KW, and research operations are typically up to 500KW.

4. *What is located in the area up to 8 km (4.8 miles) radius from the reactor site? What are the demographics in the area up to 8 km (4.8 miles) radius from the reactor site? Are there any hazardous industries located in the area up to 8 km (4.8 miles) radius from the reactor site? Are there any major rail lines located in the area up to 8 km (4.8 miles) radius from the reactor site? Please assess the affect on the safety of the reactor facility if any.*

Within an eight kilometer radius of the reactor building there are numerous manufacturing and processing facilities, machine shops, shopping centers/malls, research and development facilities, laboratories, educational, and medical facilities. These include The Ohio State University, Battelle Memorial Institute, A GM stamping plant and a fertilizer plant. This area also encompasses residential communities including portions of Columbus, Upper Arlington, Grandview, and Bexley. In addition the fiscal and administrative center of the City of Columbus and the State of Ohio including the central business district are included in this radius.

Within this radius of eight kilometers lie 89 census tracts housing a resident population of about 259,304 inhabitants.

The Franklin County Emergency Response Plan lists forty-five facilities with extremely hazardous substances within this eight-kilometer radius.

The Marion line of the CSX Railway through Ohio from West Virginia to Michigan is about ¾ of a mile to the east of the Reactor and the Norfolk and Southern is about 1.5 miles to the east. The eight-kilometer radius infringes on a major rail yard to the west in Norwich Township. These carriers' estimates indicate that the average number of Hazmat carloads exceeds no more than 2% of the rail traffic traversing the county in a one-year period. The bulk of the Hazmat carried consists of Class 8 (Sodium Hydroxide) with varying amounts of petroleum distillates.

Discussions with and information gathered by the Ohio Emergency Management Agency and The Ohio State University Office of Environmental Health and Safety indicate that these facilities should have no affect on the safety of the reactor.

5. *What is the location of nearest major airport? How close are the flight paths to the reactor site? Please assess the affect on the safety of the reactor facility if any. (Reference: NUREG 1537, Part I, Section 2.2)*

   The OSURR is located about 12 kilometers west of Port Columbus International Airport (CMH). It is about 0.4 kilometers south of the west flight path for both arrivals and departures. Arrival altitude at 12 kilometers is about 2700 feet and departure altitude is about 3500 feet. The probability that any aircraft could affect the reactor's safety is extremely small. This is based on statistics provided by www.airline-safety-records.com, which shows the ten major U.S. airlines averaged only 2.77 accidents per one million takeoffs for the five-year period 1996-2001.

6. *Discuss postulated accidents at identified facilities in the area up to 8 km (4.8 miles) radius from the reactor site that may affect the safety of the reactor facility. (Reference: NUREG 1537, Part I, Section 2.2)*

   There are accidents that could affect the daily operations of the facility but it is not likely they would affect safety of the reactor. The only postulated accidents that could reach the reactor building would have to involve an airborne exposure plume. While there is the remote possibility these could cause evacuation of the reactor building there would be no physical effect on the building itself. Before evacuation the reactor would be shutdown and the facility secured. We have had communications with the Ohio Emergency Management Agency and The Ohio State University Office of Environmental Health and Safety. From these communications we have not postulated any accidents that should lead to adverse affects on reactor safety from events within an eight-kilometer radius.

7. *Are the 65,000 students, staff and faculty included in the population figures in the table of Section 2.2.1 of the SAR?*

   The 65,000 population for The Ohio State University represents a daytime resident population only. This populace consists of faculty, staff and students many of whom commute on a daily basis to the facility. It is not possible to determine how many of them remain within an eight-kilometer distance after they leave the university. There are about 8,500 individuals on campus during the evening and night hours. This figure represents on-campus dormitory residents, night maintenance and security staff and residents and patients of the OSU Hospital complex. The reactor building is about two kilometers west of the main campus population.

8. *What is the site history for severe weather such as tornados and high winds? What is the frequency and severity of such weather? Is there any possibility of surface flooding that would affect the OSURR site?*

   The site history for severe weather is as follows.

- There were 112 instances of high winds ($\geq$ 50 knots) in Franklin County from 1/1/1950 to 4/31/2002.[1]
- There were 22 tornado events in Franklin County from 1/1/1950 to 4/31/2002.[1]
- There have never been any instances of damaging high winds or tornadoes on the OSURR site since its construction.

The frequency of severe weather for the Central Ohio Area is as follows.

- There are approximately 8 high wind days per year.[2]
- There are approximately 0.8 tornado days per year.[2]

The OSURR site is above the 500 year flood level for both the Scioto and Olentangy Rivers; therefore, the possibility of surface flooding on the OSURR site is minimal.[3]

There has never been any instance of surface flooding on the OSURR site since its construction.

1. National Climatic Data Center, Franklin County weather data from 1950 to 2002
2. National Severe Storm Laboratories, Frequency data from 1980 to 1999
3. Federal Emergency Management Agency, Insurance flood maps 39049C0235 and 39049C0231

9. *Discuss the possibility of sky shine from the reactor with a loss of coolant. (Reference: SAR Section 2.3.3)*

   Since gamma-radiation doses resulting from an exposed reactor core are not currently estimated in the SAR, calculation methods and results can be added to Section 8.5.1 to estimate the direct and indirect (sky shine – radiation reflected from the ceiling) dose rates resulting from an exposed reactor core. Refer to Attachment A of this document for the text and data to be added to the SAR.

10. *Discharge from the tertiary loop is to a floor drain on the main floor of the reactor building. What is the pathway of water after entering the reactor building floor drains? Are the drains monitored for radiological releases? Discuss the need for such monitoring. (Reference: SAR Sections 3.2.2.2 and 6.2.4.2)*

    First, we should note that quarterly monitoring of the secondary coolant over approximately the last 10 years has never indicated the presence of liquid-borne radionuclides. Release of radionuclides via the tertiary loop is considered an unlikely pathway because it would require leakage from the primary to the secondary loop, and concurrent leakage from the secondary to the tertiary loop. Since we have never encountered evidence of leakage from the primary to the secondary loop, we consider monitoring of the tertiary loop unnecessary at this time. We also consider monitoring of discharge drains unnecessary for the same reasons.

    We should also note that neither the primary-secondary or secondary-tertiary coolant loops are in continuous operation for every operation of the OSURR. First, the OSURR is not operated on a fixed duty cycle. Second, the primary-secondary coolant loop is not operated for every OSURR run, but only those wherein operating power in excess of 120 KW is intended. Finally, even for those runs involving higher-power operations requiring the operation of the primary-secondary coolant loop, the tertiary loop is operated at significant flow rates only when added heat removal capacity is necessary. These conditions typically arise on warm days with relatively high ambient air temperatures. High-power reactor operations on those days are occasional at best.

If significant concentrations of radionuclides were detected in the secondary coolant, it is likely that the facility staff would develop and implement a plan and procedures for monitoring the tertiary loop. This would likely involve sampling as is currently done for the primary and secondary coolant, wherein samples are collected and assayed for radionuclide content and concentration.

Radionuclide content in the primary coolant (pool water) is most often undetectable. The only instances where measurable quantities are detected are those cases where samples are taken immediately after operation of the reactor for an extended period of time at high power, and prior to operation of the pool water cleanup system, which removes the radionuclides typically encountered with pool reactors using aluminum plate fuel and aluminum core structures, which are ^{24}Na and ^{41}Ar. Thus, we might have a source term in the range of nanocuries per liter.

Should this water be circulated through the primary-secondary coolant loop, and if leakage were to occur, one would have to have knowledge of the leak rate to determine the actual concentration in the secondary coolant. When in operation, the primary-secondary coolant loop does not involve significant generation of fluid pressures, so we are unlikely to challenge the integrity of the heat exchanger seals. The plate-and-frame design of the heat exchangers make the leakage path from primary to secondary loop unlikely. A more likely pathway is from the coolant loop to the floor of the building. If leakage were observed on the floor, the OSURR staff would undertake activities to, first, stop the leak, and, second, to contain any leaked material, and, third, assay fluid leaked from the system for radionuclide content, and, if necessary, collect any radionuclide-bearing liquid and decontaminate any surfaces or areas involved in the spill.

Similar arguments apply to the secondary-tertiary loop. The tertiary loop flow rate is variable from zero through about 100 gallons per minute. It is likely that any leakage from the secondary to the tertiary loop would experience a very large dilution prior to discharge to the floor drains.

Any discharge reaching the floor drains would be directed into the overall sewage drainage network for the University area. Eventually, the outfall of drainage from the University area is to the regional sewage discharge, which is processed by the city sewage processing plant prior to discharge to natural drainage systems, which are the area's rivers and streams. The Radiation Safety Office performs monthly monitoring of the discharge volumes in connection with regulatory activities related to the Broadscope License. In the past twelve month period from July 2001 through June 2002, the lowest monthly sewer volume occurred in June 2002 and the volume of water was 1.93×10^8 liters. Thus, for a source term measured in the nanocuries per liter range, we are probably achieving dilution of that concentration by at least a factor of 10^8, so it is unlikely that any discharge from the

OSURR facility floor drains, unlikely as they are to begin with, would result in a significant environmental impact.

11. *Are there any private or otherwise owned wells in the vicinity of OSURR site that may be affected by reactor operations? If so, discuss the need to monitor for radioactivity.*

    There are no privately owned wells or otherwise owned wells in the vicinity of the OSURR. The closest well is about one kilometer to the northeast. It is not affected by reactor operations and is not monitored for radioactivity.

12. *Are there any systems or practices used to minimize the activation of the pool wall? What affect does radiation have on the pool wall and liner integrity? What surveillance is done to assess damage of the pool wall and liner? Discuss aging effects of the fiberglass-reinforced epoxy paint in the reactor pool and the bulk shielding pool. (Reference: SAR Section 3.1.1.1)*

    There are no systems or practices used to minimize activation of the pool walls other than existing structural materials. As described in the SAR Section 3.1.3.1 the reactor core is surrounded on the south and west sides by thermal column extensions and 0.125-inch thick stainless steel plates. These walls have shown no signs of deterioration. The east wall is eight feet and nine inches from the core centerline and it too has shown no signs of deterioration. The north wall is one foot and 10.25 inches from the core centerline and there has periodically been evidence of deterioration of the epoxy-based liner near the beam port extensions. Our records indicate the reactor pool liner has been repaired or repainted several times including 1968, 1972, 1984, 1991, 1995, and 2000. The BSF pool was last repainted was in 1984. Each pool is visually inspected every time the reactor is operated in a pre-start checkout procedure. The volume of make up water is also checked weekly and compared to previous amounts to help evaluate water loss which could be indicative of a damaged pool liner.

    In June of 1991 the fuel was placed in the storage pit and the reactor pool was drained prior to installation of the cooling system necessary for our power increase from 10KW to 500KW. At that time we repaired the damaged epoxy around the beam port extensions on the north wall and recoated the remaining pool walls. We repeated this process in September of 1995 and September of 2000. It appears we are on about a five-year cycle for repair and repainting of the reactor pool liner and perhaps a twenty-year cycle for the BSF pool.

13. *This quote is taken from SAR Section 3.1.1.2: "Blister tests conducted in the RERTR program on high burnup fuel have provided data which can be used to show that no fission product releases will occur from the fuel in the OSURR core even under maximum credible accident conditions that do not involve direct mechanical damage to the fuel plates." Please provide the reference for this conclusion?*

    Testing on fuels of the type used in the OSURR was conducted as part of the RERTR fuels development program. The reference document is ANL/RERTR/TM-10, "Performance of Low-Enriched U3Si2-Aluminum Dispersion Fuel Elements in the Oak Ridge Research Reactor", G. L. Copeland, R. W. Hobbs, G. L. Hofman, and J. L. Snelgrove, October, 1987, Argonne National Laboratory, pp. 17-18.

These tests showed that fuel blistering generally occurred in the range of 530° C. to 550° C. for the uranium silicide dispersion fuel. These tests were conducted by heating the fuel plates in a furnace, which results in less convective heat transfer than for fuel plates immersed in water. Chapter 8 of the OSURR SAR notes that for the maximum credible accident reactivity insertion and power burst without scram, the maximum fuel plate surface temperature in the hot channel will be about 146° C. For loss of pool water accidents, Chapter 8 of the OSURR SAR provides an analysis of possible fuel plate temperatures following pool draining. For non-instantaneous draining of the pool, we expect a fuel plate surface temperature of about 259° C. Both of these estimates are less than the blister temperature threshold measured by the RERTR studies.

14. *Discuss the Wigner effect with the graphite in column, the graphite filled central irradiation facility (CIF), the graphite isotope irradiation elements (GIIE), beam plugs, and the solid graphiteflectors. Also discuss the possibility of gas buildup in the thermal columns, CIF, GiIEs, beam plugs, and the solid graphite reflectors. (Reference: SAR Sections 3.1.1.3, 3.1.2.1, 3.1.3.1, 3.8.1, 3.8.4, and 3.8.6)*

The Wigner effect is a result of the interaction of high-energy neutrons with the nuclei of atoms forming a graphite lattice. Stored energy and changes in some material properties result. Uncontrolled adiabatic release of stored energy can result in sudden temperature rise, and material effects such as dimensional changes can affect enclosures or surrounding materials.

The OSURR system includes some graphite components. We should note that the GIIE assemblies and the graphite-filled CIF are not currently used and there are no plans to use them. The graphite-filled CIF has not been used in the OSURR since about the late 1970s, and the GIIEs were removed when the LEU core was loaded in the late 1980s and have not been reloaded, although nothing in our operating procedures precludes their use in the future. Those graphite-bearing components routinely exposed to fast neutron exposures are the thermal column extensions along the west and south faces of the OSURR core, and the beam port plugs normally installed in the two ports positioned along the north core face.

Estimates of the Wigner effect in the graphite components begin with specification of the fast neutron flux at those locations. The thermal column extensions and beam port plugs are all at ex-core locations. Based on measurements of the OSURR neutron energy spectrum in an ex-core experimental facility (pneumatic "rabbit" tube) the neutron flux above 1 MeV at ex-core locations near the fuel boundary is about 2×10^{11} nv at full power.

Next, we estimate the fast neutron fluence necessary to accumulate significant stored energy in a graphite lattice at or near room temperature. We should note that at the position of the thermal column extensions and beam port plugs the temperature during full-power operation will likely be elevated, but likely less than the boiling point of water, since we observe no generation of steam voids at those locations during operation. For estimation of stored energy, we reference measured data reported in IAEA-TECDOC-1154, "Irradiation Damage in Graphite due to Fast Neutrons in Fission and Fusion Systems". Figure 4.1 on page 86 of this document shows curves for accumulation of total stored energy in graphite at various irradiation temperatures. Using the lowest irradiation temperature of all those shown (72° C.) we estimate that stored energy in the range of 1000 J/gram is attained with a fast neutron fluence of about 1×10^{20} nvt. The use of the 1000 J/gram limit for stored energy is to assume that the Wigner energy release results in heating of the graphite to a temperature close to its thermal oxidation temperature, which is the minimum temperature for which deleterious effects occur. Consideration of higher stored energy is possible, but it would take longer to reach these conditions.

Using the estimate of ex-core fast flux noted above, we see that the time to accumulate this much fluence represents about 16 years of continuous operation of the OSURR at full power, or about 138,889 hours of effective full-power operation. The average number of effective full-power hours of operation of the OSURR in recent years has been about 130 per year. Thus, one would estimate, based on recent power history, that the OSURR could operate for about another 1,070 years before fast neutron fluences are accumulated in the graphite components that would result in significant stored energy.

Initiating the release of Wigner energy generally requires elevated temperatures. Figure 4.2 on page 88 of the reference document notes that peak energy releases occur at about 200° C., which would be unlikely to be attained during normal operation of the OSURR. These temperatures would also be unlikely to be reached even under accident conditions as described in Chapter 8 of the OSURR, since heat transfer from the graphite components to the surrounding materials would limit internal temperatures. The thermal column extensions are surrounded by the pool water and also mechanically attached to the walls of the pool, which provides a large capacity heat sink. The beam port plugs are normally located within the boundary of the pool wall and in thermal contact with the surrounding materials, which also provides a high-capacity heat sink. Even if some of the stored energy were released, it is likely that these same heat sinks would provide sufficient conductive and convective heat transfer to limit the rise of the internal temperature of the graphite components.

These considerations lead us to conclude that Wigner energy is a not a significant concern for the graphite components of the OSURR.

We are aware of buildup of gases in the graphite reflectors of TRIGA reactors. It is our understanding based on this effect as observed at the University of Texas reactor that the gases resulted from radiolytic dissociation of water within the graphite reflector housing. We should first note that throughout the operational history of the OSURR no water intrusion has been found or suspected in the graphite thermal column extensions, which would be the most likely places where this would occur. When these have been inspected (which has been done on occasion when the pool has been drained) we have found no evidence of cracking of the housing or leakage of water from the enclosures.

If moisture were to intrude into the enclosures, it would be possible for effects similar to those observed in the Texas reactor to occur. The water molecules would break down into their constituent atoms. Over time, the oxygen atoms might form into O_2 molecules. Thus, a mixture of H_2 and O_2 gases might accumulate in the aluminum enclosures. Some have noted that this represents an explosive mixture. However, we note that there is no ignition source within or near the thermal column extensions. These are sealed enclosures covered by about 15 feet of water. If the gases were released, they would dilute into the relatively large air volume of the reactor building and their concentrations would quickly drop below that necessary to form an explosive mixture.

Gas pressures would build up within the aluminum enclosures of the graphite thermal column extensions if gases were evolved within them. If internal pressures are high enough, forced leakage could occur. However, this is unlikely to result in an explosive release of energy. More likely would be slow formation of cracks around or through areas of structural weakness of the aluminum enclosures. Once sufficient crack size was attained, the internal pressures would be relieved through release of the gases. The formation of gas bubbles at or near the location of the thermal column extensions would not damage or otherwise adversely affect the core structures or fuel integrity.

15. *If coolant temperatures can be as high as 145 °F what is the expected maximum cladding surface temperature with heat transfer? In SAR Sections 4.8.1 and 4.8.2 the average fuel plate and channel is used to calculate the coolant outlet temperature and cladding surface temperature. What is the expected maximum coolant outlet temperature and cladding surface temperature for the hottest channel? Verify these valves are acceptable through reference or analyses. (Reference: SAR Sections 4.8.1 and 4.8.2)*

The following (or similar) text will be added to the end of Section 4.8.1:

A conservative estimate can be determined for the maximum coolant outlet temperature and cladding surface temperature using analysis similar to that done above but with per-plate power distribution values determined by modeling performed in the design of the reactor (Seshadri, 1988). Five potential core configurations were modeled, and all of these configurations use seventeen fuel element and four control elements as the current core configuration does (one of the modeled core configurations is nearly identical to the current core configuration). Each of the core configurations was analyzed with control rods out and with various combinations of control rods completely inserted into the core for a total of seven control-rod conditions, yielding a total of 35 models. With any one of the shim-safety control rods completely inserted into the core, the reactor will be subcritical, meaning that many of the models assume a non-realistic condition for operating. However, using the condition of completely inserted control rod(s) can give a very conservative estimate for maximum per-plate power generation with great flux tilt in the core. The flatter flux profiles resulting from the models with all control rods out of the core are much closer to actual power generation conditions because the control rods are generally kept at similar heights during reactor operation.

As part of the design analysis, the element-averaged percentage per-plate power distributions were estimated for each model. The highest hot-channel per-plate power estimate of 0.4764% resulted from a model with shim safeties 1 and 2 completely inserted into one of the core configurations. For operation at 500 kW, this corresponds to 2.38 kW of power generation in each of the fueled plates in the hot-channel element. A flow channel between two of the fueled plates in this element will therefore remove 2.38 kW at steady state, assuming that half of the generated heat in each fueled plate is discharged into the coolant channel on either side. This can be used with the maximum allowable inlet temperature (reactor scram on 95 °F inlet coolant) to estimate the maximum hot channel outlet temperature.

$$Q = mC_P\,(T_2 - T_1)$$

where Q = channel heat generation (2.38 kW = 8127 BTU/hr)
m = mass flow rate (101.3 pounds mass/hr)
C_P = 1 BTU/lbm/°F
T_1 = Coolant Temperature at Core Inlet (95 °F)
T_2 = Coolant Temperature at Core Outlet

The value used above for the mass flow rate through the core assumes the core-average flow velocity used previously for the average coolant outlet temperature estimate. Because greater heat generation in the hot channel will cause a greater buoyancy force and therefore greater flow rate than the average, use of the average flow velocity makes this calculation more conservative. Solving the above equation for T_2 yields an estimate of 175.2 °F for outlet temperature in the hot channel, which is well below the boiling point of light water.

The following (or similar) text will be added to the end of 4.8.2:

Following a procedure similar to that used above but with an outlet temperature of 175.2 °F, the maximum cladding surface temperature can be estimated for the hottest channel. To be conservative in this calculation, the maximum coolant temperature and maximum fuel power density for the channel will be assumed. This is not realistic in that the maximum coolant temperature should occur at the outlet and the channel maximum fuel power density should occur near the axial center. The average channel fuel power density is estimated assuming that 2.38 kW is generated in the 19.43 cm^3 volume of fuel in a fueled plate.

$$q'''_{ave} = P / V = \frac{2.38\ \text{kW}}{19.43\ \text{cm}^3} = 122.5\,\frac{\text{W}}{\text{cm}^3}$$

where q''' = power density in watts/cm^3
P = power generated in hot-channel plate (2.38 kW)
V = fuel volume in the hot-channel plate (19.43 cm^3)

As seen in Section 4.9.1, the maximum channel fuel power density is 1.292 times the average channel fuel power density, yielding a maximum channel fuel power density of

$$q'''_{max} = 122.5 \frac{W}{cm^3} \cdot 1.292 = 158.3 \frac{W}{cm^3}$$

This power-density estimate can then be used to estimate cladding surface temperature.

$$T_s - T_{fl} = (r_f^2 q''')/(2h_S [r_f + t_C])$$

where q''' = power density (158.3 W/cm^3)
t_c = cladding thickness (0.15" = 0.0381 cm)
r_f = fuel radius (0.01" = 0.0254 cm)
h_s = convective constant of coolant fluid (0.05 W/cm^2/K = 0.028 W/cm^2/°F)
T_s = surface temperature of the fuel plate
T_{fl} = coolant fluid temperature (175.2 °F)

As discussed above, the value chosen for the convective coefficient represents the minimum of the possible range, resulting in further conservatism for the cladding temperature estimate. Solving for T_s yields a maximum hot-channel cladding-surface temperature of 203.9 °F. Even given the conservative assumptions of maximum allowable inlet coolant temperature, average core flow rate in the hot channel, minimum possible convective coefficient, maximum coolant temperature with maximum fuel power density, and (more than) maximum flux tilt, the estimated maximum cladding temperature is still below the point of causing onset to nucleate boiling. If the hot-channel per-plate power generation of the flatter flux profile resulting from the model with all control rods removed is assumed (discussed in Section 4.8.1), the outlet coolant temperature drops to 169.4 °F, and the maximum cladding temperature estimate drops to 198.1 °F.

16. *What is the estimate of the temperature rise across the hot channel? (Reference: SAR Section 4.9.2)*

Delete the last sentence in Section 4.9.2 and add the following text:

The same approach can used to estimate the temperature rise across the hot channel. We can start with the estimate for temperature rise of 80.2 °F calculated in Section 4.8.1. This starting estimate should be too high because it has assumed the average core flow rate through the hot channel. Since the following method will be balancing pressure drop through the hot channel with buoyancy to estimate temperature rise, a better estimate should result.

Assuming the maximum allowed inlet temperature of 95 °F, the average temperature is

$$T_{ave} = 95 \text{ °F} + (80.2 \text{ °F})/2 = 135.1 \text{ °F}.$$

Using the value of 2.38 kW (8127 BTU/hr) power generation in a hot-channel plate from

Section 4.8.1, the mass flow rate is

$$\dot{m} = (8127\ \text{BTU/hr/channel})/(80.2\ °\text{F})/(1\ \text{BTU/lb}_m/°\text{F}) = 101.3\ \text{lb}_m/\text{hr}.$$

At 135.1 °F, the density of water is 61.461 lb_m/ft^3, yielding a volumetric flow rate of

$$Q = (101.3\ \text{lb}_m/\text{hr}) / (61.461\ \text{ft}^3/\text{hr}) = 1.648\ \text{ft}^3/\text{hr}$$

and a velocity

$$v = Q/A = (1.648\ \text{ft}^3/\text{hr}) / (0.00211\ \text{ft}^2) = 781.4\ \text{ft/hr} = 0.2170\ \text{ft/s}.$$

The dynamic viscosity of water at 135.1 °F is $1.012\text{x}10^{-5}$ $\text{lb}_f\text{-s/ft}^2$, giving a pressure drop

$$\Delta P = \frac{3 \cdot \mu \cdot v \cdot L}{B^2} = \frac{(3)\left(1.012\text{x}10^{-5}\ \frac{\text{lb}_f \cdot \text{s}}{\text{ft}^2}\right)\left(0.217\ \frac{\text{ft}}{\text{s}}\right)(2\ \text{ft})}{(0.00483\ \text{ft})^2} = 0.565\ \frac{\text{lb}_f}{\text{ft}^2}$$

The volumetric buoyancy forces resulting from inlet and average water densities are

$\gamma_{\text{inlet}} = 62.023\ \text{lb}_f/\text{ft}^3$ (95 °F core inlet), and
$\gamma_{\text{ave}} = 61.461\ \text{lb}_f/\text{ft}^3$ (135.1 °F hot channel average)

yielding a buoyancy

$$\Delta P_b = 2\ \text{ft} * (62.023\ \text{lb}_f/\text{ft}^3 - 61.461\ \text{lb}_f/\text{ft}^3) = 1.124\ \text{lb}_f/\text{ft}^2.$$

The estimates for pressure drop and buoyancy resulting from a hot-channel temperature rise of 80.2 °F are nearly a factor of two from each other. Assuming a 70 °F temperature rise yields the following:

$T_{\text{ave}} = 95\ °\text{F} + (70\ °\text{F})/2 = 130\ °\text{F}.$
$\dot{m} = (8127\ \text{BTU/hr/channel})/(70\ °\text{F})/(1\ \text{BTU/lb}_m/°\text{F}) = 116.1\ \text{lb}_m/\text{hr}$
$\rho_{\text{ave}} = 61.544\ \text{lb}_m/\text{ft}^3$
$Q = (116.1\ \text{lb}_m/\text{hr}) / (61.544\ \text{ft}^3/\text{hr}) = 1.886\ \text{ft}^3/\text{hr}$
$v = (1.886\ \text{ft}^3/\text{hr}) / (0.00211\ \text{ft}^2) = 894.1\ \text{ft/hr} = 0.2483\ \text{ft/s}$
$\mu_{\text{ave}} = 1.059\text{x}10^{-5}\ \text{lb}_f\text{-s/ft}^2$
$\Delta P = 0.676\ \text{lb}_f/\text{ft}^2$
$\Delta P_b = 2\ \text{ft} * (62.023\ \text{lb}_f/\text{ft}^3 - 61.544\ \text{lb}_f/\text{ft}^3) = 0.958\ \text{lb}_f/\text{ft}^2$

These new estimates for pressure drop and buoyancy are closer, and using linear extrapolation, a new estimate for temperature rise can be generated.

$\Delta T_1 = 80.2\ °\text{F}$ $\quad$ $(\Delta P_b - \Delta P)_1 = 1.124\ \text{lb}_f/\text{ft}^2 - 0.565\ \text{lb}_f/\text{ft}^2 = 0.559\ \text{lb}_f/\text{ft}^2$

$\Delta T_2 = 70.0\ °F$ $\quad (\Delta P_b - \Delta P)_2 = 0.958\ lb_f/ft^2 - 0.676\ lb_f/ft^2 = 0.282\ lb_f/ft^2$
$\Delta T_3 = ?$ $\quad (\Delta P_b - \Delta P)_3 = 0$

$(\Delta P_b - \Delta P) = 0.0272 * \Delta T - 1.619 \quad \Rightarrow \Delta T_3 = 59.5\ °F$

This new estimate for temperature rise results in the values

$\Delta P = 0.835\ lb_f/ft^2$
$\Delta P_b = 0.794\ lb_f/ft^2$

Performing an interpolation between the values calculated for 70 °F and 59.5 °F and repeating the process yields the final estimate of 61 °F for the hot-channel temperature rise. This value is much lower than the 80.2 °F predicted in Section 4.8.1. However, this is to be expected since the value calculated in that section assumed the core-average flow rate through the hot channel, which is a conservative, rather than realistic, assumption.

17. *Discuss the difference in the estimated and the measured void coefficient. How will this difference affect the safety analyses of this reactor? (Reference: SAR Section 4.5)*

One must use care in examining the data presented in section 4.5 of the SAR. The predicted values for the various LEU core configurations analyzed are listed in units of percent reactivity per one percent void, whereas the measured value listed at the bottom of page 109 of the SAR is shown in units of absolute reactivity per one percent void. To make them consistent, one would either have to divide the predicted values by 100, or multiply the measured value by 100. Doing so consistently gives us:

Minimum Predicted Void Coefficient:	-1.8×10^{-3}/1% void
Maximum Predicted Void Coefficient:	-4.5×10^{-3}/1%void
Measured Void Coefficient:	-7.92×10^{-3}/1% void

Thus, the measured value for void coefficient is actually more negative than the predicted values; meaning negative reactivity feedback is stronger in the actual core geometry than those predicted. This implies a more conservative analysis would result if one based the shutdown mechanisms on a smaller value for void coefficient.

In any case, the most important result is that the void coefficient, both predicted and measured for the OSURR, is negative in sign. This implies a negative reactivity effect for void formation. The actual value of the void coefficient has little effect in the accident analysis for step reactivity insertions, since the primary shutdown mechanism over the short-term accident conditions is the negative temperature feedback from fuel heating. Void formation is a slower process and affects the transient behavior only after sufficient heat transfer from the fuel to the coolant-moderator has occurred.

18. *In SAR Section 3.2.2.1 the following statement is made: "The pump is protected from running dry in that if the water level drops low enough to expose the pump head, the reactor will have already tripped, blocking operation of the primary pump." Does the*

pump automatically trip if there is a reactor trip? Discuss how this protection is provided.

The OSURR cooling system controls do not now have a feature wherein pump operation is blocked in the event of water level dropping below the intake suction point. The statement referenced in the SAR may either refer to a design feature initially planned but not implemented, or it may be a general reference to pump inoperability in the event of a loss of suction, which results in no flow. If pool water level drops sufficiently, an independent level monitoring system initiates a reactor trip, but this is not currently interfaced to any primary pump controls.

We should note that run-dry of the primary pump does not compromise the function of the reactor safety system. While a run-dry event might damage the pump, the reactor safety system would initiate a trip via independent systems if such damage prevented operation of the pump or compromised its ability to maintain a minimum flow rate in the primary cooling loop. The safety system has a pump operate-nonoperate sensing circuit, and a flow monitor, which generates a reactor trip if primary flow falls below the minimum, required by the trip setpoint. These provide indications of primary pump damage in the event of a run-dry situation arising.

19. *Are there I&C comparisons that are current? Please point out any particular similarities and/or differences with the comparable systems. (Reference: SAR Section 3.3.1)*

The OSURR was supplied by Lockheed Nuclear Products and is of a design similar to the Purdue University Reactor and a research facility in Colombia, South America. The OSURR I&C system has been upgraded and modernized at various times in its history, using both in-house and vendor-supplied instrumentation.

The comments in section 3.3.1 were intended to be of a more general nature concerning common features of research reactor I&C and safety systems. These include, among others, ionization chamber-based power monitoring channels, redundant systems and readouts, and fail-safe shutdown systems using gravity as the insertion process for safety control rods. In that sense, the OSURR shares many general features of research reactor control systems used in a wide range of applications and covering a large operator and user base, which over the years has compiled a remarkable record of safety and reliability.

20. *Discuss the aging effects of the control system. If vacuum tube technology is used, discuss the challenges to maintaining such a system that may affect the ability to operate safely for the duration of the renewed license? (Reference: SAR Section 3.3.7)*

Normal aging effects of the control system would be expected to result in failures of individual components and/or subsystems. These effects might include burnout of discrete components, changes in the value of circuit elements, or degradation of insulation or cables.

The OSURR control system is designed to be fail-safe and resistant to common mode failure. Control rod holding magnet current is normally off and will only flow to the electromagnets if the safety system allows it, based on the condition of systems required for operation.

If a required system is non-functional, the safety system precludes operation by blocking magnet current. In the event of such a failure, the OSURR staff would take steps to effect repairs and return the system to an operable condition.

Currently, no component of the OSURR I&C and safety system uses vacuum tubes. The last remaining vacuum tube instrumentation, which was the original Leeds and Northrop stripchart recorder system, was replaced with IC-based chart recorders in the early 1990s.

The context of SAR Section 3.3.7 concerns the electrical power supply to the console and I&C systems. When the original version of the SAR was written in 1987 there were still some vacuum tube-based components, and the console power supply was designed to provide the appropriate voltages. That capability remains, but is no longer utilized.

21. *In SAR Section 3.3.16 the following statement is made. "A 'Fast Scram' occurs when reactor power reaches 150% of full power." This would be 750 kW. Discuss why this is different from the TS 2.2.(2).*

    The statement in the SAR Section 3.3.16 is incorrect. Technical Specification 2.2(2) is correct. A reactor safety system setting does initiate protective action so that reactor thermal power does not exceed the reactor scram point of 120% of full power during a transient. The statement that a fast scram occurs when reactor power reaches 150% of full power is a carryover from our SAR and Technical Specifications prior to 1988 when we had HEU fuel and were licensed to operate at 10KW. The incorrect statement occurred in our 1987 submittal supplemented in May 1989 and February and June of 1990 that allowed our power increase to 500KW.

22. *In SAR Section 7.3.7 the following statement is made. "Administrative controls are established to allow NRL personnel discretion in approving and carrying out experiments and operations in a safe manner." How are the requirements of 10 CFR 50.59 met?*

    The administrative controls that are used to meet the requirements of 10CFR50.59 are found in two procedures, AP-04 Approval of Requests for Reactor Operation and AP-14 OSURR Modification Requests. These two procedures assure that proposed changes, tests, and experiments are reviewed prior to implementation so there are no unreviewed safety questions or Technical Specification changes. Each procedure requires the prior review and approval by a Senior Reactor Operator as to whether the proposed change, test, or experiment constitutes an unreviewed safety question or change to the Technical Specifications. If in the judgment of the SRO there is no unreviewed safety question then the change, test, or experiment may proceed. The SRO may not review and approve his or her own submittal. The Reactor Operations Committee (ROC) normally at their next meeting reviews results of all SRO decisions. Activities that the SRO deems may constitute an unreviewed safety question or require a change to the Technical Specifications are forwarded to the Reactor Operations Committee for review and approval. If the ROC believes the change, test, or experiment constitutes an unreviewed safety question or requires a Technical Specification change the results are forwarded to the NRC for final review and approval prior to implementation.

23. *Are there facilities available to store the radioactive effluent if necessary? (Reference: SAR Section 6.2.3)*

    Currently there is a large plastic tank housed in the Reactor Building with the capacity to store the entire amount of secondary coolant. For primary coolant the fluid would simply be held in the reactor pool for decay if needed. If for some reason this were not an option, alternative storage containers could be purchased. For example, tubs as large as 1500 gallons are readily available from commercial suppliers. Four of these would hold the entire contents of the reactor pool primary coolant.

24. *Please discuss the difference between confinement and containment. Does not the use of a facility designed for confinement as containment cause uncontrolled releases at ground level? Please quantify the dose to the unrestricted areas. Compare ground release (release from a leaky building with the ventilation secured) with the controlled release from the 30 ft high release point when the ventilation is operable. Please justify such operational actions as advocated in Section 6.3.6.4 with calculations or operational data. What is the decision tree used by the staff to decide the action to take? (See question 33 also)*

    See the answer to question 33.

25. *Does the Director of the Radiation Safety Section (RSS) have authority separate and similar to the Manager of Operations such that "he or she may terminate any operation at any time if it is deemed to be or likely to adversely affect the health or safety of the public?"*

    The Director of the Radiation Safety Section (University Radiation Safety Officer) has the authority similar to the OSURR Manager of Reactor Operations to terminate any operation at any time if it is deemed to be or likely to adversely affect the health and safety of the public. This authority is found in Figure 6.1 Administrative Controls of the Technical Specifications. It indicates a path of direct responsibility from the Director of the Radiation Safety Section to the Director of the Nuclear Reactor Laboratory. This authority is independent from the Manager of Reactor Operations because of the separate reporting lines of each.

26. *The SAR states that fuel element handling tools are kept stored and locked during reactor operation, and OSURR operational procedures require that the reactor be shut down with the control rods fully inserted into the core during all fuel element movements. Please justify why this should not be a limiting condition of operation (LCO) in the TS? Please justify why this TS should not also establish the configuration of the confinement system (building and ventilation configuration) during fuel movement.*

During fuel element movements the reactor is "shutdown" according to the Technical Specification definition of "Reactor Shutdown" because the control rods are fully inserted. Limiting Conditions of Operation (LCO) are established to assure safe operation of the facility. Since during fuel movement the reactor is shutdown and thus not operating no LCOs should apply. The procedure for fuel element inspections requires a pre-start checkout of the reactor. This checkout assures that the rods are fully inserted and the building exhaust fan is operating prior to fuel movement. This checkout procedure is also completed each day the reactor is operated and assures the fuel handing tools are secured during reactor operations.

When our Technical Specifications were revised for our conversion to LEU and our subsequent power increase we did not consider these items for inclusion as Limiting Conditions of Operation. We made this decision after review of several other research reactor technical specifications including those from Lowell and Missouri-Rolla. Neither of these included LCOs for fuel movement. In our application for license renewal dated December 15, 1999 we simply did not consider adding them as LCOs since they had not been included in the Technical Specifications that had been approved earlier.

27. *Please discuss a reactivity accident with the initial conditions of low-power.*

In this case, the power transient would begin without the temperature feedback effects initially present. This will affect the initial rate of power rise until a point is reached wherein fuel temperature begins to insert negative reactivity. Eventually, conditions would be similar to those existent for the case analyzed assuming a reactivity insertion at high power, including the dominance of the fuel temperature feedback to the point of terminating the power rise and actually reducing it. Thus, the overall results, including final power level reached and fuel cladding temperatures would likely not be significantly different than those already analyzed in Chapter 8 of the SAR.

We find that this conclusion is born out by a similar analysis done for the University of Texas-Austin reactor, in their SAR of September 1984. The UT reactor analyzed a step reactivity insertion of $4 from an operating power of 1 KW and a similar transient from an operating power of 880 KW. Comparing each case (Figure 7-1 on page 7-3 of the UT SAR) we note that the power rise for the insertion at low power is faster than for the insertion at high power, but both power pulses terminate at about the same peak power, and maximum fuel cladding temperatures are approximately equal for both cases. While we acknowledge that the UT reactor is fueled with TRIGA fuel assemblies and the OSURR has plate-type LEU in the form of uranium silicide, the physical processes driving the behavior of the power transients are similar.

We should point out that the OSURR safety systems provide a measure of protection from low power reactivity insertion accidents. If a rapid power rise were initiated at low power, the first reactor trip setpoint to be exceeded would likely be the 120% overpower trip from the linear power level recorder. During startup the sensitivity of this channel is scaled so that indications of neutron levels become reliable in the range of 1 to 10 watts. A rapid power transient would drive the recorder offscale at its high range very quickly, causing a shutdown. If this did not occur, there are two power rate trips initiated from the logarithmic power channel, which typically begins responding to neutron flux from the core in a reliable manner at about 50 watts power. The period trip setpoints are a five second indication on the period stripchart recorder, or a one second period trip from an electronic, or "fast" scram monitoring channel. We should also note that a shutdown trip can be generated by a count rate indication on the startup channel stripchart recorder if it drops below an indicated reading of 2 cps. If a rapid power transient were initiated at low power, the sudden increase in incident flux to the startup channel neutron sensor (fission chamber) would likely "saturate" the pulse counting electronics, dropping the signal below 2 cps and initiating a scram. Finally, the two redundant overpower trip channels will cause a reactor trip at a setpoint of 120% of absolute full power. All of these engineered safeguards provide reasonable assurance that the effects of a reactivity insertion accident at low power can be mitigated.

Finally, we should point out that operator intervention is always available to initiate a shutdown. During the training of licensees we note the importance of observing the behavior of the startup channel count rate during initial startup of the reactor, and of what actions to take in the event of unexpected indications.

Upon recommendation of the OSURR Reactor Operations Committee on 8/21/02, the OSURR staff agreed to conduct an analysis of a reactivity insertion accident at low power using a simplified model of the OSURR core. This will likely use a point reactor kinetics model with reactivity feedback. We will report the results of this analysis to the ROC for their review and evaluation, and request a recommendation as to its inclusion in a future revision of the SAR.

28. *TS 3.2.1 specifies that the control rod drop time is 600 msec. The control rod drop time used for the analysis of design basis accident appears to be 530 msec (Table 8.3). Please discuss how the 70 msec difference will affect the results of the analysis.*

Measured rod drop times have consistently been about 500 ms, and the 530 ms rod drop time value used for analysis represents an upper bound on rod drop times likely to be encountered in actual operations.

In any case, the accident scenarios discussed in Section 8.4.3 bound the scenario proposed in this question. There are four cases analyzed, each representing different amounts of step reactivity insertion, void coefficient, and initial conditions of core inlet temperature and operating power. These four cases were analyzed under two scenarios: with and without scram. That is, assuming no scram occurs, the rod insertion time is infinite.

The question proposed above would thus fall in level of severity between any of the cases 1, 2, 3, and 4, wherein the rods fall with a 530 ms insertion time, and the scenario for these cases wherein the rod "falls" with an infinite insertion time.

The results can be compared by examining Figures 8.2 through 8.5. For an infinite rod fall time, the worst-case power peaking occurs at 49 megawatts at about 100 ms after the onset of the transient, with fuel cladding temperature approaching about 145° C. at about the same time. For the same case (Case 4) in the scenario wherein the rod falls with a 530 ms insertion time, the power peaks at a little less than 12 megawatts between 50 and 60 milliseconds after the start of the transient, and the fuel cladding temperature peaks at a little over 90° C. at about the same time. Thus, the scenario assuming a 600 ms insertion time would fall between these two cases, probably closer to the analysis wherein the rods fall with a 530 ms insertion time than that for the infinite insertion time.

29. *In TS 1.3 "Scram Time" is defined. Does control rod drop time have the same definition?*

"Scram time" has the same definition as "drop time", and along with "insertion time", these are used as interchangeable terms. "Insertion Time" is included in this list because it is used in OSURR procedures with the same meaning as the other two. To address the inconsistency of defining scram time but then setting a limit on rod drop time, all three of these terms will be given the same definition in TS 1.3. To be consistent with the text in TS 3.2.1 which sets the limit for "Rod Drop Time", the definition given for these three terms will be:

> The elapsed time from the receipt of a safety signal to when a shim/safety rod is fully inserted.

30. *In the bases for TS 3.2.1, the referenced SAR Section 4.3.3 does not exist. What is the correct reference?*

The correct reference for the bases of Technical Specification 3.2.1 is Safety Analysis Report Section 8.4.3.3.

31. *The basis for TS 3.2.2 states that the specification is based on a continuous reactivity insertion event. Is this referring to an analysis of a reactivity ramp accident? Please provide a description of that event and the analysis which leads to the maximum reactivity insertion rate specification of 0.02% Dk/k per second.*

The specification of a constant reactivity addition rate implies a ramp insertion, but we first note that such is a conservative assumption, since a linear insertion of reactivity assuming a constant control rod withdrawal rate is unlikely. The maximum reactivity insertion rate would only occur during a relatively short time wherein the highest-worth portions of the control rods were being withdrawn. Reactivity insertion rate would decrease once movement of the control rods positioned the rods in regions of lower reactivity worth.

The original specification of the 0.02% δk/k/second insertion rate dates back to the original Technical Specifications provided by the reactor supplier (Lockheed Nuclear Products) and the initial licensing actions for the OSURR in 1960. The addition rate assumes the simultaneous withdrawal of the most reactive shim safety rod concurrently with the regulating rod. Lockheed Nuclear Products specified the control rod drive speed and the likely worth of the control rods for various core geometries. These values were based on calculated and measured values for similar materials in other reactors, as well as computations based on computer codes available at the time. Page 1-6 of the original OSURR Hazards Summary Report (HSR) discusses some of these assumptions. Presumably, the engineering staff of Lockheed Nuclear Products performed the calculations and safety studies which led to the inclusion of this particular specification in the initial licensing documents. However, the present-day NRL staff has no access to records that document these calculations in detail, and have been unable to gain access to them because of the passage of time and the fact that Lockheed Corporation is no longer in the business of supplying research reactor systems.

Similar analyses were performed for the conversion of the OSURR to the LEU uranium silicide fuel, and were incorporated in the SAR submitted in 1987 that addressed these changes. These analyses included studies of the control rod worths for various LEU core geometries and the worths of various experimental facilities and fuel assemblies at various core grid positions. Based on estimates of control rods worth and the known control rod withdrawal rates, the specification of reactivity insertion rate noted in TS 3.2.2 was retained. The estimates of control rods worths for various LEU core geometries is summarized in the SAR section 4.4.3. It was found that for the known control rod withdrawal rates and the reactivity of the highest-worth shim safety rod and the regulating rod withdrawn simultaneously, the reactivity addition rate was less than that specified in the original technical specifications. Subsequent measurements have confirmed this, as noted at the end of section 4.4.3, thus assuring that the specified limit is reasonable and can be met.

32. *In TS 3.2.3 the terms slow scram and fast scram are used. Please include a definition of these two terms in Section 1.3 of the TSs.*

While we acknowledge that these terms are not explicitly defined in the TS section referenced, we note that they are defined and discussed in the SAR. However, to meet the request stated in the question, we will add the following to section 1.3 of the Technical Specifications when they are next revised:

> **Fast Scram** – the rapid insertion of the shim/safety rods into the reactor for the purpose of quickly shutting down the reactor, accomplished by turning off current to the electromagnets holding the control rods to their drive motors, wherein current is turned off by electronic means through the biasing of a current-controlling device.
>
> **Slow Scram** – the rapid insertion of the shim/safety rods into the reactor for the purpose of quickly shutting down the reactor, accomplished by turning off current to the electromagnets holding the control rods to their drive motors, wherein current is turned off by electromechanical means through the opening of relay contacts controlling the operation of the magnet current supply modules.

33. *In TSs 3.4 and 3.5 there appears to be a confusion between confinement and containment systems. Please review these TSs and clarify the wording such that it is clear which system*

you have and wish to maintain and use in your analyses and emergency response. This same confusion appears to exist in the SAR especially in the last two paragraphs of Section 8.4.4.5. If the ventilation is intended to be secured during an event the building leakage rate should be determined and used to calculate the unrestricted airborne and direct (from the building) dose with a ground release during a damaged fuel plate type accident. (See question 24 also)

There is no confusion between confinement and containment in the SAR and TSs. Under no set of conditions is the definition of containment met, nor is containment claimed. However, ambiguity exists in the SAR and TSs regarding use of the phrase "confinement isolation".

The definitions from the SAR are:

> **Confinement** - a closure on the overall facility which controls the movement of air into it and out of it through a controlled path.
>
> **Containment** - a testable enclosure which can support a defined pressure differential and which is normally closed.

The term confinement clearly pertains to the setup of the OSURR for normal operation. The reactor building has a single exhaust fan and closed doors and windows that provide a closure on the overall facility and control of the movement of air into it and out of it through a controlled path. Measurements have been made that indicate that all airflow other than at the exhaust point is flowing into the reactor bay while the exhaust fan is running.

However, in the event of a release in which the fan is shut off to limit release of radionuclides to the environment, the reactor building does not meet the definition of containment or confinement. It has neither a defined pressure differential nor a controlled path for the movement of air. Instead, it has doors and windows closed and exhaust fan shut off to minimize the quantity of radionuclides released to the outside. In TS 3.4, this action is referred to as "confinement isolation".

While this term has never been a source of ambiguity to the reactor lab staff, it may not be clear to outside parties. For the OSU Reactor Lab, "confinement isolation" meant isolating the interior of the building from the outside to the greatest extent possible, i.e. not running the exhaust fan. However, the term "confinement" is defined such that the fan must be running, making the phrase "confinement isolation" oxymoronic. To rectify this, we propose to change the phrase "confinement isolation" to "building isolation". The wording of applicable sections of the SAR and TSs (including TSs 3.4 and 3.5) will need to be changed to properly address the terms "confinement" and "building isolation" so that they are clear.

To address building leakage with the exhaust fan off, measurements were made at ground level of airflow between the reactor bay and offices and between the reactor bay and the outside with the exhaust fan turned off. In all cases, the measurements showed that at the ground level the flow of air was into the reactor bay. This indicates that natural convection causes cooler air to enter near the bottom of the building and warmer air to exit near the roofline, i.e. the reactor building acts like a chimney. An expert in heating, ventilation and cooling (HVAC) at the Office of Environmental Health and Safety at The Ohio State University confirmed that this would be the expected airflow behavior for a building such as ours.

Therefore, ground releases should not occur for the situation of "building isolation" after a release of radionuclides into the building. Instead, there would be a release similar to that with the exhaust fan running, but with the time constant of the natural exchange rate of air from the building instead of the exchange rate from running the exhaust fan.

Since no estimates for dose outside the restricted area of the reactor building are currently included in the accident analysis of Section 8.4 of the SAR, these will be added to Section 8.4.4.5. In the event of a fuel-plate rupture release into the reactor building, the recommended course of action is to shut the building exhaust fan off, but an estimate of doses outside the restricted area assuming the exhaust fan is left on provides an upper bound on the submersion dose that personnel outside the building could receive. Therefore, estimates for both situations will be added to the SAR. See Attachment B of this document for text and data that will be included in Section 8.4.4.5. This information will replace what is currently the last paragraph of the section.

Section 6.3.6.4 mentions that the building purging rate could be increased by the use of portable fans. This statement was included in the SAR to cover possible unforeseen events. Such an action has never been taken, nor is it likely that it ever would be. As a consequence, the staff does not have a formal decision tree for taking this action. If an event were to occur for which this capability was useful, it is most likely that the NRL staff would deem this situation a "Nuclear Emergency", which requires notification of the OSU Radiation Safety Section and the NRC, among others. Since such an event would fall outside the scope of Section 6.3.6, we propose to remove this reference to using extra exhaust fans from 6.3.6.4.

34. *In TS 3.4, is the analysis referenced in the "revised SAR of September 1987" also in the SAR submitted with the license renewal application?*

    This information is also in the new SAR. The statement is an editorial oversight of a carryover from the previous SAR. The text "revised SAR of September 1987" should simply read "SAR".

35. *Please discuss and justify the fact that the specification of TS 3.7.1 (4) is greater than TS 3.2.2.*

    As best we can determine at this time, these limits have been retained from earlier versions of the OSURR Technical Specifications, initially submitted in 1960, with a revision submitted and approved in 1987. Since no objections to these provisions were made at the time, we have retained them for the current submission.

We believe that the provisions of TS 3.7.1(4) must be viewed in the context of the overall TS 3.7.1, in particular with the provisions of TS 3.7.1(2), which specifies an upper limit for total reactivity worth of a movable experiment of 0.4% δk/k, and TS 3.7.1(3), which specifies that the total worth of all movable experiments be less than 0.6% δk/k. Thus, even if reactivity were added at a rate close to the limit of 0.05% δk/k/second, the total reactivity added would be less than that required to attain prompt criticality, and thus be bounded by the reactivity insertion accident analyses presented in Chapter 8 of the SAR.

TS 3.2.2 deals with reactivity insertion from the control rods, which have higher total reactivity worth than allowed for movable experiments. It is conceivable that under certain conditions movement of the control rods could add reactivity close to or exceeding that required to attain prompt criticality, and this condition could be reached in a relatively short time if higher rates of reactivity insertion were allowed. The slower rate of reactivity addition specified for the control rods provides an added measure of assurance that attainment of such a condition and its undesirable consequences could be prevented by either operator action or the initiation of a scram by the safety system.

36. *Please provide justification for not having a TS placing limits on fueled experiments.*

The original Technical Specifications for the OSURR approved in 1961 and in force through 1987 had a requirement for limitations on experiments containing Special Nuclear Material. This limit was 5 grams mass in solid form. When the OSURR operating license was amended in 1987, this specification was removed after review by the Reactor Operations Committee (ROC). The ROC was concerned that such a restriction, taken by itself, might limit the operations of the OSURR in undesirable ways. For example, at the time of the LEU core conversion, there was some discussion of obtaining an instrumented LEU fuel assembly for experiments related to the LEU core conversion. It was felt that this could be precluded if the Technical Specification requirement were retained. Also, in our review of other research reactor Technical Specifications, we found that limitations on fueled experiments were not always included. For example, at the time, the reactor at The University of Lowell (now The University of Massachusetts at Lowell) did not have such a requirement.

The OSURR staff believes that current procedures and requirements for experiment review and approval can adequately encompass considerations related to fueled experiments. For example, experiment review currently requires, among other things, consideration of reactivity effects (total experiment worth, effects on excess reactivity and shutdown margin), shadowing of control rods and/or instrumentation, corrosions and possible effects on pool water purity, evolution of gases or liquids during irradiation, physical integrity of the irradiated material, local heating, and the production of activation and/or fission products and the possible dose rates resulting from them.

Finally, if necessary, the ROC can review the proposed experiment and advise the OSURR staff as to whether or not the proposed experiment would constitute an unreviewed safety question, or otherwise offer suggestions or impose requirements as to the conduct of fueled experiments.

37. *TS 6.2.4(7) refers to the TS Section 6.6.4. This section is not part of the TSs. What is the proper reference?*

Technical Specification 6.2.4(7) should refer to Technical Specification Section 6.6.2 Special Reports.

38. *In TS 6.3.1 it is indicated that procedures are not meant to preclude the use of "independent judgement." Please discuss what is meant by "independent judgement" in association to approved and reviewed procedures and the limits that are placed on this "independent judgement."*

The language in question in TS 6.3.1 simply recognizes that it is not possible to anticipate all situations and circumstances that might arise in the course of the operation of the facility, now or in the future. That is, situations may arise that are not specifically addressed by the written procedures. In those cases, inaction resulting from the lack of a written procedure specific to the situation may be undesirable. The language in the TS recognizes that actions not specifically delineated in the procedures may be necessary in some instances.

Limitations placed on such actions will likely be based on individual initiative and related to the specific circumstances at the time. It is important to recognize that persons involved in such situations are likely to be trained individuals, likely holding RO or SRO licenses, and their actions will be guided by their knowledge, experience, and common sense. During the training of these individuals, they are instructed as to the need to observe legal requirements of appropriate regulatory authorities, and the overall concern for the safety of individuals and the protection of property.

39. *Please provide justification for not including electrical interaction with reactor instrumentation as a valid consideration in experimental evaluation in TS 6.4.2(3).*

The list of items (1) through (4) in TS 6.4.2 is intended to represent some of the possible items for consideration in the approval of an experiment. It is not intended as a comprehensive and exhaustive list of considerations. Certainly electrical interactions, if credible, would be considered in the approval process, as would any other relevant feature or phenomenon associated with the experiment. Such considerations would be considered under the broader scope of the language stated in TS 6.4.2, regarding whether or not a proposed experiment represents a threat to the integrity of the reactor. We believe this provides a sufficiently broad purview to encompass evaluation of possible electrical interaction with reactor system instrumentation.

Attachment A – Information to Be Added to Section 8.5.1

In the event of a significant earthquake or other event that caused a loss of pool water and exposed the reactor core, personnel in the reactor building could be exposed to elevated levels of direct and indirect (sky shine – radiation reflected from the ceiling) radiation. Calculations to conservatively estimate the direct and indirect dose rates resulting from such a loss of pool water follow. The following assumptions have been made for this estimation.

1) The reactor has a 30-year operating history of 56,359 kW-hr thermal energy production per year (and accompanying fission-product accumulation). This value of thermal energy production is the average for the period 1991-2001, which is the period of time to date with a maximum operating power of 500 kW. The value for operating history was chosen to represent the period from 1991 to 2020, which is the year in which the current license extension will expire. Prior to 1991, the reactor had a maximum operating power of 10 kW, which led to thermal energy production values of 1000-2000 kW-hr per year. These low-level, long-term contributions to fission-product inventory are insignificant compared to those made with the maximum power at 500 kW, and they will therefore be ignored.
2) In the year prior to the uncovering of the core, the reactor will produce 160,000 kW-hr of thermal energy. This number is the greatest quantity of reactor thermal energy produced in a fiscal year during the period 1991-2001. It is highly unlikely that this quantity of utilization will be exceeded in a given fiscal year of the license extension.
3) The reactor will be run 9 hrs/week. This is the average utilization of the reactor during from 1991 to 2001.
4) Each reactor run has occurred in a single block of time at the end of the week. For operations that occurred more than a week before, this will have little impact on the source term because the magnitude of long-lived isotopes still emitting radiation from these runs will not change appreciably over a few days. However, for the operation immediately preceding the loss of coolant, this will over-predict the quantity of short-lived isotopes, making this assumption conservative.
5) The safety system will trip the reactor when the water level drops below 20 feet above the reactor core. Whether the reactor pool lost its water from a beam-port rupture or cracking from an earthquake, it would take some finite amount of time for the pool to empty. In section 8.4.2.2 of the SAR, it has been determined that a complete rupture of an open Beam Port #2 (a 6.125" ID port that is 16.75' below the pool surface) would cause the pool to drain in 199 seconds. This will be taken as the first time for dose estimates from the uncovered core.

Values for buildup factors, densities, mass coefficients, and gamma-ray spectra constants have been taken from the first edition of **Principles of Radiation Shielding** by Chilton, Fultis, and Shaw.

Source Term Estimation

The gamma photon source term is generated using the method outlined in Appendix 6 of the first edition of **Principles of Radiation Shielding** by Chilton, Fultis, and Shaw. This method uses a summation of exponential decay components for six different energy groups of gamma photons resulting from the decay of fission products. Validity of these parameters is claimed over the time range of 10^{-4} to 10^{9} seconds. After 10^{9} seconds (31.6 years), the contributions to the overall photon flux will be insignificant compared to contributions from more recently created fission products. The source term equation has the form:

$$\Gamma_j(t_o,t_s)=\sum_{i=1}^{N_j}\frac{\alpha_{ij}}{\lambda_{ij}}e^{-\lambda_{ij}t_s}\left(1-e^{-\lambda_{ij}t_o}\right),\quad j=1\text{ to }6 \tag{1}$$

where Γ_j = rate of energy release in group j normalized to unit fission rate (MeV / fission)

α_{ij} = gamma-ray spectra constants for ^{235}U fission products

λ_{ij} = gamma-ray spectra constants for ^{235}U fission products

t_o = length of operation time (s)

t_s = elapsed time since reactor operation (s)

The energy groups are:

Group	**1**	**2**	**3**	**4**	**5**	**6**
Energy Range (MeV)	5-7.5	4-5	3-4	2-3	1-2	0-1
Average Energy (MeV)	6.25	4.5	3.5	2.5	1.5	0.5

This equation produces an estimate of gamma energy in MeV in each energy group per fission, normalized to unit fission rate. Therefore, multiplying by the average fission rate and dividing by the average energy in each group gives an estimate of the number of photons released each second in each energy group at time t_s after an operation for t_o seconds. Each photon in an energy group is assumed to have the average energy of that group. To account for buildup of long-life fission products from an extended reactor operation history, a summation of this equation is performed over the number of years specified in one-week increments.

To simplify the solution of a distributed radiation source and target problem, the source term is assumed to be at a single point at the center of the reactor. This assumption is reasonable because the dimensions of the reactor (15"x18"x24") are less than 1/10 the magnitude of the distances from source to targets in this problem. In addition, this assumption is conservative in that the source point is exposed to a greater solid angle of reactor building ceiling than the actual core would be. Radiation attenuation within the core is ignored, which again is conservative. Radiation reflected from the pool walls and floor is ignored since it will be much less and lower energy than the direct radiation.

Direct Dose Rates at the Pool Top East Wall

If a reactor employee were to visually verify loss of pool water, he would go to the pool top at the east pool wall, necessitating an estimate of direct radiation at this location. To estimate the direct dose rate, the source term is multiplied by the geometric efficiency at a given distance and by the percentage of photons in each energy group that interact in a 1 cm thickness of air to determine the number of photons that interact in 1 cm^3 of air at the appropriate distance. Absorption of photons in the air between the source and target is ignored since the high-energy source of photons will experience little attenuation by air in the distance traveled. This is a conservative simplification.

The geometric efficiency is defined for this calculation as the ratio of 1 cm^2 to the surface area of a sphere ($1/[4\pi r^2]$) with a radius equal to the distance from the source point to the top of the east pool wall (561 cm). The percentage of photons/second that interact in a given thickness is determined with the equation

$$1-\exp\left(\frac{\mu}{\rho}\cdot\rho\cdot x\right) \tag{2}$$

where: μ/ρ = mass attenuation coefficient (cm^2/g)
ρ = material density (g/cm^3)
x = material thickness (cm)

Therefore, the interaction rate density at a distance 'r' is:

$$R=\frac{S}{4\pi r^2}\left[1-\exp\left(\frac{\mu}{\rho}\cdot\rho\cdot x\right)\right] \tag{3}$$

where: R = photon interaction rate density ($\#/cm^3/s$)
S = photon source strength (#/s)
r = source-to-target distance (cm)
μ/ρ, ρ, x are as defined in Equation 2

Multiplying the interaction rate density for each energy group by the average deposited energy for that group and summing gives the energy deposition rate density with units of $MeV/cm^3/s$. Since Compton scattering strongly dominates photon interactions in air (and water) for incident photon energies greater than 0.1 MeV, the energy deposited per interaction will be estimated as the maximum energy deposition for Compton scattering, which occurs for a scattering angle of 180°.

The energy of Compton-scattered photons is given by equation 2-17 in the 2^{nd} edition of ***Radiation Detection and Measurement*** by Glenn Knoll as:

$$E' = \frac{E}{1 + \frac{E}{0.511\,\text{MeV}}(1 - \cos\theta)} \qquad (4)$$

where E = energy of incident photon (MeV)
E' = energy of scattered photon (MeV)
θ = scattering angle (degrees)

Therefore, the energy deposited by an incident photon of energy E that is scattered 180° is:

$$E - E' = E \cdot \left(1 - \frac{1}{1 + \frac{2 \cdot E}{0.511\,\text{MeV}}}\right) \qquad (5)$$

where E and E' are defined the same as in Equation (4)

The dose rate in air can then be calculated with the equation:

$$\dot{D} = \dot{E}\left(\frac{\text{MeV}}{\text{cm}^3 \cdot \text{s}}\right) \cdot \frac{1}{\rho_{air}}\left(\frac{\text{cm}^3}{\text{g}}\right) \cdot \frac{\text{J}}{6.24\text{x}10^{12}\,\text{MeV}} \cdot \frac{3600\,\text{s}}{\text{hr}} \cdot \frac{1000\,\text{g}}{\text{kg}} \cdot \frac{100\,\text{rad}}{\text{J/kg}} \qquad (6)$$

where $\dot{D}$ = dose rate (rad/hr)
$\dot{E}$ = energy deposition rate density in air ($MeV/cm^3/s$)
ρ_{air} = density of air (g/cm^3)

In addition to calculating the dose rate in air with this method, the dose rate in water is calculated because water is a reasonable representation of tissue for dose calculations. Because the quality factor for gamma radiation is one, the dose rate in rad/hr will equal the effective dose rate in rem/hr.

The estimated direct-radiation dose rates are shown below in Table 1.

Time Elapsed Since Reactor Scram	Dose Rate to Air (rem/hr)	Dose Rate to Water (rem/hr)
199 sec	16116	17275
1 hour	5907	6329
10 hours	992	1059
24 hours	501	533
7 days	149	158
30 days	66	71
90 days	32	34

Table 1. Dose Rates from Direct Radiation from the Uncovered OSURR Core

It is obvious that visual verification of an uncovered core is not a good idea for personnel immediately after core uncovering is suspected to have happened.

Sky Shine Exposure Rate Estimation

To estimate the percentage of emitted photons that impinge upon the ceiling, the ratio of the surface area of the pool top to the surface area of a sphere with a radius equal to the perpendicular distance from the source point to the pool top is calculated. This is conservative because more collimation will occur than this simplification estimates (the average distance from core to pool top is greater than the perpendicular distance).

To make the distributed-source / distributed target problem of estimating the number of photons that reflect from the ceiling back towards the reactor building bay floor more manageable, some simplifications must be made. The Klein-Nishina formula (Equation 2-18 in the 2nd edition of ***Radiation Detection and Measurement*** by Glenn Knoll) gives differential scattering cross-section as a function of solid angle in spherical coordinates and photon energy.

$$\frac{d\sigma}{d\Omega} = Zr_o^2\left(\frac{1}{1+\alpha(1-\cos(\theta))}\right)^2\left(\frac{1+\cos^2(\theta)}{2}\right)\left(1+\frac{\alpha^2(1-\cos(\theta))^2}{(1+\cos^2(\theta))[1+\alpha(1-\cos(\theta))]}\right) \quad (7)$$

where $d\sigma$ = differential scattering cross section
$d\Omega$ = differential solid angle
Z = atomic number
r_0 = classical electron radius
α = ratio of photon energy to electron rest mass
θ = azimuthal scattering angle

Since scattering is uniform over polar angle, this variable can be integrated out, leaving an equation that is only a function of photon energy and azimuthal angle. The angle from the point on the ceiling above the reactor to the floor at the edge of the reactor-building bay is roughly 35°. Therefore, the number of photons/second that scatter off of the ceiling towards the floor can be estimated by multiplying the total number of photons/second that are scattered by the ceiling by the ratio:

$$\frac{\text{Klein-Nishina formula numerically integrated from } \theta = 145° \text{ to } 180°}{\text{Klein-Nishina formula numerically integrated from } \theta = 0° \text{ to } 180°}$$

This simplification is conservative because some photons that are scattered in the angle range 145°-180° from the ceiling away from the point directly above the reactor will not head towards the floor, even though they are counted by this approximation.

The total number of photons/second that are scattered by the ceiling is determined by multiplying the number of photons/second that impinge upon the ceiling by Equation 2 (using values for μ/ρ, ρ and x for the ceiling material). This then gives a formula for an estimate of photons/second in each energy group that reflect from the ceiling back towards the reactor building bay floor as:

$$Nr = Nc \cdot \left(1 - \exp\left(\frac{\mu}{\rho} \cdot \rho \cdot x\right)\right) \cdot \frac{\int_{145°}^{180°} d\theta \left(\frac{1}{1+\alpha(1-\cos(\theta))}\right)^2 \left(\frac{1+\cos^2(\theta)}{2}\right)\left(1 + \frac{\alpha^2(1-\cos(\theta))^2}{(1+\cos^2(\theta))[1+\alpha(1-\cos(\theta))]}\right)}{\int_{0°}^{180°} d\theta \left(\frac{1}{1+\alpha(1-\cos(\theta))}\right)^2 \left(\frac{1+\cos^2(\theta)}{2}\right)\left(1 + \frac{\alpha^2(1-\cos(\theta))^2}{(1+\cos^2(\theta))[1+\alpha(1-\cos(\theta))]}\right)} \quad (8)$$

where Nr = number of photons/second that are reflected to the floor
Nc = number of photons/second that impinge on the ceiling
μ/ρ, ρ, x are as defined in Equation 2
α, θ are as defined in Equation (7)

The photon flux can be estimated by dividing the number of photons/second that are reflected to the floor by the area of a circle formed by the 35° azimuthal angle from the point on the ceiling above the reactor. Then, the interaction rate density is estimated using Equation 2 to determine the number of photons/second that Compton scatter in a cm^3 per second at five feet from the floor. Five feet from the floor is chosen as a reasonable height for estimating whole-body dose rate.

The energy of photons reflected to the floor can be determined using Equation 4, which gives scattered photons of ~ 0.2 MeV for all of the energy groups of source photons. Since significant absorption will occur for these low-energy photons in the 30 feet between the ceiling and five feet above the floor, attenuation will not be ignored. For this distance and energy, approximately 13.4% of the photons will be absorbed, adding a factor of 0.866 to the calculation of interaction rate density.

The energy deposition rate density is then determined by multiplying the photon interaction rate density by the amount of energy deposited per interaction, which is determined to be 0.088 MeV by Equation 5 with an incident energy of 0.2 MeV. The dose rate is calculated by using this estimate of energy deposition rate density in Equation 6. Table 2 shows estimated dose rates to air and water from indirect radiation scattered from the ceiling to 5 feet above the reactor bay floor. It appears that personnel should be able to reenter the building after evacuation as long as ALARA is observed.

Time Elapsed Since Reactor Scram	Dose Rate to Air (mrem/hr)	Dose Rate to Water (mrem/hr)
199 sec	3027	3133
1 hour	1144	1185
10 hours	229	237

24 hours	132	136
7 days	37	39
30 days	18	19
90 days	10	10

Table 1. Dose Rates from Scattered Radiation from the Uncovered OSURR Core

Attachment B – Addition to Section 8.4.4.5

Doses to persons outside the building will come from submersion in a cloud of released radionuclides and from radiation emitted from the reactor building. The submersion dose results from the diluted radionuclide stream from the exhaust fan or from natural flow of air through the building that exits at the roofline (if the exhaust fan has been shut off).

An analysis for the activity concentration released from the building can be performed using the same method used in Section 6.3.5, which describes Ar-41 release from the exhaust fan.

$$A_D = A \cdot q \cdot \Psi(x)$$

where A_D = effective exposure concentration in curies/m^3,
q = building exhaust rate in m^3/second,
$\psi(x)$ = dilution factor at distance x, in sec/m^3, and
A = activity concentration in the exhaust stream.

The dilution factor was calculated in Section 6.3.5 at x = 0 as 9.921×10^{-3} s/m^3 for a release from the roofline of the building. Using this value for $\psi(0)$ with the appropriate values for building exhaust rate gives values for activity concentrations outside the restricted area. These concentrations can then be used to calculate estimates for accumulated doses from the nuclides of interest as was done for immersion dose inside the building. For isotope 'i',

$$D_i = \frac{A_i \cdot q \cdot \Psi}{\rho V} \cdot \frac{1 - e^{-(\lambda_i + \Lambda)t}}{\lambda_i + \Lambda} E_i \cdot F_1 \cdot F_2 \cdot F_3 \cdot F_4 \cdot F_5 \cdot F_6$$

where all variables are as defined previously in this section.

Tables 8.16 and 8.17 show the results of this calculation. We can see that turning the building fan off should keep submersion doses to personnel outside the building to very low levels. Even if the exhaust fan were left running, accumulated dose to persons outside would be reasonable, particularly after the first day.

To calculate the direct dose from the reactor building to someone standing at ground level at the middle of one of the outer walls, assume a half-hemisphere with a volume equivalent to that of the reactor building. The direct dose can then be calculated as a submersion dose from a finite hemisphere divided by two, since the dose only comes from half of a hemisphere.

$$D_i = \frac{A_i}{\rho \cdot V} \frac{1 - e^{-(\lambda_i + \Lambda)t}}{\lambda_i + \Lambda} E_i \cdot F_1 \cdot F_2 \cdot F_3 \cdot F_4 \cdot F_5 \cdot F_7 \cdot \mu \cdot R$$

F_7 = factor to account for half-hemispherical geometry = ¼
μ = gamma absorption coefficient in air (m^{-1})
R = radius of half hemisphere (m)
and all other variables are as defined above

Tables 8.18 and 8.19 show the results of this calculation. This estimate is very conservative in that it assumes that a person is standing up against a building wall for an extended period of time, and it does not take into account absorption from building walls or concrete in the building. In the event of such an accident, personnel would be prevented from receiving integral doses from direct radiation such as those seen in Table 8.18 by restricting access to the building.

Table 8.16 Integral Whole-Body Gamma Doses From Submersion Outside of the Restricted Area Assuming a Leakage Fraction of 0.0042 Hr^{-1} (Purge Fan Off)

	Dose In Rem									
Isotope symbol	Exposure Times									
	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	24 Hours	48 Hours	168 Hours	720 Hours
^{131}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{132}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{133}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{134}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{135}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{85m}Kr	0.0000	0.0000	0.0000	0.0000	0.0001	0.0002	0.0006	0.0006	0.0006	0.0006
^{87}Kr	0.0000	0.0001	0.0001	0.0003	0.0005	0.0007	0.0011	0.0011	0.0011	0.0011
^{88}Kr	0.0003	0.0005	0.0007	0.0014	0.0027	0.0048	0.0120	0.0121	0.0121	0.0121
^{131m}Xe	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0001	0.0002	0.0003
^{133m}Xe	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0003	0.0005	0.0008	0.0009
^{133}Xe	0.0000	0.0000	0.0001	0.0001	0.0002	0.0004	0.0043	0.0077	0.0167	0.0208
^{135m}Xe	0.0000	0.0000	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001
^{135}Xe	0.0000	0.0001	0.0001	0.0003	0.0006	0.0011	0.0063	0.0073	0.0074	0.0074
TOTALS	0.0003	0.0007	0.0011	0.0022	0.0042	0.0073	0.0247	0.0295	0.0390	0.0433

Table 8.17 Integral Whole-Body Gamma Doses From Submersion Outside of the Restricted Area Assuming a Leakage Fraction of 0.857 Hr^{-1} (Purge Fan On)

Dose In Rem										
Isotope symbol	Exposure Times									
	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	24 Hours	48 Hours	168 Hours	720 Hours
^{131}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{132}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{133}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{134}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{135}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{85m}Kr	0.0016	0.0030	0.0043	0.0077	0.0123	0.0167	0.0193	0.0193	0.0193	0.0193
^{87}Kr	0.0098	0.0185	0.0263	0.0449	0.0674	0.0841	0.0897	0.0897	0.0897	0.0897
^{88}Kr	0.0498	0.0952	0.1366	0.2402	0.3783	0.5034	0.5651	0.5651	0.5651	0.5651
^{131m}Xe	0.0000	0.0001	0.0001	0.0002	0.0003	0.0004	0.0005	0.0005	0.0005	0.0005
^{133m}Xe	0.0002	0.0005	0.0007	0.0013	0.0021	0.0029	0.0035	0.0035	0.0035	0.0035
^{133}Xe	0.0033	0.0064	0.0092	0.0167	0.0276	0.0392	0.0477	0.0477	0.0477	0.0477
^{135m}Xe	0.0055	0.0096	0.0126	0.0178	0.0209	0.0215	0.0215	0.0215	0.0215	0.0215
^{135}Xe	0.0097	0.0187	0.0271	0.0485	0.0789	0.1099	0.1300	0.1300	0.1300	0.1300
TOTALS	0.0799	0.1520	0.2169	0.3773	0.5878	0.7781	0.8773	0.8773	0.8773	0.8773

Table 8.18 Integral Whole-Body Gamma Doses From Direct (From the Building) Dose Assuming a Leakage Fraction of 0.0042 Hr^{-1} (Purge Fan Off)

Isotope symbol	Dose In Rem									
	Exposure Times									
	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	24 Hours	48 Hours	168 Hours	720 Hours
^{131}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0003	0.0005	0.0012	0.0017
^{132}I	0.0000	0.0000	0.0000	0.0000	0.0001	0.0001	0.0003	0.0003	0.0003	0.0003
^{133}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0001	0.0006	0.0009	0.0011	0.0011
^{134}I	0.0000	0.0000	0.0000	0.0000	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001
^{135}I	0.0000	0.0000	0.0000	0.0000	0.0001	0.0002	0.0008	0.0008	0.0009	0.0009
^{85m}Kr	0.0075	0.0149	0.0222	0.0434	0.0835	0.1544	0.5437	0.5546	0.5548	0.5548
^{87}Kr	0.0403	0.0788	0.1156	0.2166	0.3822	0.6055	0.9195	0.9195	0.9195	0.9195
^{88}Kr	0.2044	0.4045	0.6005	1.1640	2.1889	3.8861	9.7232	9.7449	9.7449	9.7449
^{131m}Xe	0.0002	0.0003	0.0005	0.0010	0.0020	0.0040	0.0443	0.0822	0.2028	0.3001
^{133m}Xe	0.0012	0.0024	0.0037	0.0073	0.0145	0.0288	0.2896	0.4832	0.8210	0.8729
^{133}Xe	0.0145	0.0289	0.0434	0.0867	0.1730	0.3443	3.7214	6.6699	14.4060	17.9015
^{135m}Xe	0.0292	0.0525	0.0713	0.1078	0.1362	0.1457	0.1464	0.1464	0.1464	0.1464
^{135}Xe	0.0479	0.0955	0.1428	0.2827	0.5543	1.0659	6.1429	7.0391	7.1922	7.1922
TOTALS	0.3452	0.6778	1.0000	1.9095	3.5349	6.2352	21.5331	25.6424	33.9912	37.6364

Table 8.19 Integral Whole-Body Gamma Doses From Direct (From the Building) Dose Assuming a Leakage Fraction of 0.857 Hr^{-1} (Purge Fan On)

Dose In Rem										
Isotope symbol	Exposure Times									
	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	24 Hours	48 Hours	168 Hours	720 Hours
^{131}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{132}I	0.0000	0.0000	0.0000	0.0000	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001
^{133}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
^{134}I	0.0000	0.0000	0.0000	0.0000	0.0000	0.0001	0.0001	0.0001	0.0001	0.0001
^{135}I	0.0000	0.0000	0.0000	0.0000	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001
^{85m}Kr	0.0072	0.0139	0.0200	0.0355	0.0568	0.0774	0.0891	0.0891	0.0891	0.0891
^{87}Kr	0.0389	0.0735	0.1043	0.1780	0.2669	0.3333	0.3554	0.3554	0.3554	0.3554
^{88}Kr	0.1973	0.3773	0.5414	0.9519	1.4991	1.9945	2.2391	2.2391	2.2391	2.2391
^{131m}Xe	0.0002	0.0003	0.0004	0.0008	0.0013	0.0019	0.0023	0.0023	0.0023	0.0023
^{133m}Xe	0.0012	0.0023	0.0033	0.0059	0.0098	0.0139	0.0169	0.0169	0.0169	0.0169
^{133}Xe	0.0140	0.0270	0.0391	0.0706	0.1165	0.1656	0.2015	0.2015	0.2015	0.2015
^{135m}Xe	0.0282	0.0492	0.0649	0.0919	0.1076	0.1108	0.1109	0.1109	0.1109	0.1109
^{135}Xe	0.0462	0.0890	0.1286	0.2305	0.3750	0.5226	0.6182	0.6182	0.6182	0.6182
TOTALS	0.3332	0.6325	0.9020	1.5651	2.4332	3.2203	3.6337	3.6337	3.6337	3.6337