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Safety Analysis for Operating the Annular Core Research Reactor with Cintichem-Type Targets Installed in the Central Region of the Core

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Safety Analysis for Operating the Annular Core Research Reactor with Cintichem-Type Targets Installed in the Central Region of the Core

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Abstract

Production of the molybdenum-99 isotope at the Annular Core Research Reactor requires highly enriched, uranium oxide loaded targets to be irradiated for several days in the high neutron-flux region of the core. This report presents the safety analysis for the irradiation of up to seven "Cintichem-type" targets in the central region of the core and compares the results to the Annular Core Research Reactor Safety Analysis Report. A 19 target grid configuration is presented that allows one to seven targets to be irradiated, with the remainder of the grid locations filled with aluminum "void" targets. Analyses of reactor neutronic, thermal hydraulics, and heat transfer calculations are presented. Steady-state operation and accident scenarios are analyzed with the conclusion that the reactor can be operated safely with seven targets in the grid, and no additional risk to the public.

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CONTENTS

EXECUTIVE SUMMARY	9
1. INTRODUCTION.....	11
1.1 <i>Mo-99 Demand and Production Requirements</i>	11
1.2 <i>Mo-99 Production Using the ACRR</i>	13
1.3 <i>Proposed Activity</i>	13
2. TARGET ISSUES	15
2.1 <i>Target Description</i>	15
2.2 <i>Target Quality</i>	17
2.3 <i>Target Receipt, Inspection, Storage, Handling, and Transfer</i>	17
2.4 <i>Target Grid</i>	18
3. NEUTRONIC ANALYSIS	19
3.1 <i>MCNP Results for 7 Targets and 12 Aluminum "Void" Targets</i>	20
3.2 <i>Fission Densities and Peaking Factors for the Fuel Elements</i>	24
3.3 <i>Target Coupling Factors</i>	30
3.4 <i>Summary of Neutronic Results</i>	30
4. THERMAL ANALYSIS	32
4.1 <i>Steady-State Heat Transfer Analysis - Fuel Elements</i>	32
4.2 <i>Steady-State Heat Transfer Analysis - Targets</i>	33
5. CRITICAL HEAT FLUX ANALYSIS	36
5.1 <i>Steady-State Critical Heat Flux Analysis - Fuel Elements</i>	36
5.2 <i>Steady-State Critical Heat Flux Analysis - Targets</i>	36
6. STRESS ANALYSIS.....	39
6.1 <i>Steady-State Stress Analysis of the ACRR Fuel</i>	39
6.2 <i>Steady-State Stress Analysis of the ACRR Fuel Element Clad</i>	39
6.3 <i>Steady-State Stress Analysis of the Target Coating</i>	39
6.4 <i>Steady-State Stress Analysis of the Target Clad</i>	40
7. ACCIDENT ANALYSIS WITH COMPARISONS TO CURRENT SAR.....	41
7.1 <i>Target Radionuclide Inventory and Release</i>	42
7.2 <i>Accident Analysis</i>	45
7.2.1 <i>Mishandled/Dropped Target</i>	45
7.2.2 <i>Misloaded Target</i>	47
7.2.3 <i>Dislodged or Removed Target at Power</i>	49
7.2.4 <i>Target Misalignment During Loading</i>	49
7.2.5 <i>Target Coating Failure at Power</i>	50
7.2.6 <i>Leaking Target</i>	51
7.2.7 <i>Flooded Target</i>	53

7.2.8 Uncontrolled Control Rod Bank Withdrawal	54
7.2.9 Control Rod Bank Overspeed	57
7.2.10 Limiting Event Pulse	58
7.3 Effects of Moderator Heating in the Target Region	59
7.4 Other Off-Normal Events	60
8. SUMMARY AND CONCLUSIONS	66
9. REFERENCES	67
APPENDIX A - Sample Input/Output for the Seven Target/177 Fuel Element Configuration	A-1
APPENDIX B - Hazard Analysis for Target Irradiation and Storage at the ACRR Facility	B-1

TABLES

Table 1	Neutronic results for 7 targets/12 aluminum “void” targets configuration.	22
Table 2	Element average to core average fission ratios for the ACRR core configuration with 7 target, 12 aluminum “void” targets, and 177 fuel elements.	25
Table 3	Element average to core average fission ratios for the ACRR core configuration with 0 targets, 19 aluminum “void” targets, and 201 fuel elements.....	26
Table 4	Target coupling factors for the 7 target, 12 aluminum “void” target configuration.....	30
Table 5	Peak and average temperatures for the inner and outer fuel pieces for a fuel element axial peaking factor equal to $F_{ze}=1.25$	33
Table 6	Peak coating and clad temperatures for a target at 20 kW, axial peaking factor equal to $F_{ze}=1.2$, and $T_{wall} = 120^{\circ}\text{C}$	34
Table 7	Accident scenarios with targets.....	41
Table 8	Noble and halogen radioisotope target inventory for a 20 kW target.	43
Table 9	Downwind effective whole body dose for the complete release of a 20 kW - saturated activity target, 100% nobles, 1% halogens, and 30 minute decay.	44
Table 10	Downwind effective whole body dose for gas release of a 20 kW - saturated activity target, 3% nobles, 0.03% halogens, 30 minute decay.	46
Table 11	Downwind effective whole body dose for gas release of a 20 kW - saturated activity target, 3% nobles, 3% halogens, and 30 minute decay.....	47
Table 12	Solubility of some noble gases and halogens in water.	52
Table 13	Downwind effective whole body dose for a leaking target at 20 kW - 14 day irradiation.	53
Table 14	Comparison of SAR events to the proposed action.....	61

FIGURES

Figure 1	Production and decay scheme for Mo-99.....	11
Figure 2	Schedule for Mo-99 delivery.	12
Figure 3	Conversion of Mo-99 activity to power generation required.	12
Figure 4	Scale target representation.	16
Figure 5	Target internals.	16
Figure 6	Elevation view of 19 target grid installed in the core.	18
Figure 7	Element and sector identification.....	19
Figure 8	Plan view of ACRR with 177 fuel elements, 7 targets, and 12 aluminum “void” targets arranged with a triangular pitch of 5.3 cm.	21
Figure 9	Elevation view of the ACRR with targets.....	21
Figure 10	Radial Fission Distribution for ACRRDCINT43APIE, 19 Targets, 147 Element Core, CR Out.....	28
Figure 11	Radial Fission Distribution for 201_CROUT, 19 Aluminum “Void” Targets, 201 Element Core, CR Out.	29
Figure 12	Target power required to reach coolant saturation at channel exit conditions and CHF conditions.	38

EXECUTIVE SUMMARY

A previous safety analysis (Parma, 1998a) presented the basis for operating the Annular Core Research Reactor (ACRR) in the steady-state mode with the central cavity liner removed, the central cavity region flooded, and aluminum "void" targets installed in a 19 target grid configuration. A 177 to 213 element core range was analyzed to envelop the neutronic characteristics of the core. An accident evaluation compared the consequences of abnormal events to those in the ACRR Safety Analysis Report (SAR) (Estes, 1996), and concluded that there was no increase in the likelihood of occurrence or consequences of the limiting event, that there were no new accident scenarios or initiating events, and that there was no impact on the margin of safety as compared to the current safety authorization. It was also concluded that reactor operations with the 19 "void" targets installed could be conducted safely under the ACRR Technical Safety Requirements (TSRs) (Boldt, et al., 1998).

The SAR contains information regarding target irradiation in the central region of the core with the cavity liner removed. Section 10.10, "Experiments Requiring Long-Term Exposure," and Section 14.4.11.7, "Long Term Experiment Hazards," present the irradiation of up to 19 targets containing 20 g of ^{235}U at power levels of 17.5 kW for up to 20 days. Also described is the operation of the reactor at 2 MW for 2 years with a comparison of the fission product inventory to the limiting event. Additional analysis, presented in this report, was performed to ensure that target irradiation could be performed safely and within the bounds of the SAR.

This safety analysis presents results from neutronic, thermal hydraulic, and heat transfer calculations associated with operating the ACRR with "Cintichem-type," 93% enriched, uranium oxide loaded targets in the central region of the core. A 19 target grid configuration is presented that allows one to seven targets to be irradiated, with the remainder of the grid locations filled with aluminum "void" targets. Consistent with the prior analysis, operation of the reactor in the steady-state mode only is considered.

Some of the neutronic characteristics of the reactor will change with the addition of targets in the central region of the core. Peaking factors and excess reactivity will change as compared to the configuration without targets. The control rod bank worth and temperature feedback effects are calculated to not change significantly from the results in Parma (1998a).

The following parameters and the ranges are assumed for this configuration:

Targets --

Configuration: 19 grid

Number: 1 to 7 targets installed in the central region of the target grid; row 1 (center location) and row 2 (six middle locations) are allowed for targets - the empty locations are filled with aluminum "void" targets (solid aluminum targets).

Loading: 20 g ^{235}U loading nominal, in the form of UO_2 .

Range allowed: 18 - 22 g ^{235}U (to be expanded to 10 - 40 g in the future SAR update).

Target Movement: Target movement into and out of the target grid will be performed with the reactor shutdown. Irradiated target transfer out of the ACRR pool or transfer to the Hot Cell Facility is not addressed in this analysis. If transfer to the hot cell facility is required, it will be performed by a separate analysis and using approved facility procedures.

Target Inspection: Target inspection/acceptance will be performed using approved procedures for target inspection.

Irradiated Target Storage: Irradiated or unirradiated targets may be stored in either the 27 storage holster or the 10 storage holster in the ACRR pool.

Core Configuration -

Configuration and Number: Commensurate with the target loading. Enough excess reactivity is required to allow for 100% reactor power level to be reached with equilibrium ^{135}Xe concentration in the core (~\$5-\$7 estimated), but limited by the TSRs to less than \$12. A core size of 177 elements (five rows of fuel), and seven 20 g, ^{235}U loaded targets has a calculated excess reactivity of ~\$5. The same configuration with a 201 element core has a calculated excess reactivity of ~\$8.

Ni Elements: Nickel elements are not required in the outer row of the core.

90% elements: The ten 90% loaded fuel elements are not excluded from use in the proposed configuration.

Aluminum/Water elements: The hollow aluminum elements (water filled) are not excluded from use in the proposed configuration.

Operating Parameters -

Power Level: The maximum power per element allowed by the TSRs is 25 kW per element. Nominal operating power is 21.74 kW per element. For a 201 fuel element core with 7 targets and 12 aluminum "void" targets installed in the 19 target grid, and the peak fuel element operating at 21.74 kW, the average target power is expected to be 13.7 kW. The total core and target power level (100% power level) is expected to be 2.65 MW. However, the current heat rejection system is limited in capacity to ~2 MW. Therefore it is uncertain if the 2.65 MW can be reached and maintained. There is no TSR limit on total reactor power. The maximum pool water inlet temperature to the core allowed by the TSRs is 50°C. The maximum measured fuel temperature allowed by the TSRs is 1400°C.

Excess Reactivity: Limited to \$12 by TSRs.

Duration at Power: The duration at power will be limited to 8 days per run, since the limiting event, described in the SAR, uses this duration to determine the dose at the 3 km boundary. With the pulse mode disabled, the limiting event is considered incredible.

This evaluation does not address the heat rejection system upgrade to 4 MW, the airlock addition, or other upgrades to auxiliary systems. Reactor operations will be conducted under the TSRs. The conclusions drawn from this analysis are that, with seven targets installed in the 19 target grid, the reactor can be operated safely and within the authorization basis established in the SAR.

1. INTRODUCTION

Currently there are approximately 40,000 nuclear diagnostic procedures performed daily in the U.S. The most important radioisotope used in nuclear medicine diagnostic procedures is technetium-99m which results from the decay of molybdenum-99. ^{99m}Tc is involved in 70-80 percent of all nuclear medicine procedures in the U. S. and its use is expected to continue to grow. Its widespread usage in the nuclear medicine community is due to the availability of the ^{99}Mo supply, the transportability of ^{99}Mo , the parent/daughter half-life relationship of the two isotopes, and the many valence states, short half-life, and low-energy gamma emitted from ^{99m}Tc . These factors allow ^{99m}Tc to be used in a wide variety of diagnostic tests while maintaining a low body burden to the patient.

There are two basic methods for the production of significant quantities of ^{99}Mo ; 1) nuclear fission, and 2) neutron activation of ^{98}Mo , a stable isotope with an atomic abundance of about 24 percent. Fission is the most widely used method, since a very high specific activity of ^{99}Mo can be achieved, allowing for smaller dispensers, or generators, for the ^{99m}Tc . In the fission process, target capsules are loaded with uranium that has been highly enriched in the ^{235}U isotope. The targets are placed in a reactor core in high flux areas and irradiated for several days. As shown in Figure 1, the fission product ^{99}Mo is produced with a probability of about 6 percent. It decays with a half-life of 66 hours to ^{99m}Tc and ^{99}Tc .

Figure 1 Production and decay scheme for Mo-99.

Until its shutdown in 1991, Cintichem, Inc. provided approximately half of the U. S. demand for ^{99}Mo , with Nordion International, Inc. of Canada providing the balance of the supply. Both providers used the fission process to produce the ^{99}Mo , although the chemical extraction processes were significantly different. With the shutdown of Cintichem, Inc., the DOE purchased the rights to the "Cintichem process" after being commissioned by Congress to facilitate finding or creating a backup supplier. Nordion currently supplies all of the ^{99}Mo for the U. S.

1.1 Mo-99 Demand and Production Requirements

The current U. S. demand for ^{99}Mo is approximately 3,000 invoice, or six day curies (Ci), per week. World demand is roughly twice that of the U. S. As shown in Figure 2, for a processing and shipping time of 30 hours, the production requirement at the source to meet the U. S. demand is approximately 19,700 Ci/week. Figure 3 shows the formulation for converting the required production activity to the required power generation in the targets. A seven-day irradiation time, chosen arbitrarily, provides about 83 percent of the saturated activity of ^{99}Mo . For a seven-day irradiation, approximately 42.4 Ci of ^{99}Mo are produced as fission products per kilowatt (kW) of

target power. Thus, to meet the U. S. demand, about 465 kW of target fission power must be generated continuously, seven days per week, twenty-four hours per day, with all of the targets processed weekly. If 25 kW of fission power were generated in each target, 19 targets could be processed weekly to meet the U..S. demand. A lower target power would require the irradiation and processing of more targets to meet the full U. S. demand.

Figure 2 Schedule for Mo-99 delivery.

Time rate of change of ^{99}Mo = Production rate - Decay rate - Absorption rate

$$\frac{d^{99}\text{Mo}}{dt} = \gamma \Sigma_f \Phi V - \lambda^{99}\text{Mo} - \sigma_a \Phi^{99}\text{Mo}$$

Saturated Activity = 51.1 Ci/kW

$$^{99}\text{Mo Activity (t)} = \lambda^{99}\text{Mo (t)} = \frac{\lambda \gamma \Sigma_f \Phi V}{\lambda + \sigma_a \Phi} (1 - e^{-(\lambda + \sigma_a \Phi)t})$$

for ^{99}Mo , σ_a is small, $\sigma_a \Phi \Rightarrow 0$

7 day irradiation is 0.83 of saturated activity

U.S. demand = 19,700 Ci/wk

Minimum nuclear power required for targets = 465 kW

Meeting the U.S. demand would require operating the reactor 7 days per week and processing targets every day.

Figure 3 Conversion of Mo-99 activity to power generation required.

1.2 Mo-99 Production Using the ACRR

Hundreds of neutronic calculations were performed using the computer code MCNP (Briesmeister, 1993) in order to determine the optimal configuration for production of ^{99}Mo in the ACRR using "Cintichem-type" targets. The conclusion reached was that, although a significant quantity of ^{99}Mo could be produced at different locations within the core and reflector regions, producing tens of thousands of curies weekly in an efficient manner would require irradiation in the central region of the core where the neutron flux levels are the highest (Coats and Parma, 1994). Removal of the central cavity liner and the placement of targets in that region were found to allow for the largest neutronic coupling between the targets and the core and also allowed for natural convection cooling of the targets (Parma, 1995a, Parma, 1997a).

Under steady-state conditions, the maximum power that can be achieved in a target is not limited by the constraints on target coating temperature or heat flux. Instead the target power is limited by the maximum allowable heat flux for the peak fuel element. A target power-to-peak fuel element power ratio of less than one, and the increased flow channel size of the targets, ensures that the peak fuel element heat flux will always be the limiting condition under steady-state operating conditions. The current TSRs limit the peak fuel element to 25 kW maximum and 21.74 kW nominal. Using the 21.74 kW peak element power, a 19 target arrangement could produce ~65% of the U. S. demand for 20 g ^{235}U loaded targets and a seven day irradiation. Processing 38 targets using the 19 target grid and a three and four day cycle could produce ~93% of the U. S. demand. The average target power would be 15.7 kW. For seven targets irradiated in the 19 target grid, a production level of 22 to 31% could be maintained (Parma, 1997a).

1.3 Proposed Activity

This safety analysis presents the calculative results associated with operating the reactor in the steady-state mode with the central cavity liner removed and the central region of the core filled with one to seven "Cintichem-type" uranium loaded targets and aluminum "void" targets in the 19 target grid configuration. This analysis describes operation of the reactor in the steady-state mode only. Accident scenarios involving power transients governed by reactivity insertions are also presented. Consistent with Parma (1998a), pulse-mode operations are not presented.

For this analysis, a configuration using the 19 target grid and a one to seven target arrangement is proposed. The target loading will be limited to 20 ± 2 g of ^{235}U in the form of 93% enriched uranium oxide. Irradiation of greater than seven targets and different target loadings will be analyzed in a future updated ACRR SAR. The core size will be dictated by the amount of excess reactivity required to reach the maximum operating power level with the equilibrium xenon concentration in the core. The maximum operating power level is limited by the TSR limit on the peak fuel element power of 25 kW maximum and 21.74 kW nominal; the excess reactivity is limited to \$12. The current physical limitations of the heat rejection system will, in effect, limit the long-term, steady-state power level to ~2 MW. The duration at power will be limited to 8 days per run, since the limiting event uses this duration to determine the dose at the 3 km boundary. With the pulse mode disabled, the limiting event is considered incredible.

In the steady-state mode, the most important issues regarding reactor safety are power peaking, changes in reactivity, and off-normal events associated with the core configuration changes. Since the reactor is operated at a relatively low power level and is cooled by natural convection,

most credible accident-initiating events are driven by reactivity excursions. This evaluation addresses these issues by presenting the results of detailed calculative information and concludes that the reactor can be operated safely and within the authorization basis established in the SAR. Reactor operations will be conducted under the ACRR TSRs.

No additional modifications to the reactor are required to irradiate targets other than inserting targets into the target grid. This analysis does not include any reference to the heat rejection system upgrade, airlock installation, or upgrades to any auxiliary systems. Also omitted from this analysis are irradiated target transfer issues and target inspection details. Irradiation of greater than seven targets and higher and lower uranium loadings will be analyzed in the future.

2. TARGET ISSUES

A "Cintichem-type" target is one that is fabricated using the same dimensions, materials, and processing techniques used previously at Cintichem, Inc. Target fabrication and quality control (QC) inspection are performed at Los Alamos National Laboratory (LANL) in Wing 9 of the Chemistry and Metallurgy Research (CMR) facility.

2.1 Target Description

A scale drawing of the target is shown in Figure 4, and the target internals in Figure 5. The target is made up of three main components: the target tube and coating, the bottom end fitting, and the top end fitting.

The target tube is welded stainless steel (Type 304) tubing, 1.25 in. diameter by 18 in. long, and 30 to 35 mils thickness. The tubing is purchased through a commercial vendor. The internal surface is coated with UO_2 by an electrodeposition process. Before electrodeposition takes place, the internal surface of the tube is chemically etched. This process reduces the thickness of the tube from ~35 mils to 25 to 30 mils, and leaves an oxide residue that allows for a well-adhering coating to be deposited. If the etching process reduces the target tube mass to less than 240 g (25 mils), the target tube is rejected. The coating is distributed uniformly over the target tube surface due to the nature of the electrodeposition coating process. Approximately 0.75 in. on both ends of the tube is not coated, hence the coated length is ~16.5 in. For a 20 g ^{235}U coated target (~24.4 g of UO_2), the coating thickness for a uniformly coated tube would be ~60 μm , assuming that the coating is deposited at full density. Since the coating is not deposited at full density, the thickness is ~120 μm thick. After the electrodeposition process, the target is "baked out" in a furnace with dry nitrogen to 500°C for one hour and then vibrated to determine any coating fall off. Typically, very little fall off occurs. Differences in the target mass before and after the "bake out" step indicate that the coating porosity is ~50%.

The bottom end fitting is a stainless steel plug that incorporates a 0.5 in. diameter void cylinder that acts as a coating catcher. The design allows for the collection of any failed coating material. The fission density is reduced in the catcher region due to a smaller neutron flux and neutron absorption in the stainless steel. The fitting is welded to the tube.

The top end fitting is also stainless steel and incorporates a stainless steel CAJON VCR® type fitting. The VCR® fitting is welded to the end fitting and the end fitting subsequently welded to the tube. The cap incorporates a stainless steel VCR® plug, a VCR® gasket, and a VCR® female nut. The VCR® plug is machined with a pintle that allows for target handling in the ACRR pool using specially designed handling tools.

For a target coated with 18 g of ^{235}U (~22 g UO_2), the minimum mass of the assembled target, including the VCR® components, is 560 g. The mass (weight) of a coated and assembled target submerged in water is 160 g (Wemple, 1996). This allows for a sufficient negative force to maintain the target in the grid during irradiation. The natural convection flow by the target contributes only a few grams to the buoyancy effect (see Section 7.4).

For a target operating at 15 kW for 14 days (0.21 MW-days), the ^{235}U burnup in the target would be ~0.3 g.

Figure 4 Scale target representation.

Figure 5 Target internals.

2.2 Target Quality

Quality control of the targets is primarily the responsibility of LANL. The main issues of concern for irradiation and processing at Sandia National Laboratories (SNL) are as follows: 1) the target meets dimensional and material specifications; 2) the target is leaktight; 3) the coating is well adhered to the surface; 4) the mass loading of the ^{235}U is within specification; and 5) the coating is uniform along the surface of the target. LANL's QC inspection program is designed to ensure that the targets sent to SNL will meet these and other self-imposed requirements. Since each target contains a quantity of highly enriched uranium greater than one gram, the regulations governing the control and transfer of special nuclear materials must also be followed. Each target transferred to SNL maintains a unique serial number that is stamped on the flat of the lower fitting. A data package, which contains the quality information associated with the target, accompanies each target.

Before a target is irradiated in the reactor, a QC inspection is performed at SNL. This inspection is governed by an approved procedure, and is designed to ensure that target transport did not adversely affect the target or target coating integrity. The inspection includes: 1) a visual inspection of the target externals; 2) a visual internal inspection of the coating using a borescope; 3) a pressure test of the target; and 4) a leak check of the target. As part of the leak check inspection, the target is backfilled with 9 psia of helium. As more data is accumulated, and confidence in the target quality is established, some or all of these quality checks may be terminated, or randomly performed on a selected number of targets.

An etched uncoated target has been pressure-tested to rupture in the Force and Pressure Laboratory at SNL. The target was water filled and pressurized at approximately 10 psi per second until it ruptured at 2600 psi (Shields, 1996). The target failed axially along the weld seam of the tubing. Target pressures under normal operating conditions are ~25 psia, and ~100 psia under severe transient conditions (see Sections 6.4 and 7.0)

2.3 Target Receipt, Inspection, Storage, Handling, and Transfer

The targets are shipped from LANL to SNL in Department of Transportation-approved 55 gallon drum containers of ten per drum. After receipt by the SNL Special Nuclear Materials department, the drums are transferred Area V. The targets are surveyed, inventoried and placed in a locked cabinet that has been approved for target storage (Martin and McCrory, 1998). Other than the Special Nuclear Material (SNM) requirements, no other special handling is required for the unirradiated targets. Targets that have passed the SNL QC inspection are stored in a separate locked cabinet.

A target transfer system is being developed which will allow fresh targets to be lowered into the ACRR pool, and irradiated targets raised to the level of the pass-through port. Target handling tools have been fabricated that can attach to targets or the aluminum "void" targets. Currently, fresh targets may be lowered in a basket and placed in the target grid using the "long" target handling tool. Targets used in testing may be kept in the target grid or in storage areas in the ACRR pool. Two storage holsters are available in the ACRR pool for storing up to 37 irradiated or unirradiated targets or fuel elements. If targets must be transferred to the HCF before the target transfer system and cask have been approved for use, the previously approved methods used in the target validation experiment campaign may be employed, with further analysis.

2.4 Target Grid

The target grid and aluminum “void” targets were described in detail in Parma (1998a). Design and QA requirements for the grid are included in the records package associated with the quality plan PEQP-COREMODS (Parma, 1995b). Figure 6 shows the 19 target grid assembly with a 5.3 cm triangular pitch. The grid is fabricated from 6061 aluminum for low neutron absorption. The grid is self-supporting and is indexed in the core using the upper and lower core grid plates, and a bottom pin. A grid is composed of a bottom plate with a pin, a tri-form lower structure, a support plate that indexes on the lower core grid plate, a lower target grid plate that supports and aligns the bottom of the targets, an upper alignment target grid plate that aligns the top portion of the targets, and an upper support plate that aligns the grid to the upper core grid plate. The lower structural portion of the grid with the tri-form shape is welded together. The upper portion that holds the targets is secured to the lower structure using six aluminum tie rods. All of the plates are designed with numerous holes to allow for water flow. The grid is placed in the core using a handling tool that is inserted in the center target location and screwed to the lower structural portion of the grid. The tool extends to the pool surface. The target “nose” sits in a hole in the lower target grid plate. The upper alignment target grid plate captures the upper edge of the target tube. The targets, including the aluminum targets, will be handled using tools that grip the pintle on the end of the target. The same tool is used for both targets and aluminum “void” targets and is designed to work within the confines of the upper support plate to prevent misalignment of the targets. The tool is designed so that it cannot grip a fuel element.

Figure 6 Elevation view of 19 target grid installed in the core.

3. NEUTRONIC ANALYSIS

The computer code MCNP (Briesmeister, 1993) was used to perform the neutronic analysis of the core. A discrete, three-dimensional model of the core was used to investigate reactivity and peaking effects associated with core configuration changes. Calculations were performed to determine temperature and density coefficients, radial peaking among elements, axial peaking, and radial and azimuthal peaking within elements. The same model used to perform the analysis with the cavity liner removed (Parma, 1998a) was used for this report. The model was also modified to incorporate one sixth core symmetry, using specular reflection at the interfaces. This modification did not enhance the run time of the code, but did allow for a simpler geometric representation.

The neutronic model of the core was described in Parma (1998a) for operating the reactor with the cavity liner removed. Also, several core configurations were presented with the calculative results for reactivity changes, feedback coefficients, radial element-to-element fission distributions, axial fission distributions, and radial fission distributions within the elements.

Figure 7 shows the identification of fuel elements, control elements, safety elements and transient rods used for determining peaking factors in the core. Locations "1-c", "1-e", and "2-c" were further sectorized in the radial and azimuthal directions for some of the MCNP calculations so that the peaking within an element could be studied. For these positions, a five-centimeter-long section at the center of the element was divided into six pie shapes and ten radial sections.

Figure 7. Element and sector identification.

3.1 MCNP Results for 7 Targets and 12 Aluminum "Void" Targets

Figures 8 and 9 show the core configuration for a 19 aluminum target grid with 7 targets, 12 aluminum "void" targets, and 177 fuel elements. The MCNP input, and a portion of the output, are included in Appendix A. The aluminum targets are modeled as solid aluminum cylinders, 1.25 inches in diameter and length similar to an actual target. The "Cintichem-type" targets are modeled as closely as possible to the actual target design. The bottom end fitting with the internal catcher is included in the model. The standard target tubing used in the model is 35 mils of stainless steel and the standard coating is modeled as 93% enriched UO_2 with 20 g of ^{235}U . The coating is assumed to be uniform and extends ~16 in. within the tube. The targets are axially centered about the centerline of the fuel and are arranged in a triangular 5.3 cm pitch. The aluminum grids and axial connecting rods are included in the model.

Some of the cases analyzed for this core configuration (7 standard targets with 12 voids) are included in the following list. The transient rods are in the "full out" position unless otherwise specified.

1. 177 fuel elements - 5 rows, 300 K fuel, 300 K moderator, control rods at "full out" position, 7 standard targets - 20 g ^{235}U , 35 mils SS tubing (19G_7T_177_CROUT).
2. Case 1. - control rods "full in" (19G_7T_177_CRIN).
3. 177 fuel elements - 6 rows, 900 K fuel (core only), 300 K moderator, control rods at "full out" position, (19G_7T_177_1_300_900).
4. 177 fuel elements - 6 rows, 900 K fuel (core only), 400 K moderator (core only), 94% moderator density (core only), control rods at "full out" position, (19G_7T_177_0.94_400_900).
5. 177 fuel elements - 6 rows, 300 K fuel, 400 K moderator (cavity only), 94% moderator density (cavity only), control rods at "full out" position, (19G_7T_177_0.94C_400C_300C).
6. 231 fuel elements - 6 rows, 300 K fuel, 300 K moderator, control rods at "full out" position, (19G_7T_231_CROUT).
7. Case 1. - standard targets with tubing thickness of 25 mils (19G_7T_177_25MIL).
8. Case 1. - standard targets with ^{235}U loading of 25 g (19G_7T_177_35MIL_25GM).
9. Case 1. - standard targets with both tubing thickness of 25 mils and ^{235}U loading of 25 g (19G_7T_177_25MIL_25).
10. Case 1. - targets flooded (19G_7TFLOOD_177).
11. Case 1. - targets removed and replaced with water (19G_7TOUT_177).
12. Case 1. - targets have complete coating fall off (19G_7TFLAKED_177).
13. Case 1. - fuel with 10 times the estimated equilibrium ^{135}Xe concentration at full power (19G_7T_177_XEFUEL).
14. Case 1. - targets with 10 times the estimated equilibrium ^{135}Xe concentration at full power (19G_7T_177_XETAR).

The MCNP results for K_{eff} and changes in reactivity are presented in Table 1. For all of the cases presented, the neutron lifetime was calculated to be between 42 to 47 μs . No calculation of the effective delayed neutron fraction (β_{eff}) was performed. It is assumed that for this core configuration, $\beta_{\text{eff}} = 0.0073$.

Figure 8 Plan view of ACRR with 177 fuel elements, 7 targets, and 12 aluminum "void" targets arranged with a triangular pitch of 5.3 cm.

Figure 9 Elevation view of the ACRR with targets.

Table 1 Neutronic results for 7 targets/12 aluminum “void” targets configuration.

#	Case	Keff	Reactivity ρ (\$) ¹	$\Delta\rho = \rho - \rho(\#)$ (\$) ²
1	19G_7T_177_CROUT – 177 Ele Core, 7 Std Targets - 20 g ²³⁵ U, 35 mils SS, CR Out	1.0380 ± 0.0010	+5.01	–
2	19G_7T_177_CRIN - 177 Ele Core, CR In	0.9672 ± 0.0010	-4.65	(1) -9.66
3	19G_7T_177_1_300_900 - 177 Ele Core, 900 K Fuel, 300 K Mod	1.0213 ± 0.0010	+2.86	(1) -2.15
4	19G_7T_177_0.94_400_900 - 177 Ele Core, 900 K Fuel, 400 K Mod Core, 94% Mod Dens	1.0064 ± 0.0010	+0.88	(3) -1.98
5	19G_7T_177_0.94C_400C_300C - 177 Ele Core, 300 K Fuel, 400 K Mod Cavity, 94% Mod Dens Cavity	1.0571 ± 0.0010	+7.39	(1) +2.38
6	19G_7T_231_CROUT – 231 Ele Core, CR Out	1.0992 ± 0.0010	+12.37	(1) +7.36
7	19G_7T_177_25MIL - 25 mils SS Tubing	1.0404 ± 0.0010	+5.32	(1) +0.31
8	19G_7T_177_35MIL_25GM - 25 g ²³⁵ U	1.0452 ± 0.0010	+5.93	(1) +0.92
9	19G_7T_177_25MIL_25GM - 25 mils SS Tubing, 25 g ²³⁵ U	1.0450 ± 0.0010	+5.90	(1) +0.89
10	19G_7TFLOOD_177 – Flooded Targets	1.0360 ± 0.0010	+4.76	(1) -0.25
11	19G_7TOUT_177 – Targets Removed	1.0069 ± 0.0009	+0.94	(1) -4.07
12	19G_7TFLAKED_177 – Coating Fall Off	1.0098 ± 0.0009	+1.33	(1) -3.68
13	19G_7T_177_XEFUEL – 10x ¹³⁵ Xe Fuel	0.9358 ± 0.0009	-9.40	(1) -14.4
14	19G_7T_177_XETAR – 10x ¹³⁵ Xe Targets	1.0106 ± 0.0010	+1.44	(1) -3.57

¹ Uncertainty in reactivity values is less than ± \$0.14

² Uncertainty in change in reactivity values is less than ± \$0.19

The following conclusions are drawn from the MCNP calculations:

- **Excess Reactivity and Target Worth** - For seven targets with 20 g ²³⁵U, 35 mil thickness cladding, and 177 fuel elements, the excess reactivity at 300 K is \$5. Comparing this value to the results in Parma (1998a) for a 177 core and 19 “void” targets, the excess reactivity was \$1.76. The target worth (i.e., replacing a void with a target) is +\$0.46 ± 0.028.
- **Control Rod Bank Worth** - For the 177 fuel element/7 target/12 void core, the control rod bank worth from “Full In” to “Full Out” is \$9.66 ± 0.19. Compare this value to the results in Parma (1998a) for the 177 core with 19 aluminum “void” targets of \$11.52 ± 0.19, and for the 201 core with “voids” of \$10.94 ± 0.19. This reduction in worth of \$1.30 to \$1.90 is likely due to a slight flux tilt toward the center of the core. There is no minimum control rod bank worth.
- **Fuel Element Worth** - For a 231 fuel element/7 target/12 void core, the excess reactivity is \$12.37. This is a difference of \$7.36 ± 0.19 for 54 fuel elements, or \$0.136 ± 0.004 per element in the outer row.

- **Target Tube Thickness Variation Reactivity Effect** - Reducing the target tube thickness from 35 mils to 25 mils, for a 20 g ^{235}U loading, changes reactivity by about $+\$0.04 \pm 0.03$ per target. For a 25 g loaded target, the difference may be smaller.
- **Target ^{235}U Loading Variation Reactivity Effect** - Increasing the target loading from 20 g to 25 g of ^{235}U changes reactivity by $+\$0.13 \pm 0.03$ per target or $+\$0.025 \pm 0.005$ per target per gram.
- **Flooded Target Worth** - Flooding one target changes reactivity by about $-\$0.04 \pm 0.03$.
- **Target Removal Worth** - Removing one target from the grid and replacing it with water changes reactivity by $-\$0.58 \pm 0.03$. Parma (1998a) reported that removing one aluminum "void" target from the grid and replacing it with water changed reactivity by about $-\$0.15$.
- **Flaked Target Worth** - Coating failure (i.e. complete flaking into the catcher region) of one target changes reactivity by about $-\$0.53 \pm 0.03$.
- **Xenon Poisoning Effects in the Core** - The change in reactivity from ^{135}Xe poisoning with the core operating at 2.26 MW (2.35 MW core and targets) is calculated to be $-\$1.44$ for the Xe concentration evenly distributed in the fuel and the ^{135}Xe in equilibrium. Since more of the Xe is created in the peak regions of the core, and the neutron importance is larger in those regions, the actual reactivity effect could be larger by 20% to 30%. Similar calculations on the pulse configuration predicted the effect to be $-\$1.10$ at 2 MW (Parma, 1997b). Actual measurements performed on the pulse configuration (Nelson, et al., 1995) and during the target validation campaign (Bragg, 1996) showed the ^{135}Xe equilibrium to occur at ~ 48 hours during a continuous operation, and the reactivity effect to be $-\$1.50$. No shutdown peak was found. This result is consistent with calculations performed by solving the Xe/I equations using ^{135}Xe absorption results from MCNP. For the core, Xe equilibrium is calculated to occur at ~ 48 hours. After shutdown, the Xe concentration peaks at ~ 2 hours with a value only 1.5% greater than the equilibrium concentration.
- **Xenon Poisoning Effects in the Target** - The change in reactivity from ^{135}Xe poisoning with the seven targets each operating at 14 kW is calculated to be $-\$0.36$ with the ^{135}Xe in equilibrium. For the targets, Xe equilibrium is calculated to occur at ~ 48 hours. After shutdown, the Xe concentration peaks at ~ 7 hours with a value $\sim 50\%$ greater than the equilibrium concentration.
- **Reactivity Effects of Heating the Target Region With Seven Targets Installed** - Heating the complete cavity region to saturated conditions (temperature change of 100°C) would add $\$2.38 \pm 0.19$ worth of reactivity, or $\$0.024/^\circ\text{C}$. In actuality, if exit saturation conditions could occur in the target region, a temperature gradient in the moderator would exist from the bottom to the top of the core. For a 20°C inlet temperature, exit saturation conditions (112°C) would correspond to an average change in the temperature in the target moderator region of $\sim 45^\circ\text{C}$, equivalent to $\sim \$1.08$ worth of reactivity added to the core.
- **Core Temperature Feedback Effects** - The effect of heating the fuel to 900 K and moderator to 400 K, with a density decrease in the water moderator by 5.7%, is $-\$4.13$ compared to $-\$4.49$ found in Parma (1998a). Separate calculations changing fuel temperature, water moderator density, and water moderator temperature indicate that the three effects are separable; Doppler-broadening effects contribute 52%, while density effects in the water moderator (5.7% lower density) and spectral effects in the water moderator (300 K to 400 K) contribute 48% to the overall feedback coefficient. The fuel temperature feedback can be formulated as

$$\Delta\rho_{\text{fuel}} = -\$0.1699 (T(K)^{1/2} - 300^{1/2})$$

and the moderator feedback (including density and spectral effects) as

$$\Delta\rho_{\text{moderator}} = -\$0.3483 \Delta\text{density}(\%)$$

where $\Delta\rho$ is the change in reactivity in dollars, $T(K)$ is the average fuel temperature in degrees Kelvin, and $\Delta\text{density}(\%)$ is the percent change in water density. These formulations are close to the results for the pulse configuration.

No MCNP calculations have been performed for the seven targets in other geometric configurations. MCNP calculations have been performed but are not presented for missing aluminum “void” targets around the seven target arrangement. Having the voids in the grid is required to maintain the maximum coupling factor between the targets and the core. A pitch of 5.3 cm was found to give the optimized coupling between the targets and the core.

No MCNP calculations were performed for less than seven targets in the grid. However, Parma (1998a) did present the 19 target configuration filled completely with the aluminum “void” targets. It is assumed that the incremental changes for these loadings are bounded by the analyses presented in this report and Parma (1998a).

Other calculations were performed to determine the axial fission density in the targets and fuel, the radial fission density within the fuel, the fission coupling in a target where the coating has failed and has fallen into the catcher, and the fission product poisoning absorption rate.

3.2 Fission Densities and Peaking Factors for the Fuel Elements

Element-to-element fission ratios are shown in Table 2 for the 177 element core with 7 standard targets and 12 aluminum “void” targets. Peak element to average element values of 1.71 and 1.94 are calculated for the control rods “full out” and “full in”, respectively. The peak location is in the center of the first row. Comparing the results to Parma (1998a), shown in Table 3 for the 19 void, 201 element core, the peak to average values are 1.59 and 1.79 for the control rods “full out” and “full in”, respectively. The 7 target, 177 element core radial element peaking factors are ~8% larger.

Element-to-element fission ratios were calculated for a 7 target, 12 void, 231 element core configuration. For the control rods “full out,” the peaking occurs at the same location with a value of 1.76, within 3% of the peak value for the 177 element core.

Calculations varying the target loading (20 g to 25 g ^{235}U) and wall thickness (35 mils to 25 mils) resulted in peaking factors within 2% of the 7 standard target, 177 element core configuration.

Table 2 Element average to core average fission ratios for the ACRR core configuration with 7 target, 12 aluminum “void” targets, and 177 fuel elements.

7 Targets, 12 Aluminum “Void” Targets and 177 Fuel Elements ²			
Element Position ¹	# of Elmts	Control Rods Out	Control Rods In
ACRR Run		19G_7T_177_CROUT	19G_7T_177_CRIN
12 (1-c)	3	1.59	1.86
11	6	1.66	1.87
15 (1-e)	6	1.33	1.47
14	6	1.70	1.94
13	3	1.71	1.94
Row 1 Avg	24	1.59	1.79
22 (2-c)	6	1.05	1.19
21	6	1.00	1.06
25	6	0.98	0.90
24	6	1.01	1.07
23	6	1.03	1.14
Row 2 Avg	30	1.01	1.07
TR	0	Void	Void
32	6	0.99	1.05
31	6	0.89	0.82
CR	6	0.84	0.02
34	6	0.91	0.82
33	6	0.94	0.97
SR	3	0.93	1.00
Row 3 Avg	33	0.91	0.76
43	6	0.90	0.96
42	6	0.68	0.54
44	6	0.82	0.87
41	24	0.78	0.71
Row 4 Avg	42	0.79	0.75
Row 5 Avg	48	0.94	0.94

¹ Fuel element position identification numbers are identified in Figure 7.

² Fission ratio uncertainties are ~1.0% for the individual elements and ~0.4% for row averages. Values were rounded to two decimal places.

Table 3 Element average to core average fission ratios for the ACRR core configuration with 0 targets, 19 aluminum “void” targets, and 201 fuel elements.

		0 Targets, 19 Aluminum “Void” Targets and 201 Fuel Elements ²	
Element Position ¹	# of Elmts	Control Rods Out	Control Rods In
ACRR Run		201_CROUT	201_CRIN
12 (1-c)	3	1.57	1.72
11	6	1.58	1.72
15 (1-e)	6	1.27	1.35
14	6	1.58	1.74
13	3	1.59	1.76
Row 1 Avg	24	1.50	1.64
22 (2-c)	6	1.09	1.18
21	6	1.01	1.02
25	6	1.00	0.90
24	6	1.01	1.04
23	6	1.04	1.11
Row 2 Avg	30	1.03	1.05
TR	0	Void	Void
32	6	1.04	1.09
31	6	0.94	0.84
CR	6	0.90	0.02
34	6	0.96	0.87
33	6	0.98	1.03
SR	3	0.99	1.10
Row 3 Avg	33	0.97	0.80
43	6	0.96	1.05
42	6	0.76	0.62
44	6	0.90	0.96
41	24	0.85	0.79
Row 4 Avg	42	0.86	0.83
Row 5 Avg	48	0.89	0.90
Row 6 Avg	38	0.98	1.06

¹ Fuel element position identification numbers are identified in Figure 7.

² Fission ratio uncertainties are ~1.5% for the individual elements and ~0.7% for row averages. Values were rounded to two decimal places.

The axial peaking factor within an element (F_{ze}) and the radial peaking factor within the fuel element at the axial centerline (F_{re}) were not calculated for the 7 standard target, 12 void target configuration. Instead, calculations were performed for the 19 target grid filled with 19 standard targets. This result and the results found in Parma (1998a) are assumed to bound the possible variations due to differences in the target configurations.

Figure 10 presents the calculated results for the radial fission density for a 19 standard target, 147 element core with the control rods "full out" (ID# ACRRDCINT43APIE). The results are shown for the six pie sections within fuel elements identified as 1-c, 1-e, and 2-c in Figure 7. The peak/avg (radial) values printed at the top of each graph are different from the values found on Table 2 since this calculation includes 19 targets. The peak/avg (axial) values are the calculated values from the tally information collected in the five-centimeter-long center section of the fuel. The results are nearly identical to those presented in Figure 11 for the 19 void, 201 element configuration presented in Parma (1998a). For element 1-c the fission density peaks at the surface with a peak to average value of ~ 1.75 . The conclusion from these results is that targets do not significantly change the radial peaking distributions within an element.

Notice that the peak to average axial value for the 1-c element is 1.37 ± 0.04 . This value is about three sigma (approximately 9%) greater than the value of 1.25 used in Parma (1998a) to determine fuel temperature at the peak location in the core. The effect of a higher axial peaking factor will be discussed in Section 4 of this report.

Figure 10 Radial Fission Distribution for ACRRCINT43APIE, 19 Targets, 147 Element Core, CR Out.

Left - 1-c; Middle - 1-e; Right - 2-c

Figure 11 Radial Fission Distribution for 201_CROUT, 19 Aluminum "Void" Targets, 201 Element Core, CR Out.
Left - 1-c; Middle - 1-e; Right - 2-c

3.3 Target Coupling Factors

The coupling factors for the target to the peak fuel element are shown in Table 4 for the 7 target, 12 void configuration with variations in core size, target loading, and target tube thickness. Some variation in the coupling does occur. However, all of the results are within 8% of an average value of $0.0312 W_{\text{avg tar}}/W_{\text{peak fuel}} - g^{235}\text{U}_{\text{tar}}$. This table may be used to determine the average target power and the total core power for a given configuration. For instance, for a peak element power limit of 21.74 kW, the average target power for the 7 standard target configuration would be 13.71 kW. The average fuel element power would be 12.71 kW. For a 177 fuel element core, the total core power would be 2.25 MW, the total target power would be 96 kW (0.096 MW), and the total core and target power would be 2.3°C 5 MW. For a 201 fuel element core, the core power would be an additional 24 times 12.71 kW (0.30 MW), for a total core and target power of 2.65 MW. The current 2 MW heat rejection system may effectively limit the steady-state power level to a value lower than that limited by the peak fuel element.

Table 4 Target coupling factors for the 7 target, 12 aluminum "void" target configuration.

Configuration		Target Avg/ Peak FE ($W_{\text{tar}}/W_{\text{peak fuel}} - g^{235}\text{U}_{\text{tar}}$)	Peak FE/ Avg FE (F_{rc})	Target Avg/ Avg FE ($W_{\text{tar}}/W_{\text{avg fuel}} - g^{235}\text{U}_{\text{tar}}$)	Target Avg Power (kW) at Peak FE Power of 21.74 kW	Total Reactor and Target Power (MW)
1	20 g ^{235}U , 35 mil targets 177 fuel element core	0.0316	1.71	0.0540	13.7	2.35
2	20 g ^{235}U , 25 mil targets 177 fuel element core	0.0338	1.72	0.0581	14.7	2.34
3	25 g ^{235}U , 35 mil targets 177 fuel element core	0.0291	1.72	0.0501	15.8	2.35
4	25 g ^{235}U , 25 mil targets 177 fuel element core	0.0304	1.74	0.0529	16.5	2.33
5	20 g ^{235}U , 35 mil targets 231 fuel element core	0.0309	1.76	0.0544	13.4	2.95

Uncertainty in the coupling factors is $<\pm 1\%$.

It is assumed that for a single target and 18 voids, the peaking and coupling factors would be similar to those presented.

Additional analysis comparing the results of individual target performance to the average target reveals that the outer six targets operate at a few percent over the average, and the center at ~15% below the average. Axial peaking within a target has been calculated to be about 1.20.

3.4 Summary of Neutronic Results

Replacing "void" targets with "Cintichem-type" targets increases the excess reactivity of the core, slightly increases the fuel element peaking factor of the core (F_{rc}), and may increase the axial peaking factor of an element (F_{ze}). The radial peaking factor and radial/azimuthal fission density profiles within the fuel elements (F_{re}) are unchanged. For the core sizes analyzed in this document, the control rod bank worth decreased slightly.

The excess reactivity associated with different core configurations has no added safety implications in operating the reactor other than those already analyzed in the ACRR SAR. It is estimated that \$6 to \$7 of excess reactivity is required to reach the maximum core power level (constrained by the peak fuel element power of 21.74 kW) with equilibrium ^{135}Xe concentration. For a 177 fuel element, 7 standard target, 12 aluminum "void" target core configuration at room temperature and unpoisoned, the excess reactivity is calculated to be ~\$5. With the same target configuration, a 201 element core is calculated to have ~\$8.3 of excess reactivity. For this configuration, with the peak element operating at 21.74 kW, the average target power would be 13.7 kW, and the total core and target power would be 2.65 MW.

The temperature feedback of the core is calculated to be close to that calculated in Parma (1998a) for a similar configuration with 19 aluminum "void" targets. Consistent with Parma (1998a), the density and spectral coefficient in the cavity region is calculated to be positive. If the cavity coolant were to reach exit saturation conditions, ~\$1.08 of reactivity could be added to the system if the inlet temperature was 20°C. However, under steady-state conditions, with the core operating at the limit imposed on the peak fuel element, the cavity coolant temperature is significantly lower than the saturation temperature. The analysis in the following section shows that for a peak fuel element power of 21.74 kW (2.65 MW total core and target power), and 13.7 kW in the targets, the exit coolant temperature of the target region is ~80°C for a 40°C inlet temperature. From an initial coolant temperature of 20°C, the reactivity added due to cavity coolant heating would be ~\$0.96. This result assumes that the twelve unheated channels would also maintain an 80°C water temperature at the exit, which is a very conservative assumption. The additional reactivity is smaller than the overall negative temperature feedback effect that is calculated to be ~\$4.55 for a steady-state power level of 2.65 MW. Hence a significant negative reactivity balance is maintained in the power operating range.

For the 201 element, 19 aluminum "void" target core presented in Parma (1998a), the peaking occurred in the first row at the center location; $F_{rc}=1.59$, $F_{ze}=1.28 \Rightarrow F_{rc}F_{ze}=2.04$. For the 177 element, 7 target, 12 aluminum "void" core, the peaking occurs in the same location: $F_{rc}=1.71$, $F_{ze}=1.37 \Rightarrow F_{rc}F_{ze}=2.34$.

The TSRs (3.2.9 Limitation on Experiments) require that for all operating modes, the reactivity worth of an experiment that is not secured shall be more positive than -\$1.00. Since adding a target adds ~\$0.60 of reactivity, and a void ~\$0.15 of reactivity, this requirement is fulfilled.

4. THERMAL ANALYSIS

The Safety Limit (SL) is identified in the TSRs as follows: "The maximum equilibrium reactor power level shall not result in any element power greater than 32 kW per fuel element." Limiting Control Settings (LCS) are identified for the steady state mode as follows: "The PPS Safety Channel SCRAM settings shall not exceed the maximum allowable settings in the applicable modes of operation; Fuel Temperature - 1400°C, Percent Power - 115% (calculated equivalent of 25 kW for peak fuel element)." The Plant Protection System (PPS) will activate, causing a SCRAM when the temperature in the instrumented fuel element at the peak location in the core is at or below 1400°C, or when the neutron detectors measure an equivalent neutron flux in the core that indicates that the power level is at 115% of full power (i.e., the peak element reaches 25 kW). Nominal operating power for the peak element is $25 \text{ kW}/1.15 = 21.74 \text{ kW}$.

The basis for the limitations on power and temperature is to ensure that the peak fuel locations will maintain a temperature well below the physical limitations of the fuel, and that a sufficient safety margin exists between the trip point and the safety limit. For $\text{UO}_2\text{-BeO}$, the $\alpha\text{-}\beta$ solid phase transition occurs at $\sim 2100^\circ\text{C}$ and the eutectic melt at 2150°C , from Table 14.2-2 of ACRR SAR.

Limitations on target power and temperature are not discussed in the TSRs. It will be shown in this analysis that the limitations imposed on the fuel and fuel elements represent the limiting conditions at steady-state power, and therefore no additional TSRs are needed to ensure safe operation of the reactor.

4.1 Steady-State Heat Transfer Analysis - Fuel Elements

A steady-state heat transfer analysis for the fuel elements was performed in Parma (1998a) for operating with only aluminum "void" targets. The results from that analysis still apply, since the only factors that could affect the outcome are the radial and azimuthal fission densities within the element. These fission densities were shown in Section 3 to be virtually the same as the results presented in Parma (1998a).

With the addition of targets to the core, the fuel element peak location will be experimentally determined by a core temperature mapping procedure, and the power trip adjusted to meet the requirements imposed by the TSRs on peak element power and temperature. Hence, the element-to-element peaking factors do not affect the element temperature-to-power formulation derived in Parma (1998a):

$$T(^{\circ}\text{C}) = 187.91 P_{ele}(\text{kW})^{0.636}$$

or

$$P_{ele}(\text{kW}) = 2.6858 \times 10^{-4} T(^{\circ}\text{C})^{1.572}$$

where P_{ele} is the total element power in kilowatts, and T is the inner pellet temperature in degrees Celsius at the axial center line of the fuel element.

The axial peaking factor within the peak element does, however, affect the formulation, since one of the assumptions in the analysis was that the axial peaking factor was 1.25. If the axial peaking factor is greater than this value, which is the case for a 19 target core with a value of 1.37, then using the formulation will allow for a more conservative setting of the power trip. That is, if the axial profile is steeper than originally predicted, a given temperature will be reached at the axial centerline at a lower power level than predicted.

Table 5 presents the results from Parma (1998a) for an element power of 32, 25, and 21.74 kW representing the safety limit, limiting control setting, and operating value, respectively. The axial peaking factor for the element was $F_{ze}=1.25$. The average temperature for the inner radial section of fuel is expected to correspond to the instrumented fuel element measured temperature.

Table 5 Peak and average temperatures for the inner and outer fuel pieces for a fuel element axial peaking factor equal to $F_{ze}=1.25$.

Element Power (kW)	Peak Fuel Temp (°C)	Avg Temp for Inner Fuel (°C)	Avg Temp for Outer Fuel (°C)
32	1825	1730	1469
25	1538	1469	1267
21.74	1390	1336	1163

The SL of 32 kW provides an adequate margin of safety such that if it is reached, the phase transition temperature is not exceeded. At 32 kW, there remains a margin of $\sim 275^{\circ}\text{C}$ below the phase transition temperature.

4.2 Steady-State Heat Transfer Analysis - Targets

Cintichem, Inc. experimentally measured the internal peak coating temperature in situ and found that for a 13 kW target, the peak internal temperature was 345°C . From this result, the temperature differential across the coating was calculated to be 229°C . The mass of the target coating was not reported (Franzen, 1985).

The target has a very simple geometry with heating in only one region, and an analytical solution to the steady-state heat equation can be readily found if it is assumed that the conductivity does not have a significant temperature dependence. However, in order to properly include the temperature-dependent nature of the conductivity, the steady-state heat transfer code CYLSS (CYLindrical Steady-State heat transfer) (Parma, 1998b) was used to determine the peak clad and coating temperatures. The code is multi-region and one- or two-dimensional in R and Z or R and Phi (azimuthal direction). The code was run in the R direction for the analysis with an insulated boundary condition on the internal coating surface of the fuel. The clad boundary was fixed at 120°C , which represents a typical wall temperature under subcooled nucleate boiling conditions. It was assumed that all of the energy from fission was deposited in the coating. Results from prior work (Parma, 1988) (Naegeli, 1998a) indicate that for the average fission fragment in UO_2 , 3% of the fragments would be emitted from the coating into the gas, depositing $<1\%$ of the total energy into the gas.

Since the actual coating thickness and conductivity is unknown, several calculations were performed in order to determine the range of variability in the results. The target power was arbitrarily chosen as 20 kW with a peaking factor of 1.2. Coating thickness was allowed to vary from 80 to 160 μm representing full and 50% density UO_2 . The cladding thickness was 30 mils (750 μm). Several coating conductivities were examined including: 1) a particle bed made up of UO_2 and He gas with a particle size of 1 μm , porosity of 50%, and emissivity of 0.5; 2) UO_2 at full density; 3) UO_2 pressed powder (100 psi) in air; and 4) U_3O_8 pressed powder (100 psi) in air. The results for the peak coating and clad temperatures are presented in Table 6.

Table 6 Peak coating and clad temperatures for a target at 20 kW, axial peaking factor equal to $F_{ze}=1.2$, and $T_{wall} = 120^\circ\text{C}$.

Condition	Peak Coating Temp ($^\circ\text{C}$)	Avg Coating Temp ($^\circ\text{C}$)	Peak Clad Temp ($^\circ\text{C}$)
UO_2 Particle Bed - 80 μm	184	173	150
UO_2 Particle Bed - 160 μm	216	195	150
UO_2 Full Density - 80 μm	155	153	150
UO_2 Full Density - 160 μm	158	156	150
UO_2 Pressed Powder - 80 μm	381	305	150
UO_2 Pressed Powder - 160 μm	611	458	150
U_3O_8 Pressed Powder - 80 μm	266	230	150
U_3O_8 Pressed Powder - 160 μm	365	298	150

These calculated results are found to be consistent with the analytical solution in rectangular coordinates. The peak coating temperature occurs at the internal surface with the solution

$$T_{coating}(0) = \dot{q}''' a \left[\frac{a}{2k_{coating}} + \frac{b-a}{k_{clad}} \right] + T_{wall}$$

where $\dot{q}'''$ is the volumetric heat rate, k is the conductivity, T_{wall} is the temperature at the clad surface boundary (120°C), a is the coating thickness, and $b-a$ is the cladding thickness.

For the cladding, the peak occurs at the cladding/coating interface with the solution

$$T_{clad}(a) = \dot{q}''' a \left[\frac{b-a}{k_{clad}} \right] + T_{wall}$$

Table 6 and the analytical solution show that the peak cladding temperature is dependent only on the wall temperature, the heat flux, the cladding thickness and the cladding conductivity. For a

30 mil cladding thickness, and a 20 kW target, the peak clad temperature will not exceed 150°C as long as the critical heat flux, analyzed in the next chapter, is not exceeded.

The peak coating temperature is higher than the peak clad temperature by the heat flux times half of the coating thickness, divided by the coating conductivity. Doubling the coating thickness, while keeping the heat flux and conductivity constant, will double the peak coating temperature minus the peak clad temperature.

The experimental results from Cintichem, Inc., can be used with the above formulation to estimate the peak temperature in a 20 kW target. Using the experimental result of 229°C coating temperature differential at 13 kW, and assuming that the coating thickness is the same, the differential temperature in a 20 kW target coating would be 352°C. Assuming the clad temperature is 150°C, the peak coating temperature would be 502°C.

The particle bed formulation for the conductivity was expected to represent the most valid representation of the coating conductivity, since the electrodeposition process does not produce a full density coating and helium is maintained in the matrix. The UO₂ pressed powder in air was expected to represent the most conservative values. However, in comparing the analytical results to the Cintichem experimental values, the pressed-powder conductivity seems to be more appropriate. At 20 kW, the peak coating temperature for a 160 μm thick coating is calculated to be 216°C and 611°C for the particle bed and UO₂ powder, respectively. Since the melting point of UO₂ is 2500°C, there exists at least a factor of four from the melt point.

5. CRITICAL HEAT FLUX ANALYSIS

5.1 *Steady-State Critical Heat Flux Analysis - Fuel Elements*

The ACRR SAR and TSRs discuss the critical heat flux (CHF) issues associated with steady-state operation of the reactor. Section A2.1 of the TSRs plot the data presented in Table 4.4-2 of the ACRR SAR for the minimum critical heat flux ratio (MCHFR) as a function of inlet temperature, using the correlation developed at the University of New Mexico (Rao and El Genk, 1994). The correlation was developed for the ACRR using similar geometry, inlet temperatures, pressures, and flow conditions maintained in the core. Previous correlations significantly under-predicted the value of the MCHFR. The TSR plot shows that a MCHFR greater than two can be maintained for a peak element power of 25 kW and core coolant inlet temperatures less than 50°C.

For the core configurations proposed, no change to the MCHFR analysis is required, since the fuel and grid configuration remain the same and the natural convection cooling properties of the core are unaltered. With the central cavity liner removed, a larger flow area exists on one side of the fuel element with the 19 aluminum "void" targets installed. This geometry change will provide for greater cooling capability for the first row of fuel elements. For these elements, the MCHFR predictions are conservative.

The computer code MASSFLR (Parma, 1996) was used to determine the coolant flow rate and exit temperature as a function of the ACRR fuel element power and coolant inlet temperature under steady-state conditions. The code numerically solves the coupled continuity, momentum, and energy equations as applied to a one-dimensional, incompressible, subcooled, vertical flow channel. The flow is driven by natural circulation caused by the buoyant force associated with the decrease in the water density in the flow channel as the water is heated. The code was validated using measured data taken for the ACRR in 1983. The results were in agreement with the measured data for a 20°C inlet temperature at reactor powers of 2 and 4 MW. For higher inlet temperatures, the calculated results were a few degrees higher (more conservative) as compared to the measured values.

For a 50°C inlet temperature, the coolant channel exit temperature is calculated to reach saturation (112°C) at a fuel element power of ~24 kW. The flow rate in the channel under these conditions is ~0.09 kg/s (5.4 l/min or 1.4 gal/min). For lower inlet coolant temperatures, exit saturation would be reached at a higher power level. For a 20°C coolant inlet temperature, the element power required to reach saturation temperature is calculated to be ~38 kW and the flow rate is ~0.10 kg/s.

5.2 *Steady-State Critical Heat Flux Analysis - Targets*

The Cintichem Nuclear Reactor operated at a power of 5 MW in an open pool, was light water moderated, forced convection cooled, and used 93% enriched plate type fuel (Burn, 1988). Single targets, or bundles of four, were loaded into a fuel assembly location. Due to constraints on the flow rate, the tight pitch arrangement of the targets, and the "no-boiling" criteria of the reactor, a target power of 13 kW was established as the limit (Cintichem).

For the ACRR, the thermal hydraulic issues for the fuel elements and targets are significantly different than for the Cintichem Reactor. The ACRR fuel elements are cooled by natural convection with subcooled nucleate boiling at element power levels above ~5 kW. As described in Section 5.1, the exit coolant temperature is calculated to reach saturation for a fuel element at ~24 kW for a 50°C coolant inlet temperature, and ~38 kW for a 20°C inlet temperature.

The targets are also cooled by natural convection, with subcooled nucleate boiling acting as the primary heat removal mechanism from the target surface. The targets are spaced on a very open pitch of 5.3 cm for the 19 target grid, allowing for a 2.1 cm gap between the targets, compared to the fuel elements, which are on a relatively tight pitch with a 0.42 cm gap. The larger pitch allows for a higher flow rate and lower temperature rise in the target coolant channel, as compared to the fuel elements. This phenomenon occurs since the friction and acceleration pressure-drop terms are inversely dependent on the square of the flow channel area, allowing for lower pressure drops and higher flow rates for larger flow channels. However, for the target channels, the minor losses (entrance and exit) dominate the pressure drop due to the grid flow holes, and the full effect of the larger flow channels is not achieved. If a target was operated at 24 kW with a coolant inlet temperature of 50°C, the exit temperature would be ~104°C with a channel flow rate of ~0.11 kg/s. For 38 kW and 20°C inlet temperature, the exit temperature would be ~101°C with a channel flow rate of ~0.11 kg/s. The larger flow area for the targets ensures that the target exit flow conditions will always operate at a greater margin below the exit saturation conditions than for the peak and average fuel element.

The target power for a given peak element power can be found using the coupling factors from Table 4. For a 25 g ^{235}U target with a 25 mil thick cladding, the coupling factor is $0.0304 W_{\text{target}}/W_{\text{peak fuel}} - g^{235}\text{U}$ or $0.76 W_{\text{target}}/W_{\text{peak fuel}}$. At a peak fuel element power of 25 kW, the target power would be 19 kW. For a 50°C inlet coolant temperature, the exit temperature for the peak fuel element channel is calculated to be above the saturation temperature of 112°C; the target channel exit temperature is calculated to be ~97°C and the channel flow rate ~0.10 kg/s.

The CHF conditions as a function of inlet temperature can be calculated for the targets using the same methodology as for the fuel elements, using the CHF correlation developed at the University of New Mexico (Rao and El Genk, 1994). Figure 12 shows the target power required to reach the saturation conditions for the coolant channel exit, and CHF conditions. Both the total power method and local conditions method, described in Rao and El Genk, 1994, are plotted using the correlation. The figure also shows the target power required to reach a coating melt temperature of 2500°C, using the conservative conductivity for UO_2 pressed powder and 160 μm coating thickness. For a 50°C inlet coolant temperature, a target power of 55 kW is required to reach CHF conditions, or a MCHFR of 2.9 compared to the expected maximum steady-state target power of 19 kW.

This analysis shows that for the seven targets in the 19 target grid configuration with ^{235}U loadings of up to 25 g, the target fuel temperature, heat flux, and exit channel temperature maintain significant margins of safety. Under steady-state conditions, the target conditions are found to lag significantly behind those of the peak fuel element. Hence, there are no additional requirements to be imposed on the targets, with respect to the TSRs, under steady-state operating conditions.

Figure 12 Target power required to reach coolant saturation at channel exit conditions and CHF conditions.

6. STRESS ANALYSIS

6.1 Steady-State Stress Analysis of the ACRR Fuel

Steady-state stresses for the ACRR fuel were examined in Parma (1998a) for operating with only aluminum "void" targets. The results from that analysis still apply, since the only factors that could affect the outcome are the radial and azimuthal fission densities within the element. These fission densities were shown in the previous section to be virtually the same as for Parma (1998a). The result of the analysis was that, for a fuel element at 25 kW, the maximum hoop stress occurs at the inner pellet surface and is about a factor of six below the fracture stress.

6.2 Steady-State Stress Analysis of the ACRR Fuel Element Clad

The ACRR SAR uses the formulation for a thin walled pressure vessel to determine the hoop stress in the cladding. At 20°C the 0.2% offset yield is 33,000 psi; at 120°C the yield is 26,400 psi. This corresponds to a rupture pressure of 920 psi and 743 psi, respectively, for the two temperatures. At 25 kW, the average temperature of the gas is ~800°C. Using an initial gas pressure loading of 2 atm of helium, the element pressure at 25 kW would be 7.3 atm or ~110 psi, a factor of six less than the values required to reach the 0.2% offset yield. Removal of the central cavity will not affect this result. These calculations do not include future gas generation due to fuel burnup.

6.3 Steady-State Stress Analysis of the Target Coating

For UO_2 , the ultimate tensile strength (UTS) (tension) is ~18 ksi and the fracture strength (FS) (compression) is ~110 ksi. The Young's Modulus (E) is strongly dependent on the porosity of the material (Olander, 1976). At 50% porosity, E is 4% of its full density value of 32,400 ksi.

As described in Section 2.1, one of the steps in the target fabrication process is a "bake out" process at 500°C to remove water from the coating. If the coating was at full density, the stresses in the coating would be ~125 ksi tension, exceeding the UTS by a factor of seven. However, as described earlier, the target coating density is not at full density but is calculated, from moisture loss during "bake out", to be ~50%. For this porosity, the stress in the coating is significantly reduced to ~4 ksi tension, 4.5 times less than the UTS.

During irradiation of a 25 gram ^{235}U coated target, the target power would be ~19 kW for a peak element power of 25 kW (from Section 5.2). Assuming that the clad temperature at the coolant interface is 120°C, the temperature at the coating/clad interface would be ~150°C. The peak coating temperature, using the conductivity for UO_2 pressed powder, would be 371°C for a porosity of 0% (80 μm), and 591°C for a porosity of 50% (160 μm). For the full density, 0% porosity case, the stress at the coating/clad interface would be ~37 ksi tension, and at the coating surface, ~85 ksi compression. The compression is within the FS for UO_2 . For the more realistic 50% porosity case, the stress at the coating/clad interface would be ~1 ksi tension, and at the coating surface, ~6 ksi compression. These values are well below the UTS and FS for UO_2 .

The effects of a flaked coating are presented in the accident analysis section of this report.

6.4 Steady-State Stress Analysis of the Target Clad

The formulation for a thin walled pressure vessel can be used to determine the hoop stress in the target cladding. At 20°C the 0.2% offset yield is 33,000 psi; at 120°C the yield is 26,400 psi. For a cladding thickness of 25 mils, the 0.2% offset yield corresponds to internal target pressures of 1320 psi and 1056 psi, respectively, for the two temperatures. At 19 kW, the average temperature of the gas would be 518°C, using the UO₂ pressed-powder conductivity and a porosity of 50%. Using an initial gas pressure loading of 9 psia (0.6 atm) of helium, the target pressure would be ~24 psia (1.6 atm), a factor of 44 times less than the values required to reach the 0.2% offset yield. As described in Section 2.2, the target has been tested to failure, with a rupture occurring at 2600 psi along the weld seam of the tube.

7. ACCIDENT ANALYSIS WITH COMPARISONS TO CURRENT SAR

The accident analysis presented in Parma (1998a), for operating with aluminum "void" targets in the target grid, remains valid with the addition of targets to the grid. However, with the addition of targets to the target grid, other accident scenarios become plausible which must be evaluated to determine their safety significance. These scenarios were determined by performing a hazard analysis for target irradiation and target handling and storage at the ACRR Facility. This analysis is presented in Appendix B. No events are identified that pose a high or moderate risk to the public. Several events that could release fission products from the targets are identified, but these pose only a minor, or very minor, risk to the public. A subset of these events was chosen for further analysis in order to define the envelope of accident conditions to which the operation could be subjected. These events are presented in Table 7. Events associated with irradiated target transfer out of the ACRR pool and transfer to the Hot Cell Facility were not considered in this analysis.

Table 7 Accident scenarios with targets.

Event	Description
1. Mishandled/Dropped Target (HS-FR-3)	An irradiated target is mishandled or dropped in the ACRR pool resulting in a breach of cladding and fission product release.
2. Misloaded Target (TI-FR-5)	A target is inserted into a target grid location which has a greater than expected neutronic coupling to the core, or a target or group of targets are loaded into the target grid with a higher than expected U-235 loading, resulting in target(s) operating at higher than expected powers, leading to target overheating, clad failure, and fission product release.
3. Dislodged or Removed Target at Power (TI-FR-9)	A target is dislodged from the target grid or is inadvertently removed from the grid while the reactor is at power, causing a possible reactivity anomaly.
4. Target Misalignment During Loading (TI-FR-4)	A target is not positioned properly in the target grid and does not completely seat in the lower target grid plate, resulting in it touching a fuel element or other target and overheating the cladding at that location.
5. Target Coating Failure at Power (TI-FR-8)	A partial or complete uranium oxide coating failure occurs in a target while the reactor is at full power resulting in a hot spot and cladding failure.
6. Leaking Target (TI-FR-6, HS-FR-1)	A target develops a leak due to unsealing of the cap, corrosion, or damage, resulting in a release of fission products.
7. Flooded Target (TI-FR-7)	A target develops a leak due to unsealing of the cap, corrosion, or damage, resulting in water ingress into a target, which leads to possible target pressurization, clad rupture, and fission product release.
8. Uncontrolled Control Rod Bank Withdrawal (TI-FR-3)	The control rod bank is inadvertently continuously withdrawn, resulting in a power increase and possible overheating of the targets and/or the fuel elements if unmitigated by a scram.
9. Control Rod Bank Overspeed (TI-FR-13)	The clock pulse generator board fails, increasing the clock speed to levels that allow for greater than 20 cents per second reactivity addition, which results in a power increase and possible overheating of the targets and/or the fuel elements if unmitigated by a scram.
10. Inadvertent Pulse or Limiting Event Pulse (TI-FR-14)	The pulse or transient rod withdrawal mode is inadvertently entered and a pulse, transient rod fast withdrawal, or limiting event pulse is performed, resulting in target coating vaporization and cladding rupture, possible fuel element cladding rupture, and a release of fission products.

In general, for steady-state research-type reactors, credible accident scenarios, in which a significant quantity of fission products could potentially be released, are limited to uncontrolled reactivity excursions. The only events, in a steady-state mode, that could add enough reactivity to cause significant fuel and/or target damage are an uncontrolled control rod bank withdrawal incident without SCRAM or a control rod bank overspeed incident.

Other accident scenarios associated with the reactor, listed in Table 14.1-1 of the ACRR SAR, are presented to show that they are not affected by the addition of targets to the grid.

7.1 Target Radionuclide Inventory and Release

The target radionuclide fission product inventory can be calculated using a computer code such as ORIGEN2 (Croff, 1983) or by direct calculation using the cumulative yields (Meek and Rider, 1974). For the analyses presented in this report, the radionuclide inventory was calculated using the cumulative yields in an EXCEL spreadsheet. This allowed for a smaller group of isotopes to be examined quickly without having to re-run and post-process ORIGEN2 for every permutation of power and operating history. The results were compared to ORIGEN2 to ensure agreement between the results.

For releases from the target to the environment, the isotopes of concern include the noble gases (Kr and Xe) and the halogens (Br and I). These chemical groups have the greatest potential for escaping to the atmosphere since they are either non-reactive gases or have relatively high vapor pressures, compared to the other chemical groups. The halogens, however, are very chemically reactive, and will readily form compounds. All of the other chemical groups have very low vapor pressures at room temperature and would not be expected to escape from the pool. For a rupture-type incident in the pool, Restrepo, 1992 suggests using pool decontamination factors of 1 for the noble gases, 100 for the halogens, and 1400 for all other groups. These decontamination factors were used in determining the doses associated with the limiting event presented in the ACRR SAR.

Table 8 shows the calculated noble and halogen radioisotope target inventory using the cumulative yields from Meek and Rider, 1974, and a target power of 20 kW, for a 7 day, 14 day, and infinite irradiation (i.e. saturated activity). The isotopes that are used in the MACCS dose code (Chanin and Young, 1997) are identified in the MACCS column.

Note that after a seven day irradiation, all but Kr-85, I-131, Xe-131m, I-132, and Xe-133 are at their saturated activity values. The difference from the seven day results and the saturated values for these isotopes makes up only ~10% of the total activity.

Results from prior work (Parma, 1988) (Naegeli, 1998a) indicate that for an average fission fragment created in UO₂, 3% of the fragments are emitted from the coating into the gas. However, the range of a fission fragment in 9 psia He is ~22 cm. Therefore, a significant fraction of the fragments are implanted back into the coating on the opposite side of the target. It is expected that less than 3% of the nobles and halogens would be in the gas and available for release under normal operating conditions and after removal of the target from the reactor. Results from tests during the target processing verification campaign showed that only ~1% of the Kr and Xe, and $5 \times 10^{-7}\%$ of the iodine atoms were available for release (Parma and Naegeli, 1997). The small iodine fraction was due to plate-out on the coating surface.

Unless otherwise specified, the results that will be presented in this analysis assume that only 3% of the nobles and halogens are available for release from a target. All of the other chemical groups are assumed to be trapped within the target.

Table 8 Noble and halogen radioisotope target inventory for a 20 kW target.

Isotope	Chemical Group	MACCS	Half Life	7-day Irradiation (Ci)	14-day Irradiation (Ci)	Saturated Activity (Ci)
Br-82	H		1.47 d	0.03	0.03	0.03
Br-83	H	x	2.40 h	88.9	88.9	88.9
Kr-83m	N	x	1.86 h	88.9	88.9	88.9
Br-84	H	x	31.8 m	162.0	162.0	162.0
Kr-85m	N	x	4.48 h	220.0	220.0	220.0
Kr-85	N	x	10.7 y	0.06	0.12	48.3
Kr-87	N	x	1.27 h	426.0	426.0	426.0
Kr-88	N	x	2.84 h	601.0	601.0	601.0
I-130	H		12.4 h	0.04	0.04	0.04
I-131	H	x	8.04 d	215.0	333.0	475.0
Xe-131m	N	x	11.9 d	2.23	3.71	6.65
I-132	H	x	2.28 h	560.0	686.0	723.0
I-133	H	x	20.8 h	1130.0	1130.0	1130.0
Xe-133m	N	x	2.19 d	28.6	31.7	32.1
Xe-133	N	x	5.24 d	685.0	956.0	1130.0
I-134	H	x	52.6 m	1280.0	1280.0	1280.0
I-135	H	x	6.57 h	1070.0	1070.0	1070.0
Xe-135m	N	x	15.3 m	184.0	184.0	184.0
Xe-135 ¹	N	x	9.10 h	1110.0	1110.0	1110.0
Xe-138	N	x	14.1 m	1050.0	1050.0	1050.0
Total				8900.0	9430.0	9840.0

¹ the Xe-135 inventory is conservative since it has a large absorption cross section, but is not adjusted for transmutation. MCNP calculations indicate that at equilibrium, the Xe-135 inventory is ~30% of this value.

Under a hypothetical condition where a reactor pulse occurred with targets in the target grid, the targets would rupture due to the extreme amount of energy deposited in the coating in a very short period of time. This event is assumed to be incredible since the pulse mode is disabled by hardware changes. However, the analysis of the downwind doses from this event would bound the consequences of all of the in-pool reactor accidents. Using the pulse scenario, the target coating would vaporize, rupture the targets, and allow all of the fission products to be dispersed into the pool. The downwind doses for a single target rupture can be calculated using the same assumptions used for the limiting event described in Section 14.3 of the ACRS SAR: 1) 30 minute decay period; 2) pool decontamination factors of 1 for the noble gases, 100 for the halogens, 1400 for all other chemical groups; and 3) unfiltered high-bay. With the added assumption that only the nobles and halogens contribute significantly to the downwind dose (it

can be shown that, using the pool decontamination factor of 1400, the downwind dose contribution from the "other" chemical groups contributes only tens of microrem to the total dose at 3000 m.), Table 8 and dose conversion factors as a function of distance (Naegeli, 1998b) calculated from the MACCS dose code (Chanin and Young, 1997) were used to calculate the downwind dose.

The downwind dose results for a vaporized target coating, ruptured target, and inventory release into the pool are shown in Table 9 for a target power of 20 kW irradiated for an infinite time (saturated activity) for distances of 0-0.1 km and 2.9-3.1 km. The doses presented are effective whole body doses. The shaded boxes identify the isotopes contributing greater than 1% of the total dose. For the exclusion boundary of 3.0 km, the dose is calculated to be 5.1 mrem. If all seven targets were to rupture, the dose is calculated to be 36.0 mrem. This value is within the operating envelope for the ACRR SAR of 40 mrem (whole body) to an individual at the exclusion area boundary (Page 14-14 of the ACRR SAR). The onsite dose is calculated to be 57.3 mrem for one target rupture and 401 mrem for a seven target rupture. The largest dose contributors are Kr-88, I-131, I-133, and I-135.

Table 9 Downwind effective whole body dose for the complete release of a 20 kW - saturated activity target, 100% nobles, 1% halogens, and 30 minute decay.

Isotope	0-0.1 km (mrem)	2.9-3.1 km (mrem)
Br-83	0.01	0.00
Kr-83m	0.00	0.00
Br-84	0.10	0.002
Kr-85m	0.25	0.012
Kr-85	0.00	0.00
Kr-87	2.47	0.45
Kr-88	11.11	2.68
I-131	14.94	0.34
Xe-131m	0.00	0.00
I-132	2.15	0.046
I-133	11.37	0.23
Xe-133m	0.00	0.00
Xe-133	0.35	0.094
I-134	1.93	0.031
I-135	7.40	0.11
Xe-135m	0.16	0.008
Xe-135	2.30	0.57
Xe-138	2.77	0.50
Total	57.33	5.14

7.2 Accident Analysis

The accident scenarios presented in Table 7 have been further analyzed along with the radionuclide inventory for the target presented in Table 8, the assumptions presented in Section 7.1, and downwind dose conversion factors. The results show that for many of the events, there are no consequences. For the events where a breach in the target cladding does occur, the downwind doses are small due to the inventory available for release from the target, and the mitigating factors associated with pool decontamination.

7.2.1 Mishandled/Dropped Target

Targets will be handled routinely and on a daily basis once a production mode is established. Target handling tools have been built to add and remove targets and aluminum "void" targets to and from the target grid. Other equipment and tooling will be required to actually move a target to the Hot Cell Facility.

With the number of movements required for the targets in a production-type environment, dropping a target from the handling tool is an anticipated event. Although the target does have negative buoyancy, it is not heavy enough to cause damage to the reactor, other equipment in the pool, or to itself. That is, it would be an incredible event for a target to be punctured by dropping it on the reactor or tank floor. A possible event would be that the VCR[®] fitting (target cap) would loosen due to dropping of the target. Dropping a target and the target cap loosening is an unlikely event. Since the target is loaded with helium to a pressure of 9 psia (see Section 2.1 for a description of the target), if the event did occur, water would penetrate the target and allow the gasses to escape into the pool.

Assuming that 3% of nobles and 3% of halogens are entrained in the target gas and released to the pool, the downwind doses can be calculated using the same methodology used in the previous section. Downwind effective whole body doses are presented in Table 10 assuming a 30 minute decay, with pool decontamination factors of 1 for the nobles and 100 for the halogens. The results are 3% of the values presented in Table 9. The largest dose contributors near the facility are Kr-88, I-131, I-133, and I-135. At the 3.0 km boundary, the noble gases play a larger role in contributing to the dose. The doses at the 0.1 km and 3.0 km boundaries are low - 1.72 mrem and 0.15 mrem, respectively.

Table 11 presents the same scenario, but with a halogen pool decontamination factor of 1, i.e., all of the halogens are released to the high bay atmosphere. This scenario would correspond to a target failure or cap loosening as the target cask is removed from the GIF or ACRR pool and is placed on the high-bay floor. This result is presented for comparative purposes only, since details on irradiated target transfer are not presented in this analysis. In reality the time delay for removal of a target from the reactor and transport to the hot cell facility would be much longer than 30 minutes, hence the radioactive isotope inventory would be significantly reduced. Both the 0.1 km and 3.0 km doses are dominated by the halogens, but still remain relatively low - 114 mrem and 2.4 mrem, respectively.

Note that if the experimental measurement of the halogen release from the target gas ($5 \times 10^{-7}\%$) is correct, the downwind dose results only from the noble gas release - 0.6 mrem and 0.12 mrem for 0.1 km and 3.0 km, respectively.

In conclusion, although a dropped target in the reactor pool is an anticipated event, the likelihood of any release of fission products into the pool is unlikely. Puncturing a target due to a drop is extremely unlikely. A more likely event would be the loosening of the target cap during or after a drop. None of the halogens and only ~1% of the noble fission product inventory would be expected to be released to the pool resulting in less than a mrem at 0.1 km.

Table 10 Downwind effective whole body dose for gas release of a 20 kW - saturated activity target, 3% nobles, 0.03% halogens, 30 minute decay.

Isotope	0-0.1 km (mrem)	2.9-3.1 km (mrem)
Br-83	0.00	0.00
Kr-83m	0.00	0.00
Br-84	0.00	0.00
Kr-85m	0.01	0.002
Kr-85	0.00	0.00
Kr-87	0.07	0.014
Kr-88	0.33	0.080
I-131	0.45	0.010
Xe-131m	0.00	0.00
I-132	0.06	0.001
I-133	0.34	0.007
Xe-133m	0.00	0.00
Xe-133	0.01	0.002
I-134	0.06	0.001
I-135	0.22	0.003
Xe-135m	0.01	0.00
Xe-135	0.07	0.017
Xe-138	0.08	0.015
Total	1.72	0.15

Table 11 Downwind effective whole body dose for gas release of a 20 kW - saturated activity target, 3% nobles, 3% halogens, and 30 minute decay.

Isotope	0-0.1 km (mrem)	2.9-3.1 km (mrem)
Br-83	0.02	0.00
Kr-83m	0.00	0.00
Br-84	0.30	0.006
Kr-85m	0.01	0.002
Kr-85	0.00	0.00
Kr-87	0.07	0.014
Kr-88	0.33	0.080
I-131	44.81	1.03
Xe-131m	0.00	0.00
I-132	6.46	0.14
I-133	34.11	0.69
Xe-133m	0.00	0.00
Xe-133	0.01	0.002
I-134	5.80	0.093
I-135	22.21	0.34
Xe-135m	0.01	0.00
Xe-135	0.07	0.017
Xe-138	0.08	0.015
Total	114.31	2.42

7.2.2 Misloaded Target

The configuration used for the seven target irradiation is for the center and first row of the target grid to be filled with one to seven targets and the outer row filled with aluminum "void" targets. If less than seven targets are irradiated, the empty locations are filled with "void" targets. The purpose for this arrangement is twofold: 1) replacing the empty locations with voids allows for the coupling factor in the targets to remain large; and 2) using only the center seven locations allows for logistically simple target identification. There are no safety implications associated with the target arrangement in the grid. To remove and add a target to the grid, the reactor would be made subcritical by several dollars. A mix-up could occur that would allow for the wrong location to be filled or emptied, or left open with no target. If the error was not discovered, the reactor would again be made critical at a cold condition. Depending on the nature of the error, the reactor temperature, and the xenon buildup, the critical condition may not indicate a significant reactivity difference in the core. However, there would be no safety consequence for operating with this configuration. The fuel element peaking factors vary by less than 10% when

comparing the seven target configuration to the all “void” configuration, hence the fuel integrity would not be jeopardized.

Another possible loading error is if a target with a larger or smaller amount of ^{235}U was inadvertently placed in the target grid at a location previously occupied by a standard target. The current loading variation for this analysis is 18 to 22 grams of ^{235}U . In Table 4, variations in ^{235}U loading from 20 to 25 grams, and cladding thickness from 25 to 35 mils, were analyzed. Fuel element peaking factors do not change significantly for these variations.

Since the targets are manufactured to specifications and are controlled at SNL using SNM and quality verification procedures, it is unlikely that a mass loading error could occur for a given target. With regard to target transfers into and out of the grid, a procedure will govern the target movements in the same manner that fuel elements are tracked. Also, it is unlikely that a variety of target configurations would be used in a production mode scenario. That is, it is anticipated that one target/reactor configuration will be used for an extended period of time. Hence a loading error, where targets are left out, more added, or are not replaced with an aluminum “void” targets, is an unlikely event.

However, in order to quantify the effects of a loading error, four possible conditions were examined: 1) seven 20 g targets are replaced with 25 g targets, the positive reactivity addition of \$0.92 is not detected, and the reactor is operated at the same power level (assuming that the neutronic instrumentation is a measure of the total power); 2) seven 20 g targets are replaced with 0 g targets, the negative reactivity addition of \$3.68 is not detected, and the reactor is operated at the same power level; 3) seven 20 g targets are replaced with water, the negative reactivity addition of \$4.07 is not detected, and the reactor is operated at the same power level; 4) seven 20 g targets are replaced with nineteen 20 g targets, the positive reactivity addition of ~\$6 is not detected, and the reactor is operated at the same power level. Note that for all of these conditions, it is assumed that the neutronic instrumentation used is a measure of the total core power. The instrumented elements are not considered in the analysis, although the scram limitation on fuel temperature would act to limit the peak element power that could be achieved.

For the original condition - seven 20 g ^{235}U targets - full power would result in the peak fuel elements operating at the nominal power of 21.74 kW. Using the coupling factors from Table 4, the total core and target power would be 2.35 MW, of which 4.1% would be due to the target power. For case 1, at a total power of 2.35 MW, the peak element power remains the same, 21.74 kW, and the target power becomes 4.7% of the total power. For case 2, the element peak to average is 1.55, and using the same methodology, the peak element power becomes 20.5 kW, with no target power. For case 3, the peak element power becomes 20.7 kW. For case 4 - a 19 target, 177 fuel element core - the fuel element peak to average is 1.77 and the target coupling factor is $0.0334 W_{\text{tar}}/W_{\text{peak fuel}} g^{235}\text{U}_{\text{tar}}$. At 2.35 MW the peak fuel element power becomes 20.9 kW, and the target power becomes 11.3% of the total power.

This simple analysis agrees with a similar conclusion that can be determined from the neutronic results comparing the peak element power to total core power. For a given reactor fuel loading configuration, the ratio of the peak element power to the total power is within ~5% for any variation in the target loading configuration.

In conclusion, for the range of conditions analyzed in this report, target loading errors are found to have no effect on the overall safe operation of the reactor. If a target loading error occurred, breach of the target or fuel element cladding would not occur due to excessive heat flux, nor would excessive fuel element temperatures be reached. No consequence would result from a fuel loading error.

7.2.3 Dislodged or Removed Target at Power

This analysis assumes that the reactor will be made subcritical by several dollars before a target is removed or added to the target grid. This requirement must be incorporated into the facility operating procedures. There are two possible accident scenarios by which a target could be inadvertently removed at power: 1) operator error, where an operator thinks the reactor is shut down, and removes a target or aluminum "void" from the grid; or 2) a target is pulled out of the grid or becomes dislodged due to the drag forces along the surface of the target caused by the natural convection flow.

For the flow rates associated with a 20 kW target, the drag forces can be calculated from the pressure drops to be ~3 grams. Also, a fully assembled uncoated target was tested in a water jacket to experimentally measure the effects of buoyant and friction forces on the target that would be experienced under natural convection flow conditions (Wemple, 1996). For the same velocity conditions, the lifting force was found to be ~10 grams. Since the equivalent mass of a submerged target is 160 g, dislodging a target is a very unlikely event. Dislodging a target would not be expected to have a significant reactivity effect.

Removal of a target or an aluminum "void" target from the grid at the critical condition would be an unlikely operator error. When the reactor is at full power, a pronounced blue glow, due to Cerenkov radiation, exists in the core. Also at full power, eddy currents below the surface of the water are strong and do not allow for a good visual image of the grid. For an inadvertent removal to occur, the reactor would have to be critical at a low power level. However, for the 19 grid configuration, a negative reactivity would be added to the core. Removal of a 20 g target would decrease the reactivity by \$0.58. Removal of an aluminum "void" target would decrease the reactivity by \$0.15.

In conclusion, inadvertent target or aluminum "void" target removal at the critical condition is found to have no effect on the overall safe operation of the reactor.

7.2.4 Target Misalignment During Loading

The target handling tool is used to add and remove targets and aluminum "void" targets from the target grid. The handling tool is positioned directly over the grid location from above the island that supports the control, safety, and transient rod drives. The tool is designed to work with the grid to properly align a target in the grid. The upper alignment target grid plate and the upper support plate are approximately 4 in. apart and act to lock the target in place laterally as the target is lowered into the grid. Once the target is below the upper support plate, the outer sleeve of the release mechanism on the handling tool mates with the target hole in the plate. This feature coupled with loading the targets from directly above the grid ensure that the target's nose will seat properly in the lower target grid plate. With some difficulty, a target or aluminum "void" target could be misaligned such that the nose of the target ends up resting on the lower target grid plate instead of properly seating. This event is considered unlikely. For the degrees of freedom

involved, the most that the target can be misaligned using the handling tool is ~0.79 in. using the design tolerances of the tool and upper grid plate. (Note that the sleeve on the handling tool can be refit to allow for a much closer tolerance within the hole of the top target grid plate.) For the 19 target grid with a pitch of 5.3 cm, the nose of the target would have to be misaligned by at least 0.84 in. before the target would contact another target and 0.71 in. before an outer row target could contact a fuel element.

The change in reactivity for a misaligned target is expected to be negligible. If a misaligned target came in contact with another target, the location of the contact would be ~0.8 in. below the fuel region on the target being contacted. The contact region would be a very loose point contact. If a misaligned target contacted a fuel element, the location of the contact would be about 1.1 in. above the lower region of the fuel. Again, the contact region would be a very loose point contact and would not cause the clad to overheat at the contact location.

Damage to the cladding of a contacted fuel element or target is considered incredible since a significant impact is required to cause even a dent in a 20 mil stainless steel tube 1.5 in. in diameter.

In conclusion, target misalignment is found to have no effect on the overall safe operation of the reactor. Although target misalignment is an unlikely event, the occurrence of such an event is not expected to cause any damage to the cladding of an adjacent target or fuel element.

7.2.5 Target Coating Failure at Power

Complete failure of a target coating is an unlikely event since each target must pass quality control vibration and temperature tests before being accepted (see Section 2.2). Cintichem Inc. had never experienced a coating failure while a target was being irradiated, and had no provision in their targets for a fuel catcher, in the event that the coating flaked and fell to the bottom of the target (McGovern, 1998). However, since the coating/cladding interface is a mechanical bond, and some targets are rejected due to coating fall off during fabrication, partial coating failure is an anticipated event. In order to ensure that the failed portion of the coating would not overheat, melt, vaporize, or cause a local hot spot in the cladding to occur, a fuel catcher feature was analyzed and added to the target design in the early stages of the program development. The catcher is designed into the bottom stainless steel plug of the target (see Section 2.1) and acts as a neutron absorber, effectively lowering the power level in the fuel to a level that can be cooled.

In the event that the complete coating failed and fell into the catcher region, -0.53 of reactivity would be added to the core. If the coating failed during a shutdown cycle, the reactivity change might not be detected as being associated with a coating failure. MCNP calculations were performed with the coating of the center target completely powdered as UO_2 in the catcher. The powdered UO_2 was modeled in radial and axial sections to determine the power variations within the catcher. The coupling factor of the powder to the peak element is $0.00184 \text{ W/W}_{\text{peak fuel}} - \text{g } ^{235}\text{U}$. For a 20 g ^{235}U target, and the peak fuel element at 21.74 kW, the powdered coating would operate at 800 W.

The steady-state heat transfer code CYLSS (Parma, 1998b) was used to determine the two-dimensional temperature profile in the powdered coating. The code was run in the R-Z direction for the analysis, with an insulated boundary condition on the bottom surface of the cladding one

centimeter below the powder. The upper surface of the powder was allowed to radiate. The powder density was ~60% of full density and was modeled as a particle bed with helium as the fill gas. The power variation results from the MCNP calculation were used in the analysis. The clad boundary was fixed at 120°C, which represents a typical wall temperature under subcooled nucleate boiling conditions. The peak temperature in the powder was found to be 1790°C in the center just below the surface, well below the melting point for UO₂ of 2500°C. The peak temperature at the powder/stainless steel interface was found to be 355°C, well below any eutectic formation temperature.

For the case where only a piece of coating begins to flake from the internal surface of the cladding, the temperature of the flake can be calculated using radiation heat transfer. Assuming an emissivity of 0.5, an average internal wall temperature of 250°C, and a heat flux in the coating of 46 W/cm² (corresponding to a 19 kW, 16.5 in. coating length), the temperature of the flake is 1420°C, well below the melting point for UO₂ of 2500°C.

In conclusion, although Cintichem never experienced a coating failure in their targets during irradiation, a fuel catcher has been incorporated into the current target design. This feature will effectively reduce the power in the fuel catcher region to an acceptable level to ensure that overheating conditions will not occur. No consequence is expected to occur for this event.

7.2.6 Leaking Target

A leaking target could occur if the target had a pinhole leak, or if the target cap was inadvertently loosened during handling. Since the welds on every target are leak checked before shipment to SNL, a loosened cap is a more plausible event. A leaking target is an anticipated event.

The initial loading pressure of helium in a target is 9 psia. For a 25 g ²³⁵U target operating at 19 kW, the pressure within the target would be ~22 psia (1.5 atm), using the UO₂ powder conductivity and a coating porosity of 50%. This assumes the gas temperature is equal to the average coating temperature on the internal surface, 443°C. The water pressure at 20 feet below the pool surface is ~22 psia (1.5 atm). If a pinhole leak or a loose fitting occurred in a target while it was at power, the leakage would be internal since the targets would not be at the maximum pressure. However, for analytical purposes, this analysis will assume that as the nobles and halogens are created, 3% become entrained in the helium fill gas and are immediately deposited into the pool. The isotopes are assumed to be retained in the pool water and escape to the environment only through pool water evaporation. The integrated release to the atmosphere can be calculated for each isotope using results from the continuous release downwind dose code CAP88-PC (Parks, 1992).

Table 12 shows the solubility limits for krypton, xenon, bromine, and iodine in water at 50°C and atmospheric pressure (CRC, 1973). Although the solubility for these elements is not extremely large, with the exception of bromine, the volume of the pool - 56,000 liters - allows for tens of kilograms of the elements to be held in solution. For a 20 kW, 14 day irradiated target, approximately 30 mg of noble gases and halogens are created; this quantity of material should be easily maintained in solution in the pool water.

The available mechanisms for the radioactive isotopes to leave the pool are by decay, pool water cleanup through the resin beds, pool water evaporation with the isotopes entrained in the vapor,

and diffusion where the isotopes diffuse to the surface and escape, and plate out to surfaces in the pool. For the halogens and other chemically reactive groups, all of the mechanisms are possible. For the noble gases, only decay, evaporation, and diffusion are possible. Of these, diffusion is the most difficult mechanism to quantify, since the pool is not stagnant, but is mixed due to the natural circulation and pump mixing. However, a somewhat stagnant layer does exist from the surface to about 1.5 m below the surface. Since the diffusion mechanism is difficult to quantify, it will be ignored for this analysis, and evaporation will be assumed to be the primary release mechanism to the high bay atmosphere. With the reactor operating at full power in the near future, a better understanding of the diffusion parameters can be developed.

Table 12 Solubility of some noble gases and halogens in water.

Element	Solubility at 50 °C (g/l)	Total Solute in 56,000 l of Water (kg)
Krypton	0.18	10.1
Xenon	0.50	28.0
Bromine	35.2	1971.0
Iodine	0.78	43.7

Assuming that decay, evaporation, and water cleanup are the only methods for the radioactive isotopes to leave the pool, the integrated release from the building can be calculated using the formulation (Massey, 1995)

$$R_i = S_i \frac{m_p}{V_p} \left(\frac{1}{\lambda_i + (m_p + \gamma m_c)/V_p} \right) \frac{m_s}{V_s} \left(\frac{1}{\lambda_i + m_s/V_s} \right)$$

where R_i is the total activity released from the stack of isotope i ; S_i is the activity of isotope i initially released from the target or element into the pool; m_p is the evaporation rate from the pool; m_c is the cleanup rate with efficiency γ ; m_s is the stack release rate; V_p is the pool and piping volume; V_s is the high bay room volume; and λ_i is the decay constant for isotope i .

Using an evaporation rate of 9.31 liters per hour from the pool (typical value at a pool temperature of 45 to 50 °C), a pool cleanup rate of 6,800 liters per hour with an efficiency of 0.95 for removing the halogens and an efficiency of zero for removing noble gasses, a pool volume of 56,000 liters, a stack release rate of 4,600 liters/s, and a high bay room volume of 2,330,000 liters, the total activity released from the ACRR high bay stack can be calculated for each noble gas and halogen radioisotope deposited in the pool.

The effective whole body dose for a leaking target is presented in Table 13 at distances of 0.3, 1.6, and 3.0 km with the wind direction towards the northwest. The target is assumed to be operating at 20 kW and has been leaking for 14 days. The inventory of nobles and halogens in

the target released to the pool is assumed to be 3%. Two conditions are presented for the halogens, without and with pool water cleanup. None of the values approach one mrem. The radioisotopes that dominate the doses are I-131, followed by I-133 and I-135. If no pool water retention is assumed, the doses would increase by three orders of magnitude for the nobles and two orders of magnitude for the halogens.

**Table 13 Downwind effective whole body dose for a leaking target
at 20 kW - 14 day irradiation.**

Condition	0.3 km (mrem)	1.6 km (mrem)	3.0 km (mrem)
Nobles 3% release to pool, 100% retention	0.00049	0.00018	0.000079
Halogens 3% release to pool, 100% retention, no cleanup	0.124	0.0296	0.0067
Halogens 3% release to pool, 100% retention, with cleanup	0.0136	0.00315	0.000714

In conclusion, although a leaking target is an anticipated event, the downwind doses that would be associated with a leak would be ~0.1 mrem on-site, and ~0.03 mrem at the nearest receptor, assuming that the nobles and halogens are completely absorbed in the water, and evaporation of the pool water is the release mechanism to the high bay atmosphere. If the nobles are not absorbed in the water, the on-site and nearest receptor doses would be ~0.6 mrem and ~0.2 mrem, respectively.

7.2.7 Flooded Target

An extension of the leaking target scenario is that of a flooded target. Since a pinhole leak is unlikely, the most plausible event would be a loosened target cap during handling. This event, like the leaking target event, is anticipated.

If a large leak formed in a target, water could fill the target since the target is typically at a lower pressure than the pool. Completely flooding one target would decrease the reactivity by a few cents. When the reactor and target were brought to full power, the water would heat up due to contact with the coating. If the target was completely filled initially with water, the water would expand and the internal target pressure would be relieved by forcing water out through the leaking area. If the leak were small enough that the target pressure could not be relieved promptly, the pressure would continue to increase and a target rupture could occur. A target rupture would be very unlikely.

Using the results of the target rupture test of 2600 psi presented in Section 2.2, and the properties of saturated water - temperature table (Wark, 1977), the water temperature can be identified where a target rupture would occur. This value is found to be 360°C. In order to exceed the

0.2% offset yield of 26,400 psi, a pressure of 1056 psi would be required, which corresponds to a water temperature of 290°C.

The results presented in Section 7.2.6 for the leaking target showed that for a 19 kW target with a 160 μm coating thickness, the average coating temperature was 443°C for UO_2 powder. For a particle bed the temperature is 193 °C. If the coating behaves as UO_2 powder, a flooded target could rupture if the leak becomes sealed after flooding. If the coating behaves as a particle bed, then the target will not rupture, nor will the 0.2% offset yield be reached.

In the very unlikely event that a target rupture occurred due to a flooded target, the damage would be localized to only the single target, due to the separation between the targets and the fuel element. The coating would remain intact except for possibly the region surrounding the damaged area. The downwind dose would be the same as for the case where a target is dropped or punctured and is presented in Table 10. The largest dose contributors near the facility are Kr-88, I-131, I-133, and I-135. At the 3.0 km boundary, the noble gases play a larger role in contributing to the dose. The doses at the 0.1 km and 3.0 km boundary are low - 1.72 mrem and 0.15 mrem, respectively.

In conclusion, like the leaking target event, a flooded target is anticipated. However, target rupture is very unlikely since the pressure would be relieved through the leaking area. For this case the downwind doses would be the same as for those presented in leaking target event.

7.2.8 Uncontrolled Control Rod Bank Withdrawal

Parma (1998a) examined the uncontrolled control rod bank withdrawal transient for the 19 target grid configuration with 19 aluminum "void" targets. The number of failures required to have an uncontrolled rod withdrawal incident where a SCRAM does occur is at least two. If the PPS operates and a SCRAM occurs at 115% of full power, the event would be terminated with no adverse effect on the core. The first failure is either a human failure where the operator continues to shim rods out without regard to the instrumentation, or a hardware failure where the Rod Up/Down switch fails closed. The second failure is either a hardware failure in both Gamma-Metrics drawers, where the startup rate signal is incorrect or has failed, or a software failure in the DAC 4 computer, where the Control Rod Withdraw Enable logic fails (Chapter 7.4.12 of the SAR). In order to have the event occur without a SCRAM, (uncontrolled rod bank withdrawal without SCRAM (URWWOS)) the PPS system must fail, which would include at least two hardware failures where both PPS drawers completely fail. Since at least two independent failures are required to have an uncontrolled rod withdrawal occur, this event is very unlikely. This event coupled with a PPS failure is extremely unlikely.

For the same transient with up to seven targets installed in the 19 grid configuration, the same number of failures are required for the incident to occur, with at least one human or hardware failure plus a second failure of the Control Rod Withdraw Enable logic or startup rate signal from both Gamma-Metrics drawers.

Given that the two failures occur, the next issue is whether the power trip PPS set point will prevent target ruptures from occurring. The difference between a fuel element and a target for this type of transient, starting from a low power level, is that a fuel element maintains a large heat

capacity and will not have excessive fuel temperatures or exceed the critical heat flux under the zero flow conditions. Contrast this result with a target that has only a small heat capacity, dictated by the cladding thickness, and which under zero flow conditions could more easily exceed the critical heat flux.

The computer code KIFLHE (KInetics, FLow, Heat) (Parma, 1998c) was used to analyze reactor transients where the heat flux and fuel temperatures were required to be calculated as a function of the reactor power and flow conditions. KIFLHE is a transient code that is similar to PK1D (Pickard and Odom, 1978). It uses zero-dimensional (i.e., point) neutron kinetics, two-dimensional (R-Z) heat transfer, and one-dimensional (Z) flow. The code monitors heat flux and calculates the CHF using the Rao, El Genk 1994 data. Reactivity feedback effects are included by monitoring the "average" element in the core throughout the transient, and determining the proper fuel and moderator temperature reactivity feedback. The code only models single phase flow. If the flow channel experiences saturation conditions, the code does not model the increased mass flow in the channel, as would be expected for two phase flow. The code allows for other elements or experiments to be monitored during the transient by using element peaking factors or a power coupling factor.

Assuming that a power trip occurs at 115% of full power, a SCRAM would be initiated and rods would begin dropping into the core within 0.3 s. Calculations were performed using KIFLHE with an initial critical power level of one watt and reactivity insertion rates of 10 and 20 ¢/s . A target coupling factor of $1.1 W_{\text{tar}}/W_{\text{avg fuel}}$ and a target peaking factor of 1.2 was used in the calculations. For each of these cases, a 115% full power trip with a 0.3 s delay was used. The calculations were performed in a conservative manner by assuming that if the zero flow CHF was exceeded for the targets, then recovery of the heat transfer coefficient was not possible, even if the heat flux dropped below the critical value. Under zero-flow conditions, the target power required to reach the CHF was calculated to be 40 kW. For the ramp rates analyzed, the CHF in the targets was reached, but the amount of energy deposited in the targets was small, ensuring that a rupture would not occur in the targets. The melting point of stainless steel is $\sim 1500^\circ\text{C}$; the temperature at which the cladding has essentially zero tensile strength is $\sim 1100^\circ\text{C}$. The heat capacity required to reach the cladding integrity loss can be calculated to be ~ 160 kJ. For the 10 and 20 ¢/s ramp rate calculations with SCRAM, the amount of energy deposited in a target is 42 kJ (clad temperature = 350°C) and 120 kJ (clad temperature = 900°C), respectively. Starting at a condition of -2 in reactivity and one watt, a 20 ¢/s ramp insertion resulted in 100 kJ of energy deposited in the target. Hence, the cladding will remain intact and the fuel temperature will remain well below the melting point for this transient.

If a power trip SCRAM is not initiated by the PPS, the target heat capacity is not adequate to allow the cladding to survive the "mini-pulse" portion of the transient. The target cladding would be expected to lose its integrity and rupture. However, fuel melt is not expected to occur. Calculations show that at the time when the cladding would be breached, the temperature of the fuel is $\sim 1200^\circ\text{C}$, significantly lower than the UO_2 melt point of 2500°C .

Assuming that the seven targets rupture as a result of an uncontrolled rod withdrawal without SCRAM, the damage would be confined to the target region since the amount of stored energy in the helium fill gas would be small, and the targets are confined to the center region of the target grid. Since the coating would not be expected to melt, the inventory release to the pool

would be 3% of the nobles and halogens. The results presented in Table 10 for the downwind dose for a single target puncture would apply. Multiplying the results by seven, the effective whole body dose at 0.1 km would be 12.0 mrem, and at 3.0 km would be 1.1 mrem. If a more conservative assumption is made that all of the target inventory is released to the pool, then Table 9 would apply. For a seven target release the dose at 0.1 km would be 401 mrem, and at 3.0 km would be 36.0 mrem. As discussed in Section 7.1, this value is within the 40 mrem (whole body) operating envelope presented in the ACRR SAR, Page 14-14.

Similar URWWOS analyses performed for the condition where the initial core power was in the kilowatt range gave similar results. Since the flow conditions are still low for the target in this power range, the target cladding behavior determines the outcome of the event. However, this would not be the case for the condition where the transient begins with the initial core power in the megawatt range. For this condition the flow would be well established in the target region, and the cladding behavior of the fuel would determine the outcome of the event. The results would be the same as those presented previously in Parma (1998a). If a SCRAM did not occur, the CHF of the fuel elements would be exceeded tens of seconds after the event was initiated. The dose consequence of cladding failure in the first row of fuel elements would be a 2.2 mrem effective whole body dose at 3.0 km.

A similar case was analyzed where the startup rate Control Rod Withdraw Enable logic was assumed to operate as designed. Under this scenario, the control rod bank adds positive reactivity until the startup rate reaches six decades per minute, which is equivalent to a period of 4.3 s or \$0.54 of reactivity. Once this condition is reached, additional positive reactivity cannot be added by control rod motion. However, the control rods could be driven into the core adding negative reactivity if operator intervention was taken. Assuming no operator intervention is taken to slow down the transient as full power is reached, the feedback effects due to the core heatup would lag behind the water heatup rate in the central cavity region, since the heat capacity of the fuel elements is large compared to the targets. The transient code KIFLHE was used to simulate this event with the positive feedback from the central cavity region included in the analysis. The event was found to terminate by a PPS initiated SCRAM at the maximum power level of 115% of full power. No excessive temperatures or heat fluxes were found to exist for the targets or fuel elements. If a SCRAM did not occur, damage to the targets similar to that describe above would result. The target cladding would be expected to lose its integrity and rupture due to the heat flux exceeding the CHF. The same dose consequences described above would result.

In conclusion, although an uncontrolled control rod withdrawal event coupled with a rod block enable failure is very unlikely, no consequence is expected if the PPS power trip functions properly. This event coupled with a PPS failure is extremely unlikely. However, dose estimates at the 3 km boundary for a seven target rupture would be within the operating envelope of 40 mrem identified in the SAR.

7.2.9 Control Rod Bank Overspeed

The uncontrolled control rod bank withdrawal assumed that the maximum reactivity addition rate for the control rod bank was 20 ϕ /s. This addition rate is limited by a TSR requirement. The reactivity addition rate is governed by the speed at which the control rod bank is withdrawn from the reactor. Rod drive motion is controlled by a stepper motor, which uses a Clock Pulse Generator Board (see Chapter 7.4.16.1 of the SAR), and a Rod Controller that is a programmable logic controller (PLC) (see Chapter 7.16.2 of the SAR). The Clock Pulse Generator Board generates the pulses and the Rod Controller provides the logic to drive the stepper motors. The Rod Controller maintains a frequency monitor that trips the PLC if the clock frequency is exceeded in the "slow" or "fast" speeds and disables the rods from moving out of the core.

The number of failures that would be required to allow the control rod bank to be withdrawn at a rate greater than the 20 ϕ /s addition limit is two. The first is a failure in the Clock Pulse Generator Board. A number of failures can be identified that could cause a higher clock frequency to be generated by the Clock Pulse Generator Board. The second is a failure of the Rod Controller to properly detect the high-frequency condition and disable the rods from moving out of the core. In the most adverse condition, the clock frequency could reach a maximum of 1800 Hz. At this frequency, a reactivity addition rate of $\sim$ \$1.20/s could be achieved by the control rods. Since the sampling time for the Rod Controller is less than 150 ms, only $\sim$ \$0.20 of reactivity would be added to the reactor, which is an acceptable quantity. Hence if the Rod Controller operates properly, this accident scenario would not continue.

The startup rate rod block functioning properly, as described in Section 7.2.8, could also prevent the accident sequence from continuing. However, the response of the startup rate rod block could be too slow if dollars per second of reactivity could be added as a result of a failure in the Clock Pulse Generator Board. If the reactor was at the critical condition, and the rod block enable logic used the entire watchdog timer limit of up to 1.2 seconds (see Chapter 7.4.11 of the SAR) before disabling the rod motion, a significant quantity of reactivity could be added to the reactor.

Since at least two hardware failures are required in order to have an overspeed condition, and since the proper operation of the rod drives is checked at startup, this event is very unlikely. If the event did occur, and the PPS operated with a SCRAM occurring at 115% of full power, the reactor would shut down, but the amount of energy deposited in the targets could be sufficient to exceed the melt point of the target cladding. Section 7.2.8 noted that for a 20 ϕ /s rate addition with the rod block logic failing, but the PPS operating properly, the amount of energy deposited in the targets would be insufficient to rupture the targets. However, for larger reactivity addition rates the energy deposition is larger. Thus at some point above 20 ϕ /s, the target cladding would fail. The resulting dose consequence would be the same as that presented in Section 7.2.8 for the case where the PPS fails to operate. The effective whole body dose at 0.1 km would be 12.0 mrem, and at 3.0 km, 1.1 mrem.

If the PPS failed to operate properly, the reactivity added could be significantly greater than \$1, resulting in a large power spike. This event would be extremely unlikely. For a significantly larger power spike the coating melt point would be exceeded resulting in a larger noble and halogen release to the pool. The dose consequence of this event would be the same as described in the following section for a pulse condition.

7.2.10 Limiting Event Pulse

The pulse mode has been disabled by hardware and software changes to the control system. As such, an inadvertent pulse is considered an incredible event. A pulse of any significant size would cause the target coating to vaporize, since the heat capacity of the targets is relatively small and the excursion would be too rapid to allow for heat transfer to the water. In order to quantify the dose consequences of such an event, the limiting event, described in Section 14.3 of the ACRR SAR, is analyzed with seven targets. This event represents a bounding case for postulated credible and incredible accidents. It requires the addition of reactivity by the transient rods coupled to an unplanned removal of a large negative worth experiment in the central cavity in the same time frame. The accident is considered the Maximum Credible Accident. An adiabatic analysis is used to calculate the energy deposition and temperature of the fuel material as a function of reactivity insertion.

The limiting event analysis used an 850 MJ pulse caused by the insertion of \$10.25 of reactivity to disrupt the core by melting cladding and fuel under adiabatic conditions. The core inventory at the time of the disruption corresponded to a reactor operation of 2 MW for 8 days. Cladding failure of 48% of the fuel elements was postulated, with 3.0% of the nobles and 5.1% of the halogens released into the pool. As discussed in Section 7.1, a decay period of 30 minutes was assumed with pool decontamination factors of 1 (100% release) for the nobles and 100 (1% release) for the halogens. The dose consequence at 3.0 km was calculated to be 5.57 mrem, assuming an unfiltered exhaust from the ACRR stack and 3.74 mrem for a filtered (mitigated) exhaust (ACRR SAR Table 14.5-2 and 14.1-2).

The SAR allows for an evaluated dose of up to 40 mrem (whole body) at the exclusion area boundary of 3 km for a beyond design basis accident. Page 14-14 of the SAR states

“... SNL’s operating envelope for this SAR is any operation (and associated “beyond design basis accident” scenario/source) whose dose consequence, as evaluated by these or similar techniques, is <40 mrem (whole body) above background to an individual at the exclusion area boundary.”

For the configurations considered in this document, the pulse mode is disabled, and the consideration of adiabatic conditions is incredible. Therefore, this limiting event is highly hypothetical and is considered incredible for steady-state mode operations with the central cavity liner removed. However, it can be shown that if this accident did occur, the event would not produce an evaluated dose consequence at the boundary of greater than 40 mrem.

Two cases will be analyzed: Case 1) no targets, 19 aluminum voids, reactor operating at 3 MW; Case 2) seven 20 g U-235 targets operating at 15 kW each, 12 aluminum voids, reactor core operating at 2.8 MW. Case 1 represents the largest power level that could be achieved for a 213 element core with no targets and the peak fuel element operating at 21.74 kW as limited by the TSRs. Case 2 represents a maximum core loading of 213 elements and configuration 2 of Table 4 of this report, with the peak fuel element operating at 21.74 kW.

The assumptions for this analysis will be the same as described above with the following additions:

- 1) No credit for filters in the high bay ventilation system will be taken.
- 2) Instead of 48% cladding failure of the fuel, 60% will be assumed. This will conservatively bound the ~25% larger peaking factors that are found for the core configurations with the central cavity liner removed. However, it can be argued that actually less fuel element cladding failures will occur since a larger portion of the core has lower peaking factors than for the pulse core configuration.
- 3) The duration at power is assumed to be 8 days for the reactor fuel. However, it can be shown that the increase in dose consequence at the 3 km boundary using 8 days as the operating history versus infinite is only ~7%, when the noble and halogen isotopes are considered.
- 4) The target fuel is assumed to be vaporized, allowing for 100% of the nobles and 1% of the halogens to escape. A 30 minute decay is assumed, as for the above analysis, before release from the high bay.
- 5) The irradiation time assumed for the targets is assumed to be infinity (saturated activity for the noble and halogen isotopes), which allows for the most conservative analysis to be made.

For Case 1, the evaluated whole body dose at the 3 km boundary can be calculated by simply multiplying the unmitigated limiting event dose of 5.57 mrem by the appropriate values derived from the assumptions. The result is a 3 km whole body dose of 13.05 mrem. The evaluated dose for Case 2 can be derived in a similar fashion and using the results from Table 9. The evaluated dose at 3 km for the reactor fuel release is 9.75 mrem, and 26.99 mrem for the target release, for a total of 36.74 mrem.

In conclusion, the evaluated dose consequence for the limiting event with seven targets in the target grid, using conservative assumptions, is within the 40 mrem authorization limit imposed by the SAR.

7.3 Effects of Moderator Heating in the Target Region

Targets located in the central region of the core allow for direct heating of the water and cause density changes and flow within the region. Also, some of the cavity water will be heated directly by the first row of fuel elements. Section 3.4 presented the expected water exit temperature for the cavity region, for inlet conditions of 40°C, for a peak fuel element power of 21.74 kW, and presented the expected reactivity effect due to the cavity water heating. This result was for a single target channel, with the assumption that other unheated channels also maintained a similar exit water temperature. Using this conservative assumption, and an initial startup coolant temperature of 20 °C, the reactivity added due to cavity heating would be ~\$0.96. The additional reactivity is smaller than the overall negative temperature feedback effect of the core that is calculated to be ~\$4.55 for a steady-state power level of 2.65 MW. Hence a significant negative reactivity balance is maintained in the power operating range.

In conclusion, since this effect is small when compared to the overall negative reactivity temperature feedback effects, and since the effect is directly dependent on reactor power under steady-state conditions, it cannot be an accident initiator at power. Section 7.2.8 showed that for a startup transient condition at the maximum startup rate of six decades per minute, the reactivity contribution from the cavity water heating was not sufficient to jeopardize the safe shutdown of the reactor.

7.4 Other Off-Normal Events

The ACRR SAR accident section was thoroughly reviewed to determine if there were other abnormal events requiring additional analysis. Table 14 presents all of the abnormal events and corresponding section numbers found in the ACRR SAR, a short description of the event, the potential radiological impact (as identified in the SAR, Table 14.1-1), the result interpreted from the SAR, and the applicability to the proposed action. All of the other events were found either to be not applicable or to be sufficiently bounding with no added probability or consequence by the proposed modifications, thus not warranting any further analysis.

The SAR contains information regarding target irradiation in the central region of the core with the cavity liner removed. Section 10.10 "Experiments Requiring Long-Term Exposure" and Section 14.4.11.7 "Long Term Experiment Hazards" present the irradiation of up to 19 targets containing 20 g of ^{235}U at power levels of 17.5 kW for up to 20 days. Also described is the operation of the reactor at 2 MW for 2 years with a comparison of the fission product inventory to the limiting event. A comparison of a target failure to the water logged fuel element (i.e., a ruptured fuel element) is made in Section 14.4.11.7. The conclusion reached is as follows:

"Both the temperatures and fission product inventories for the long term experiment materials are less than those for the core fuel elements in the peak flux regions, hence the calculations for the core fuel elements bound the long term experiment materials. Safety concerns for the long term steady-state experiments containing fissionable materials are bounded by the consequences evaluated for the Limiting Event and do not represent a hazard beyond those analyzed."

In conclusion, all other events reported in the SAR remain bounded by the SAR. Additional analysis presented in this report was performed in much greater detail than that presented in the SAR to ensure that target irradiation could be performed safely and within the bounds of the SAR.

Table 14 Comparison of SAR events to the proposed action.

EVENT	DESCRIPTION	POTENTIAL RADIO-LOGICAL IMPACT	RESULT OF SAR ANALYSIS	APPLICABILITY TO PROPOSED ACTION
REACTIVITY EVENTS				
Limiting Event (Beyond Design Basis Accident) (14.3)	Insertion of \$10.25 excess reactivity by simultaneous removal of a high negative worth experiment from central cavity and the transient rods in 80 ms.	Low to Moderate off-site; potential High off-site	An 850 MJ pulse results that disrupts the core. Clad failure was 48 percent. Whole body dose at the 3.0 km exclusion boundary is 3.7 and 5.6 mrem for the filtered and unfiltered conditions respectively.	The pulse and transient rod withdrawal modes are disabled and the central cavity is removed. This event is incredible since there is no pulse capability. The dose consequence of this event with seven targets present is calculated to be 37 mrem. This value is higher than that calculated for the SAR but still within the 40 mrem authorization basis. See Section 7.2.10
Control Rod Bank Withdrawal Without SCRAM from Subcritical or Low Power Delayed Critical (14.4.6)	Range of reactivity rates and excess reactivity: 5 to 20 cents/s and \$8.00 to \$12.00 of excess reactivity. Assumes Control System and PPS fails to operate. (See Section 8.1 of this report.)	Potential for Low on-site	A small pulse (tens of MW) occurs after >\$1.00 of reactivity is added. Only tens of MJ of energy are deposited resulting in no adverse consequences. The power level drops after the pulse due to the temperature feedback, then rises as reactivity continues to be added. Cladding failure may occur after \$7 of reactivity has been inserted. If reactivity limited, a steady state power will be reached that does not result in cladding failure	Since control rod bank worth and temperature coefficients remain the same as for the pulse configuration, the same range of insertion rates and excess reactivities applies. See Section 7.2.8.
Control Rod Bank Withdrawal Without SCRAM from High Power Delayed Critical (14.4.7)	Range of reactivity rates, excess reactivity, and initial power levels: 5 to 20 cents/s, \$8.00 to \$12.00, and 1 to 4 MW.	None	This event has the same consequences as for the previous event from a low power delayed critical. The initial pulse results in no adverse consequences. The steady state power achieved is dependent on the excess reactivity available.	Same implications as those in the previous event. See Section 7.2.8.

Table 14 (continued)

EVENT	DESCRIPTION	POTENTIAL RADIO-LOGICAL IMPACT	RESULT OF SAR ANALYSIS	APPLICABILITY TO PROPOSED ACTION
FUEL ELEMENT				
Metal/Water Reactions (coupled to Limiting Event) (14.3.3)	Metal/Water reactions could occur if extremely hot material comes in contact with water.	None	The fuel is protected by the niobium cup and the stainless steel cladding. Under steady state and nominal pulsing conditions the temperatures of the cladding are too low for this type of reaction to occur. For the limiting event, the potential exists for having such conditions present.	Bounded by the SAR.
Fuel Element Pressure - Internal (14.4.5)	Helium production and noble gas generation from fission for 1 MW-yr of operation. Pressure based on a \$6.00 pulse and 4 MW steady state.	None	Gas generated from helium production by (n, α) reactions in the beryllium is small. Noble gas production from fission is also small. For 1MW-yr of operation, no significant pressure increase will occur for pulse or steady state conditions.	Bounded by the SAR.
Waterlogged Fuel Element (14.4.8)	For fuel element in peak location, a leak in the cladding results in the element filling with water.	Low to Moderate on-site, Low off-site	A pulse operation with a waterlogged fuel element at the peak location in the core would cause it to rupture and fail three other elements.	Bounded by the SAR.

Table 14 (continued)

EVENT	DESCRIPTION	POTENTIAL RADIO-LOGICAL IMPACT	RESULT OF SAR ANALYSIS	APPLICABILITY TO PROPOSED ACTION
REACTOR COOLANT				
Cold Water Injection (14.4.9.1)	A 4°C slug of water is injected into core region with reactor at full power.	None	A step addition of a 4°C cold water slug could add ~\$1.02 in reactivity. A small pulse would occur followed by a rise in the power level to a steady state condition. A \$1.02 pulse would result in tens of MJ energy deposition and hence not result in any adverse consequence.	Temperature feedback effects are calculated to be similar to the pulse core configuration. Hence there is no difference from consequence presented in the SAR.
Partial Loss of Flow (14.4.9.2.2)	Flow blockage in peak power channel.	None	Partial loss of flow through a fuel element coolant channel is not expected to result in conditions that would challenge the integrity of the cladding since cross flow would be maintained.	Bounded by the SAR.
Total Loss of Flow (14.4.9.2.1)	Prolonged operation without heat removal capability.	Potential for Low on-site	Component failure resulting in loss of heat removal and prolonged operation without heat removal capability are of no consequence if the reactor is shutdown before the CHF is reached. At 4 MW the MCHFR is 1.7 for an inlet temperature of 60°C.	Bounded by the SAR.
Total Loss of Coolant (14.4.9.2.3)	A 0.41 m pipe rupture at the bottom of the reactor tank allowing for all of the water to drain in 0.9 minutes.	Potential for Low to Moderate on-site, Low off-site	The maximum fuel temperature is 880°C and occurs 1.2 minutes following the loss of coolant for a power level of 4 MW. No mechanical boundaries are challenged.	Bounded by the SAR. Peaking factors will not change the result of this event.

Table 14 (continued)

EVENT	DESCRIPTION	POTENTIAL RADIO- LOGICAL IMPACT	RESULT OF SAR ANALYSIS	APPLICABILITY TO PROPOSED ACTION
FUEL HANDLING				
Dropped Fuel Element during Loading, Unloading, or Transfers (14.4.10.1)	Removal of one fuel element from the core at power.	None	Removing an element from the core could add \$0.50 of reactivity to the core. The core would be on a positive period without incident.	Bounded by the SAR. The worth of a fuel element remains approximately the same.
Damaged Fuel Shipping Cask (14.4.10.2)	Release of radioactive material as a result of a fuel cask or load being dropped into the reactor pool onto the core.	Low on-site	Damage to the core as a result of a dropped cask or load is unlikely since these units are not typically handled over the tank. If this event occurs, the core could be damaged resulting in the release of fission products bounded by the limiting event.	Bounded by the SAR.
Drop Single Irradiated ACRR Fuel Element (14.4.10.3)	Dropped element results in a rupture of the cladding.	Low on-site	Assuming 100% of the nobles and 10% of the halogens are released, the dose consequence at 3 km is 0.5 mrem.	Bounded by the SAR.
EXPERIMENTS				
Unirradiated Fuel (14.4.10.4)	UO ₂ test fuel pins as part of an experiment.	Low on-site	Dose consequences are well below guideline values.	Not applicable to the proposed action.
Experiment Containment Failure (14.4.10.5)	See Chapter 11.	None	Dose consequences are well below guideline values.	See Section 7.2.6 Leaking Target and Section 7.2.7 Flooded Target.
Fissile-Experiments Containment Failure (14.4.10.6)	Radioactive materials released from containment.	No off-site release; some local contamination	Dose consequences are well below guideline values.	See Section 7.2.6 Leaking Target and Section 7.2.7 Flooded Target.
Long Term Experiment (14.4.11.7)	Evaluates target samples of 20g of U-235 irradiated in the reactor.	Low on-site	Dose consequences are well below guideline values.	See text in Section 7.4.

Table 14 (continued)

EVENT	DESCRIPTION	POTENTIAL RADIO-LOGICAL IMPACT	RESULT OF SAR ANALYSIS	APPLICABILITY TO PROPOSED ACTION
MISCELLANEOUS EVENTS				
Fire in Reactor Room (14.4.11.1)	A selected inventory of materials in the High Bay ignites and burns.	Potential for Moderate on-site, Low to Moderate off-site	No affect to core.	Bounded by the SAR.
Collapse of Bridge Crane (14.4.11.2)	Seismic event causes bridge crane to fall on core and 24 elements are assumed to rupture.	Low to Moderate on-site	10% of the total core inventory is available for release. Dose consequences are within those analyzed for the Limiting Event.	Bounded by the SAR.
Detonation of Explosive Material (14.4.11.3)	Unplanned detonation of 250 grams of high explosive	None, potential for physical damage	Explosive material must be fully contained. Hence damage possible only to experiment.	Bounded by the SAR.
Hydrogen Hazards (14.4.11.6)	Evaluates explosive effects of 1 gram of H ₂ .	Potential Low on-site	Damage possible to experiment.	Bounded by the SAR.
Aircraft/Projectile Impact (14.4.11.4)	Crash of aircraft into facility analyzes both military and civilian types.	Potential for Moderate to High on-site and off-site	Bounded by the Limiting Event.	Bounded by the SAR.
Natural Disaster (14.4.11.5)	Evaluated (qualitatively) effects of wind, flooding, tornadoes, and earthquakes.	Potential for Low on-site, local physical damage	Structural damage could occur allowing for the collapse of the bridge crane.	Bounded by the SAR.
Loss of Electric Power (14.4.11.8)	Loss of commercially supplied electric power.	None	No backup power is required for this type of reactor. Reactor is designed to fail safe.	Bounded by the SAR

8. SUMMARY AND CONCLUSIONS

The following list summarizes the operating conditions of the reactor that this analysis supports:

- Reactivity control and safety systems require no additional modifications other than those modifications from previous USQDs required for disabling the pulse and transient rod withdrawal modes.
- Operating limits are those established by the TSRs for the steady-state mode. The reactor power is limited by the TSRs to 21.74 kW for the peak fuel element.
- 177 to 213 fuel element loading is acceptable; nickel elements are not required; 90% elements and water-filled aluminum elements may be used in the outer core regions.
- 19 target grid with up to seven "Cintichem-type" targets in the first row (center location) and second row (six middle locations) of the target grid. The balance filled with aluminum "void" targets.
- Target loading: 18 - 22 g ^{235}U ; minimum cladding thickness of 25 mils.
- Irradiation time limited to 8 days as analyzed in the limiting event.
- Irradiated target transfer is limited to within the ACRR pool. Storage of irradiated targets is limited to any of the 37 holster locations in the ACRR pool.

This safety evaluation has presented the analysis associated with operating the ACRR in the steady-state mode with the central cavity liner removed and one to seven "Cintichem-type" targets installed in the central region of the 19 target grid. The main issues associated with safety and operation of the reactor with the central cavity liner removed are found to be reactivity effects and peaking factors. A core configuration range from 177 to 213 fuel elements has been presented that will allow for an excess reactivity of \$5 to \$9. It is expected that a 201 core configuration will allow for physics testing to be performed on the full range of target loadings, for zero to seven.

For the uncontrolled regulating rod bank withdrawal transient, the rod enable logic and the PPS power trip will protect the targets from rupturing. If the logic or associated hardware fails and the PPS fails, target rupture could occur, but the damage would be localized to the targets only. Release of the radionuclide inventory (nobles and halogens) in the target fill gas to the pool, and subsequently to the environment, would allow for the downwind doses at the 3.0 km boundary to be less than 1 mrem.

The limiting event presented in the ACRR SAR uses a 850 MJ pulse to disrupt the core by melting cladding and fuel under adiabatic conditions. For the configurations considered in this document, the pulse and transient rod withdrawal modes are disabled by hardware changes. The limiting event as presented in the SAR is incredible for the conditions presented in this analysis.

In conclusion, operating the reactor at steady-state power with core configurations that include up to seven "Cintichem-type" targets installed in the 19 target grid has the same or greater margin of safety as compared to that described in the ACRR SAR. The reactor can be operated safely and within the authorization basis established in the SAR.

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APPENDIX A - Sample Input/Output for the Seven Target/177 Fuel Element Configuration


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c
c
c 60 Degree Core & Target Segment Model of the ACRR
c for Mo-99 Production SAR
c
c 177 Fuel Elements (5 Full FE Rows + 0-extra FE on each flat)
c
c Control Rods: Full Out
c Safety Rods: Full Out
c Shim Rods: Full Out
c
c 7 Targets (20 g U-235)
c
c 12 Aluminum "Void" Rods
c
c
c
c -----
c cell cards
c -----
c
c -----
c Targets
c -----
c
9101 0 -909 -905 -916 u=5 $ fuel flake space
9102 0 916 -905 -915 u=5 $ catcher space
9103 0 -904 905 -915 u=5 $ catcher space
9104 3 -8.03 -907 908 (909:905) 904 -915 -901 u=5 $ catcher ss304 portion
9105 3 -8.03 902 -901 915 -912 u=5 $ cladding
9106 9 7.4104e-2 -902 903 910 -911 u=5 $ fuel
9107 0 -902 -912 915 #9106 u=5 $ internal void
9108 0 -906 912 -913 u=5 $ end cap void
9109 3 -8.03 906 -901 912 -913 u=5 $ top end cap
9110 3 -8.03 -905 913 -914 u=5 $ top fitting
9111 10 -2.19 901 918 -913 u=5 $ top target grid (70% dense aluminum)
9112 10 -2.19 907 908 -917 u=5 $ bottom target grid (70% dense aluminum)
9113 2 -1.0 -908:(907 917)
:(901 917 #9111):(905 913):914 u=5 $ outside of target
c
c
9201 0 -909 -905 -916 u=6 $ fuel flake space
9202 0 916 -905 -915 u=6 $ catcher space
9203 0 -904 905 -915 u=6 $ catcher space
9204 3 -8.03 -907 908 (909:905) 904 -915 -901 u=6 $ catcher ss304 portion
9205 3 -8.03 902 -901 915 -912 u=6 $ cladding
9206 9 7.4104e-2 -902 903 910 -911 u=6 $ fuel
9207 0 -902 -912 915 #9206 u=6 $ internal void
9208 0 -906 912 -913 u=6 $ end cap void
9209 3 -8.03 906 -901 912 -913 u=6 $ top end cap
9210 3 -8.03 -905 913 -914 u=6 $ top fitting
9211 10 -2.19 901 918 -913 u=6 $ top target grid (70% dense aluminum)
9212 10 -2.19 907 908 -917 u=6 $ bottom target grid (70% dense aluminum)
9213 2 -1.0 -908:(907 917)
:(901 917 #9211):(905 913):914 u=6 $ outside of target
c
c
9301 0 -909 -905 -916 u=7 $ fuel flake space
9302 0 916 -905 -915 u=7 $ catcher space
9303 0 -904 905 -915 u=7 $ catcher space
9304 3 -8.03 -907 908 (909:905) 904 -915 -901 u=7 $ catcher ss304 portion
9305 3 -8.03 902 -901 915 -912 u=7 $ cladding
9306 9 7.4104e-2 -902 903 910 -911 u=7 $ fuel
9307 0 -902 -912 915 #9306 u=7 $ internal void
9308 0 -906 912 -913 u=7 $ end cap void
9309 3 -8.03 906 -901 912 -913 u=7 $ top end cap
9310 3 -8.03 -905 913 -914 u=7 $ top fitting
9311 10 -2.19 901 918 -913 u=7 $ top target grid (70% dense aluminum)
9312 10 -2.19 907 908 -917 u=7 $ bottom target grid (70% dense aluminum)
9313 2 -1.0 -908:(907 917)
:(901 917 #9311):(905 913):914 u=7 $ outside of target
c

```

```

c
9401 0      -909 -905 -916      u=8 $ fuel flake space
9402 0      916 -905 -915      u=8 $ catcher space
9403 0      -904 905 -915      u=8 $ catcher space
9404 3 -8.03  -907 908 (909:905) 904 -915 -901 u=8 $ catcher ss304 portion
9405 3 -8.03  902 -901 915 -912      u=8 $ cladding
9406 9 7.4104e-2 -902 903 910 -911 -      u=8 $ fuel
9407 0      -902 -912 915 #9406      u=8 $ internal void
9408 0      -906 912 -913      u=8 $ end.cap void
9409 3 -8.03  906 -901 912 -913      u=8 $ top end cap
9410 3 -8.03  -905 913 -914      u=8 $ top fitting
9411 10 -2.19  901 918 -913      u=8 $ top target grid (70% dense aluminum)
9412 10 -2.19  907 908 -917      u=8 $ bottom target grid (70% dense aluminum)
9413 2 -1.0   -908:(907 917)
      : (901 917 #9411):(905 913):914 u=8 $ outside of target

```

```

c
c
9501 0      -909 -905 -916      u=9 $ fuel flake space
9502 0      916 -905 -915      u=9 $ catcher space
9503 0      -904 905 -915      u=9 $ catcher space
9504 3 -8.03  -907 908 (909:905) 904 -915 -901 u=9 $ catcher ss304 portion
9505 3 -8.03  902 -901 915 -912      u=9 $ cladding
9506 9 7.4104e-2 -902 903 910 -911 -      u=9 $ fuel
9507 0      -902 -912 915 #9506      u=9 $ internal void
9508 0      -906 912 -913      u=9 $ end cap void
9509 3 -8.03  906 -901 912 -913      u=9 $ top end cap
9510 3 -8.03  -905 913 -914      u=9 $ top fitting
9511 10 -2.19  901 918 -913      u=9 $ top target grid (70% dense aluminum)
9512 10 -2.19  907 908 -917      u=9 $ bottom target grid (70% dense aluminum)
9513 2 -1.0   -908:(907 917)
      : (901 917 #9511):(905 913):914 u=9 $ outside of target

```

```

c
c
c -----
c BeO-UO2 Fuel Elements
c -----
c

```

```

c
31000 0      -31 39 -40      u=10 $ void center hole
31001 1 -3.342 31 -32 39 -40 u=10 $ beo-uo2 fuel
31002 0      32 -33 39 -40 u=10 $ void gap
31003 4 -8.4   33 -34 39 -40 u=10 $ niobium cup
31004 0      34 -35 39 -40 u=10 $ void gap
31005 7 -2.80  -37 38 -39 u=10 $ beo plug
31006 7 -2.80  -37 40 -41 u=10 $ beo plug
31007 3 -8.03  -35 -39 #31005 u=10 $ end cap
31008 3 -8.03  -35 40 #31006 u=10 $ end cap
31009 3 -8.03  35 -36      u=10 $ ss304 cladding
31010 2 -1.0   36      u=10 $ water

```

```

c
c
31100 0      -31 39 -40      u=11 $ void center hole
31101 1 -3.342 31 -32 39 -40 u=11 $ beo-uo2 fuel
31102 0      32 -33 39 -40 u=11 $ void gap
31103 4 -8.4   33 -34 39 -40 u=11 $ niobium cup
31104 0      34 -35 39 -40 u=11 $ void gap
31105 7 -2.80  -37 38 -39 u=11 $ beo plug
31106 7 -2.80  -37 40 -41 u=11 $ beo plug
31107 3 -8.03  -35 -39 #31105 u=11 $ end cap
31108 3 -8.03  -35 40 #31106 u=11 $ end cap
31109 3 -8.03  35 -36      u=11 $ ss304 cladding
31110 2 -1.0   36      u=11 $ water

```

```

c
c
31200 0      -31 39 -40      u=12 $ void center hole
31201 1 -3.342 31 -32 39 -40 u=12 $ beo-uo2 fuel
31202 0      32 -33 39 -40 u=12 $ void gap
31203 4 -8.4   33 -34 39 -40 u=12 $ niobium cup
31204 0      34 -35 39 -40 u=12 $ void gap
31205 7 -2.80  -37 38 -39 u=12 $ beo plug
31206 7 -2.80  -37 40 -41 u=12 $ beo plug
31207 3 -8.03  -35 -39 #31205 u=12 $ end cap
31208 3 -8.03  -35 40 #31206 u=12 $ end cap

```

31209	3 -8.03	35 -36	u=12	\$ ss304 cladding
31210	2 -1.0	36	u=12	\$ water
c				
c				
31300	0	-31 39 -40	u=13	\$ void center hole
31301	1 -3.342	31 -32 39 -40	u=13	\$ beo-uo2 fuel
31302	0	32 -33 39 -40	u=13	\$ void gap
31303	4 -8.4	33 -34 39 -40	u=13	\$ niobium cup
31304	0	34 -35 39 -40	u=13	\$ void gap
31305	7 -2.80	-37 38 -39	u=13	\$ beo plug
31306	7 -2.80	-37 40 -41	u=13	\$ beo plug
31307	3 -8.03	-35 -39 #31305	u=13	\$ end cap
31308	3 -8.03	-35 40 #31306	u=13	\$ end cap
31309	3 -8.03	35 -36	u=13	\$ ss304 cladding
31310	2 -1.0	36	u=13	\$ water
c				
c				
31400	0	-31 39 -40	u=14	\$ void center hole
31401	1 -3.342	31 -32 39 -40	u=14	\$ beo-uo2 fuel
31402	0	32 -33 39 -40	u=14	\$ void gap
31403	4 -8.4	33 -34 39 -40	u=14	\$ niobium cup
31404	0	34 -35 39 -40	u=14	\$ void gap
31405	7 -2.80	-37 38 -39	u=14	\$ beo plug
31406	7 -2.80	-37 40 -41	u=14	\$ beo plug
31407	3 -8.03	-35 -39 #31405	u=14	\$ end cap
31408	3 -8.03	-35 40 #31406	u=14	\$ end cap
31409	3 -8.03	35 -36	u=14	\$ ss304 cladding
31410	2 -1.0	36	u=14	\$ water
c				
c				
31500	0	-31 39 -40	u=15	\$ void center hole
31501	1 -3.342	31 -32 39 -40	u=15	\$ beo-uo2 fuel
31502	0	32 -33 39 -40	u=15	\$ void gap
31503	4 -8.4	33 -34 39 -40	u=15	\$ niobium cup
31504	0	34 -35 39 -40	u=15	\$ void gap
31505	7 -2.80	-37 38 -39	u=15	\$ beo plug
31506	7 -2.80	-37 40 -41	u=15	\$ beo plug
31507	3 -8.03	-35 -39 #31505	u=15	\$ end cap
31508	3 -8.03	-35 40 #31506	u=15	\$ end cap
31509	3 -8.03	35 -36	u=15	\$ ss304 cladding
31510	2 -1.0	36	u=15	\$ water
c				
c				
31600	0	-31 39 -40	u=16	\$ void center hole
31601	1 -3.342	31 -32 39 -40	u=16	\$ beo-uo2 fuel
31602	0	32 -33 39 -40	u=16	\$ void gap
31603	4 -8.4	33 -34 39 -40	u=16	\$ niobium cup
31604	0	34 -35 39 -40	u=16	\$ void gap
31605	7 -2.80	-37 38 -39	u=16	\$ beo plug
31606	7 -2.80	-37 40 -41	u=16	\$ beo plug
31607	3 -8.03	-35 -39 #31605	u=16	\$ end cap
31608	3 -8.03	-35 40 #31606	u=16	\$ end cap
31609	3 -8.03	35 -36	u=16	\$ ss304 cladding
31610	2 -1.0	36	u=16	\$ water
c				
c				
31700	0	-31 39 -40	u=17	\$ void center hole
31701	1 -3.342	31 -32 39 -40	u=17	\$ beo-uo2 fuel
31702	0	32 -33 39 -40	u=17	\$ void gap
31703	4 -8.4	33 -34 39 -40	u=17	\$ niobium cup
31704	0	34 -35 39 -40	u=17	\$ void gap
31705	7 -2.80	-37 38 -39	u=17	\$ beo plug
31706	7 -2.80	-37 40 -41	u=17	\$ beo plug
31707	3 -8.03	-35 -39 #31705	u=17	\$ end cap
31708	3 -8.03	-35 40 #31706	u=17	\$ end cap
31709	3 -8.03	35 -36	u=17	\$ ss304 cladding
31710	2 -1.0	36	u=17	\$ water
c				
c				
31800	0	-31 39 -40	u=18	\$ void center hole
31801	1 -3.342	31 -32 39 -40	u=18	\$ beo-uo2 fuel
31802	0	32 -33 39 -40	u=18	\$ void gap

31803 4 -8.4 33 -34 39 -40 u=18 \$ niobium cup
 31804 0 34 -35 39 -40 u=18 \$ void gap
 31805 7 -2.80 -37 38 -39 u=18 \$ beo plug
 31806 7 -2.80 -37 40 -41 u=18 \$ beo plug
 31807 3 -8.03 -35 -39 #31805 u=18 \$ end cap
 31808 3 -8.03 -35 40 #31806 u=18 \$ end cap
 31809 3 -8.03 35 -36 u=18 \$ ss304 cladding
 31810 2 -1.0 36 u=18 \$ water

c

c

31900 0 -31 39 -40 u=19 \$ void center hole
 31901 1 -3.342 31 -32 39 -40 u=19 \$ beo-uo2 fuel
 31902 0 32 -33 39 -40 u=19 \$ void gap
 31903 4 -8.4 33 -34 39 -40 u=19 \$ niobium cup
 31904 0 34 -35 39 -40 u=19 \$ void gap
 31905 7 -2.80 -37 38 -39 u=19 \$ beo plug
 31906 7 -2.80 -37 40 -41 u=19 \$ beo plug
 31907 3 -8.03 -35 -39 #31905 u=19 \$ end cap
 31908 3 -8.03 -35 40 #31906 u=19 \$ end cap
 31909 3 -8.03 35 -36 u=19 \$ ss304 cladding
 31910 2 -1.0 36 u=19 \$ water

c

c

32000 0 -31 39 -40 u=20 \$ void center hole
 32001 1 -3.342 31 -32 39 -40 u=20 \$ beo-uo2 fuel
 32002 0 32 -33 39 -40 u=20 \$ void gap
 32003 4 -8.4 33 -34 39 -40 u=20 \$ niobium cup
 32004 0 34 -35 39 -40 u=20 \$ void gap
 32005 7 -2.80 -37 38 -39 u=20 \$ beo plug
 32006 7 -2.80 -37 40 -41 u=20 \$ beo plug
 32007 3 -8.03 -35 -39 #32005 u=20 \$ end cap
 32008 3 -8.03 -35 40 #32006 u=20 \$ end cap
 32009 3 -8.03 35 -36 u=20 \$ ss304 cladding
 32010 2 -1.0 36 u=20 \$ water

c

c

32100 0 -31 39 -40 u=21 \$ void center hole
 32101 1 -3.342 31 -32 39 -40 u=21 \$ beo-uo2 fuel
 32102 0 32 -33 39 -40 u=21 \$ void gap
 32103 4 -8.4 33 -34 39 -40 u=21 \$ niobium cup
 32104 0 34 -35 39 -40 u=21 \$ void gap
 32105 7 -2.80 -37 38 -39 u=21 \$ beo plug
 32106 7 -2.80 -37 40 -41 u=21 \$ beo plug
 32107 3 -8.03 -35 -39 #32105 u=21 \$ end cap
 32108 3 -8.03 -35 40 #32106 u=21 \$ end cap
 32109 3 -8.03 35 -36 u=21 \$ ss304 cladding
 32110 2 -1.0 36 u=21 \$ water

c

c

32200 0 -31 39 -40 u=22 \$ void center hole
 32201 1 -3.342 31 -32 39 -40 u=22 \$ beo-uo2 fuel
 32202 0 32 -33 39 -40 u=22 \$ void gap
 32203 4 -8.4 33 -34 39 -40 u=22 \$ niobium cup
 32204 0 34 -35 39 -40 u=22 \$ void gap
 32205 7 -2.80 -37 38 -39 u=22 \$ beo plug
 32206 7 -2.80 -37 40 -41 u=22 \$ beo plug
 32207 3 -8.03 -35 -39 #32205 u=22 \$ end cap
 32208 3 -8.03 -35 40 #32206 u=22 \$ end cap
 32209 3 -8.03 35 -36 u=22 \$ ss304 cladding
 32210 2 -1.0 36 u=22 \$ water

c

c

32300 0 -31 39 -40 u=23 \$ void center hole
 32301 1 -3.342 31 -32 39 -40 u=23 \$ beo-uo2 fuel
 32302 0 32 -33 39 -40 u=23 \$ void gap
 32303 4 -8.4 33 -34 39 -40 u=23 \$ niobium cup
 32304 0 34 -35 39 -40 u=23 \$ void gap
 32305 7 -2.80 -37 38 -39 u=23 \$ beo plug
 32306 7 -2.80 -37 40 -41 u=23 \$ beo plug
 32307 3 -8.03 -35 -39 #32305 u=23 \$ end cap
 32308 3 -8.03 -35 40 #32306 u=23 \$ end cap
 32309 3 -8.03 35 -36 u=23 \$ ss304 cladding

32310 2 -1.0 36 u=23 \$ water
 c
 c
 32400 0 -31 39 -40 u=24 \$ void center hole
 32401 1 -3.342 31 -32 39 -40 u=24 \$ beo-uo2 fuel
 32402 0 32 -33 39 -40 u=24 \$ void gap
 32403 4 -8.4 33 -34 39 -40 u=24 \$ niobium cup
 32404 0 34 -35 39 -40 u=24 \$ void gap
 32405 7 -2.80 -37 38 -39 u=24 \$ beo plug
 32406 7 -2.80 -37 40 -41 u=24 \$ beo plug
 32407 3 -8.03 -35 -39 #32405 u=24 \$ end cap
 32408 3 -8.03 -35 40 #32406 u=24 \$ end cap
 32409 3 -8.03 35 -36 u=24 \$ ss304 cladding
 32410 2 -1.0 36 u=24 \$ water
 c
 c
 32500 0 -31 39 -40 u=25 \$ void center hole
 32501 1 -3.342 31 -32 39 -40 u=25 \$ beo-uo2 fuel
 32502 0 32 -33 39 -40 u=25 \$ void gap
 32503 4 -8.4 33 -34 39 -40 u=25 \$ niobium cup
 32504 0 34 -35 39 -40 u=25 \$ void gap
 32505 7 -2.80 -37 38 -39 u=25 \$ beo plug
 32506 7 -2.80 -37 40 -41 u=25 \$ beo plug
 32507 3 -8.03 -35 -39 #32505 u=25 \$ end cap
 32508 3 -8.03 -35 40 #32506 u=25 \$ end cap
 32509 3 -8.03 35 -36 u=25 \$ ss304 cladding
 32510 2 -1.0 36 u=25 \$ water
 c
 c
 32600 0 -31 39 -40 u=26 \$ void center hole
 32601 1 -3.342 31 -32 39 -40 u=26 \$ beo-uo2 fuel
 32602 0 32 -33 39 -40 u=26 \$ void gap
 32603 4 -8.4 33 -34 39 -40 u=26 \$ niobium cup
 32604 0 34 -35 39 -40 u=26 \$ void gap
 32605 7 -2.80 -37 38 -39 u=26 \$ beo plug
 32606 7 -2.80 -37 40 -41 u=26 \$ beo plug
 32607 3 -8.03 -35 -39 #32605 u=26 \$ end cap
 32608 3 -8.03 -35 40 #32606 u=26 \$ end cap
 32609 3 -8.03 35 -36 u=26 \$ ss304 cladding
 32610 2 -1.0 36 u=26 \$ water
 c
 c
 32700 0 -31 39 -40 u=27 \$ void center hole
 32701 1 -3.342 31 -32 39 -40 u=27 \$ beo-uo2 fuel
 32702 0 32 -33 39 -40 u=27 \$ void gap
 32703 4 -8.4 33 -34 39 -40 u=27 \$ niobium cup
 32704 0 34 -35 39 -40 u=27 \$ void gap
 32705 7 -2.80 -37 38 -39 u=27 \$ beo plug
 32706 7 -2.80 -37 40 -41 u=27 \$ beo plug
 32707 3 -8.03 -35 -39 #32705 u=27 \$ end cap
 32708 3 -8.03 -35 40 #32706 u=27 \$ end cap
 32709 3 -8.03 35 -36 u=27 \$ ss304 cladding
 32710 2 -1.0 36 u=27 \$ water
 c
 c
 32800 0 -31 39 -40 u=28 \$ void center hole
 32801 1 -3.342 31 -32 39 -40 u=28 \$ beo-uo2 fuel
 32802 0 32 -33 39 -40 u=28 \$ void gap
 32803 4 -8.4 33 -34 39 -40 u=28 \$ niobium cup
 32804 0 34 -35 39 -40 u=28 \$ void gap
 32805 7 -2.80 -37 38 -39 u=28 \$ beo plug
 32806 7 -2.80 -37 40 -41 u=28 \$ beo plug
 32807 3 -8.03 -35 -39 #32805 u=28 \$ end cap
 32808 3 -8.03 -35 40 #32806 u=28 \$ end cap
 32809 3 -8.03 35 -36 u=28 \$ ss304 cladding
 32810 2 -1.0 36 u=28 \$ water
 c
 c
 32900 0 -31 39 -40 u=29 \$ void center hole
 32901 1 -3.342 31 -32 39 -40 u=29 \$ beo-uo2 fuel
 32902 0 32 -33 39 -40 u=29 \$ void gap
 32903 4 -8.4 33 -34 39 -40 u=29 \$ niobium cup

32904 0 34 -35 39 -40 u=29 \$ void gap
 32905 7 -2.80 -37 38 -39 u=29 \$ beo plug
 32906 7 -2.80 -37 40 -41 u=29 \$ beo plug
 32907 3 -8.03 -35 -39 #32905 u=29 \$ end cap
 32908 3 -8.03 -35 40 #32906 u=29 \$ end cap
 32909 3 -8.03 35 -36 u=29 \$ ss304 cladding
 32910 2 -1.0 36 u=29 \$ water

c

c

33000 0 -31 39 -40 u=30 \$ void center hole
 33001 1 -3.342 31 -32 39 -40 u=30 \$ beo-uo2 fuel
 33002 0 32 -33 39 -40 u=30 \$ void gap
 33003 4 -8.4 33 -34 39 -40 u=30 \$ niobium cup
 33004 0 34 -35 39 -40 u=30 \$ void gap
 33005 7 -2.80 -37 38 -39 u=30 \$ beo plug
 33006 7 -2.80 -37 40 -41 u=30 \$ beo plug
 33007 3 -8.03 -35 -39 #33005 u=30 \$ end cap
 33008 3 -8.03 -35 40 #33006 u=30 \$ end cap
 33009 3 -8.03 35 -36 u=30 \$ ss304 cladding
 33010 2 -1.0 36 u=30 \$ water

c

c

33100 0 -31 39 -40 u=31 \$ void center hole
 33101 1 -3.342 31 -32 39 -40 u=31 \$ beo-uo2 fuel
 33102 0 32 -33 39 -40 u=31 \$ void gap
 33103 4 -8.4 33 -34 39 -40 u=31 \$ niobium cup
 33104 0 34 -35 39 -40 u=31 \$ void gap
 33105 7 -2.80 -37 38 -39 u=31 \$ beo plug
 33106 7 -2.80 -37 40 -41 u=31 \$ beo plug
 33107 3 -8.03 -35 -39 #33105 u=31 \$ end cap
 33108 3 -8.03 -35 40 #33106 u=31 \$ end cap
 33109 3 -8.03 35 -36 u=31 \$ ss304 cladding
 33110 2 -1.0 36 u=31 \$ water

c

c

33200 0 -31 39 -40 u=32 \$ void center hole
 33201 1 -3.342 31 -32 39 -40 u=32 \$ beo-uo2 fuel
 33202 0 32 -33 39 -40 u=32 \$ void gap
 33203 4 -8.4 33 -34 39 -40 u=32 \$ niobium cup
 33204 0 34 -35 39 -40 u=32 \$ void gap
 33205 7 -2.80 -37 38 -39 u=32 \$ beo plug
 33206 7 -2.80 -37 40 -41 u=32 \$ beo plug
 33207 3 -8.03 -35 -39 #33205 u=32 \$ end cap
 33208 3 -8.03 -35 40 #33206 u=32 \$ end cap
 33209 3 -8.03 35 -36 u=32 \$ ss304 cladding
 33210 2 -1.0 36 u=32 \$ water

c

c

33300 0 -31 39 -40 u=33 \$ void center hole
 33301 1 -3.342 31 -32 39 -40 u=33 \$ beo-uo2 fuel
 33302 0 32 -33 39 -40 u=33 \$ void gap
 33303 4 -8.4 33 -34 39 -40 u=33 \$ niobium cup
 33304 0 34 -35 39 -40 u=33 \$ void gap
 33305 7 -2.80 -37 38 -39 u=33 \$ beo plug
 33306 7 -2.80 -37 40 -41 u=33 \$ beo plug
 33307 3 -8.03 -35 -39 #33305 u=33 \$ end cap
 33308 3 -8.03 -35 40 #33306 u=33 \$ end cap
 33309 3 -8.03 35 -36 u=33 \$ ss304 cladding
 33310 2 -1.0 36 u=33 \$ water

c

c

33400 0 -31 39 -40 u=34 \$ void center hole
 33401 1 -3.342 31 -32 39 -40 u=34 \$ beo-uo2 fuel
 33402 0 32 -33 39 -40 u=34 \$ void gap
 33403 4 -8.4 33 -34 39 -40 u=34 \$ niobium cup
 33404 0 34 -35 39 -40 u=34 \$ void gap
 33405 7 -2.80 -37 38 -39 u=34 \$ beo plug
 33406 7 -2.80 -37 40 -41 u=34 \$ beo plug
 33407 3 -8.03 -35 -39 #33405 u=34 \$ end cap
 33408 3 -8.03 -35 40 #33406 u=34 \$ end cap
 33409 3 -8.03 35 -36 u=34 \$ ss304 cladding
 33410 2 -1.0 36 u=34 \$ water

```

c
c
33500 0 -31 39 -40 u=35 $ void center hole
33501 1 -3.342 31 -32 39 -40 u=35 $ beo-uo2 fuel
33502 0 32 -33 39 -40 u=35 $ void gap
33503 4 -8.4 33 -34 39 -40 u=35 $ niobium cup
33504 0 34 -35 39 -40 u=35 $ void gap
33505 7 -2.80 -37 38 -39 u=35 $ beo plug
33506 7 -2.80 -37 40 -41 u=35 $ beo plug
33507 3 -8.03 -35 -39 #33505 u=35 $ end cap
33508 3 -8.03 -35 40 #33506 u=35 $ end cap
33509 3 -8.03 35 -36 u=35 $ ss304 cladding
33510 2 -1.0 36 u=35 $ water
c
c
33600 0 -31 39 -40 u=36 $ void center hole
33601 1 -3.342 31 -32 39 -40 u=36 $ beo-uo2 fuel
33602 0 32 -33 39 -40 u=36 $ void gap
33603 4 -8.4 33 -34 39 -40 u=36 $ niobium cup
33604 0 34 -35 39 -40 u=36 $ void gap
33605 7 -2.80 -37 38 -39 u=36 $ beo plug
33606 7 -2.80 -37 40 -41 u=36 $ beo plug
33607 3 -8.03 -35 -39 #33605 u=36 $ end cap
33608 3 -8.03 -35 40 #33606 u=36 $ end cap
33609 3 -8.03 35 -36 u=36 $ ss304 cladding
33610 2 -1.0 36 u=36 $ water
c
c
33700 0 -31 39 -40 u=37 $ void center hole
33701 1 -3.342 31 -32 39 -40 u=37 $ beo-uo2 fuel
33702 0 32 -33 39 -40 u=37 $ void gap
33703 4 -8.4 33 -34 39 -40 u=37 $ niobium cup
33704 0 34 -35 39 -40 u=37 $ void gap
33705 7 -2.80 -37 38 -39 u=37 $ beo plug
33706 7 -2.80 -37 40 -41 u=37 $ beo plug
33707 3 -8.03 -35 -39 #33705 u=37 $ end cap
33708 3 -8.03 -35 40 #33706 u=37 $ end cap
33709 3 -8.03 35 -36 u=37 $ ss304 cladding
33710 2 -1.0 36 u=37 $ water
c
c
33800 0 -31 39 -40 u=38 $ void center hole
33801 1 -3.342 31 -32 39 -40 u=38 $ beo-uo2 fuel
33802 0 32 -33 39 -40 u=38 $ void gap
33803 4 -8.4 33 -34 39 -40 u=38 $ niobium cup
33804 0 34 -35 39 -40 u=38 $ void gap
33805 7 -2.80 -37 38 -39 u=38 $ beo plug
33806 7 -2.80 -37 40 -41 u=38 $ beo plug
33807 3 -8.03 -35 -39 #33805 u=38 $ end cap
33808 3 -8.03 -35 40 #33806 u=38 $ end cap
33809 3 -8.03 35 -36 u=38 $ ss304 cladding
33810 2 -1.0 36 u=38 $ water
c
c
33900 0 -31 39 -40 u=39 $ void center hole
33901 1 -3.342 31 -32 39 -40 u=39 $ beo-uo2 fuel
33902 0 32 -33 39 -40 u=39 $ void gap
33903 4 -8.4 33 -34 39 -40 u=39 $ niobium cup
33904 0 34 -35 39 -40 u=39 $ void gap
33905 7 -2.80 -37 38 -39 u=39 $ beo plug
33906 7 -2.80 -37 40 -41 u=39 $ beo plug
33907 3 -8.03 -35 -39 #33905 u=39 $ end cap
33908 3 -8.03 -35 40 #33906 u=39 $ end cap
33909 3 -8.03 35 -36 u=39 $ ss304 cladding
33910 2 -1.0 36 u=39 $ water
c
c
34000 0 -31 39 -40 u=40 $ void center hole
34001 1 -3.342 31 -32 39 -40 u=40 $ beo-uo2 fuel
34002 0 32 -33 39 -40 u=40 $ void gap
34003 4 -8.4 33 -34 39 -40 u=40 $ niobium cup
34004 0 34 -35 39 -40 u=40 $ void gap

```

34005 7 -2.80 -37 38 -39 u=40 \$ beo plug
 34006 7 -2.80 -37 40 -41 u=40 \$ beo plug
 34007 3 -8.03 -35 -39 #34005 u=40 \$ end cap
 34008 3 -8.03 -35 40 #34006 u=40 \$ end cap
 34009 3 -8.03 35 -36 u=40 \$ ss304 cladding
 34010 2 -1.0 36 u=40 \$ water

c

c

34100 0 -31 39 -40 u=41 \$ void center hole
 34101 1 -3.342 31 -32 39 -40 u=41 \$ beo-uo2 fuel
 34102 0 32 -33 39 -40 u=41 \$ void gap
 34103 4 -8.4 33 -34 39 -40 u=41 \$ niobium cup
 34104 0 34 -35 39 -40 u=41 \$ void gap
 34105 7 -2.80 -37 38 -39 u=41 \$ beo plug
 34106 7 -2.80 -37 40 -41 u=41 \$ beo plug
 34107 3 -8.03 -35 -39 #34105 u=41 \$ end cap
 34108 3 -8.03 -35 40 #34106 u=41 \$ end cap
 34109 3 -8.03 35 -36 u=41 \$ ss304 cladding
 34110 2 -1.0 36 u=41 \$ water

c

c -----

c Control Rod

c -----

c

400 5 -2.48 -49 45 -46 u=70 \$ b4c poison
 401 0 -48 45 #400 u=70 \$ void cap
 402 3 -8.03 48 -47 45 u=70 \$ poison sleeve
 403 2 -1.0 47 44 u=70 \$ water
 404 3 -8.03 -47 44 -45 u=70 \$ magneform plug
 405 0 -31 43 -44 u=70 \$ void center hole
 406 1 -3.342 31 -32 43 -44 u=70 \$ beo-uo2 fuel
 407 0 32 -33 43 -44 u=70 \$ void gap
 408 4 -8.4 33 -34 43 -44 u=70 \$ niobium
 409 0 34 -35 43 -44 u=70 \$ void gap
 410 7 -2.80 -35 42 -43 u=70 \$ beo plug
 411 0 -35 -42 u=70 \$ void gap
 412 3 -8.03 35 -36 -44 u=70 \$ ss304 cladding
 413 2 -1.0 36 -44 u=70 \$ water

c

c -----

c Safety Rod

c -----

c

500 5 -2.48 -51 55 -56 u=80 \$ b4c poison
 501 0 -50 55 #500 u=80 \$ void cap
 502 3 -8.03 50 -47 55 u=80 \$ poison sleeve
 503 2 -1.0 47 54 u=80 \$ water
 504 3 -8.03 -47 54 -55 u=80 \$ magneform plug
 505 0 -31 53 -54 u=80 \$ void center hole
 506 1 -3.342 31 -32 53 -54 u=80 \$ beo-uo2 fuel
 507 0 32 -33 53 -54 u=80 \$ void gap
 508 4 -8.4 33 -34 53 -54 u=80 \$ niobium
 509 0 34 -35 53 -54 u=80 \$ void gap
 510 7 -2.80 -35 52 -53 u=80 \$ beo plug
 511 0 -35 -52 u=80 \$ void gap
 512 3 -8.03 35 -36 -54 u=80 \$ ss304 cladding
 513 2 -1.0 36 -54 u=80 \$ water

c

c -----

c Transient Rod

c -----

c

600 0 -58 -62 63 u=90 \$ void
 601 8 -2.7 58 -57 u=90 \$ al tubing
 602 2 -1.0 57 -64 u=90 \$ water
 603 8 -2.7 64 -65 u=90 \$ al guidex
 604 2 -1.0 65 u=90 \$ water
 605 8 -2.70 -58 -61 62 u=90
 606 5 -2.48 -59 61 -60 u=90 \$ poison
 607 8 -2.7 59 -58 61 -60 u=90 \$ inner sleeve
 608 0 -58 60 u=90 \$ void
 609 8 -2.7 -58 -63 u=90 \$ end plug


```

c
c -----
c Aluminum "Void" Target
c -----
c
8001 10 -2.19 901 918 -913      - u=50 $ top target grid (70% dense aluminum)
8002 10 -2.19 907 908 -917      u=50 $ bottom target grid (70% dense aluminum)
8003 8 -2.7 (908-907-901-913);(-905 913 -914)-u=50 $ "void" target
8004 2 -1.0 -908;(907 917)
      :(901 917 #8001);(905 913):914    u=50 $ outside of target
c
c -----
c Target Region Aluminum Tie-Rod
c -----
c
99 8 -2.7 -210 75 -71
c
c -----
c Upper Target Grid Region
c -----
c
100 10 -2.19 101 102 -201 -202 71 -72
c
c -----
c Lower Target Grid Region
c -----
c
101 10 -2.19 101 102 -201 -202 74 -75
c
c -----
c Target Region Boundary
c -----
c
1 0 101 102 -201 -202 75 -71 #99 fill=2
c
c -----
c Target Region Lattice Fill (7 Targets 12 Aluminum Voids)
c -----
c
2 2 -1.0 -11 12 -13 14 -15 16 lat=2 u=2
    fill -1:1 0:3 0:0
    0 5 0 $ 0
    0 6 50 $ 1
    50 50 2 $ 2
    2 2 0 $ 3
c
c -1 0 1
c
c -----
c Driver Core Boundary
c -----
c
3 0 101 102 (201:202) -203 -204 75 -71 fill=3
c
c -----
c Driver Core Lattice Fill (177 elements TR/SR out, CR out)
c -----
c
4 2 -1.0 -21 22 -23 24 -25 26 lat=2 u=3
    fill -5:5 0:10 0:0
    0 0 0 0 0 0 0 0 0 0 0 0 $ 0
    0 0 0 0 0 0 0 0 0 0 0 0 $ 1
    0 0 0 0 0 0 0 3 14 0 0 0 $ 2
    0 0 0 0 3 3 13 19 80 0 0 0 $ 3
    0 0 0 10 11 12 18 23 30 32 0 0 $ 4
    0 0 0 15 16 17 22 29 33 3 3 0 $ 5
    0 0 90 20 21 70 28 34 3 3 0 $ 6
    0 0 24 25 26 27 35 3 3 0 0 $ 7
    0 31 33 34 35 36 3 3 0 0 0 $ 8
    0 3 3 3 3 3 3 3 0 0 0 0 $ 9
    3 3 3 3 3 3 0 0 0 0 0 0 $ 10
c

```

```

c      -5 -4 -3 -2 -1  0  1  2  3  4  5
c
c
c -----
c water reflector
c -----
c
5  2  -1.0 (101 102 72 -105 -70)      Sabove grid
      :(101 102 -74 104 -70)      Sbelow grid
      :( (203:204) 101 102 74 -72 -70 #6 #7) Sbetween grids
c
c -----
c Top Grid Plate
c -----
c
6  8 -2.7 101 102 -70 71 -72 (201:202)
c
c -----
c Bottom Grid Plate
c -----
c
7  8 -2.7 101 102 -73 74 -75 (201:202)
c
c -----
c external void
c -----
c
8  0      -101:-102:70:-104:105
c
c
c -----
c surface cards
c -----
c
c -----
c Mo-99 Target
c -----
c
901 cz 1.58750 $ or of cladding
902 cz 1.49860 $ or of fuel
c 903 cz 1.49421 $ ir of fuel 44 microns 15 g U-235
903 cz 1.49275 $ ir of fuel 58 microns 20 g U-235
c 903 cz 1.49128 $ ir of fuel 73 microns 25 g U-235
c 903 cz 1.48981 $ ir of fuel 88 microns 30 g U-235
c 903 cz 1.48834 $ ir of fuel 103 microns 35 g U-235
c 903 cz 1.48687 $ ir of fuel 117 microns 40 g U-235
c 903 cz 1.48539 $ ir of fuel 132 microns 45 g U-235
904 6 kz 3.97764 1 1
905 cz 0.635
906 cz 0.47625
907 6 kz -0.549926 0.33333333 1
908 6 pz 0.000
909 6 kz 0.84496 3.00 1
910 6 pz 5.60820
911 6 pz 46.09580
912 6 pz 48.00080
913 6 pz 48.95330
914 6 pz 49.90580
915 6 pz 5.47624
916 6 pz 2.94667
917 6 pz 1.27
918 6 pz 47.68330
c
c -----
c Target Lattice Block
c -----
c
11 px 2.65
12 px -2.65
13 p 1 1.7320508076 0 5.3
14 p 1 1.7320508076 0 -5.3

```

15 p -1 1.7320508076 0 5.3
 16 p -1 1.7320508076 0 -5.3
 c
 c -----
 c Driver Core Lattice Block
 c -----
 c
 21 px 2.0855
 22 px -2.0855
 23 p 1 1.7320508076 0 4.171
 24 p 1 1.7320508076 0 -4.171
 25 p -1 1.7320508076 0 4.171
 26 p -1 1.7320508076 0 -4.171
 c
 c -----
 c BeO-UO2 Fuel Element
 c -----
 c
 31 cz 0.2413
 32 cz 1.684
 33 cz 1.72025
 34 cz 1.77125
 35 cz 1.82225
 36 cz 1.87325
 37 cz 1.487
 38 pz -1.905
 39 pz 0.00
 40 pz 52.25
 41 pz 54.1550
 c
 c -----
 c Control Rod
 c -----
 c
 42 3 pz -2.54
 43 3 pz 0.00
 44 3 pz 52.25
 45 3 pz 54.79
 46 3 pz 107.04
 47 cz 1.74625 \$ or of poison sleeve
 48 cz 1.50495 \$ ir of poison sleeve
 49 cz 1.4605 \$ or of poison
 c
 c -----
 c Safety rod
 c -----
 c
 50 cz 0.83185 \$ ir of poison sleeve
 51 cz 0.5715 \$ or of poison
 52 4 pz -2.54
 53 4 pz 0.00
 54 4 pz 52.25
 55 4 pz 54.79
 56 4 pz 107.04
 c
 c -----
 c Transient Rod
 c -----
 c void section
 57 cz 1.27 \$ or of cladding
 58 cz 1.20 \$ ir of cladding
 c poison section
 59 cz 0.88 \$ or of poison
 60 5 pz 52.88
 61 5 pz -23.321
 62 5 pz -26.495
 63 5 pz -99.5962
 c guide tube id w/sleeve 1.18 in
 64 cz 1.49860
 65 cz 2.02438
 c
 c -----

```

c   Top Grid Plate
c   -----
c
70  cz  53.35
71  pz  57.23
72  pz  59.77
c
c   -----
c   Bottom Grid Plate
c   -----
c
73  cz  47
74  pz -11.99
75  pz -6.91
c
c   -----
c   Core Segment Boundaries
c   -----
c
*101 p -1 1.7320508076 0 0.05    $ spectral reflection surface
*102 px 0.05                      $ spectral reflection surface
104  pz -24.0
105  pz  72.0
c
201  2 px  12.285
202  2 p  1 1.7320508076 0 24.57
203  2 px  36.12
204  2 p  1 1.7320508076 0 72.24
c
210  1 cz  0.9525                Starget grid spacer rods
c
c
c   -----
c   materials
c   -----
c
c   -----
c   uo2-beo fuel (3.342 g/cc to give approx. 101 g U-235 per element)
c   -----
c
m1    4009.50c -0.28285    8016.50c -0.52773    92235.50c -6.630e-2
      92238.50c -1.2223e-1    92234.50c -4.5462e-4    92236.50c -4.3567e-4
c
c   -----
c   water (1 g/cc)
c   -----
c
m2    1001.50c  2          8016.50c  1
c
c   -----
c   ss 304 (7.95 g/cc)
c   -----
c
m3    14000.50c -0.0059    24000.50c -0.1853    25055.50c -0.017
      26000.55c -0.6884    28000.50c -0.1034
c
c   -----
c   niobium (8.4 g/cc)
c   -----
c
m4    41093.50c  1
c
c   -----
c   b4c poison (2.48 g/cc)
c   -----
c
m5    6000.50c  0.2        5010.50c  0.1592    5011.55c  0.6408
c
c   -----
c   nickel reflector (8.9 g/cc)
c   -----
c

```

```

c
c m6 28000.50c 1
c
c -----
c beo (2.8 g/cc)
c -----
c
m7 4009.50c 0.5 8016.50c 0.5
c
c -----
c al 6061 (2.7 g/cc)
c -----
c
m8 12000.50c -0.01 13027.50c -0.968 14000.50c -0.006
24000.50c -0.0035 25055.50c -0.0015 26000.55c -0.007
29000.50c -0.004
c
c -----
c cintichem fuel (UO2 10.96 g/cc 100% td - 93% U-235)
c -----
c
m9 92235.50c 2.3003e-2 92238.50c 1.3386e-3
92234.50c 2.5320e-4 92236.50c 1.0534e-4
8016.50c 4.9403e-2
c
c -----
c target grid (70 v/o Al + 30 v/o H2O -- 2.19 g/cc)
c -----
c
m10 12000.50c -0.00863 13027.50c -0.835384 14000.50c -0.005178
24000.50c -0.003021 25055.50c -0.001295 26000.55c -0.006041
29000.50c -0.003452 1001.50c -0.015330 8016.50c -0.121670
c
c -----
c s(a,b) identifiers
c -----
c
mt1 beo.01t
mt2 lwtr.01t
mt7 beo.01t
c
c -----
c transformations
c -----
c
c translate origin of target region aluminum tie-rod
c
*tr1 2.64922 10.71118 0
c
c rotate pseudo-cavity x-axis 30 degrees
c
*tr2 0 0 0 30 60 90 120 30 90
c
c movement of control rods -0.001 (full up) to -54.791 (full down)
c
c 15.56 cm out
c *tr3 0 0 -39.23
c
*tr3 0 0 -0.001
c
c movement of safety rods 0.001 (full up) to -54.789 (full down)
c
*tr4 0 0 0.001
c
c movement of transient rods 0 (full down) to 90 (full up)
*tr5 0 0 90
c
c
c Position of Mo-99 Target
c

```

```

c    z coordinate locates bottom of target
c    z = 0.2730 -- target centered about core center line
c
*tr6  0 0 0.2730
c
c
c    -----
c    Tallies
c    -----
c
f14:n  31001 31101 31201 31301 31401 t      $ Row 1
fm14   -1 1 -6
fc14   Average Fission Density in ACRR Fuel Row 1
c
f24:n  31501 31601 31701 31801 31901 t      $ Row 2
fm24   -1 1 -6
fc24   Average Fission Density in ACRR Fuel Row 2
c
f34:n  32001 32101 406 32201 32301 506 t      $ Row 3
fm34   -1 1 -6
fc34   Average Fission Density in ACRR Fuel Row 3
c
f44:n  32401 32501 32601 32701 32801 32901 33001 t $ Row 4
fm44   -1 1 -6
fc44   Average Fission Density in ACRR Fuel Row 4
c
f54:n  33101 33201 33301 33401 33501 33601 t      $ Row 5
fm54   -1 1 -6
fc54   Average Fission Density in ACRR Fuel Row 5
c
c f64:n  33701 33801 33901 34001 34101 t      $ Row 6
c fm64   -1 1 -6
c fc64   Average Fission Density in ACRR Fuel Row 6
c
c
f74:n  9106                      $ Center Target
fm74   7.4104e-2 9 -6
fc74   Average Fission Density in Center Target
c
f84:n  9206                      $ Target Ring 1
fm84   7.4104e-2 9 -6
fc84   Average Fission Density in Target Ring 1
c
c f94:n  9306 9406 9506 t          $ Target Ring 2
c fm94   7.4104e-2 9 -6
c fc94   Average Fission Density in Target Ring 2
c
c
c    imp:n 1 452r 0
imp:n 1 468r 0
c
mode  n
c
kcode 1000 1.0 3 800
c
c    Source in driver core lattice position (0,5) - Corner of Row 2
c
ksrc 10.1725 18.060960 5 10.1725 18.060960 25 10.1725 18.060960 45
c
print 10 60 100 110
c
prdump 0 0 0 4

```

estimator	cycle	800	ave of	797 cycles	combination	simple average	combined average	corr
k(collision)	1.033091		1.040697	0.0013	k(col/abs)	1.039129 0.0010	1.037944 0.0009	0.5904
k(absorption)	1.039476		1.037560	0.0009	k(abs/trk ln)	1.039270 0.0011	1.037973 0.0009	0.4889
k(trk length)	1.022949		1.040979	0.0015	k(tk ln/col)	1.040838 0.0014	1.040725 0.0013	0.8778
rem life(col)	8.2837E+03		8.0530E+03	0.0019	k(col/abs/trk ln)	1.039746 0.0011	1.037974 0.0009	
rem life(abs)	8.2500E+03		8.0556E+03	0.0018	life(col/abs)	8.0543E+03 0.0018	8.0564E+03 0.0018	0.9724
source points generated	1007							

source distribution written to file srctp cycle = 800
1problem summary

run terminated when 800 kcode cycles were done.

+ 02/04/98 00:23:35
Mo-99 Run: 177FE/07T/20G/12V/CRO/TRO/SRO probid = 02/03/98 15:52:19

neutron creation	tracks	weight	energy	neutron loss	tracks	weight	energy
	(per source particle)				(per source particle)		
source	799995	1.0000E+00	2.0300E+00	escape	19096	1.5917E-02	1.6234E-02
			energy cutoff	0 0.			0.
			time cutoff	0 0.			0.
weight window	0 0.	0.		weight window	0 0.	0.	
cell importance	0 0.	0.		cell importance	0 0.	0.	
weight cutoff	0 9.6669E-02	7.0536E-07		weight cutoff	812800	9.7133E-02	6.5705E-07
energy importance	0 0.	0.		energy importance	0 0.	0.	
dxtran	0 0.	0.		dxtran	0 0.	0.	
forced collisions	0 0.	0.		forced collisions	0 0.	0.	
exp. transform	0 0.	0.		exp. transform	0 0.	0.	
upscattering	0 0.	3.7629E-07		downscattering	0 0.	1.8464E+00	
			capture	0 5.9550E-01	6.7979E-02		
(n,xn)	63801	7.4124E-02	7.7970E-02	loss to (n,xn)	31900	3.7062E-02	1.6289E-01
fission	0 0.	0.		loss to fission	0 4.2518E-01	1.4427E-02	
total	863796	1.1708E+00	2.1079E+00	total	863796	1.1708E+00	2.1079E+00

number of neutrons banked	31901	average lifetime, shakes	cutoffs
neutron tracks per source particle	1.0798E+00	escape	9.2924E+03 tco 1.0000E+34
neutron collisions per source particle	1.3374E+02	capture	7.7454E+03 eco 0.0000E+00
total neutron collisions	106991828	capture or escape	7.7691E+03 wc1 -5.0000E-01
net multiplication	1.0371E+00 0.0002	any termination	8.9896E+03 wc2 -2.5000E-01

computer time so far in this run	511.27 minutes	maximum number ever in bank	2
computer time in mcrun	508.26 minutes	bank overflows to backup file	0
source particles per minute	1.5740E+03	field length	0
random numbers generated	934671118	most random numbers used was	14534 in history 509623

keff estimator	keff	standard deviation	68% confidence	95% confidence	99% confidence
collision	1.04070	0.00139	1.03930 to 1.04209	1.03792 to 1.04347	1.03702 to 1.044
absorption	1.03756	0.00098	1.03658 to 1.03854	1.03560 to 1.03952	1.03496 to 1.040
track length	1.04098	0.00154	1.03944 to 1.04252	1.03791 to 1.04405	1.03690 to 1.045
col/absorp	1.03794	0.00098	1.03696 to 1.03892	1.03600 to 1.03989	1.03536 to 1.040
abs/trk len	1.03797	0.00097	1.03700 to 1.03895	1.03603 to 1.03991	1.03540 to 1.040
col/trk len	1.04073	0.00139	1.03933 to 1.04212	1.03795 to 1.04350	1.03705 to 1.044
col/abs/trk len	1.03797	0.00097	1.03700 to 1.03895	1.03603 to 1.03992	1.03540 to 1.040

if the largest of each keff occurred on the next cycle, the keff results and 68, 95, and 99 percent confidence interval

keff estimator	keff	standard deviation	68% confidence	95% confidence	99% confidence
collision	1.04086	0.00140	1.03945 to 1.04226	1.03807 to 1.04364	1.03716 to 1.044
absorption	1.03766	0.00099	1.03667 to 1.03865	1.03569 to 1.03962	1.03505 to 1.040
track length	1.04117	0.00155	1.03962 to 1.04272	1.03808 to 1.04426	1.03707 to 1.045
col/abs/trk len	1.03807	0.00098	1.03708 to 1.03905	1.03611 to 1.04002	1.03548 to 1.040

the estimated collision/absorption neutron lifetimes, one standard deviations, and 68, 95, and 99 percent confidence in

type	lifetime(sec)	standard deviation	68% confidence	95% confidence	99% con
------	---------------	--------------------	----------------	----------------	---------

removal	8.0564E-05	1.4378E-07	8.0420E-05 to 8.0707E-05	8.0277E-05 to 8.0850E-05	8.0184E-05 t
capture	7.7426E-05	1.2819E-07	7.7298E-05 to 7.7554E-05	7.7171E-05 to 7.7681E-05	7.7087E-05 t
fission	4.6485E-05	8.1298E-08	4.6404E-05 to 4.6566E-05	4.6323E-05 to 4.6647E-05	4.6270E-05 t
escape	9.5533E-05	1.1610E-06	9.4371E-05 to 9.6695E-05	9.3220E-05 to 9.7845E-05	9.2465E-05 t

1tally 14 nps = 799995
 + Average Fission Density in ACRR Fuel Row 1
 tally type 4 track length estimate of particle flux.
 tally for neutrons
 number of histories used for normalizing tallies = 797000.00

volumes
 cell: 31001 31101 31201 31301 31401 total
 4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 2.27972E+03

cell 31001
 multiplier bin: -1.00000E+00 1 -6
 2.42164E-05 0.0088

cell 31101
 multiplier bin: -1.00000E+00 1 -6
 5.04065E-05 0.0062

cell 31201
 multiplier bin: -1.00000E+00 1 -6
 4.05675E-05 0.0068

cell 31301
 multiplier bin: -1.00000E+00 1 -6
 5.18091E-05 0.0061

cell 31401
 multiplier bin: -1.00000E+00 1 -6
 2.59381E-05 0.0087

cell union total
 multiplier bin: -1.00000E+00 1 -6
 3.85875E-05 0.0034

1tally 24 nps = 799995
 + Average Fission Density in ACRR Fuel Row 2
 tally type 4 track length estimate of particle flux.
 tally for neutrons
 number of histories used for normalizing tallies = 797000.00

volumes
 cell: 31501 31601 31701 31801 31901 total
 4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 2.27972E+03

cell 31501
 multiplier bin: -1.00000E+00 1 -6
 3.20498E-05 0.0078

cell 31601
 multiplier bin: -1.00000E+00 1 -6
 3.04963E-05 0.0076

cell 31701
 multiplier bin: -1.00000E+00 1 -6
 2.96640E-05 0.0077

cell 31801
 multiplier bin: -1.00000E+00 1 -6
 3.06458E-05 0.0075

cell 31901
 multiplier bin: -1.00000E+00 1 -6
 3.14080E-05 0.0078

cell union total
 multiplier bin: -1.00000E+00 1 -6

3.08528E-05 0.0036
1tally 34 nps = 799995
+ Average Fission Density in ACRR Fuel Row 3
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 797000.00

volumes
cell: 32001 32101 406 32201 32301 506 total
4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 2.73566E+03

cell 32001
multiplier bin: -1.00000E+00 1 -6
2.99973E-05 0.0077

cell 32101
multiplier bin: -1.00000E+00 1 -6
2.70789E-05 0.0079

cell 406
multiplier bin: -1.00000E+00 1 -6
2.54092E-05 0.0082

cell 32201
multiplier bin: -1.00000E+00 1 -6
2.76547E-05 0.0079

cell 32301
multiplier bin: -1.00000E+00 1 -6
2.87288E-05 0.0078

cell 506
multiplier bin: -1.00000E+00 1 -6
1.40932E-05 0.0110

cell union total
multiplier bin: -1.00000E+00 1 -6
2.54937E-05 0.0036

1tally 44 nps = 799995
+ Average Fission Density in ACRR Fuel Row 4
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 797000.00

volumes
cell: 32401 32501 32601 32701 32801 32901 33001
4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02
cell: total
3.19160E+03

cell 32401
multiplier bin: -1.00000E+00 1 -6
2.72410E-05 0.0085

cell 32501
multiplier bin: -1.00000E+00 1 -6
2.42046E-05 0.0084

cell 32601
multiplier bin: -1.00000E+00 1 -6
2.27689E-05 0.0087

cell 32701
multiplier bin: -1.00000E+00 1 -6
2.05742E-05 0.0092

cell 32801
multiplier bin: -1.00000E+00 1 -6
2.32006E-05 0.0087

cell 32901
multiplier bin: -1.00000E+00 1 -6

2.47880E-05 0.0084

cell 33001
multiplier bin: -1.00000E+00 1 -6
2.50836E-05 0.0088

cell union total
multiplier bin: -1.00000E+00 1 -6
2.39801E-05 0.0035

1tally 54 nps = 799995
+ Average Fission Density in ACRR Fuel Row 5
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 797000.00

volumes
cell: 33101 33201 33301 33401 33501 33601 total
4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 4.55943E+02 2.73566E+03

cell 33101
multiplier bin: -1.00000E+00 1 -6
1.55688E-05 0.0113

cell 33201
multiplier bin: -1.00000E+00 1 -6
1.53187E-05 0.0110

cell 33301
multiplier bin: -1.00000E+00 1 -6
6.21242E-05 0.0055

cell 33401
multiplier bin: -1.00000E+00 1 -6
5.79690E-05 0.0057

cell 33501
multiplier bin: -1.00000E+00 1 -6
5.18104E-05 0.0061

cell 33601
multiplier bin: -1.00000E+00 1 -6
2.65613E-05 0.0086

cell union total
multiplier bin: -1.00000E+00 1 -6
3.82254E-05 0.0031

1tally 74 nps = 799995
+ Average Fission Density in Center Target
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 797000.00

volumes
cell: 9106
2.22585E+00

cell 9106
multiplier bin: 7.41040E-02 9 -6
9.56887E-04 0.0157

1tally 84 nps = 799995
+ Average Fission Density in Target Ring 1
tally type 4 track length estimate of particle flux.
tally for neutrons
number of histories used for normalizing tallies = 797000.00

volumes
cell: 9206
2.22585E+00

cell 9206
multiplier bin: 7.41040E-02 9 -6
6.86259E-03 0.0063

dump no. 2 on file runtpe nps = 799995 coll = 106991828 ctm = 508.26 nm = 934671118

12 warning messages so far.

run terminated when 800 kcode cycles were done.

computer time = 511.33 minutes

mcnp version 4a 10/01/93 02/04/98 00:23:38 probid = 02/03/98 15:52:

APPENDIX B - Hazard Analysis for Target Irradiation and Storage at the ACRR Facility

Hazard Table Definitions

Activity Area: A set of events that are grouped according to a generic task or function.

Event Category: Operational Event – event occurs due to conditions and activities associated with the operation(s) being performed; Natural Phenomenon Event – event is a natural phenomenon such as earthquake, flood, etc.; External Event – event is the result of activities or operations external to the facility. Unless otherwise specified in the table, the event category is assumed to be operational.

Hazard Category: For this analysis the nuclear or radiological hazards that are considered are 1) Fission Product/Isotope Release, 2) Contamination of Personnel, 3) Radiation Exposure, and 4) Inadvertent Criticality.

Event Description: The initiating event and event scenario that would lead to the described resultant consequence. The assumption in the analysis is that the resultant consequence occurred as part of the event, as opposed to the potential for the resultant consequence.

Causes: Initiating sources for the event.

Preventative Features: Design and administrative controls that would act to preclude the initiating event or terminate the event sequence. The effect of the features would be to stop the resultant consequence from occurring.

Frequency/Probability: The likelihood or the annual probability (p) of the occurrence of the event and resultant consequence. For this analysis, the probability includes, qualitatively, the likelihood of the resultant consequence with failure of the preventative features. The frequency range is described in Table B-2

Method of Detection: The detectable phenomena that would indicate that an event and resultant consequence had occurred.

Mitigative Features: Design and administrative controls that would act to bound the consequences of the event.

Unmitigated Consequences: The effect of the unmitigated event. For this analysis, credit is taken for the passive mitigative design features that are inherent to the system of interest. For example, the fact that the fuel matrix retains most of the fission products is inherent, passive, and mitigative to their release to the pool. The fact that the pool will retain, or scrub out a large portion of the halogens is inherent, passive, and mitigative to their release to the high bay. However, the high bay ventilation system is not inherent to the reactor, and is an active system. The consequence is a qualitative estimate of the onsite and offsite impact on people and the environs resulting from the incident, unmitigated by any active system such as filters in a ventilation system. The consequence range is described in Table B-3. The degree of the consequence ranges from A to D, where D is the smallest and A is the largest.

Risk Bin: The risk bin is a qualitative ranking of the combination of the probability of the occurrence and unmitigated consequence. The risk bin matrix is described in Table B-4. The degree of risk ranges for 1 to 4 where 4 is the smallest and 1 is the largest.

The hazards addressed in this hazard analysis are only the radiological and nuclear hazards associated with target irradiation and storage in the ACRR Facility. These include the following:

1. Fission Product
2. Contamination of Personnel
3. Radiation Exposure (Direct exposure from source)
4. Inadvertent Criticality

Each of the identified activity areas was considered in order to assess the potential exposure to these hazards during the conduct of the activity. Table B-1 shows the activity/hazards matrix that has been constructed for this analysis.

Table B-1. Hazards Associated With Target Irradiation and Storage.		
	Irradiated Target Handling and Storage in the ACRR Pool	Target Irradiation in the ACRR
Fission Product Release	X	X
Contamination of Personnel	X	
Radiation Exposure	X	X
Inadvertent Criticality	X	

Each of the activity areas was considered in detail to identify initiating events and event scenarios which could result in the release of or exposure to the hazards identified. The initiating events and event descriptions are listed in the Tables B-5 and B-6. The ordering of the events is not relevant. Each initiating event was qualitatively analyzed with respect to causes, preventative features, frequency, mitigative features, unmitigated consequences, risk, and recommended action. The frequency or probability of occurrence for an event was assigned according to Table B-2. The consequence for the event was assigned according to Tables B-3. The risk was assigned according to the matrix in Table B-4. Hazard tables that developed from this analysis are included as Tables B-5 and B-6. Results and recommendations from the hazard analysis are described.

The probability of occurrence, consequences, and risk categories assigned in Tables B-2 to B-4 are consistent with the definitions used in DOE-STD-1027-92, "Hazard Categorization and Accident Analysis Techniques for Compliance With DOE Order 5480.23, Nuclear Safety Analysis Reports," and DOE-STD-3009-94, "Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports."

The consequence is a qualitative estimate of the onsite and offsite impact on people or the environs resulting from the incident, unmitigated by any active system such as filters in a ventilation system. Passive features such as reactor pool water and distance to site boundary are considered in the qualitative assessment. The lowest consequence category, Category D, also includes the possibility for no consequence (i.e. zero impact).

For this hazard analysis it is assumed, for qualitative purposes, that the release of fission products from one target in the pool could have the potential to cause an onsite dose of 0.003 to 0.06 rem and an offsite dose of 0.0003 to 0.005 rem. The release of several targets in the pool could have the potential to cause an onsite dose of 0.03 to 0.6 rem and an offsite dose of 0.003 to 0.05 rem. A complete core disruption in the pool could have the potential to cause an onsite dose of 0.5 to 10 rem and an offsite dose of 0.05 to 1 rem.

Table B-2. Frequency/Probability of Occurrence Categories.

Category	Characteristic Word	Frequency Per Year	Description
I	Normal Operations	$F \geq 1$	Normal Operations
II	Likely or Anticipated	$1 > F \geq 10^{-2}$	Incidents that may occur several times during the lifetime of the facility. Incidents that are expected to occur.
III	Unlikely	$10^{-2} > F \geq 10^{-4}$	Accidents that are not anticipated to occur during the lifetime of the facility. Natural phenomena of this frequency category include: Uniform Building Code-level earthquake, 100-year flood, maximum wind gust, etc.
IV	Very Unlikely	$10^{-4} > F \geq 10^{-6}$	Accidents that will probably not occur during the life cycle of the facility.
V	Extremely Unlikely	$10^{-6} > F$	All other accidents. Accidents too unlikely to be considered in the design basis. Some accidents in this frequency category may be evaluated as beyond design basis accidents.

Table B-3. Consequence Severity Categories.

Category	Affected Population/Entity			
	Public	Collocated Worker	Worker	Environment
A	<ul style="list-style-type: none"> • Immediate health effects 	<ul style="list-style-type: none"> • Immediate health effects 	<ul style="list-style-type: none"> • Loss of life 	<ul style="list-style-type: none"> • Significant offsite contamination requiring cleanup
B	<ul style="list-style-type: none"> • Latent health effects 	<ul style="list-style-type: none"> • Latent health effects 	<ul style="list-style-type: none"> • Severe injury or disability 	<ul style="list-style-type: none"> • Moderate to significant onsite contamination • Moderate offsite contamination
C	<ul style="list-style-type: none"> • Irritation or discomfort but no permanent health effects 	<ul style="list-style-type: none"> • Irritation or discomfort but no permanent health effects 	<ul style="list-style-type: none"> • Lost time injury but no disability 	<ul style="list-style-type: none"> • Moderate contamination of originating facility • Minor onsite contamination • Minor offsite contamination
D	<ul style="list-style-type: none"> • No significant offsite impacts 	<ul style="list-style-type: none"> • No significant impacts 	<ul style="list-style-type: none"> • Minor or no impact or disability 	<ul style="list-style-type: none"> • No or minor contamination of originating facility • No onsite contamination • No offsite contamination

Table B-4. Risk Ranking Matrix.

Consequence Category	Frequency Category				
	I	II	III	IV	V
A	1	1	2	2	3
B	1	1	2	3	4
C	2	2	3	3	4
D	3	3	4	4	4

Risk Ranking 1: Hazard or accident scenario poses a high risk to the public, collocated workers, workers, or the environment. Immediate actions should be taken by the facility manager to reduce the potential consequences or likelihood of this event. For a risk ranking of 1 for the public, the event is analyzed quantitatively in the accident analysis.

Risk Ranking 2: Hazard or accident scenario poses a moderate risk to the public, collocated workers, workers, or the environment. Near- to moderate-term actions should be taken by the facility manager to reduce the potential consequences or likelihood of this event. For a risk ranking of 2 for the public, the event is analyzed quantitatively in the accident analysis.

Risk Ranking 3: Hazard or accident scenario poses a minor risk to the public, collocated workers, workers, or the environment. Moderate- to long-term actions should be taken by the facility manager to reduce the potential consequences or likelihood of this event. For a risk ranking of 3 for the public, no further analysis is required.

Risk Ranking 4: Hazard or accident scenario poses a very minor risk to the public, collocated workers, workers, or the environment. Long-term actions should be considered by the facility manager to reduce the potential consequences or likelihood of this event. For a risk ranking of 4 for the public, no further analysis is required.

Key to the Hazard Event Numbering Scheme

Each hazard event in the activity tables below is numbered by activity area and hazard category according to the following scheme. The hazard event number format is:

TI-CP-1

for activity area **TI**, hazard category **CP**, and hazard event number **1** of that series. The identifiers for each activity area and hazard category are:

Activity Area	Identifier
Irradiated Target Handling and Storage in the ACRR Pool	HS
Target Irradiation in the ACRR	TI

Hazard Category	Identifier
Fission Product/	FR
Contamination of Personnel	CP
Radiation Exposure	RE
Inadvertent Criticality	IC

Table B-5. Irradiated Target Handling and Storage in the ACRR Pool.

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bin	Recommended Actions
			Design	Admin.			Design	Admin.			
Fission Product Release											
HS-FR-1	<u>Leaking Stored Target:</u> A target stored in the ACRR pool develops a leak due to unsealing of the cap, corrosion, or damage, resulting in a release of fission products.	Cap improperly sealed or loosened during handling Corroded target cladding Faulty target weld	Corrosion resistant target cladding and sealing materials	Target preparation procedure Target fabrication QA program at manufacturer Pool water quality maintained	II	Pools regularly sampled for radionuclides Increased radiation levels detected by radiation monitors on pool water cleanup resin, facility exhaust stack, and reactor room	Fuel matrix retains fission products Reactor pool water solubility retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdry	Emergency response procedures	Release of a fraction of the noble gases and a negligible quantity of halogens Workers: D Collocated workers: D Public: D Environment: D	3 3 3 3	Target preparation procedure to include torque specifications for target cap. Periodic check of pool water quality. Sample pools regularly for radionuclides.
HS-FR-2	<u>Heavy Load Dropped on Stored Targets:</u> A heavy load is dropped on stored targets in the ACRR pool resulting in a breach of cladding and fission product release.	Operator inattention Crane/rigging failure	Crane rating Rigging rating Target can be stored in any location in either of the two storage holsters in the ACRR pool. Targets stored in either the 27 rack or the 10 rack do not extend above the rack.	Crane maintenance and check-out procedure Testing of rigging, straps, cords, etc. Trained operators Hoisting and rigging procedure	III	Visual Increased radiation levels detected by reactor room radiation monitors and radiation monitor on facility exhaust stack Radiation alarm	Fuel matrix retains fission products Reactor pool water solubility retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdry	Emergency response procedures	Release of a fraction of the noble gases and a negligible quantity of halogens Workers: C-D Collocated workers: D Public: D Environment: D	3-4 4 4 4	Follow corporate hoisting and rigging manual requirements.
HS-FR-3	<u>Mishandled/Dropped Target:</u> An irradiated target is mishandled or dropped in the ACRR pool resulting in a puncture or unsealing of the cap and the release of fission products.	Operator inattention Faulty or improperly working handling tool	Handling tools designed with simple fault proof features, e.g. positive action required for release Small negative buoyancy of targets	Target handling procedure Trained operators	III	Visual Increased radiation levels detected by reactor room radiation monitors and radiation monitor on facility exhaust stack Radiation alarm	Fuel matrix retains fission products Reactor pool water solubility retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdry	Emergency response procedures	Release of a fraction of the noble gases and a negligible quantity of halogens Workers: C-D Collocated workers: D Public: D Environment: D	3-4 4 4 4	Follow target handling procedure.

Table B-5. Irradiated Target Handling and Storage in the ACRR Pool. (continued)

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bin	Recommended Actions
			Design	Admin.			Design	Admin.			
HS-FR-4 Also HS-RE-5 TI-FR-12	<u>Loss of Pool Water:</u> A leak in the ACRR pool liner or piping results in a loss of decay heat cooling, target overheating, cladding breach, and fission product release.	Liner/piping break or puncture due to a dropped load or corrosion Drain of water while cleaning	Pool wall and piping design and construction standards Targets maintain a low decay heat power and can be cooled in air after irradiation	Monitoring of pool water level and quality	V	Visual Increased radiation levels detected by reactor room radiation monitors Radiation alarm	Fuel matrix retains fission products Reactor pool water solubility retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdry	Emergency response procedures	None. No fission products would be released since the targets maintain a low decay heat power Workers: B-D Collocated workers: D Public: D Environment: D	4 4 4 4	Periodic check of pool water quality. Monitor ACRR and GIF pool levels.
Contamination of Personnel											
HS-CP-1	<u>Personnel Become Contaminated From Handling Contaminated Tools:</u> Tools used during the target transfer activity become contaminated from potentially contaminated ACRR or GIF pool water resulting in personnel contamination.	Falling to wear personal protective equipment	Tools are designed so that the handling portion is not immersed in the pool Pool water cleanup system	Target handling procedure Radiological work permit requirements Periodic tool cleaning Radiological worker training	II	Contamination monitoring (swipe surveys) Personnel contamination detectors	None	Contamination monitoring (swipe surveys) Personnel contamination detectors	Worker contamination Workers: D Collocated workers: D Public: D Environment: D	3 4 4 4	Follow radiological work permit procedures.
Radiation Exposure											
HS-RE-1	<u>Inadvertent Lift of an Irradiated Target Beyond Water Shield:</u> An irradiated target is inadvertently lifted to a height in the ACRR or GIF pool such that the water shielding is not sufficient to keep the exposure rate to the worker at acceptable levels.	Operator inattention Procedure violation	Target handling tools are of different lengths to allow for lateral transfer with the release mechanism located at the end of the tool	Target handling procedure Trained operators Health physics technician surveys all target transfers	III	Increased radiation levels detected by radiation area monitors Radiation alarm	Radiation area monitor near and above pools Radiation alarm	None	Radiation exposure to worker(s) Workers: A-C Collocated workers: D Public: D Environment: D	2-3 4 4 4	Follow target handling procedure. Ensure radiation area monitors are in place.

Table B-5. Irradiated Target Handling and Storage in the ACRR Pool. (continued)

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bin	Recommended Actions
			Design	Admin.			Design	Admin.			
HS-RE-2	<u>Inadvertent Lift of an Irradiated Target Assuming It to be Unirradiated:</u> A worker confuses an irradiated target with an unirradiated target, lifting the irradiated target out of the pool.	Operator inattention Procedure violation	Target handling tools are of different lengths to allow for lateral transfer with the release mechanism located at the end of the tool.	Target handling procedure Operator training Health physics technician surveys all target transfers	III	Visual Increased radiation levels detected by area radiation monitor Radiation alarm	Radiation area monitor near and above pools Radiation alarm	None	Radiation exposure to worker(s) Workers: A-C Collocated workers: D Public: D Environment: D	2-3 4 4 4	Follow target handling procedure. Ensure radiation area monitors are in place.
HS-RE-3	<u>Radiation Streaming Through Target Handling Tool:</u> A tool or other equipment used for target handling provides a radiation-streaming path to the worker.	Improperly designed handling tool	All tools are designed with holes to allow water filling	None	III	Increased radiation levels detected by health physics technician during target handling	None	Health physics technician surveys all target transfers	Radiation exposure to worker(s) Workers: C Collocated workers: D Public: D Environment: D	3 4 4 4	Restrict use of hollow tubes in pool unless sufficient number of bends are present to prevent streaming.
HS-RE-4	<u>Buoyant Tools:</u> A tool or other equipment used for target handling is buoyant resulting in an irradiated target rising too close to the pool surface.	Improperly designed handling tool	All tools are designed with holes to allow water filling	None	III	Visual observation of buoyancy effect Increased radiation levels detected by health physics technician during target handling Radiation alarm	Radiation area monitor near and above pools Radiation alarm	Health physics technician surveys all target transfers	Radiation exposure to worker(s) Workers: A-C Collocated workers: D Public: D Environment: D	2-3 4 4 4	Prohibit use of buoyant tools for fuel/target handling. Ensure radiation area monitors are in place.
HS-RE-5 also HS-FR-4 TI-FR-12	<u>Loss of Pool Water:</u> A leak in the ACRR pool liner or piping results in a loss of pool water shielding.	Liner/piping break or puncture due to a dropped load or corrosion Drain of water while cleaning	Pool wall and piping design and construction standards	Monitoring of pool water level and quality	IV	Visual Increased radiation levels detected by reactor room radiation monitors Radiation alarm	Radiation area monitor near and above pools Radiation alarm Pools are all below ground level Radiation alarm	Emergency response procedures	Radiation exposure to worker(s) Workers: A-C Collocated workers: D Public: D Environment: D	2-3 4 4 4	Periodic check of pool water quality. Monitor ACRR and Gif pool levels. Ensure radiation area monitors are in place.

Table B-5. Irradiated Target Handling and Storage in the ACRR Pool. (continued)

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bin	Recommended Actions
			Design	Admin.			Design	Admin.			
Inadvertent Criticality											
HS-IC-1	<u>Improper Storage of Targets:</u> Targets (Irradiated or unirradiated) are improperly stored within the ACRR pool resulting in a criticality event.	Procedure violation	Targets in wet storage cannot be easily made into a critical geometry. Greater than 163 targets are required in an optimized configuration to reach a critical condition.	Conservative fissile material limits for underwater storage areas Target handling procedure Trained personnel Target storage areas analyzed to ensure a subcritical geometry. Design analyses reviewed by internal safety review process.	IV	Increased radiation levels detected by radiation area monitors Radiation alarm	Radiation area monitor near and above pools Radiation alarm Water Shielding	Emergency response procedures	None. A submerged configuration of targets cannot be easily made critical. Workers: D Collocated workers: D Public: D Environment: D	4 4 4 4	Follow target handling procedure. Follow corporate requirements for nuclear criticality safety.

Table B-6. Target Irradiation in the ACRR.

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bin	Recommended Actions
			Design	Admin.			Design	Admin.			
Fission Product Release											
TI-FR-1	<u>Low Flow Condition:</u> Coolant flow to a fuel element or target channel (or channels) is decreased or blocked by debris; or a poor design of the grid results in insufficient flow resulting in possible cladding breach due to excessive heat flux.	Debris enters the reactor stand shroud and blocks one or more coolant flow channels at the bottom of the core Poor design of core grid or target grid	Cross-flow between coolant channels allows for cooling capability to be maintained Skimmer used to clean surface of pool Core and target grid designed to have adequate flow at full power	Standard practice to minimized materials around pool area	III	Potential reactor power anomalies caused by coolant voiding in the affected channels Increased radiation levels detected by continuous air monitor above pool and facility exhaust stack and reactor room radiation area monitors	Fuel matrix retains fission products Reactor pool water retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdry	Emergency response procedures	Release of some noble gases and halogens from one or several fuel elements Workers: C-D Collocated workers: D Public: D Environment: C	3-4 4 4 3	Consider keeping pool covered except during maintenance and transfers. Monitor fuel element and target exit channel temperatures during initial testing phase to ensure adequate flow conditions exist.
TI-FR-2	<u>Loss of Heat Sink:</u> ACRR pool water cooling capability is lost resulting in pool water heat up. As the pool water heats up to greater than 50°C, the potential for fuel element or target cladding failure exists if the critical heat flux is exceeded.	Reactor pool water cooling system failure (feed pump, heat exchanger, cooling tower, etc.) which halts active heat removal from the reactor pool water Loss of electrical power to pumps or pump controller failure	Heat capacity of pool water is large and precludes a rapid increase in pool temperature Reactor power anomalies, due to fuel element coolant channel boiling, will cause power drop outs to occur	Pool water cooling system maintenance and inspection	III	Cleanup loop temperature and alarm (audible and display) on Balance of Plant (BOP) instrumentation display monitor in reactor control room Reactor power anomalies occur due to fuel element coolant channel boiling Increased radiation levels detected by continuous air monitor above pool and facility exhaust stack and reactor room radiation area monitors	Fuel matrix retains fission products Reactor pool water retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdry	Emergency response procedures	Release of some noble gases and halogens from one or several fuel elements Workers: C-D Collocated workers: D Public: D Environment: C	3-4 4 4 3	Maintain alarm to alert operator to high core inlet water temperature. Regular maintenance and inspection schedule for pool water cooling system components.

Table B-6. Target Irradiation in the ACRR. (continued)

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bin	Recommended Actions
			Design	Admin.			Design	Admin.			
TI-FR-3	<u>Uncontrolled Control Rod Bank Withdrawal:</u> The control rod bank is inadvertently continuously withdrawn, resulting in a power increase and possible overheating of the targets and/or the fuel elements if unmitigated by a scram.	Operator inattention Regulating rod bank switch malfunction	Rod speed limited by a stepper motor clock in the clock pulse generator board and frequency monitor in the rod controller Interlock system in DAC4 and DAC2 limits rod motion if the power is increasing too quickly	Operator training	III	Control console readouts (power level, temperature, rod position, etc.) Increased radiation levels detected by continuous air monitor above pool and facility exhaust stack and reactor room radiation area monitors Radiation alarm	Plant protection system (PPS) initiates a reactor trip (scram) at a given power level Fuel matrix retains fission products Reactor pool water retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdy	Manual scram Emergency response procedures	Critical heat flux exceeded for targets and possibly the fuel elements, possible target fuel melting; possible fuel melting; release of nobles and halogens from targets and/or a significant fraction of the fuel elements Workers: B-D Collocated workers: D Public: D Environment: C	2-4 4 4 3	Maintain rod block interlock to stop rod motion if the power is increasing too quickly. Maintain frequency monitor/trip function in rod controller. Maintain a PPS reactor trip on power level.
TI-FR-4	<u>Target Misalignment During Loading:</u> A target is not positioned properly in the target grid and does not completely seat in the lower target grid plate, resulting in it touching a fuel element or other target and overheating the cladding at that location.	Operator inattention Poor design of target handling tool and target grid	Target handling tool and target grid designed to complement each other to ensure the target is positioned properly Targets sit on top of the target grid plate and not into an adjacent hole location. If touching occurs it will be only a point contact Touching of a nearby target will be in a non-heated region of the target	Operator training	III	Increased radiation levels detected by continuous air monitor above pool and facility exhaust stack and reactor room radiation area monitors Radiation alarm	Fuel matrix retains fission products Reactor pool water retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdy	Emergency response procedures	Critical heat flux burnout of fuel element cladding at the touched location should not occur since the contact region will be a point contact. However if it does occur, the release of nobles and halogens from a single fuel element would occur Workers: C-D Collocated workers: D Public: D Environment: D	3-4 4 4 4	Ensure target handling tool and grid perform as expected to ensure a misalignment cannot occur.

Table B-6. Target Irradiation in the ACRR. (continued)

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bln	Recommended Actions
			Design	Admin.			Design	Admin.			
TI-FR-5	<u>Misloaded Target:</u> A target is inserted into a target grid location which has a greater than expected neutronic coupling to the core, or a target or group of targets are loaded into the target grid with a higher than expected ^{235}U loading, resulting in target(s) operating at higher than expected powers, leading to target overheating, clad failure, and fission product release.	Operator inattention Procedure violation	Any target of a nominal loading may be placed at any target grid location since the neutronic coupling is essentially the same throughout the target grid	Operator training Targets are loaded with ^{235}U to within specified values by SNL. Otherwise targets are rejected Target preparation procedure Target loading procedure	IV	Increased radiation levels detected by continuous air monitor above pool and facility exhaust stack and reactor room radiation area monitors Radiation alarm	Fuel matrix retains fission products Reactor pool water retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdy	Emergency response procedures	Critical heat flux burnout of target cladding; target fuel melting; release of nobles and halogens from a single target Workers: C-D Collocated workers: D Public: D Environment: D	3-4 4 4 4	Ensure target handling and loading procedure identifies acceptable loading limitations.
TI-FR-6	<u>Leaking Target:</u> A target develops a leak due to unsealing of the cap, corrosion, or damage, resulting in a release of fission products.	Cap improperly sealed or loosened during handling Corroded target cladding Faulty target weld	Corrosion resistant target cladding and sealing materials	Target preparation procedure Target fabrication QA program at manufacturer Pool water quality maintained	II	Pools regularly sampled for radionuclides Increased radiation levels detected by radiation monitors on pool water cleanup resin, facility exhaust stack, and reactor room	Fuel matrix retains fission products Reactor pool water solubility retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdy	Emergency response procedures	Release of a fraction of the noble gases and halogens Workers: D Collocated workers: D Public: D Environment: D	3 3 3 3	Target preparation procedure to include torque specifications for target cap. Periodic check of pool water quality. Sample pools regularly for radionuclides.

Table B-6. Target Irradiation in the ACR. (continued)

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bln	Recommended Actions
			Design	Admin.			Design	Admin.			
TI-FR-7	Flooded Target: A target develops a leak due to unsealing of the cap, corrosion, or damage, resulting in water ingress into a target, which leads to possible target pressurization, clad rupture, and fission product release.	Cap improperly sealed or loosened during handling Corroded target cladding Faulty target weld	Corrosion resistant target cladding and sealing materials	Target preparation procedure Target fabrication QA program at manufacturer Pool water quality maintained	III	Pools regularly sampled for radionuclides Increased radiation levels detected by radiation monitors on pool water cleanup resin, facility exhaust stack, and reactor room Radiation alarm	Fuel matrix retains fission products Reactor pool water solubility retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdy	Emergency response procedures	Target rupture; release of nobles and halogens from a single target Workers: C-D Collocated workers: D Public: D Environment: D	3-4 4 4 4	Target preparation procedure to include torque specifications for target cap. Periodic check of pool water quality. Sample pools regularly for radionuclides.
TI-FR-8	Target Coating Failure at Power: A partial or complete uranium oxide coating failure occurs in a target while the reactor is at full power resulting in a hot spot and cladding failure.	Coating adherence not maintained	Target lower end fitting maintains a catcher feature to reduce flaked coating power to acceptable levels	Target fabrication QA program at manufacturer checks for coating fall off during vibration tests	III	Increased radiation levels detected by continuous air monitor above pool and facility exhaust stack and reactor room radiation area monitors Radiation alarm	Fuel matrix retains fission products Reactor pool water solubility retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdy None	Emergency response procedures	Critical heat flux burnout of target cladding; target fuel melting; release of nobles and halogens from a single target Workers: C-D Collocated workers: D Public: D Environment: D	3-4 4 4 4	Target preparation procedure to include borescope inspection and coating fall off check.
TI-FR-9	Dislodged or Removed Target at Power: A target is dislodged from the target grid or is inadvertently removed from the grid while the reactor is at power, causing a possible reactivity anomaly.	Operator inattention Procedure violation Buoyancy of the target is too large	Targets retain a negative buoyancy Removing a target or a void adds negative reactivity	Target preparation procedure Target loading procedure Operator training	III	Change in power due to negative reactivity insertion	None	None	None. Reactor power level changes Workers: D Collocated workers: D Public: D Environment: D	4 4 4 4	Target preparation procedure to include minimum weight of target. Target loading procedure.

Table B-6. Target Irradiation in the ACRR. (continued)

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bln	Recommended Actions
			Design	Admin.			Design	Admin.			
TI-FR-10	<u>Target Loaded Which Exceeds Acceptable Parameters:</u> A target or group of targets are loaded with higher than expected ^{235}U loading, or reduced cladding thickness, resulting in targets and/or fuel elements operating at higher than expected powers, overheating, clad failure, and fission product release.	Operator inattention Breakdown of target fabrication and inspection QA	Targets can be operated at high powers and are currently limited by the peak fuel element power	Operator training Targets are loaded with ^{235}U to within specified values by SNL. Otherwise target are rejected Target preparation procedure Target loading procedure	III	Increased radiation levels detected by continuous air monitor above pool and facility exhaust stack and reactor room radiation area monitors Radiation alarm	Fuel matrix retains fission products Reactor pool water solubility retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdry	Emergency response procedures	Critical heat flux burnout of target or fuel element cladding; target or fuel element fuel melting; release of nobles and halogens from several targets or fuel elements Workers: C-D Collocated workers: D Public: D Environment: C	3-4 4 4 3	Target preparation procedure to include ^{235}U loading check.
TI-FR-11	<u>Heavy Load Dropped on Core:</u> A heavy load (e.g., transfer cask) is dropped on the ACRR core, resulting in fuel element or target clad breach, and fission product release.	Operator inattention Crane/rigging failure	Crane rating Rigging rating Targets are located below top target grid plate.	Crane maintenance and check-out procedure Testing of rigging, straps, cords, etc. Trained operators Hoisting and rigging procedure	III	Visual Increased radiation levels detected by continuous air monitor above pool and facility exhaust stack and reactor room radiation area monitors Radiation alarm	Fuel matrix retains fission products Reactor pool water solubility retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdry	Emergency response procedures	Release of a fraction of the noble gases and a negligible quantity of halogens Workers: C-D Collocated workers: D Public: D Environment: D	3-4 4 4 4	Follow corporate hoisting and rigging manual requirements.

Table B-6. Target Irradiation in the ACRR. (continued)

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bin	Recommended Actions
			Design	Admin.			Design	Admin.			
TI-FR-12 Also HS-FR-4 HS-RE-5	Loss of Pool Water: A leak in the ACRR or GIF pool liner or piping results in a loss of coolant in the ACRR pool with the reactor at power, reducing the saturation temperature and leading to fuel element or target overheating, clad failure, and fission product release.	Liner/piping break or puncture due to a dropped load or corrosion Drain of water while cleaning	Pool wall and piping design and construction standards No pathway for coolant loss to the point of uncovering the core. Pipe break resulting in equipment room flooding would not uncover core; liner surrounded by concrete and earth	Monitor of pool water level and quality	V	Visual Pool water level indicator alarm Increased radiation levels detected by continuous air monitor above pool and facility exhaust stack and reactor room radiation area monitors Radiation alarm	Fuel matrix retains fission products Reactor pool water solubility retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdry	Emergency response procedures	Critical heat flux burnout of fuel elements or target cladding; fuel melting; target fuel melting; release of nobles and halogens from a significant fraction of the core Workers: B-D Collocated workers: D Public: D Environment: C	4 4 4 4	Maintain pool water level alarm to alert operator of loss of coolant situation and immediate actions to be taken. Periodic check of pool water quality.
TI-FR-13	Control Rod Bank Overspeed: The clock pulse generator board fails, increasing the clock speed to levels that allow for greater than 20 cents per second reactivity addition, which results in a power increase and possible overheating of the targets and/or the fuel elements if unmitigated by a scram.	Any clock board failure that could change the voltage input to a level that would cause the frequency to be greater than the set value Relay failure puts clock speed into "fast" mode	Rod speed limited by frequency monitor in the rod controller Interlock system in DAC4 and DAC2 limits rod motion if the power is increasing too quickly	Reactivity addition rate periodically checked in accordance with the TSRs	IV	Rod controller trip Control console readouts (power level, temperature, rod position, etc.) Increased radiation levels detected by continuous air monitor above pool and facility exhaust stack and reactor room radiation area monitors Radiation alarm	Plant protection system (PPS) initiates a reactor trip (scram) at a given power level Fuel matrix retains fission products Reactor pool water retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdry	Manual scram Emergency response procedures	Critical heat flux exceeded for targets and possibly the fuel elements, possible target fuel melting; possible fuel melting; release of nobles and halogens from targets and/or a significant fraction of the fuel elements Workers: B-D Collocated workers: D Public: D Environment: C	3-4 4 4 3	Maintain rod block interlock to stop rod motion if the power is increasing to quickly. Maintain frequency monitor/trip function in rod controller. Maintain a PPS reactor trip on power level.

Table B-6. Target Irradiation in the ACRR. (continued)

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bln	Recommended Actions
			Design	Admin.			Design	Admin.			
TI-FR-14	<u>Inadvertent Pulse:</u> The pulse or transient rod withdrawal mode is inadvertently entered and a pulse, transient rod fast withdrawal, or limiting event pulse is performed, resulting in target coating vaporization and cladding rupture, possible fuel element cladding rupture, and a release of fission products.	Operator error Incorrect lockout of pneumatic system Incorrect disabling of transient and pulse modes	Pulse and transient rod withdrawal modes are currently disabled by hardware and software changes	Pulse mode administratively controlled Currently there are no provisions to allow for hardware and software changes that would enable the pulse mode	V	Increased radiation levels detected by continuous air monitor above pool and facility exhaust stack and reactor room radiation area monitors Radiation alarm	Fuel matrix retains fission products Reactor pool water solubility retains fission products High bay ventilation system maintains a negative pressure differential, minimizing building contamination potential Distance to site bdry	Emergency response procedures	Target coating vaporizes; target cladding ruptures; release of nobles and halogens from all of the targets in the core. Fuel element failure is plausible depending on pulse size Workers: B-D Collocated workers: D Public: D Environment: C	4 4 4 4	None. Maintain current configuration with pulse mode disabled.
TI-FR-15	<u>Earthquake:</u> An earthquake occurs resulting in heavy debris falling into the pool, damaging fuel elements and targets leading to a release of fission products.	Natural phenomena Crane collapse Bridge collapse Building collapse	Building and crane construction standards	None	III	Visual	Fuel matrix retains fission products Reactor pool water solubility retains fission products Distance to site bdry	Emergency response procedures	Release of a fraction of the noble gases and halogens Workers: C-D Collocated workers: D Public: D Environment: D	3-4 4 4 4	None
TI-FR-16	<u>Tornado/High Winds:</u> A tornado or high wind situation occurs resulting in heavy debris falling into the pool, damaging fuel elements and targets leading to a release of fission products.	Natural phenomena Crane collapse Bridge collapse Building collapse	Building and crane construction standards	None	III	Visual	Fuel matrix retains fission products Reactor pool water solubility retains fission products Distance to site bdry	Emergency response procedures	Release of a fraction of the noble gases and halogens Workers: C-D Collocated workers: D Public: D Environment: D	3-4 4 4 4	None

Table B-6. Target Irradiation in the ACRR. (continued)

Event No.	Event Description	Causes	Preventative Features		Freq	Method of Detect.	Mitigative Features		Unmitigated Consequences	Risk Bin	Recommended Actions
			Design	Admin.			Design	Admin.			
TI-FR-17	<u>Loss of Electrical Power:</u> Electrical power (on-site or off-site) is lost resulting in all power to the console and cooling system being lost.	Off-site power outage Transmission line failure Circuit interruption due to power surge	Loss of power to reactor systems results in insertion (gravity fall) of all regulating rods to shut down reactor Decay heat removal does not require the cooling system to be operating	Visually verify after a power outage with the reactor operating that the regulating rods are all down	I	Visual	None	None	None Workers: D Collocated workers: D Public: D Environment: D	4 4 4 4	None
TI-FR-18	<u>Missile Impact:</u> A missile impacts the ACRR facility, which results in damage to the facility and debris falling into the pool, damaging fuel elements and targets leading to a release of fission products.	External event Crane collapse Bridge collapse Building collapse	Site isolation	None	IV	Visual	Fuel matrix retains fission products Reactor pool water solubility retains fission products Distance to site bdry	Emergency response procedures	Release of a fraction of the noble gases and halogens Workers: C-D Collocated workers: D Public: D Environment: D	3-4 4 4 4	None
TI-FR-19	<u>Airplane Crash:</u> An airplane crash occurs impacting the ACRR facility, which results in damage to the facility and debris falling into the pool, damaging fuel elements and targets leading to a release of fission products.	External event Crane collapse Bridge collapse Building collapse	Site isolation	None	IV	Visual	Fuel matrix retains fission products Reactor pool water solubility retains fission products Distance to site bdry	Emergency response procedures	Release of a fraction of the noble gases and halogens Workers: C-D Collocated workers: D Public: D Environment: D	4 4 4 4	None

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