15. ACCIDENT ANALYSES

15.0 Introduction

15.0.1 History

The Advanced Test Reactor (ATR) started power operation in mid-1969 within an envelope established in the Safety Analysis Report (SAR) and three supplements (deBoisblanc and Cohen 1965a; 1965b; 1967; deBoisblanc, Cohen, and Walker 1966). The SAR included descriptions of the facility, the reactor and its support systems, accident analysis, hazards analysis, and various appendices which described support for the SAR.

The SAR provided the basis for operation of the ATR at a maximum power level of 250 MW (thermal) with a core power imbalance[a] not to exceed 60-50-40 (relative MW). The term relative power is used to designate lobe powers normalized to 250 MW.

Program requirements in Fiscal Year 1974 and beyond identified the need for installation of an inpile test facility with a larger test space. A need for reactor operation with larger power imbalances was also identified. The larger power imbalance was required to allow operation of the reactor with high power in the lobes away from the large inpile tube (LIPT) with its limited power level capability. To determine the safety effects of operation with a LIPT, and to determine a maximum lobe operating power and power imbalance, the "ATR Operating Envelope Extension" program was initiated (ANC 1973). This program involved an extensive analysis effort to predict the power distributions in the reactor core, void worths and the distribution of the worth, and reactivity feedback coefficients. This was also the first time that the Monte Carlo technique for predicting the probability of a result and the variance on that result was used for evaluating compliance to safety margins for the ATR fuel element cladding. Previous studies had used the hot spot - hot stripe methodology.

This work demonstrated that the reactor could operate with an LIPT and with a greater power imbalance. The effects of power imbalance (e.g., reactivity variation with power distribution) were identified and enveloped in the analysis or limited as needed by operational controls.

The U.S. Atomic Energy Commission issued criteria and requirements for Reactor Development and Technology Division (RDT) reactor plant protection systems in 1969. An upgrade was started to meet the new standards contained in the standard RDT C.16-1T (AEC 1969) shortly after it was issued. In response, the "ATR PPS and Technical Specifications Design Basis Report" (DBR) was prepared and issued in 1977 (Atkinson and Hanson 1977) to provide a technical basis for the ATR Plant Protection System (PPS) and ATR Technical Specifications. This report identified the spectrum of credible reactor accidents of potential safety concern, classified the accidents by a qualitative frequency of occurrence, and analyzed the associated consequences to specify the mitigating PPS subsystems with their requirements. The DBR was prepared using the guidelines of the U.S. Nuclear Regulatory Commission's Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power

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[a] The ATR core consists of 40 fuel elements arranged in a clover-leaf pattern forming nine flux traps. The flux traps are in a 3×3 square array. The eight fuel elements surrounding each corner flux trap and the eight elements surrounding the center flux trap are referred to as lobes. The core control components can shift the azimuthal power distribution to/from non-uniform power distributions, or power imbalances. The lobe powers are specified as high-center-low when there is symmetry and as NW-NE-C-SW-SE, if symmetry does not exist. Refer to Chapter 4 (Reactor) for further description of core design and lobe power split designations.

Plants," (Chapter 15) (NRC 1978) and an American Nuclear Society (ANS) guide for the development of technical specifications for research reactors (ANSI 1974). The DBR was periodically updated with the last issue dated in 1994 (SAR-39).

As part of an effort to reduce costs, a study was initiated in 1978 to determine the feasibility of plant operation with two primary coolant pumps (PCPs). The method used to calculate electrical power costs for the Idaho National Laboratory (INL) was based on peak demand. Running a third PCP resulted in a peak demand which, in turn, resulted in a significant electrical power cost increase. Analyses supported two-pump operation, and the DBR and Technical Specifications were revised for operation with either two or three PCPs.

In 1989, a program plan was developed to provide an Upgraded Final Safety Analysis Report (UFSAR) for the ATR. At the time, new requirements for safety analysis reports for Department of Energy (DOE) nuclear facilities were being prepared. Draft requirement documents were available but were not finalized. As a result, the original plan was based on guidance for commercial reactor safety analysis reports. The plan was updated to produce a UFSAR that met the requirements of DOE O 5480.23, "Nuclear Safety Analysis Reports," when it was first issued in 1992. The current requirements for nuclear safety analysis reports are contained in DOE O 5480.23 and 10 CFR 830. The plan provided for upgraded evaluation of off-normal sequences in Chapter 15 of the UFSAR. The format and content for the first eight sections of Chapter 15, Sections 15.0 through 15.7, follows the NRC Regulatory Guide, RG 1.70, Format for Safety Analysis Reports. However, to make Chapter 15 comprehensive and stand-alone, Sections 15.8 through 15.16 includes information previously presented in the DBR (SAR-39). While this information is not necessarily required by RG 1.70 in a SAR, it provides useful material for understanding the present safety basis and provides additional documentation required by DOE O 5480.23. These additional sections contain analyses for failure of supporting systems, cask and fuel handling events, depressurized operation, fuel element failures, and high pressure and high flow events not previously discussed in Sections 15.1 through 15.7. The high flow and high-pressure events were discussed separately in the DBR and, since none of the events were limiting, the discussion of these events was also separated in Chapter 15. Finally, since the anticipated transients without scram were analyzed as part of the probabilistic risk assessment (PRA) severe accident analysis program, they are discussed in Section 15.12 with the severe accidents.

15.0.2 Methodology

The purpose of Chapter 15 in the UFSAR is to provide a current accident analysis and safety evaluation, identify the ATR safety criteria, and identify any safety areas requiring upgrade.

The approach used for accomplishing this objective involved four major areas: (a) provide physics data required for analyses that were fully documented and reviewed, (b) perform/model experiments to enhance and/or validate the correlations/codes used for ATR analyses, (c) reanalyze selected events to demonstrate compliance with the reactor protective criteria, and (d) provide an operating envelope within which reactor operation would be allowed upon demonstration that critical parameters met the analysis basis.

15.0.2.1 Physics Data. Extensive physics analyses were performed for the ATR core in support of the LIPT insertion and for the envelope extension. Capability for three-dimensional modeling with PDQ[a] was not available at the time this work was completed (two-dimensional results were extrapolated to three dimensions as needed). With the development of a three-dimensional PDQ, an extensive analysis effort was completed to update and verify the power distributions and other parameters required to develop content for Chapter 4 (Reactor) and data for the analyses in Chapter 15 (Accident Analyses). The analyses were reviewed and documented under the quality guidelines. The results, presented in Chapter 4, are used extensively in the discussion of off-normal events.

15.0.2.2 Experiments. Several experiments were performed during the initial reactor startup to provide verification of the reactor characteristics. The evaluations were performed for flow instability, flow reversal into natural convection cooling, oxide buildup, and forced convection heat transfer coefficients. The data from these tests and current reactor operation were used to benchmark the computer codes to the conditions (Wagoner 1995).

The possibility that subatmospheric pressure will occur in the reactor vessel during some transients was identified in previous studies. Since these conditions are outside the normal range evaluated in the derivation of correlations used to predict thermal-hydraulic behavior, a test program to experimentally evaluate critical heat flux (CHF), channel flooding, and flow excursion was started. Data from this program were used to evaluate applicability of correlations, or to generate new correlations.

15.0.2.3 Reanalysis. The DBR was reviewed along with results from the Reactivity Insertion Accident (RIA) and PRA programs to define transient analyses which should be reanalyzed for the UFSAR (Atkinson 1995). The identified scope for Chapter 15 of the UFSAR was the reanalysis of the limiting events associated with each category of accidents from the above programs. The DBR was used to identify the limiting event(s) in each accident category and with results from the RIA and PRA programs to ascertain the classification for event frequency. The identified limiting events were then analyzed to predict the reactor response. The analyzed response was then evaluated against the required protective margins.

The reanalyzed limiting events are in the following categories of potential off-normal events: loss of heat sink (LOHS), loss of flow, loss of coolant, reactivity/power anomalies, and high pressure (water hammer).

The reanalysis results and the existing DBR analysis were used to develop a discussion of the limiting events and bounded events.

15.0.2.4 Operating Envelope. Analyses were completed with reactor and experiment parameters/configurations which maintain the minimum required margins to the protection criteria. The evaluated range is expected to bound future operations; however, some case-basis evaluation of experiment loadings or reactor core configurations using the formalism and criteria established in Chapter 15 may be necessary. The ATR Core Safety Assurance Package (CSAP) and the experiment safety analysis documentation will provide verification that the proposed operation for each reactor cycle is within the bounds set by Chapter 15 (including any necessary analyses to show compliance to the reactor protection criteria established by Chapter 15).

[a.] PDQ is a reactor physics computer code; see Section 15.0.11.4 for a description of PDQ version 7.

15.0.3 Classification of Plant Conditions

The accident frequency classifications in the DBR were made using the guidelines contained in RDT C.16-1T (AEC 1969). Frequency classifications were: anticipated, unlikely, and extremely unlikely. The general qualitative criteria for classification of the ATR accident frequencies used in the DBR are listed in Table 15.0-1 (SAR-39). As discussed in Section 15.0.2.3 (Reanalysis), the accident frequency classifications of the reanalyzed events were ascertained from the results of the RIA and PRA programs. In addition, the results of the RIA and PRA programs confirmed the accident frequency classifications of the previously analyzed events. These classifications of frequency generally correspond to the ANS categories. The four ANS categories (Condition 1 through 4) and the assumed correspondence to the DBR classification are as follows:

- Condition 1 - Normal operation and operational transients: Normal Operation
- Condition 2 - Faults of moderate frequency: Anticipated
- Condition 3 - Infrequent faults: Unlikely
- Condition 4 - Limiting faults: Extremely Unlikely.

The basic principle applied in relating plant protective requirements to each of the conditions is that the most probable occurrences should yield the least radiological risk to workers and the public and the least risk to continued operation of the facility. Those extreme events having the potential for the greatest risk to workers and the public and to facility operability should be least likely to occur. Reactor trip and engineered safety system functioning is assumed as applicable.

The frequencies of accidents and faults listed in the sections for Conditions 2, 3 and 4 events were classified based on the DBR criteria or the results of the RIA and PRA programs as noted in Section 15.0.2.3.

15.0.3.1 Condition 1: Normal Operation. Condition 1 operations are expected to occur frequently or regularly in the course of reactor operation, refueling, or maintenance. Condition 1 includes normal operation and operational transients such as reactor startup, shutdown, recovery from power reductions, reactor power division adjustments, and experiment positioning. Maintenance activities such as refueling, experiment handling, facility modifications, overhaul, preventive and corrective maintenance, and core internals changeouts are normal operations. Reactor parameters are controlled within a conservative set of initial conditions which are bounded by reactor parameters assumed in the accident analyses. Maintenance activities are administratively controlled with the level of control commensurate with maintaining the assumed frequency of failure of the control.

A typical list of Condition 1 events follows:

A. Steady-State and Shutdown Operations.

1. Power Operation (N_L to N_F).
2. Shutdown (subcritical, safety rods in seat) maintenance.

B. Operation with Permissible Deviations.

Deviations that may occur during continued operation as permitted by the Technical Safety Requirements (or other formal contractor documentation) must be considered in conjunction with other operational modes. These include:

1. Operation with components or systems out of service (such as an inoperable safety rod).
2. Operation with an initial condition outside of tolerance.
3. Leakage from fuel or experiment with a limited clad defect.
4. Operation with water chemistry out of specification.
5. Excessive radioactivity in the primary coolant.
6. Testing.

C. Operational Transients.

1. Plant startup and shutdown.
2. Experiment repositioning.
3. Adjustment of power division.

15.0.3.2 Condition 2: Anticipated Faults. An anticipated fault is an off-normal condition that is expected to occur once or more during the lifetime of the facility due to an expected single fault. These faults, at worst, result in reactor trip with no plant equipment damage [some plant equipment (e.g., reactor fuel elements) may require requalification prior to reuse]. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition 3 or 4 events. Failure of the fuel clad is included in this category only if the rupture is localized and is the initiating event.

For the purposes of this report, the following events are included in this category:

- Secondary coolant system (SCS) flow control valve (FCV) fails to the open position (Section 15.1.1)
- Excessive heat removal from the SCS (Section 15.1.3)
- SCS FCV closed against stop (Section 15.2.2.1)
- Cooling tower failure (Section 15.2.2.6)
- Loss of commercial power [Sections 15.2.2.8 (loss of heat sink) and 15.3.1 (loss of primary flow) (see also supporting system events in various Sections of 15.11)]
- Trip of one (two-pump operation) or two (three-pump operation) of the operating PCPs (Section 15.3.5.1)
- Sleeve failure in 16-in. outlet pipe (Section 15.3.5.2)
- Failure of one operating PCP due to mechanical failure (Section 15.3.5.1)

- Closure of the primary coolant system (PCS) butterfly valve from operating positions to 38% (Section 15.3.5.3)
- Withdrawal of all outer shims and neck shims from 10^{-5} N_F (Section 15.4.2)
- Slow power increase or lobe power balance shift due to shim or operator failure (Section 15.4.3)
- Inpile tube (IPT) decompression with subsequent voiding due to failure of the 1/2-in. piping (Section 15.4.4)
- Rapid regulating rod withdrawal (Section 15.4.7.1)
- Withdrawal of all outer shims from 10^{-10} N_F (Section 15.4.7.2)
- Withdrawal of all outer shims and one neck shim from N_L (Section 15.4.7.3)
- Cold water injection (Section 15.4.7.4 refers to Section 15.1)
- IPT voiding due to a rupture disk or relief valve failure or an open and accessible valve (reactivity insertion in Section 15.4.7.5, dose consequences in Section 15.7.1)
- Limited perched fuel element (Bounded by 0.5$ step reactivity insertion in Section 15.4.5)
- Loop flow coastdown or loop loss of temperature control with a loop instrumentation initiated reactor trip (Section 15.4.7.6)
- Powered axial locating mechanism (PALM) drive system failures (Section 15.4.7.7)
- A loss of instrument air closes back-pressure control valve (PCV) PCV-1-1 and opens pressurizing flow control valve FCV-1-8 (Section 15.5.1)
- Back pressure valve PCV-1-1 closes due to operator error or controller failure (Section 15.5.3.1)
- Pressurizing flow control valve FCV-1-8 opens due to operator error or controller failure, or a second pressurizing pump is started (Section 15.5.3.2)
- Excessive high pressure air (HPA) is valved into the surge tank (Section 15.5.3.3)
- Diesel generator overspeed (≤ 15%) causes pressurizing pump overspeed and increased flow (Section 15.5.3.4)
- Heat exchanger tube rupture (Section 15.6.3)
- An opening in the PCS due to opening a drain valve, relief valve, or vent valve (Section 15.6.4)
- ATR Liquid Waste System release (Section 15.7.2)
- Dropping an irradiated fuel element during fuel handling in either the reactor vessel or in the working canal with no significant damage to the fuel element (Section 15.8.5)
- Withdrawal of all outer shims from 10^{-5} N_F [during depressurized operation] (Section 15.9.3.1)
- Partial fuel channel blockage from small debris in the PCS (Section 15.10.1)
- Unexpected high peaking in the fuel due to an analysis or loading error (Section 15.10.8)

- Small defects in fuel element or minor handling damage to fuel (Section 15.10.9)
- Failure of one power supply unit in Radiological Control Office (Section 15.11.1.1)
- [Failure of cubicle exhaust system due to] [a] Loss of commercial power (Section 15.11.2.1)
- [Failure of cubicle exhaust system due to] Loss of diesel power (Section 15.11.2.2)
- [Failure of cubicle exhaust system due to] Momentary loss of commercial-diesel power (Section 15.11.2.3)
- High high-pressure demineralized water (HDW) coolant temperature (Section 15.11.4.1)
- Partial loss of HDW flow to IPT cubicles (Section 15.11.4.3)
- Momentary total loss of HDW flow to IPT cubicles (Section 15.11.4.5)
- Single utility cooling water (UCW) pump failure (Section 15.11.5.1)
- Inadvertent closure of flow control valve FCV-4-5 (Section 15.11.5.3)
- [Failure of UCW due to] Loss of diesel power (Section 15.11.5.9)
- Inadvertent error in HDW-UCW heat exchanger bypass flow (Section 15.11.5.13)
- [Failure of Low pressure demineralized water (LDW) due to] Block valve closure (Section 15.11.6.1)
- LDW pump failure (Section 15.11.6.2)
- Failure of instrument air system (Section 15.11.7)
- Excessive HPA in surge tank due to operator error (Section 15.11.8.1)
- Excessive HPA in surge tank due to valve failure (Section 15.11.8.2)
- Failure to place one HPA compressor on automatic control (Section 15.11.8.3)
- [HPA] Control system failure (Section 15.11.8.4)
- [HPA] Relief valve failure (Section 15.11.8.5)
- Loss of power from any one motor control center (MCC) or bus, including the diesel-commercial bus (Section 15.11.9.1)
- Momentary total commercial power failure (for less than one hour) (Section 15.11.9.2)
- Loss of power from the 250 V dc utility bus (Section 15.11.9.4)
- Momentary diesel power failure (until the standby diesel generator is started) (Section 15.11.9.5)
- Momentary loss of diesel power with a loss of commercial power and a standby pump fault which causes a diesel bus overload (Section 15.11.9.6)

[a.] Square brackets indicate wording provided for clarification that does not appear in the event title in the referenced section.

- Loss of power to the instrument ac bus or the utility ac bus (Section 15.11.9.7)
- Commercial power overvoltage or overfrequency < 3% (Section 15.11.9.8, Section 15.14.4.1)
- Diesel generator overspeed (diesel power overfrequency < 15%) (Section 15.11.9.10)
- 27X relay in 4160 V commercial switchgear energized without an under-voltage condition (Section 15.11.9.11)
- Dropped or stuck safety rod (Section 15.11.11.1)
- Neck shim control rod unintentionally partially withdrawn (Section 15.11.12.1)
- Neck shim withdrawn or inserted, but indicated to be in the opposite position (Section 15.11.12.2)
- Mispositioning of one set of outer shim control cylinders associated with high power lobe (Section 15.11.13.1)
- Opening of flow control butterfly valve to full open or the starting of third PCP with two PCP already in operation (Section 15.14.1).

15.0.3.3 Condition 3: Unlikely Faults. Condition 3 events are faults that may occur infrequently during the life of the plant. As shown in Table 15.0-1, Condition 3 faults result from single low frequency failures and combinations of independent high frequency failures. By definition a Condition 3 event will not propagate to cause a Condition 4 event or result in damage to reactor components not readily replaceable.

- Starting a standby PCP (Section 15.1.2)
- Failure of secondary pumps (Section 15.2.2.2)
- Failure of cooling tower fans (Section 15.2.2.4)
- Loss of cooling tower basin water (Section 15.2.2.5)
- Excessive heat exchanger tube fouling (Section 15.2.2.7)
- Loss of secondary flow from either a secondary flow control valve complete closure or closure of all six motor operated cooling tower return valves (Section 15.2.2.3)
- Slow closing primary pump discharge check valve (Section 15.3.2)
- Complete closure of the flow control butterfly valve (Section 15.3.3)
- Condition 3 events bounded by 0.50$ step reactivity insertion (Section 15.4.5)
    - Loop experiment hardware failure
    - Loss of reflector coolant
    - Reflector movement toward core
    - Perched fuel drop of two fuel elements from within measurement accuracy

- Cold water addition from opening of primary pump discharge valve
- Gas release into the core.

- Withdrawal of all outer shims and neck shims from 10^{-10} N_F (Section 15.4.7.8)
- Inpile tube voiding due to opening of a normally inaccessible valve (Section 15.4.7.9)
- Pressure tube flow bypass due to a maximum useful capacity holder (MUCH) test train failure with loop scram (Section 15.4.7.10)
- Loss of loop temperature control due to heat exchanger failure or line heaters sticking on without loop scram (Section 15.4.7.11)
- Slow lobe power balance shift due to shim, lobe power indicating system or operator failure with operator compensation (Section 15.4.7.12)
- FCV-1-8 fails to close when PCV-1-1 fails closed or PCV-1-1 fails to respond to increased pressurizing flow (Section 15.5.3.6)
- Rupture of PCS outside radiographic limits (Section 15.6.4)
- A loss of coolant accident (LOCA) equivalent to a three-in. opening in the PCS with a concurrent outage of one of the Emergency Firewater Injection systems (Section 15.6.5)
- A LOCA equivalent to a three-in. opening in the PCS with a concurrent failure of one primary coolant pump to trip in response to the pump trip ESF (Section 15.6.5)
- Very small seismic LOCA (Section 15.6.7)
- Two PCS relief valves fail open (Section 15.6.9.1)
- Two vessel vent valves fail open (Section 15.6.9.2)
- Opening valve outside radiographic limits or at radiographic boundary [during reactor shutdown] (Section 15.6.10.1)
- Piping rupture outside radiographic limits [during reactor shutdown] (Section 15.6.10.2)
- Multiple drain valves opened [during reactor shutdown] (Section 15.6.10.3)
- Effluent Processing waste tank (TRA-605) release (Section 15.7.3)
- Minor damage to one fuel element in the canal, with a minor fission product release (Section 15.8.6)
- Dropping a heavy cask from an elevation of less than one ft above the canal floor, limited PCS leakage with the drop chute cover removed or other small or limited failure of the storage canal (Section 15.8.8)
- Dropping a heavy cask from one ft above the parapet within the restricted cask-lifting areas of the canal (Section 15.8.9)
- Dropping a heavy cask onto the floor north of the canal (Section 15.8.10)

- Dropping a cask onto the canal cask transfer station from within the lift height limit (Section 15.8.11)
- Dropping a cask onto the vessel or shielding over and around the vessel from within the lift height limit (Section 15.8.13)
- Dropping a cask onto the main floor not above a loop cubicle or above the circular structural and shielding wall at the 98-ft elevation (Section 15.8.15)
- A cask lowering failure over the ATR vessel in which an experiment cask impacts the transfer plate at a fast lowering speed (Section 15.8.16)
- A cask lowering failure over the canal experiment cask transfer station resulting in the experiment cask impacting the station at a fast lowering speed (Section 15.8.17)
- Momentary loss of flow due to an emergency coolant pump failure or inadvertently turned off [during depressurized operation] (Section 15.9.1.1)
- Opening valve outside radiographic limits or at radiographic boundary [during depressurized operation] (Section 15.9.2.1)
- Rapid reg-rod withdrawal [during depressurized operation] (Section 15.9.3.2)
- Withdrawal of all neck shims and outer shims from 10^{-5} N_F [during depressurized operation] (Section 15.9.3.3)
- Withdrawal of all outer shims from 10^{-10} N_F [during depressurized operation] (Section 15.9.3.4)
- Cold loop draining due to open and accessible valve [during depressurized operation] (Section 15.9.3.5)
- Loop heaters accidentally on [during depressurized operation] (Section 15.9.3.6)
- Slow power increase or lobe power balance shift [during depressurized operation] (Section 15.9.3.7)
- Perched fuel drop from within measurement accuracy (1 in.) [during depressurized operation] (Section 15.9.3.8)
- Cold water addition due to inadvertent startup of a PCP or inadvertent actuation of emergency firewater injection [during depressurized operation] (Section 15.9.3.9)
- Fuel channel blockage from a sight glass or other inventoried items left in the reactor system after a shutdown (Section 15.10.2)
- Fuel channel blockage due to failure of small structural elements in the ATR vessel above the core or inlet screen failure (Section 15.10.2)
- Gas leakage into the fuel element coolant channels due to an insulating jacket failure during normal pressurized operation (Section 15.10.4)
- Structural fuel element defects affecting a few fuel elements in the core due to design or fabrication errors or handling damage (Section 15.10.5)

- Fuel plate buckling due to an excessive buildup of oxide on fuel plate surfaces (Section 15.10.7)
- [Radiation Monitoring System (RMS) failure due to] momentary loss of diesel and commercial power (Section 15.11.1.2)
- [Failure of cubicle exhaust system due to] mechanical failure of the system (Section 15.11.2.4)
- [Failure of cubicle exhaust system due to] inadvertent shutoff of the system (Section 15.11.2.5)
- [Failure of the evacuation system due to] momentary complete loss of power to the system (Section 15.11.3.1)
- Extended total loss of HDW flow to IPT cubicles (Section 15.11.4.6)
- Single UCW pump failure with concurrent loss of commercial power (Section 15.11.5.5)
- Simultaneous failure of the operating and the standby UCW pumps (Section 15.11.5.11)
- Extended commercial power failure (more than one hour) (Section 15.11.9.3)
- Failure of PPS battery-backed power (Section 15.11.9.14)
- [Firewater system] piping failure (Section 15.11.10.1)
- Inadvertent break in the [firewater] supply lines during construction or excavation (Section 15.11.10.2)
- Mispositioning of a pair of outer shim cylinders associated with the high power lobe and simultaneous mispositioning of a pair of outer shim cylinders associated with the low power lobe (Section 15.11.13.2)
- A primary pump discharge valve or check valve closes abruptly (Section 15.13.1)
- The FCV (butterfly) fails (Section 15.13.2)
- An emergency coolant pump discharge check valve is open with PCPs in operation (Section 15.13.3.1)
- Startup of a PCP against a fully closed discharge valve with suction pressure significantly higher than 230 psig (Section 15.13.3.2)
- Startup of a fourth PCP (Section 15.14.2)
- Significant leakage out of the top closure plate due to failure of a loop closure assembly seal (Section 15.6.8).

15.0.3.4 Condition 4: Extremely Unlikely Faults. Condition 4 events are low-probability faults that are not expected to occur but are postulated because their consequences include the potential for release of significant quantities of radioactive material. A single Condition 4 event is not to cause a consequential loss of systems required to cope with the fault. The following faults have been classified in this category:

- SCS rupture upstream of primary coolant heat exchangers (HXs) (This event is also called complete loss of secondary and complete loss of heat sink in the text) (Section 15.2.1)

- Long-term loss of commercial power (Section 15.2.2.8)
- Extended loss of commercial, diesel, and utility DC power (Section 15.2.2.9)
- Long-term complete loss of flow (Section 15.3.4)
- Closure of a 24-in. reactor inlet isolation valve (Section 15.3.5.4)
- Rupture of a 16-in. outlet pipe inside the ATR vessel (Section 15.3.5.5)
- Failure of the first actuating PPS subsystem during Condition 2 event (Section 15.3.5.6)
- IPT decompression with subsequent voiding due to failure of piping >1/2-in. in diameter (reactivity insertion in Section 15.4.6, dose consequences in Section 15.7.1)
- Withdrawal of all shims and safety rods from 10^{-10} N_F (Section 15.4.7.13)
- Withdrawal of all outer shims from 10^{-10} N_F with failure of the wide range subsystem (WRS) (Section 15.4.7.14)
- Withdrawal of all outer shims and neck shims from 10^{-5} N_F with failure of the WRS (Section 15.4.7.15)
- Rapid regulating rod withdrawal with failure of the WRS (Section 15.4.7.16)
- Withdrawal of all outer shims and neck shims from 10^{-5} N_F coupled with rapid regulating rod withdrawal at 10^{-3} N_F (Section 15.4.7.17)
- Driven test loop blowdown with experiment hardware failure (Section 15.4.7.18)
- Voiding in all IPTs due to a simultaneous flow coastdown in all loops (Section 15.4.7.19)
- Loss of loop temperature control in the ATR high temperature loop (AHTL) due to heat exchanger failure or line heaters stuck on without a loop scram (Section 15.4.7.20)
- Loop flow coastdown without a loop scram (Section 15.4.7.21)
- Pressure tube cooling flow bypass due to a MUCH test train failure without a loop scram (Section 15.4.7.22)
- Perched fuel element (greater than measurement uncertainty) (Bounded by 0.5$ step reactivity insertion in Section 15.4.5)
- Movement of two reflector blocks (Bounded by 0.5$ step reactivity insertion in Section 15.4.5)
- Voiding in all IPTs due to a simultaneous loss of temperature control in all loops (Section 15.4.7.19)
- Loss of instrument air closes PCV-1-1 and opens FCV-1-8 with concurrent failure of the inlet pressure subsystem (Section 15.5.2)

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- A LOCA event equivalent to a three-in. opening in PCS with a concurrent failure of the inlet pressure subsystem (Section 15.6.5)
- A LOCA equivalent to a three-in. opening in the PCS with a concurrent failure of one primary coolant pump check valve to close (Section 15.6.5)
- A seismic-induced LOCA (bounded by a two-in. opening in the PCS with a concurrent failure of ac power) (Section 15.6.6)
- PCS rupture due to a crack inside the radiographic limits (Section 15.6.9.3)

FOIA Exemption 3 UCNI, Exemption 7f

- A large failure of the storage canal, canal bellows or a rapid draining of the PCS with the drop chute cover removed (Section 15.8.4)
- Significant melting of one entire fuel element (or partial melting in more than one fuel element) due to crushing or other accidents (Section 15.8.7)
- Dropping a cask onto the canal cask transfer station from above the height limit (Section 15.8.12)
- Dropping a cask onto the main floor above a loop cubicle or above the circular structural and shielding wall at the 98 ft elevation from above the height limit. (Section 15.8.14)
- Extended loss of flow due to extended failure of both emergency coolant pumps [during depressurized operation] (Section 5.9.1.2)
- Butterfly valve closes when shaft fails or valve is closed and stop fails [during depressurized operation] (Section 5.9.1.3)
- Core flow bypass [during depressurized operation] (Section 5.9.1.4)
- Piping rupture outside radiographic limits [during depressurized operation] (Section 5.9.2.2)
- Multiple drain valves opened [during depressurized operation] (Section 5.9.2.3)
- Pressure tube penetration or bottom head closure plug failure [during depressurized operation] (Section 5.9.2.4)
- Withdrawal of all outer shims and neck shims from 10^{-10} N_F [during depressurized operation] (Section 5.9.3.10)
- Cold experiment loop draining due to an open normally inaccessible valve [during depressurized operation] (Section 5.9.3.11)
- Step reactivity insertions [during depressurized operation] (Section 5.9.3.12)

- Fuel channel blockage or fuel damage due to the failure of large structural elements above or within the core (Section 15.10.3)
- Structural fuel element defects affecting many fuel elements in the core due to design or fabrication errors or severe handling damage (Section 15.10.6)
- [RMS failure due to] extended loss of diesel and commercial power (Section 15.11.1.3)
- [Failure of the evacuation system due to] extended complete loss of power sources to the system (Section 15.11.3.2)
- High HDW coolant temperature with concurrent IPT experiment coolant instrumentation failure (Section 15.11.4.2)
- Partial loss of HDW flow to IPT cubicles with concurrent IPT experiment coolant instrumentation failure (Section 15.11.4.4)
- Momentary total loss of HDW flow to IPT cubicles with concurrent IPT experiment coolant instrumentation failure (Section 15.11.4.7)
- Extended total loss of HDW flow to IPT cubicles with concurrent IPT experiment coolant instrumentation failure (Section 15.11.4.8)
- Single UCW pump failure with concurrent failure of the IPT experiment coolant instrumentation (Section 15.11.5.2)
- Inadvertent closure of flow control valve FCV-4-5 with concurrent failure of the IPT experiment coolant instrumentation (Section 15.11.5.4)
- Single UCW pump failure with concurrent loss of commercial power and failure of the IPT experiment coolant instrumentation (Section 15.11.5.6.)
- Extended single UCW pump failure with concurrent loss of commercial power (Section 15.11.5.7)
- Extended single UCW pump failure with concurrent loss of commercial power and failure of the IPT experiment coolant instrumentation (Section 15.11.5.8)
- Loss of diesel power with concurrent failure of the IPT experiment coolant instrumentation (Section 15.11.5.10)
- Simultaneous failure of the operating and standby UCW pumps with concurrent failure of the IPT experiment coolant instrumentation (Section 15.11.5.12)
- Inadvertent error in HDW-UCW heat exchanger bypass flow with concurrent failure of the IPT experiment coolant instrumentation (Section 15.11.5.14)
- Extreme commercial power overfrequency (will not exceed 10% with a duration of a few seconds) (Section 15.11.9.9)
- Momentary (15 min to 1 hour) complete loss of power combinations (loss of commercial and diesel power) (Section 15.11.9.12)
- Extended (longer than 1 hour) complete loss of power combinations (loss of commercial and diesel power) (Section 15.11.9.13)

- Commercial power failure with sufficiently high demand on firewater system (Section 15.11.10.3)
- Commercial and commercial-diesel power failure with sufficiently high demand on firewater system (Section 15.11.10.4)
- Firewater pump mechanical failure with sufficiently high demand on firewater system (Section 15.11.10.5)
- Simultaneous failures of position indicator, limit switches and interlocks, and seat switch (Section 15.11.11.2)
- Simultaneous failures of position indicator, limit switches, and safety rod interlocks (Section 15.11.11.3)
- Twenty-two neck shim rods 50% withdrawn (Section 15.11.12.3)
- Extreme commercial power overfrequency >3% (Section 15.14.3).

15.0.4 Plant Characteristics and Initial Conditions Assumed in Accident Analyses

15.0.4.1 Principal Plant Conditions. Table 15.0-2 lists the principal plant operating conditions and configurations important to the reactor safety analysis. The presentation of the data in Table 15.0-2 requires some discussion. Some data are nominal or typical (e.g., number of safety rods) while other data are limiting maximum or minimum values. The listing reflects data for pressurized and depressurized operation and also has information for pressurized operation with two or three PCPs in operation.

15.0.4.2 Initial Conditions. For accident evaluation, the initial conditions (reactor power, core inlet temperature, vessel inlet pressure, and core differential pressure for pressurized operation; reactor power, vessel inlet temperature, and core flow for depressurized operation) are derived from the principal plant conditions shown in Table 15.0-2. Initial conditions and other plant parameters and conditions were selected to provide a conservative analysis for the event being analyzed. For example, minimum safety rod reactivities are used to model the safety rod scram reactivity while maximum safety rod reactivities are used in the analysis of safety rod withdrawal accidents. Because the values selected vary for different events, the data in Table 15.0-2 are provided for information only. Uncertainties in the rated values are considered in the Monte Carlo sampling during analyses for compliance to reactor protection criteria.

15.0.4.3 Power Distribution. The transient response of the reactor system is dependent on the initial power distribution. The design of the reactor core utilizes shims adjustable during operation to establish a power distribution which is amenable to the experimenter requirements for each cycle of operation. The power peaking in the fuel elements is dependent on the core power distribution. The power peaking factors for three lobe power splits representing typical operation are contained in Figures 4.3-4 through 4.3-27 of Chapter 4 (Reactor). The power distributions are normally displayed as point-to-average power densities; these densities are also referred to as peaking factors. The point-to-average power density is defined by $(\gamma\Sigma_f\phi)/(\gamma\Sigma_f\phi)_{ave}$ where γ is the energy release per fission, Σ_f is the macroscopic fission cross section, and ϕ is the neutron flux resulting in fissions. The denominator is an average over the core.

Since there is essentially no variation in flow distribution among the forty fuel elements, the peak temperature will occur at the location of peak power. The peak power occurs in the lobe with the highest power. The power must be unbalanced to obtain the maximum lobe (and local) power. Therefore, the enveloping choice for fuel plate temperature evaluations is the unbalanced power distribution. The fuel plate temperature analysis for the off-normal events is completed using the 70/20 (SAR 70) power distribution for operation with three coolant pumps and the 60/40 (E60/40) power distribution for operation with two coolant pumps. As an additional conservatism, the fuel element power distributions included the effects of voiding the worst-case loop facility IPT. These power distributions (shown in Figures 15.0-1 through 3) are used in the ATR-SINDA[a] model to determine temperature distributions in the fuel plate. The more balanced power (BAL50/50) provides a conservative feedback from heating the fuel region and is used in the RELAP5[b] model for evaluation of power transients (Terry 1994).

Several power-distribution parameters have been defined for the ATR. The first is the product of the total core power and the point-to-average power density; the product is the "effective point power." (Note that this quantity would be the point power density if the core average power density had been used in the product rather than the total core power.) The second defined parameter is the "average axial peaking factor"; it is the average value of the axial peak-to-average power density for the fuel plate normalized to a peak value of one (Polkinghorne 1994b). The third parameter is the product of the effective point power and the average axial peaking factor; this product is the "effective plate power."

The peak value for the point-to-average power density occurs near the core midplane. This peak value will change through the core lifetime because the axial distribution will flatten as the uranium depletes (Figures 15.0-1 and 15.0-2 indicate the flattening). Since this change in axial distributions affect the peak temperature of the cladding, it is necessary to consider the power distributions at all times in the operating cycle. In order to ensure that the effect of the changing axial distribution is considered in the evaluation of the operation against the analysis, a limit is established including the effect of the axial distribution.

As noted above, the effective point power differs from the point power density since it (the effective point power) is not divided by the volume of the core. As a result, the value of the parameter will be large relative to other parameters such as the core power. However, the effective point power has been in use for the ATR for some time to establish limits on operation. Similarly, the effective plate power which is obtained from the effective point power will be large relative to other parameters. Although the magnitude of these parameters may appear unrealistic, they represent an effective means of conservatively analyzing and limiting the operation of ATR.

15.0.5 Reactivity Coefficients Assumed in the Accident Analyses

The transient response of the reactor is dependent on reactivity feedback effects. The fueled portion of the reactor core has negative temperature and void coefficients, while the flux traps have positive temperature and void coefficients. Nominal reactivity coefficients are discussed in Chapter 4 (Reactor).

[a.] ATR-SINDA is a thermal-hydraulics computer model; see Section 15.0.11.2 for a description of the ATR-SINDA model.

[b.] RELAP5 is a thermal-hydraulic computer code; see Section 15.0.11.1 for a discussion of RELAP5.

In the evaluation of the accident sequences, the negative feedback reactivity was included using conservative coefficients (Terry 1994). The feedback shows some dependence on power division; the more the unbalance in the power division, the stronger the feedback from the fuel element region. With the exception of the flow coastdown evaluation, the analysis assumed feedback with either a balanced power division or no feedback at all. For the loss of flow events, the negative feedback corresponded to the power unbalance being considered (either the SAR 70 or the 60/40). However, the effect on the power was only a slight decrease in the total (on the order of seven percent) before the scram occurred. With such a small effect, the difference for the more realistic feedback used and the conservative feedback from a balanced power is not considered significant.

15.0.6 Safety Rod Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the axial position of the safety rod at the time of trip, acceleration of the safety rods, and safety rod worth as a function of core position. For accident analysis, the critical parameter for safety rods is the time for insertion of reactivity to make the core subcritical or begin to balance a positive condition. For ATR, this reactivity is typically realized within the first 12 in. or approximately 33% of the safety rod travel. A trip signal to the safety rod subsystem will release the safety rods within 25 msec for pressurized operation and within 35 msec for depressurized operation. The safety rods will insert 12 in. into the core and effectively terminate power operation within 150 msec (for three pump operation) and 170 msec (for two pump operation). The safety rods will insert 12 in. into the core within 300 msec for depressurized operation.

Test data with two and three PCPs in operation are shown in Figure15.0-4 (Durney 1993). The two-pump data are generated assuming a total worth of 6.4$, and the three-pump data are generated assuming a total worth of 5.8$. Also shown is the enveloping reactivity curve which was used for the Chapter 15 analyses. The data for the enveloping curve shown in this figure is listed in Table 15.0-5; the data used for the safety rod worth in depressurized operation is also contained in Table 15.0-5. The total worth of the safety rods assumed for the analysis inserts 5.8$ in reactivity. The worth shown in other UFSAR sections show nominal or actual worths which are greater and demonstrate some of the conservatism in the analysis.

Axial power skewing is not a factor in the ATR since the neck shims are operated full in or full out, and the outer shim control cylinders rotate a uniform axial poison toward or away from the reactor fuel. There is a small local perturbation near the operating regulating rod since it is only partially withdrawn.

The analyses used the reactivity data in Table 15.0-5 for the safety rod worth, and other sources of positive or negative reactivity were summed using additional reactivity tables as allowed by RELAP5. Specific input data are included in the analysis discussion.

15.0.7 Trip Points and Time Delays to Trips Assumed in Accident Analysis

A signal from the PPS detectors will cause a trip of the 2 out of 3 logic modules when the setting is exceeded (either high or low depending upon the particular subsystem). The output from these modules provide signals to the rod clutch coil controllers which are configured as a 2 out of 4 system. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in module trip, and in release of the magnets and the safety rod mechanisms. The total delay to trip is defined as the time delay from the time trip conditions are reached to the time the rods are free and begin to insert. Limiting (worst case) trip setpoints assumed in the accident analyses, and the assumed response time (delay time) for each trip function are listed in Table 15.0-6.

15.0.8 Plant Systems and Components Available for Mitigation of Accident Effects

The plant is designed to afford proper protection against the possible effects of natural phenomena, postulated environmental conditions, and dynamic effects of the postulated accidents. In addition, the design incorporates features which minimize the probability and effects of fires and explosions. The incorporation of these features in the plant, coupled with the reliability of the design, ensures that the normally operating systems and components listed in Table 15.0-7 are available for mitigation of the events discussed in Chapter 15.

In the analysis of the Chapter 15 events, control system action is generally considered only if that action results in more severe accident results. However, some partial control system failures are considered in order to obtain a full spectrum of events; in these cases, the control system action is considered in evaluation of the events. For some accidents, the analysis is performed both with and without control systems operation to determine the worst case. Specific examples will be discussed in the analysis assumptions. In the case of sequences involving the experiment loops, the determination of frequency of occurrence considers the action of the control system.

15.0.9 Fission Product Inventories

The time-dependent fission product inventories in the reactor core are calculated by the ORIGEN[a] code using a data library based on ENDF/B-IV. Core inventories are shown in Table 15.12-7 for operation at 250 MW_t for 60 days. The time-dependent fission product inventories in the canal were calculated by the ORIGEN code. The results are discussed in Section 15.12.9. The inventories were calculated using a conservative operating schedule to maximize the inventory of fission product that would be available for release (McCracken 1986). The time-dependent fission product inventories for experiments contained in the IPTs were based on calculations using ORIGEN code. The results are documented in Section 15.7.

[a.] ORIGEN is a fission-product inventory computer code; see Section 15.12.10.2 for a discussion of ORIGEN.

15.0.10 Residual Decay Heat

Decay heating in the subcritical core is calculated from the RELAP5 point kinetics model. The selected model is the ANSI/ANS 5.1 (1979). An infinite irradiation time is assumed. Limiting events were reanalyzed using a decay heat multiplier of 1.0. Transients with substantial thermal margins were analyzed with a multiplier of 1.2. Since power is one of the parameters that is treated statistically by SINDA-SAMPLE (Liebelt 1993; Polkinghorne 1994a), the additional 20% decay heat provided by the 1.2 multiplier was not necessary. However, to avoid unnecessary costs and due to the large margins available with the 1.2 multiplier, those cases using the 1.2 multiplier with acceptable results were not rerun.

15.0.11 Computer Codes Utilized

An extensive validation and verification effort was performed for the codes used to establish the thermal limits for ATR. These efforts are summarized in Wagoner (1996). Efforts were also taken to (1) ensure retrieval of the analysis completed for Chapter 15 and (2) to maintain configuration control of models and codes used in the analysis. The approach taken to ensure this retrieval is described in Wagoner (1995). Calculations performed using these computer codes were reviewed by the analysts, a peer not involved in the analysis, and approved by management (Wagoner 1995) to ensure reliability of the results. Summaries of the principal computer codes used in reanalysis of transients for the UFSAR are given below; additional descriptions are contained in Chapter 4 (Reactor) of the UFSAR. As discussed in Section 15.0.2.3, the existing DBR (SAR-39) analyses was used to develop the discussions of limiting events. The DBR analyses typically used earlier versions of the computer codes discussed below. The computers codes used in the DBR analyses are discussed in the DBR and related references. The codes used in the analyses are listed by section in Table 15.0-8. There are no codes in Table 15.0-8 listed for Section 15.16 which is excerpted from the PRA (Eide, Khericha, and Thatcher 1991; Thatcher et. al. 1994) which summarizes the codes used. There are no codes in Table 15.0-8 listed for Section 15.12 which discusses the results of the severe accident analyses. The codes used in the severe accident analyses are discussed in Section 15.12.

15.0.11.1 RELAP Computer Code.

15.0.11.1.1 RELAP5 Computer Code —RELAP5/MOD2.5 is a system code for the analysis of transient, one-dimensional, single- and two-phase fluid dynamics (Allison et al. 1989). The code includes the nonhomogeneous and nonequilibrium characteristics of two-phase flow. The types of transients that can be analyzed include loss of coolant, pump failure, loss of heat sink, and reactivity initiated events. To validate its use in the ATR Safety analysis, an ATR RELAP5/MOD2.5 input model was benchmarked against in-core hydraulic data, flow coastdown data, and steady-state operating data (Lenglade and Pafford 1993).

RELAP5 represents the reactor or experiment loop facility systems as connected fluid volumes that describe plenums, piping, pumps, core (reactor), test region (loop) and heat exchangers. Calculated parameters are mass flow, mass inventory, energy inventory, pressure, temperature, and quality. Also evaluated are reactor power, heat transfer rates and mode, and response to control systems input.

Additional details for RELAP5 can be found in the SCDAP/RELAP5/MOD2 Code Manual (Allison et al. 1989).

15.0.11.1.2 RELAP4 Computer Code —RELAP4 (MOD3 and MOD5) is an earlier version of the RELAP computer code which was used previously for analysis of ATR transients and behavior (Atkinson and Hanson 1977). Some of those analyses are still valid, e.g., Section 15.5.1, A Loss of Instrument Air Closes Back-Pressure Control Valve PCV-1-1 and Opens Pressurizing Flow Control Valve FCV-1-8.

Further information on the earlier model and results is available (Varacalle, 1975, 1976). General information on the RELAP4 code can be found in Moore and Rettig (1973).

15.0.11.2 ATR-SINDA Computer Model. ATR-SINDA is a computer model with three-dimensional heat conduction and one-dimensional hydraulic capability (Ambrosek, Liebelt, and Merrill 1993). The model has the capability of analyzing heat transfer processes in water coolant during all transient conditions, including:

1. Subcooled convection for natural circulation, transition, and forced flow conditions

2. Subcooled nucleate boiling for both transition and fully developed regimes

3. Transition-to-dryout and film boiling conditions.

The hydraulic model is capable of predicting steady state and transient behavior, including hydraulic instability and flow reversal.

SINDA, a generalized finite-difference thermal analyzer code, is used to model the system as a three-dimensional construction. One steady state solver and one transient solver are used from those in SINDA to solve the conduction heat transfer, while user subroutines model the hydraulics. The user subroutines consist of three basic parts:

1. Water property routines assembled and used throughout INL for various codes including RELAP5;

2. The RELAP5 PILOT code, a transient, one-dimensional homogeneous fluids code, which solves the conservation equations for momentum, density, and energy in the fluid;

3. Heat transfer routines written to incorporate all of the flow conditions discussed above.

The ATR-SINDA model consists of a portion of a single fuel plate and a segment of the side plate to which it is attached. This structure is surrounded by three flow channels, one on each side of the fuel plate, and one adjacent to the side plate. The nodalization and schematic for one of the nine axial levels is shown in Figure15.0-5. As shown in Figure15.0-5, half of a fuel plate (cut axially at the azimuthal centerline) and the associated side plate (cut axially at the midpoint of the water channel on each side of the fuel plate) is modeled. Each fuel plate coolant channel is divided into six subchannels, and the side plate channel into three subchannels.

The network consists of nodes at nine axial elevations. Each level has a total of 102 nodes: 59 structural nodes with heat capacity (diffusion nodes), 28 structural nodes without heat capacity (arithmetic nodes), and 15 coolant nodes. The coolant nodes are defined as boundary nodes within the ATR-SINDA finite-difference solvers, their temperatures being updated by the user subroutines between steady state iterations and between transient time steps. All of the diffusion nodes are meshed with a three-dimensional conductor network, while the arithmetic nodes are linked with diffusion nodes in only one direction, that of predominant heat flow.

The results from ATR-SINDA are used in SINDA-SAMPLE for evaluating reactor protection margins.

15.0.11.3 SINDA-SAMPLE Computer Model. SINDA-SAMPLE is a combined statistical and simplified thermal-hydraulic model of an ATR fuel plate (Liebelt 1993). Using thermal-hydraulic input from ATR-SINDA, a Monte Carlo sampling is performed to determine the standard deviations of the parameter of interest and the margin to the reactor protection criteria. The input parameters have unique variances which are input with the nominal value of the parameter. Each parameter is sampled, and fuel plate conditions are calculated using the variable values from the sampling. The heat transfer mode is determined the same as in ATR-SINDA using the conditions calculated from the ATR-SINDA boundary values and the sampled variables.

15.0.11.4 PDQ Computer Code. The PDQ 7 program solves the neutron diffusion-depletion problem in one, two, or three dimensions (Pfeifer 1971). The three-dimensional spatial calculation may be either explicit or discontinuous trial function synthesis. Up to five lethargy groups are permitted, the fast group treatment may be simplified P_3, and the thermal neutrons may be represented by a single group or a pair of overlapping groups. Adjoint and boundary value calculations may be performed, and the depletion may be point or block. Control searches, thermal feedback, and xenon feedback are optional. The geometries available are rectangular, cylindrical, spherical, and hexagonal. All geometries allow variable mesh in all dimensions.

15.0.11.5 RSAC-4/5 Computer Code. The Radiological Safety Analysis Computer Program (RSAC-5) calculates the consequences of the release of radionuclides to the atmosphere. RSAC-5 can generate a fission product inventory; decay and grow the inventory during transport through processes, facilities, and the environment; model the downwind dispersion of the activity; and calculate doses to downwind individuals. The code has undergone independent verification and validation (Shonka 1993). As described in Wenzel (1993), the code calculates inhalation doses using the ICRP 30 model (International Commission on Radiological Protection). The committed dose equivalent (CDE) is calculated for individual organs and tissues over a 50-year period after inhalation. The various options available for dose calculations are described in the reference (Wenzel 1993). The code calculates atmospheric diffusion parameters for three different conditions or allows the user to input the parameters directly. The specific options used depend upon the accident sequence considered.

15.0.11.6 MicroShield 6.20 Computer Code. MicroShield® is a comprehensive photon/gamma ray shielding and dose assessment software program that is used for designing shields and estimating source strength from radiation measurements (Grove 2005). It can be used to develop a model geometry and material composition. MicroShield® 6.20 is maintained under configuration control number 198549 in the Engineering and Scientific Analysis Software and is fully validated, verified, and controlled in accordance with applicable INL requirements.

15.0.12 Limiting Single Failures

The most limiting single failure (where one exists) of safety-related equipment is identified in each analysis description; the consequences of this failure are described therein. In some instances, because of redundancy in protection equipment, no single failure which could adversely affect the consequences of the transient has been identified. The failures assumed are listed in Table 15.0-9.

15.0.13 Operator Actions

For most of the faults in Chapter 15, the plant will be in a safe and stable shutdown condition following an automatic actuation of reactor trip. This condition will be similar to plant conditions following any normal shutdown. The actions taken by the operators may be the same as the ones taken during normal shutdown. However, there are several events that may require additional operator action than what would be expected following a normal shutdown. These events include, a complete loss of flow accident, a loss of heat sink accident, a loss of coolant accident (LOCA) at power, a shutdown (LOCA) and a draining of the fuel storage canal.

Those events which result in complete loss of flow accident or loss of the heat sink accident require that the operator open the vessel vent valves before the reactor water temperature exceeds 200°F and maintain coolant inventory by assuring the availability of the firewater injection system. Other actions may be taken prior to these conditions, but are not considered to be required for plant protection and Chapter 15 analyses.

Events which result in a loss of coolant accident (LOCA) do not consider operator action to secure the plant. The shift supervisor will initiate an area evacuation when the evacuation criteria are met for a LOCA or other off-normal condition.

For a shutdown LOCA the operator will have to open isolation valves on the firewater system that are closed when the plant pressure is reduced below 100 psia. Opening the isolation valves ensuring firewater system actuation is required to keep the core covered with water. Operators can be made aware of level changes either by level instrumentation, located in the control room, or by vessel low-level alarm.

While there is an automatic makeup system to provide water to the canal, that uses water from the firewater system to provide sufficient water to the canal to keep the irradiated fuel in the canal covered, a manual backup has also been provided.

For recovery from LOCA events or from a canal draining event long-term operator intervention may be required. Operator action is needed to sustain firewater injection by maintaining raw water inventories via deep well pumps and keeping firewater system fuel tanks replenished.

15.0.14 ATR Plant Protection Criteria

ATR plant protection criteria have been chosen to meet the basic principle stated in Section 15.0.3 regarding release of radioactivity and plant operability. The protection criteria and the protective margins or the margins to safety for the corresponding accident frequency categories are:

1. Condition 1

   For Condition 1, off-site exposures do not exceed the limits of 100 mrem/year effective dose equivalent (EDE) and 10 mrem/year EDE from airborne releases in DOE O 5400.5 “Radiation Protection of the Public and the Environment.” On-site exposures do not exceed the limit of 5 rem/year total effective dose equivalent (TEDE) in 10 CFR 835, “Occupational Radiation Protection.” Normal operating procedures and routine monitoring of parameters ensure dose limits are not exceeded and plant operability is maintained.

The integrity of the fuel cladding is not challenged in Condition 1 except for limited clad defects (e.g., pin hole corrosion). The integrity of the fuel cladding is maintained by controlling operations within a conservative set of parameters that maintain fuel temperatures within the initial condition assumptions used in the accident analyses. For reactor operation, margins from CHF and flow instability (FI) are at least 3.0 standard deviations.

2. Condition 2

For Condition 2 faults, off-site exposures do not exceed 0.5 rem/year TEDE. “On-site” personnel exposures do not exceed the limits of 5 rem/year TEDE in 10 CFR 835.

For reactor accidents, the reactor fuel clad is the protective barrier and must be protected. No rupture of the fuel plate cladding is allowable unless the clad failure is the initiating event. The protective criteria for the reactor fuel plate clad have been chosen to be CHF and FI. The protective margin requires maintaining at least 3.0 standard deviations from CHF or FI (the term CHF may be used in this chapter to denote both critical heat flux and flow instability). The protective margins are evaluated using the Monte Carlo sampling technique in the SINDA-SAMPLE computer model.

For canal accidents no melting of the fuel plate cladding is allowed.

For overpressure events, the primary coolant system pressure shall not exceed 110% of its design pressure.

3. Condition 3

For Condition 3 faults, off-site doses are limited to 25% of the limits in 10 CFR 100, “Reactor Site Criteria,” that is, 6.25 rem whole body and 75 rem thyroid dose. Doses to evacuating on-site personnel (excluding personnel considered directly at the location of the accident) are limited to 25% of 10 CFR 100 limits.

For reactor accidents, Condition 3 faults may result in failure of the fuel element cladding as a consequence of the initiating fault. The barrier to release of radioactivity is the primary coolant system, and it becomes the protective barrier that must be maintained (unless failure is the initiator). The protective criteria is that no large releases of uranium or fission products to the primary coolant system will occur. This requires that the reactor core maintain a coolable geometry. The protective margin for the reactor fuel cladding is that a minimum of 2.0 standard deviations to the predicted fuel plate clad temperature which results in fuel plate buckling (buckling is assumed to occur at a temperature of 710°F) must be maintained. An exception is made for a beryllium induced partial flow blockage of channel 20 in a reflector within 2 standard deviation of cracking stresses or with reflector cracks. For this case, the acceptance criteria is that a coolable geometry be maintained (Miller 1998). The protective margins are evaluated using the Monte Carlo sampling technique in the SINDA-SAMPLE computer model.

For overpressure events, the primary coolant system pressure shall not exceed 120% of its design pressure.

4. Condition 4

For Condition 4 faults, off-site doses are limited to the limits in 10 CFR 100, that is, 25 rem whole body and 300 rem thyroid dose. Doses to evacuating on-site personnel (excluding personnel considered directly at the location of the accident) are also limited to 10 CFR 100 limits.

For reactor accidents, Condition 4 faults may involve failure of the primary coolant pressure boundary and result in significant melting of the reactor core. The protective criteria is that the primary coolant pressure boundary must be maintained unless its failure is the initiator and the reactor confinement must not be damaged. The potential aluminum-water reaction energy in the ATR, combined with a transient energy release, could result in damage to the primary coolant pressure boundary and the reactor confinement. The protective margin for the reactor fuel cladding is that a minimum of 3.0 standard deviations to the aluminum-water ignition threshold (AWIT) temperature (2140°F) be maintained.

A consequence-limiting protective thermal margin to limit the annual probability or estimated frequency for hot channel dryout or DNB, and thus limit the probability for significant core melt consequences, is also defined for Condition 4 events (Atkinson 1999). This thermal margin limits, to $< 1 \times 10^{-6}$/yr. with a 95% confidence the probability for fuel melt for limiting Condition 4 events with an initiating event frequency of 1×10^{-4}/yr. By use of this consequence-limiting thermal margin, the Condition 4 events protective margin for approach to AWIT is satisfied with additional margin and the reactor vessel integrity is protected. This thermal margin for Condition 4 events is expressed as:

> A margin ≥ 1.1 σ (sample standard deviations) from predicted flow instability or from DNB (critical heat flux) conditions for margin ≥ 1200 statistical samples and for a fraction of ATR power operation at high lobe power (> 47 MW) that is ≤ 0.07, or
>
> A margin ≥ 1.2 σ (sample standard deviations) from predicted flow instability or from DNB (critical heat flux) conditions for ≥ 1200 statistical samples and for a fraction of ATR power operation at high lobe power (> 47 MW) that is ≤ 0.08.

The fraction of ATR power operation at high lobe power (> 47 MW) is tracked during ATR operation as required by the implementation of the ATR UFSAR.

The protective margins are evaluated for acceptance using the Monte Carlo sampling technique in the SINDA-SAMPLE computer model (additional specific case analyses using ATR-SINDA are also available).

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The basic principle is that the most probable occurrences should yield the least radiological risk to workers and the public and the least risk to continued operation of the facility. Those extreme events having the potential for the greatest risk to workers and the public and to facility operability should be least likely to occur. The ATR Plant Protection Criteria are established in terms of on-site and off-site dose consequences for each plant condition. Additionally, the analyses of faulted conditions in the reactor have established acceptance criteria for protection of plant equipment. Maintaining these acceptance

criteria provides assurance that the radiological consequences of a fault will not exceed the dose criteria. In the reactor, the fuel element cladding is the first barrier to release of the fission product source term; therefore, the most significant protective criteria are defined in terms of margins to specified thermal hydraulic conditions for the reactor fuel element cladding. These reactor fuel cladding protective criteria are established for each faulted condition.

For stored fuel elements, the fuel cladding is the barrier to fission product release. The integrity of the cladding of stored fuel elements is the primary concern in the evaluations of fuel handling and storage accidents. Since there are both potential mechanical and thermal challenges to the stored fuel element cladding, the protective margins are described in the specific analysis of the accident conditions.

The re-analysis of events for the UFSAR mainly involved recalculating the reactor fuel cladding protective margins during the limiting faults during power operation. Except for the calculation to support the severe accident analysis in Section 15.12, radiological consequences were not re-analyzed. The discussions of radiological consequences were either taken from the DBR or other references or scaled from the bounding calculation in Section 15.12. The discussion of the potential accidents from other facility operations (e.g., canal operations) were taken from the DBR.

The DBR stated dose criteria by reference to the DOE Order source document rather than specific values. Dose criteria have been explicitly stated in the UFSAR for clarity with reference to the source requirement. The current requirements for off-site dose for Condition 1 (DOE O 5400.5) are less restrictive than those used in the derivation of ATR stack effluent limits and PCS and SCS radioactivity limits in Chapter 12 (Radiation Protection). The limits on these parameters were not re-evaluated and are maintained from the DBR analyses. The off-site dose criteria for Condition 2 faults were taken from the DBR. It is the same as the value in the INL nonreactor facility risk evaluation guidelines (Stallman 1997). The off-site dose criteria for Condition 3 are ≤ 25% of the 10 CFR 100 limits, and for Condition 4, the off-site dose criteria are the 10 CFR 100 limits. According to 10 CFR 100, the exclusion area is the area surrounding the reactor in which authority exists to control all activities including the removal of personnel and property. Because the DOE has such authority within the INL borders, the INL site boundary was assumed to be the ATR exclusion boundary. For Condition 3 and 4 events, the off-site exposures at the nearest site boundary (NSB) and the low population zone (LPZ) can differ in the time allowed to receive the dose. The exposure at the NSB (10.8 km from ATR) can be limited to two hours while the exposure at the LPZ (55.5 km from ATR) must be based on the entire passage of the radioactive plume.

The on-site worker dose criteria are specified as the 10 CFR 835 limits for Conditions 1 and 2. The on-site worker dose criteria for Conditions 3 and 4 are specified as the same as those for off-site exposures. Worker dose analyses typically take credit for worker evacuation (either local area, general ATR Complex evacuations, and INL Laboratory-wide evacuations). On-site worker dose criteria for Conditions 1 and 2 are applied to both in-facility workers and to workers in neighboring facilities. Condition 3 and 4 on-site dose criteria are applied to evacuating workers in neighboring facilities. The term “evacuating workers” is used to describe the general population of INL workers. Workers involved directly at the location of certain Condition 3 and 4 events with radiological consequences (e.g., a serious experiment handling accident resulting in exposure to an unshielded irradiated experiment or an unshielded criticality) may receive doses higher than the plant protection criteria for evacuating workers.

15.0.15 Limitations Related to Power Distributions

The power distributions selected for the analysis are severe unbalances that were expected to establish minimum margins during off-normal events for operation with at least one lobe at 70 MW or at least one lobe at 60 MW. The power divisions selected were the 70/20 (SAR 70) unbalanced and the 60/40 (E60/40) unbalanced. The results of analyses with these power divisions indicate the following:

1. Operation at 70-MW lobe power is not supported.
2. Effective plate power limits are the significant item for control of operation.
3. Lobe power limits are necessary to ensure piping design temperatures are not exceeded (other component designs also require a lobe power limit; e.g., IPT).

The bases for these observations are provided below:

1. The results of low-probability events indicate that substantial fuel melt can occur for certain events initiated at high power. The progression of the melt and interaction with the environment (core support structures and water) could not be described with high confidence using available information and techniques. The consequences were avoided by conservatively limiting the plate power.

   Although operation at 70-MW lobe power is not supported, the analytical basis includes the 70/20 results scaled to a lower power. It was not cost effective to revise all the calculations using a single power division.

2. The acceptance criteria are based on the margins to selected conditions (CHF, fuel plate buckling, and AWIT). The margins are calculated using a model that represents a single hot fuel plate in the 7F fuel element. The analysis determines the conditions for which this single plate will meet the acceptance criteria. By comparing subsequent operation to effective power of this plate and the determined conditions, it can be ensured that the operation is within the bounds of the analysis. If the initial effective plate powers throughout a proposed operation do not exceed the effective plate power analyzed, one of the conditions of the analysis is met. (Others to be met include initial conditions such as coolant temperature, flow, and system pressure.) The effective plate power is a sufficient control on the power generation for fuel element performance.

   The limiting thin plate determined by the analysis is Plate 15. This plate may increase in power as the fuel element depletes at constant power. In order to conservatively ensure that the increase is considered, the effective plate power is specified rather than a midplane distribution. The effective plate power limits are

   Two pump:

   Interior plates (between 5 and 95% of width) - 417 MW
   Plate 19 (between 10 and 90% of width) - 417 MW

   Three pump:

   Interior plates (between 5 and 95% of width) - 443 MW
   Plate 19 (between 10 and 90% of width) - 445 MW

Overpower transients that are slow with respect to the fuel plate thermal time constant (for example, the flow coastdown) were evaluated for compliance to reactor protection criteria using the axial and azimuthal power profiles for Plate 19. Those transients that are fast with respect to the fuel plate thermal time constant (for example, the reactivity initiated accidents) were evaluated for compliance using the axial and azimuthal power profiles for Plate 15. Plate 15 is typically the highest powered interior plate. The inner plates are thinner and thus have a lower heat capacity with a resultant higher temperature for the same reactor transient.

A special fuel element (YA) with an all-aluminum plate rather than a fueled plate for Plate 19 is used in selected locations during high power operation as the reflector ages. As noted above, the minimum margins for fuel element performance are determined from a model that considers only the hot plate of the 7F fuel element. Since the modeling of fuel plates in the YA fuel element would be essentially the same as the model for Plate 15 considered in the limits above, those limits will apply to the YA. However, the YA element is used when there is a potential for cracking in the adjacent beryllium reflector. The effect of the crack must be considered in evaluating the planned operation against the above limits. The potential cracking of the beryllium could increase water adjacent to the fuel plate and increase power slightly. The 7F fuel element may be used in place of the YA element if lobe powers are limited such that the effective plate power in Plate 19 does not exceed 358 MW for two-pump operation and 357 MW for three-pump operation. The effective plate powers for the inner plates are 406 MW for two-pump operation and 431 MW for three-pump operation.

The operation also utilizes non-borated fuel elements as needed to accomplish programmatic goals. The above arguments for effective plate power limitations also apply to the non-borated fuel element. The effect of this loading on the reactivity feedback from the fuel annulus assumed in the RELAP5 model has been shown to envelop that from the non-borated element (Polkinghorne 1994b). Therefore, the plate power limits shown above for the 7F fuel element also apply to the non-borated fuel element.

The amount of power unbalances in the core has some effect on the power distribution within a fuel element. The distribution within the fuel plate can have some effect on the amount of azimuthal conduction. This azimuthal conduction can affect the location of the hot stripe within the coolant channel. The two power distributions considered in the analysis resulted in nearly identical locations for the hot stripe. Since the minimum margins occur at this hot stripe, the comparisons to the analysis are completed at the hot stripe. The two power divisions considered had no apparent effect on the hot stripe location. Less severe unbalances would reduce individual plate powers and would be expected to have increased margins relative to the higher power; therefore, it is concluded that any significance to the amount of power unbalance is captured by the analysis.

The kinetics analysis demonstrated a sensitivity to power unbalance. The feedback from the fuel element region (negative reactivity) is important in reducing power in off-normal events. The balanced power division provided the least feedback and therefore the highest power for positive power transients. The feedback behavior of the balanced power division was imposed on the analysis to add conservatism to those power-transient events.

3. The analysis for loss of heat sink considered the performance of the fuel elements (approach to thermal-hydraulic events) as well as the increase in piping and vessel temperatures. The power distribution has some importance to the piping temperatures for this event since quadrant powers

influence the temperature of the outlet piping. The lobe power considered in the analysis of this event (60 MW) is a control parameter to ensure piping temperatures are properly limited.

Analyses of other components such as the IPTs consider the lobe power in determining the heating. These considerations are developed in other chapters of the UFSAR.

The discussion above applies to operation in the pressurized mode. The analysis for depressurized operation considers an unbalanced 70/20 power division. This is a very conservative approach since depressurized operation is performed with a nearly balanced core (fuel loading) and shim positions.

Table 15.0-1. General criteria for classifying Advanced Test Reactor accident frequencies.

I.	**ANTICIPATED FAULTS**
A.	Single operator errors with unsecured and accessible devices.
B.	Single control failures.
C.	Single operating failures of an active device. An active device is defined as an operating device, such as pumps, valve actuators, relays, which may fail while operating (while in motion) or fail to operate (to move).
D.	Loss of a single power source, such as an electrical bus or instrument air supply.
E.	Loss of a single supporting system, that is a system which in itself includes single controls and active devices which can fail the supporting system.
F.	An operational upset known to occur, such as a commercial power failure.
G.	Failure of a device known to have a finite probability of being defective.
II.	**UNLIKELY FAULTS**
A.	A combination of two independent anticipated faults, or failure of two components each with anticipated fault failure modes.
B.	A single failure of an occasionally active device, but when it is in a passive state (when no action or motion is being requested), such as failure of a valve part in a normally fully open or closed valve.
C.	The occurrence of an anticipated type of failure during some short and specific type of operation.
D.	The occurrence of an anticipated type of failure in combination with failure of a single interlock or mitigating device.
E.	Failure of a passive device that has not had sufficient quality assurance to guard against its failure, such as welds that have not been adequately inspected for flaws.
F.	A failure due to a human error when an independent check or review is involved, or anticipated faults when sufficient time is available for independent checks to detect and correct the condition.
G.	A single operator error with a secured or normally inaccessible device.
H.	Failure of an active device that is in a low stress state when it has had an acceptable amount of quality assurance to guard against its failure. Failure of a valve shaft in a PCS control valve is an example.
III.	**EXTREMELY UNLIKELY FAULTS**
A.	A combination of two independent unlikely faults, and other combinations of two or more unlikely and anticipated faults not previously identified in II.
B.	The occurrence of an unlikely type of failure during some short and specific type of operation.
C.	The occurrence of an unlikely type of failure in combination with failure of a single interlock or mitigating device.
D.	Failure of a passive device that has had an acceptable amount of quality assurance to guard against its failure.
E.	Failures due to very low probability events, such as large earthquakes (see Chapter 3 [Design of Structures, Components, Equipment, and Systems)].
F.	A failure due to a human error when a chain of two or more independent checks and reviews are in operation to prevent the failure.

Table 15.0-2. Principal plant conditions. **(For information only)**[a]

Thermal Power Output	
1. Maximum Total Reactor	
(a) Pressurized Operation (Maximum lobe power 60 MW)	250 MWt
(b) Depressurized Operation	0.5 MWt
2. Distribution	
(a) Fuel Plate and Plate Coolant	93%
(b) Reflector and Structure	7%
Maximum Experiment Fission Power	200 kW
Nominal Fuel Assembly Information	
Number of assemblies in core	40
Number of fuel plates per assembly	18 or 19
Active length of fuel assemblies	4 ft
Fuel matrix volume/assembly	0.02793 ft^3
Assembly flow area	0.03099 ft^2
Nominal Control Element Information	
1. Safety Rods	
(a) Number (typical)	6
(b) Total worth (nominal)	17.0$
(c) Incremental worth per rod (average)	0.08$/in
2. Outer Shim Control Cylinders (OSCC)	
(a) Number	16
(b) Total worth (beginning of life)	12.1$
(c) Incremental worth per four cylinder set (average)	0.02$/degree
3. Neck Shim Rods	
(a) Number	22
(b) Total worth (beginning of life)[b]	5.2$
(c) Incremental worth per rod (average)	0.005$/in
4. Regulating Rods	
(a) Number (1 in operation)	2
(b) Total worth (beginning of life)	0.26$
(c) Incremental worth (average)	0.005$/in
Representative Core Neutronics[c] Information	
Moderator Temperature Coefficient of Reactivity	$-6.68 \times 10^{-3}$$/°F
Fuel Temperature Coefficient of Reactivity	$-2.76 \times 10^{-4}$$/°F
Effective Neutron Lifetime	0.38×10^{-4} sec
Effective Delayed Neutron Fraction	0.0072
Flux Trap Void and Temperature Coefficient of Reactivity	Variable but positive

Table 15.0-2. (continued).

Fuel Element Void Coefficient of Reactivity	Variable but net negative
Axial Power Distribution	Figures 15.0-1 and -2
Azimuthal Power Distribution (Plate)	Figure15.0-3
Lobe Power Distributions	Table 15.0-3
Primary Coolant System - Pressurized Operation	
Operating Primary Coolant Pumps	2 or 3
Nominal Primary Coolant System Flow Rate	43,500 gpm (2 PCP) or 50,000 gpm (3 PCP)
Maximum Core Coolant Inlet Temperature	125°F
Nominal PPS Vessel Inlet Pressure Tap	371.9 psig[d]
Minimum Core Differential Pressure	77 psid (2 PCP) or 100 psid (3 PCP)
Surge Tank Coolant Inventory	97% (970 ft^3)
Primary Coolant System Inventory (Makeup and Letdown)	
Pressurizing System Flow Rate (nominal)	220 gpm
Degassing System Flow Rate (nominal)	220 gpm
LDW In Flow From Seal System (nominal)	35 gpm
Bypass Demineralizer to Gland Seal Feed Tank (nominal)	35 gpm
Primary Coolant System - Depressurized Operation	
Minimum Operating Emergency Coolant Pumps	1
Reactor Vessel Inlet Pressure (upper plenum)	Atmospheric
Maximum Core Coolant Inlet Temperature	125°F
Minimum Primary Coolant System Flow Rate	≥ 3,600 gpm
Secondary Coolant System	
Secondary Coolant System Flow Rate	Variable
Secondary Coolant System Inlet Temperature	Variable
Heat Transfer Parameters	
Maximum Corrosion Product (oxide) Thickness	0.001 in.
Nominal Coolant Channel Thickness (Channels 2 through 20)	0.078 in.
Material Properties	Table 15.0-4

a. This table is marked for information only to avoid potential use of inappropriate data in future accident analyses. Initial conditions and other plant parameters and conditions were selected to provide a conservative analysis for the event being analyzed. For example, minimum safety rod reactivities are used to model the safety rod scram reactivity while maximum safety rod reactivities are used in the analysis of safety rod withdrawal accidents.

b. Measured reactivity following 1994 CIC. Represents typical fuel loading, new reflector, and new hafnium components in the core.

c. See Chapter 4 (Reactor) for additional information on the derivation of these parameters.

d. The core inlet pressure is not directly indicated. The pressure control is based on pressure taps in the inlet piping. The RELAP5 model indicates pressure at the tap location of the PPS inlet pressure subsystem is 385.4 psia (373.2 psig) for three-pump operation and 384.1 psia (371.9 psig) for two-pump operation (Pafford 1994).

Table 15.0-3. Lobe power splits (absolute power) for three power distributions modeled in RELAP5.

LOBE	SAR 70 (MW)	BAL50/50 (MW)	E60/40 (MW)
NE	26.0	50.0	41.6
NW	26.3	50.0	41.6
C	45.7	50.0	52.8
SW	62.0	50.0	60.0
SE	70.0	50.0	54.0
Total core	230.0	250.0	250.0

Table 15.0-4. ATR fuel element material properties.

Fuel		
Heat of fusion (J/kg)	Plate 19	2.83×10^5
	Plate 15	2.39×10^5
Density (kg/m^3)	Plate 19	3409
	Plate 15	3933
Thermal conductivity (W/mK)		
Temperature (K)	Plate 19	Plate 15
273	53.0	36.1
300	53.0	36.2
400	52.9	36.0
500	52.4	35.7
600	51.9	35.4
700	51.4	35.1
800	50.9	34.8
900	50.3	34.4
933 solidus	50.2	34.4
933 liquidus	40.3	30.2
1000	40.5	30.2
1100	40.8	30.3
1200	41.0	30.3
Specific heat capacity (J/kgK)		
Temperature(K)	Plate 19	Plate 15
273	648	557
300	666	573
400	697	599
500	731	629
600	767	660

Table 15.0-4. (continued).

700	803	692
800	839	725
900	878	761
933 solidus	889	771
933 liquidus	879	764
1000	887	775
1100	899	791
1200	910	807
Clad		
Heat of fusion (J/kg)	3.95×10^5	
Density (kg/m^3)	2702	
Thermal conductivity (W/Mk)		
Temperature (K)	Al 6061-T6	Al 6061-O
273	163	177
300	167	180
350	173	186
400	177	189
450	179	190
500	188	191
550	191	191
600	190	190
650	188	188
700	185	185
750	182	182
800	179	179
855 solidus	175	175
925 liquidus	86.7	86.7
1000	88.7	88.7
1100	91.5	91.5
1200	94.2	94.2
Specific heat capacity		
Temperature (K)	Specific Heat (J/kgK)	
290	896	
400	942	
500	988	
600	1034	
700	1080	
800	1126	
855 solidus	1151	

Table 15.0-4. (continued).

925 liquidus	1084	
1000	1084	
1100	1084	
1200	1084	
Oxide		
Heat of fusion (J/kg)	1.07×10^6	
Density (kg/m^3)	3010	
Thermal conductivity (W/Mk)	2.25	
Specific heat capacity		
Temperature (K)	Specific Heat (J/kgK)	
273	1011	
300	1100	
400	1349	
500	1538	
600	1704	
700	1844	
800	1959	
900	2049	
1000	2113	
1100	2151	
1200	2164	

Table 15.0-5. Safety rod reactivity envelope.

Pressurized operation		Depressurized operation	
Time After Trip Signal (ms)	Reactivity ($)	Time After Trip Signal (ms)	Reactivity ($)
25	0	0	0.0000
75	0.08	20	0.0004
100	0.17	40	0.0012
125	0.37	60	0.0023
150	0.67	80	0.0048
175	1.25	100	0.0093
200	2.12	120	0.02
225	2.95	140	0.04
250	3.50	160	0.08
275	3.85	180	0.14
300	4.10	200	0.26
350	4.42	220	0.39
400	4.59	240	0.62
500	4.70	260	0.92
600	4.75	280	1.24
700	4.80	300	1.76
800	4.85	340	2.65
900	4.90	380	3.52
1000	4.95	460	4.44
1100	5.00	540	4.97
1200	5.05	620	5.26
1300	5.10	700	5.45
1400	5.15	800	5.59
1500	5.20	900	5.68
2000	5.45	1000	5.73
2500	5.70	1120	5.80
2700	5.80		

Table 15.0-6. Plant protection system parameters.

Parameter	Analysis Trip Point	Response Time, s
High Neutron Level	120%	0.025 (0.035 for depressurized)
Wide Range		
High Level	120%	0.025 Top two decades
Period at range change	1 second	0.035 other ranges
Low Vessel Inlet Pressure		
Three - PCP operation	312.6 psig	0.1
Two - PCP operation	312.6 psig	0.1
Low Vessel Outlet Pressure		
Three - PCP operation	88.1 psig	0.1
Two - PCP operation	88.1 psig	0.1
Low Vessel Differential Pressure		
Three - PCP operation	45.8 psid	0.4
Two - PCP operation	38.0 psid	0.4
High Vessel Inlet Temperature	150°F	40
High Vessel Outlet Temperature		
Three - PCP operations	188°F	25
Two - PCP operations	185°F	25
High Quadrant Differential Temperature	1.24 Normal ΔT	15
Low Emergency Coolant Pump Recirculation Flow	90 gpm	10
Low Emergency Coolant Flow Discharge (depressurized)	2,000 gpm	10
Low Vessel Water Level	91’10”	2.0
High Vessel Differential Pressure	146 psid	0.6
High Vessel Inlet Pressure	408.6 psig	0.5
High Vessel Outlet Pressure		
Three - PCP operation	289.9 psig	0.5
Two - PCP operation	321.6 psig	0.5
High Seismic Acceleration	0.017 g	1.0
High Primary-Coolant Radiation Level		
startup	150% Normal at $N_F/3$	0.5
full power	150% Normal at N_F	0.5
Low Upper Plenum Pressure	28 psia	7.8
High Radiation	350 mr/hr	60.0
Relief Valve Setting	390 psig	NA[a]

a. Not applicable, no delay time on relief valves.

Table 15.0-7. Plant systems and equipment available for transient and accident conditions.

UFSAR Section	Bounding Incidents	Reactor Trip Functions	Engineered Safety Feature (ESF) Actuation Functions	ESF Equipment
15.1 Increase in Heat Removal by Secondary	Cold water addition (15.1.1)	High neutron level	Not required	None
15.2 Decrease in Heat Removal by the Secondary	Complete loss of heat sink (15.2.1)	High vessel inlet temperature	High vessel pressure Low vessel level Manual	PCS relief valves EFIS Vessel vent valves
15.3 Decrease in Reactor Primary Coolant Flow Rate	Long-term complete loss of flow (15.3.4)	Low vessel inlet pressure	High vessel pressure Low vessel level Manual	PCS relief valves EFIS Vessel vent valves
15.4 Reactivity and Power Distribution Anomalies	IPT voiding due to > 1/2-in. pipe break (15.4.6)	High neutron level	Not required	None
	Withdrawal of all outer shims and neck shims from 10^{-5} N_F (15.4.2)	High rate of change in neutron level	Not required	None
	Slow lobe power balance shift. (15.4.7.12)	Quadrant differential temperature	Not required	None
15.5 Increase in Primary Coolant Inventory	Loss of instrument air.(15.5.1)	High vessel inlet pressure	High vessel pressure	PCS relief valves Pressurizing pump shut off
15.6 Decrease in Primary Coolant Inventory	A 3-in. Opening in the PCS...(15.6.4) Bottom head closure plug failure (15.6.10.4)	Low vessel inlet pressure None	Low vessel pressure Low vessel level Manual (following PPS or EFIS vessel level alarm)	EFIS LOCA Primary Coolant Pump Shutoff EFIS
15.7 Radioactive Release from a Subsystem or Component	Loop experiment fuel melt (15.7)	High neutron level	High air activity	Radiation monitoring and seal system (RMSS)
15.8 Fuel Storage Canal and Cask Handling Events	Large crack or other failure of storage canal (15.8.4)	None	None	None

Table 15.0-7. (continued).

UFSAR Section	Bounding Incidents	Reactor Trip Functions	Engineered Safety Feature (ESF) Actuation Functions	ESF Equipment
15.9 Depressurized Operations Events	Momentary loss of flow... (15.9.11) Opening valve outside radiographic... (Section 15.9.2.1) Loop heater accidentally on (15.9.3.6)	Low emergency coolant pump discharge flow Low vessel level High neutron level	None Manual[(c)] None	None EFIS[(c)] None
15.10 Fuel Failures	Large fuel channel blockage... (15.10.3)	High PCS coolant radioactivity	High air activity	RMSS
15.11 Failures of Supporting Systems	(a)	(a)	(a)	(a)
15.12 Severe Accidents	(a)	(a)	(a)	(a)
15.13 PCS High Pressure Anomalies	Butterfly valve failure (15.13.2)	Low vessel inlet pressure	None	None
15.14 PCS High Flow Anomalies	High flow (15.14.1)	High vessel differential pressure	High vessel differential pressure	Primary coolant pump shutoff
15.15 Reflector Aging	(b)	(b)	(b)	(b)
15.16 External Events		(a)	(a)	(a)

(a) No new requirements for PPS or ESF are established by this analysis.

(b) The limiting events and mitigating systems are unchanged from the new reflector.

(c) Manual actuation of EFIS is only required with irradiated fuel in the core (as discussed in Section 15.9.2.1.2).

Table 15.0-8. Summary of computer codes used for analyses of credible events.

Section	Analytical case	Computer Codes Utilized
15.1	Cold water addition	RELAP4
15.2	Complete loss of secondary	RELAP5/MOD2.5 ATR-SINDA SINDA-SAMPLE
15.3	Loss of commercial power	PDQ RELAP5/MOD2.5 ATR-SINDA SINDA-SAMPLE
	Slow closing primary discharge check valve	RELAP5/MOD2.5
	Long-term complete loss of flow	RELAP5/MOD2.5 ATR-SINDA SINDA-SAMPLE
	Complete closure of the flow control butterfly valve	RELAP5/MOD2.5 ATR-SINDA SINDA-SAMPLE
15.4	Uncontrolled outer shim and neck shim withdrawal from 10^{-5} N_F	RELAP4
	0.50$ step insertion	RELAP5/MOD2.5 ATR-SINDA SINDA-SAMPLE
	Inpile tube decompression with subsequent voiding due to failure of the 1/2-in. piping	RELAP5/MOD3 ATR-SINDA SINDA-SAMPLE
	IPT decompression with subsequent voiding due to a failure of piping < 1/2-in. in diameter	RELAP5/MOD3 ATR-SINDA SINDA-SAMPLE
15.5	Loss of instrument air to PCV-1-1	RELAP4
	Loss of instrument air to PCV-1-1 and FCV-1-8 with failure of the inlet pressure subsystem	None

Table 15.0-8. (continued).

Section	Analytical case	Computer Codes Utilized
5.6	LOCA break spectrum	RELAP5/MOD2.5
		RELAP5/MOD3
	Break or open valve outside radiographic boundary.	RELAP5/MOD2.5 ATR-SINDA SINDA-SAMPLE
	Shutdown bottom plug failure	RELAP5/MOD3
15.7	Experiment fuel melting	RSAC 4/5
	Waste Tank Release (TRA-605)	RSAC-5.2, MicroShield 6.20
15.8	NA	NA
15.9	Startup events	RELAP5/MOD 2.5, ATR-SINDA, SINDA-SAMPLE
15.10	Insulation jacket failure Loading errors Fuel blockage	COMPFLO and RELAP4 ATR-SINDA and SINDA-SAMPLE RSAC-5
15.11	Events are enveloped by previous analysis for loss of heat sink, loss of flow, and power transients	NA
15.13	A primary pump discharge valve, check valve, or butterfly control valve slams closed.	RELAP5/MOD 2.5
15.14	Opening of flow control butterfly valve	RELAP4
15.15	Reflector aging	RELAP5/MOD2.5 ATR-SINDA SINDA-SAMPLE
15.16	External events	See PRA (Eide, Khericha, and Thatcher 1991)

Table 15.0-9. Limiting single failures assumed in the UFSAR reanalysis of accidents.

Event description	Failure assumed[a]
Complete loss of SCS (Loss of Heat Sink)	Upper Emergency Firewater Injection System (EFIS) train and one Lower EFIS Actuation Valve
Loss of commercial power	Standby emergency coolant pump
Slow closing primary discharge check valve	No active component required
Complete loss of flow	DG-powered emergency coolant pump Battery backup emergency coolant pump[b]
Complete closure of the flow-control butterfly valve	No active component required
Uncontrolled outer shim and neck shim withdrawal from 10^{-5} N_F	No active component required
Inpile tube decompression with voiding due to failure of the 1/2-in. piping	One safety rod fails to insert
0.50$ Step	One safety rod fails to insert
IPT decompression with subsequent voiding due to a failure of piping > 1/2-in. in diameter	One safety rod fails to insert
Shutdown bottom plug failure	Upper EFIS actuation valve
Break or open valve outside radiographic boundary	Upper or lower EFIS out of service One PCP continuing to run One PCP check valve failing to shut

a. Failure assumed is the most limiting single failure (where one exists) of safety-related equipment which is assumed in the accident analysis of the particular event.

b. Batteries assumed to deplete after 30 minutes (see Chapter 8 (Electric Power)).

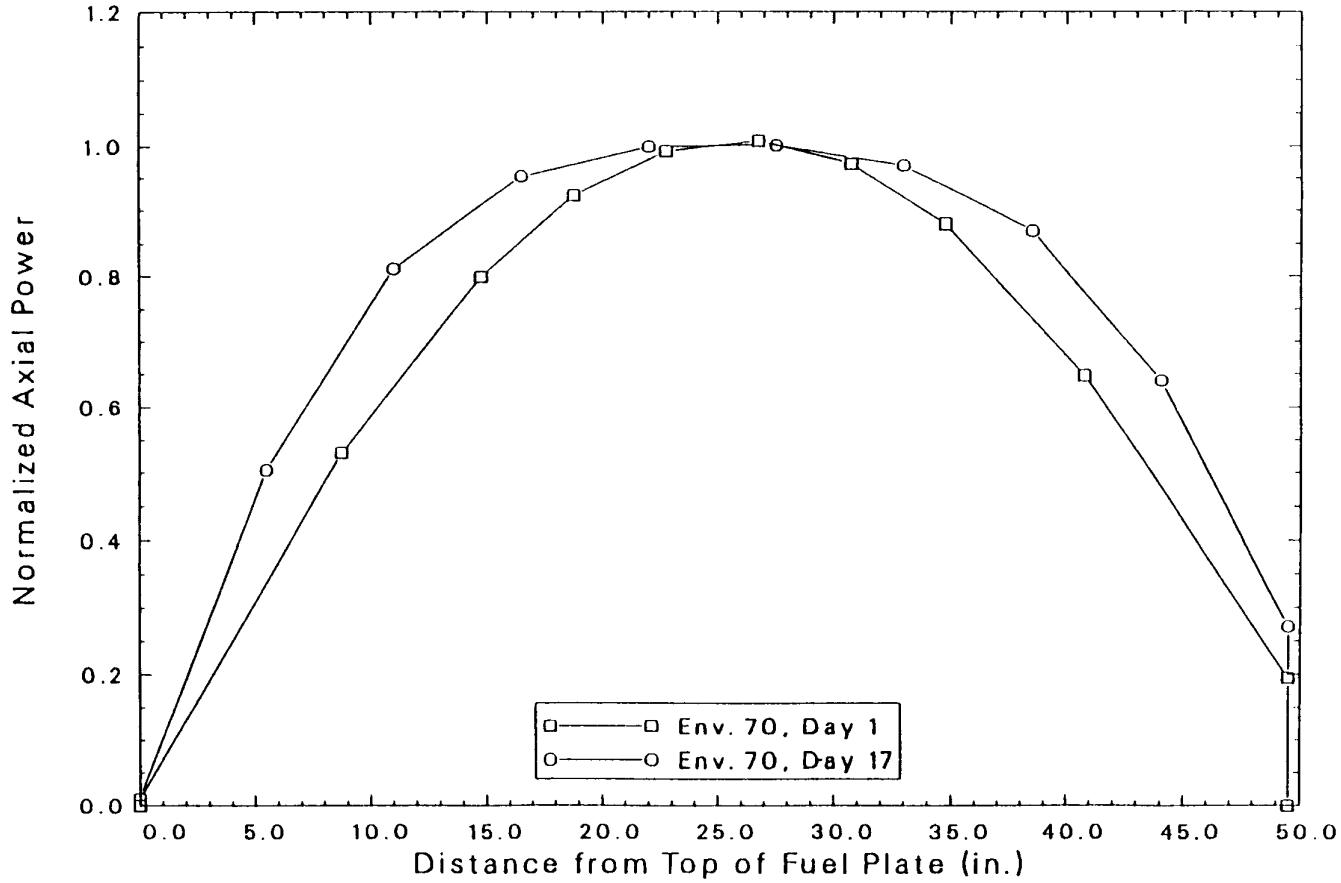

Figure 15.0-1. Normalized axial power distributions for the SAR 70 power split. **(For Information Only)**

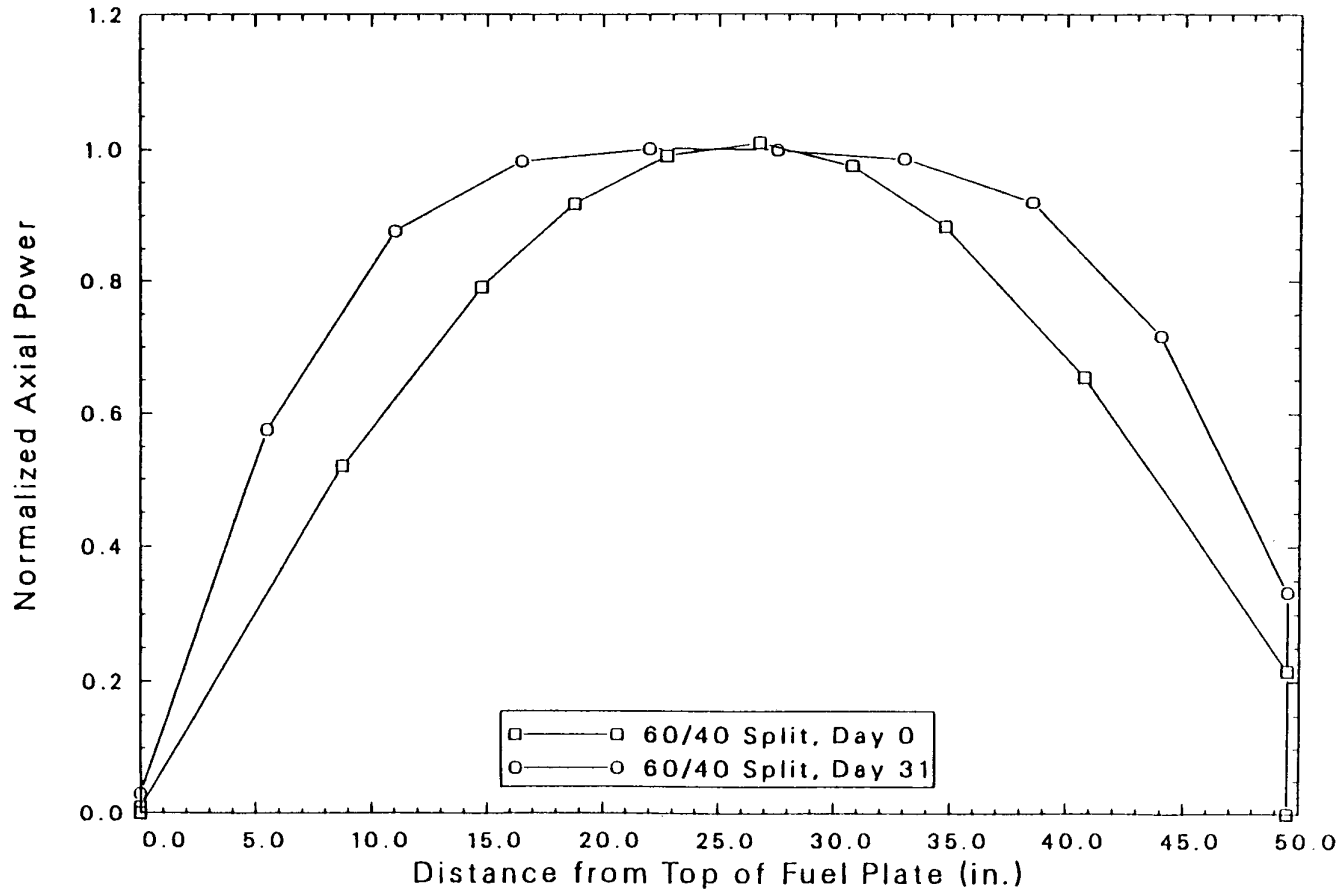

Figure 15.0-2. Normalized axial power distributions for the 60/40 power split. **(For Information Only)**

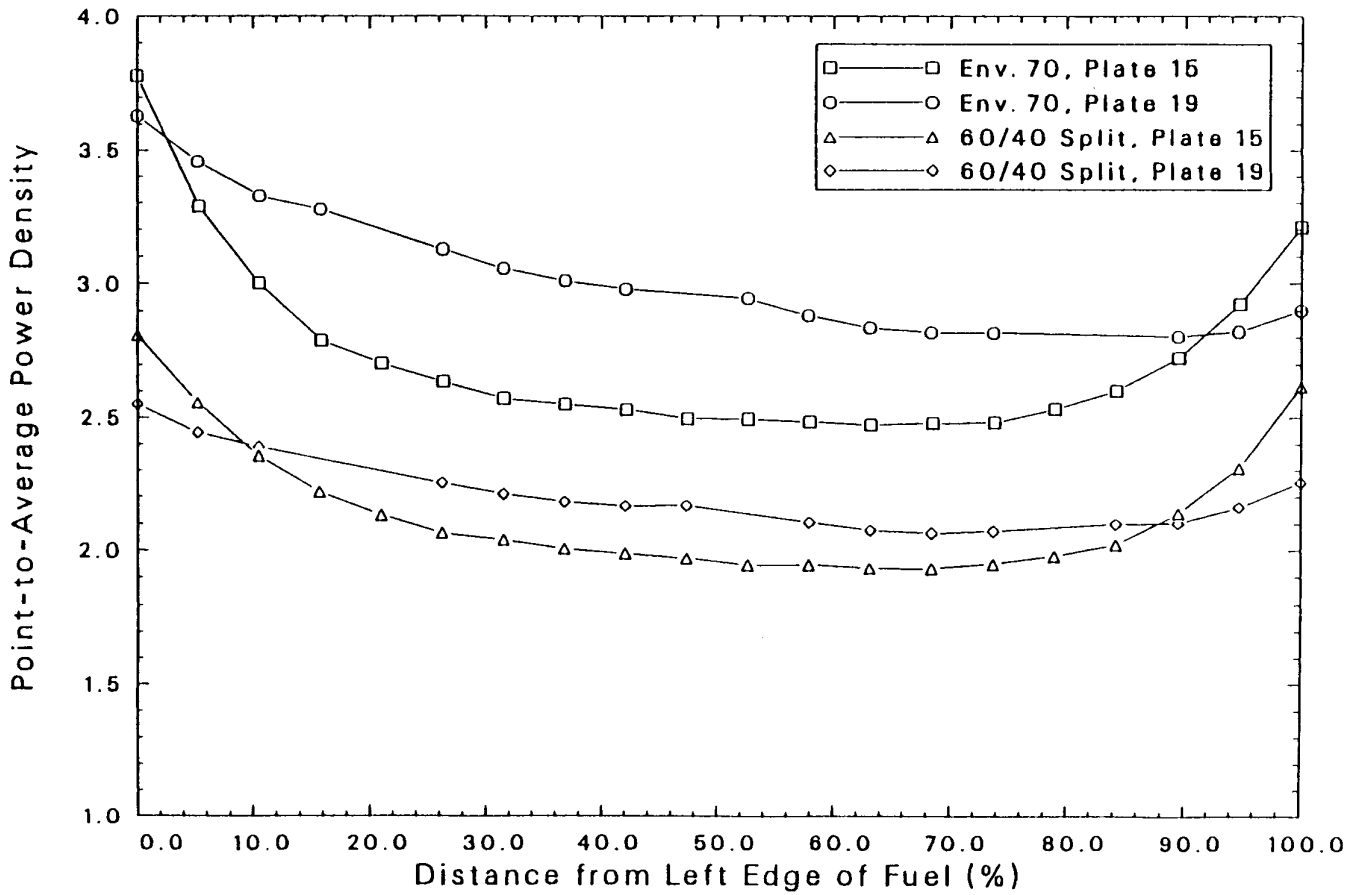

Figure 15.0-3. Azimuthal power distributions for Plate 15 and Plate 19. **(For Information Only)**

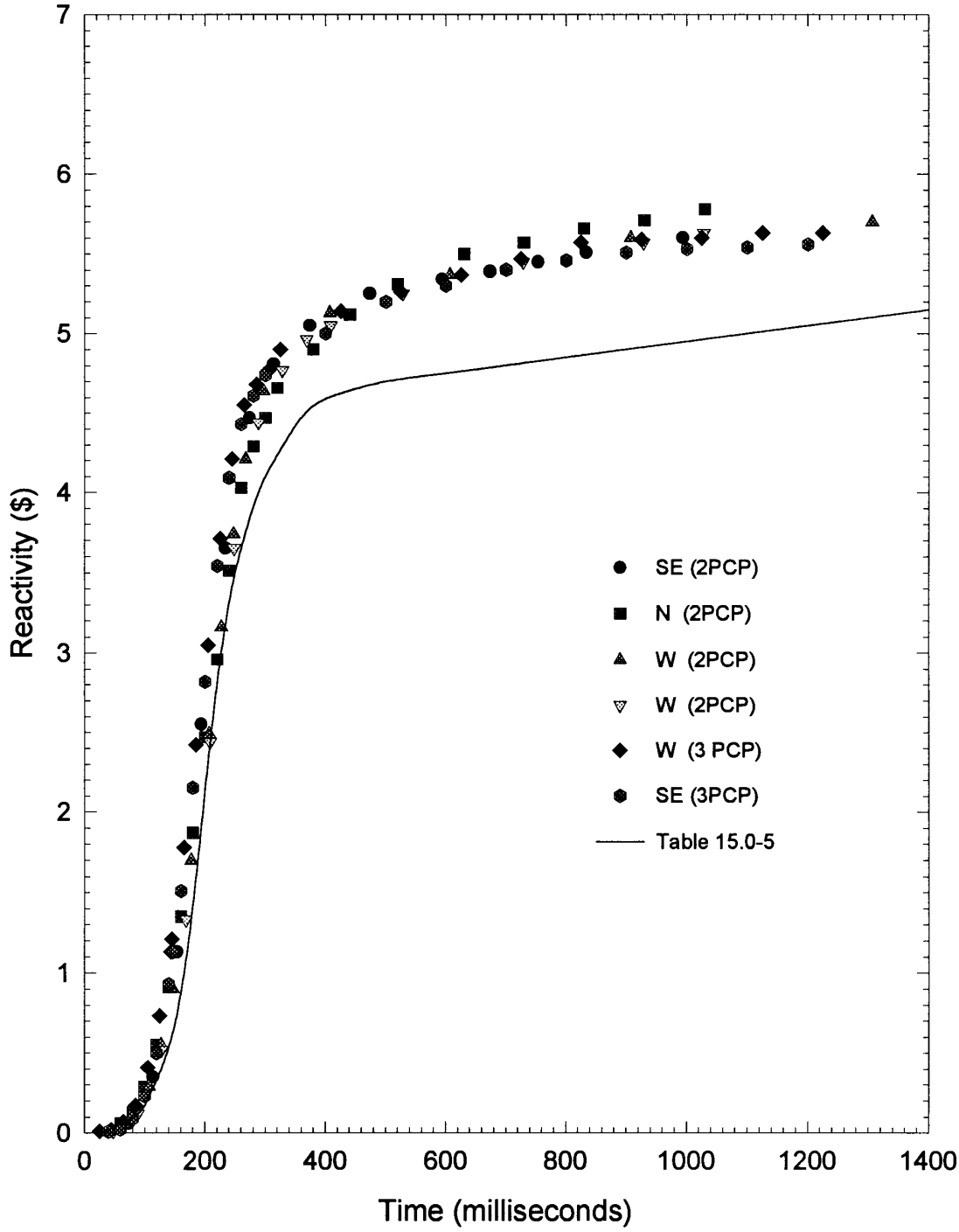

Figure 15.0-4. Safety rod reactivity time dependence. **(For Information Only)**

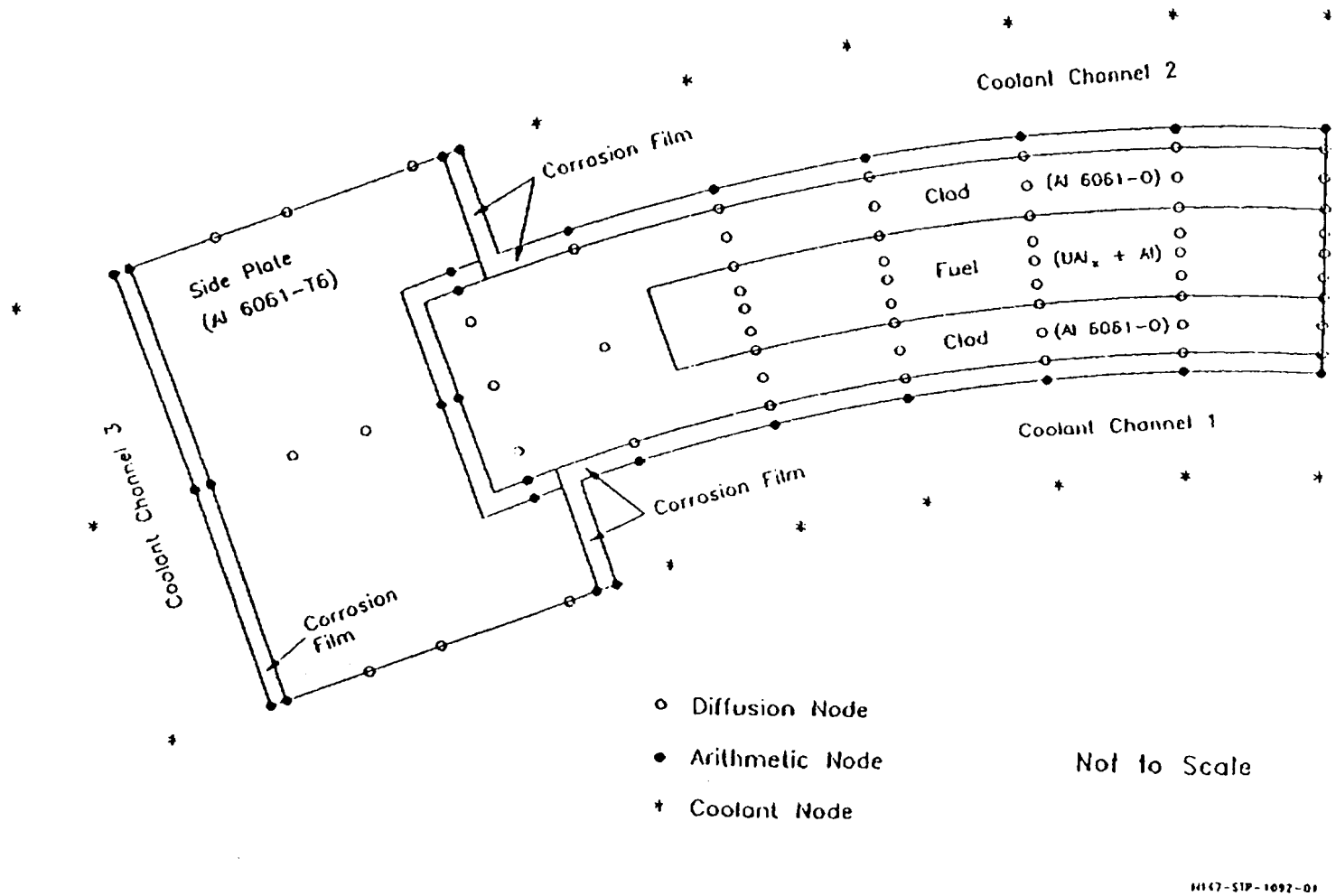

Figure 15.0-5. Cross-section of the ATR-SINDA fuel plate model. **(For Information Only)**

15.0.16 References

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