15.9 Depressurized Operations Events

The ATR is designed for operation in both the pressurized and depressurized modes. The depressurized mode is available to perform special tests (e.g., flux measurements) as needed to support pressurized operation. It has been used for requalification of the facility following major changes such as replacement of core internals. Since major changes occur infrequently (e.g., core internals replacement has occurred at approximately seven to eight year intervals), operation in the depressurized mode is relatively rare.

Depressurized or open head operation of the reactor is conducted at a desired reactor power level up to 500 kW with forced flow through the core provided by the emergency coolant pumps. Forced primary coolant flow is normally provided by two emergency coolant pumps, although the safety analyses conservatively assume flow is provided by a single pump. Depressurized operation may be with a new core or with a partially depleted core, provided that decay heat generation following reactor scram is shown to be within the conditions evaluated for the loss of coolant accident presented in Section 15.9.2. Reactor control is manual by the operator. Automatic reactor control by the servo control system is not used. The experiment loop heaters are also not used during depressurized operation. Historically, depressurized operation was allowed up to 2.5 MW; however, power levels greater than 500 kW are not required. Limiting the power level to 500 kW minimizes the accident analysis required to support the safety basis for depressurized operation without restricting required reactor operation.

The accident analysis basis supporting depressurized operation has been updated in preparation for the next major core change. The updated reactivity insertion accident analyses for depressurized operation are documented in Polkinghorne (2003a). The updated reactivity insertion analyses were performed using the ATR version of RELAP5/MOD2.5 (Davis 1999). Verification and validation of the RELAP5 reactor kinetics model is discussed in Davis (2001). Evaluations of postulated low primary coolant flow and loss of coolant accidents are documented in Sharp (2004).

Discussion in this section applies specifically to reactor operation in the depressurized configuration. The postulated loss of coolant accidents during normal reactor shut down vary significantly from those during depressurized operation. Postulated loss of coolant accidents during shutdown are discussed in Section 15.6.

15.9.1 Low Flow, Open Head Operation

Forced primary coolant flow during depressurized operation is provided by the emergency coolant pumps (ECPs). The initial flow conditions assumed in the accident analyses are conservatively based on operation of a single emergency coolant pump, although flow may be provided by 2 pumps.

Sharp (2004) identifies the following low flow events that may occur during depressurized operation.

A. Momentary loss of flow due to an emergency coolant pump failure or inadvertently turned off

B. Extended loss of flow due to extended failure of both emergency coolant pumps

C. Butterfly valve closes when shaft fails or valve is closed and stop fails

D. Core flow bypass.

Sharp (2004) categorizes each of these events with respect to their expected frequency of occurrence. Event A is categorized as Condition 3 and Events B, C, and D are categorized as Condition 4. Each event is discussed below.

15.9.1.1 Momentary Loss of Flow Due to an Emergency Coolant Pump Failure or Inadvertently Turned Off

15.9.1.1.1 Identification of Causes and Accident Description—Momentary loss of flow due to failure of the operating emergency coolant pump (or pumps), resulting in an emergency coolant pump flow coastdown, is considered a Condition 3 loss of flow event. For this event, operators are assumed to be able to restore flow from at least one ECP (e.g., restoring commercial and/or diesel power) within several hours and before adequate core cooling is lost (see Section 15.9.1.1.2). An extended loss of flow event occurring during depressurized operation, wherein operators are unable to re-establish ECP flow, is a Condition 4 event (Sharp 2004).

An emergency coolant pump coastdown would result in a rapid reduction in the primary coolant flow rate to below the scram setpoint. Following reactor scram, decay heat must be removed from the core in order to ensure no fuel damage occurs. The loss of flow conditions evaluated for this flow coastdown event also bound a pump seizure event.

The minimum required flow during depressurized operation (3600 gpm) can be provided by one ECP, however, depressurized operation is typically conducted with both ECPs operating. A simultaneous loss of commercial and diesel power to both ECPs would result in immediate coastdown of the M-10 ECP. With both ECPs initially running (typical configuration), the M-11 ECP would continue to operate for 30 minutes on battery power. Following depletion of battery power, the M-11 ECP would coastdown and reactor scram would occur on either low Emergency Pump Circulation Flow or low Emergency Flow (conservatively assuming that the loss of power does not directly initiate reactor scram). This scenario, with both ECPs initially operating, is more limiting than the case where M-10 only is initially operating. The case with only the M-10 ECP initially operating would result in an immediate reactor scram due to M-10 coastdown while the M-11 ECP remained available to provide 30 minutes of continued flow following reactor scram.

15.9.1.1.2 Analysis of Effects and Consequences—The evaluation of a momentary loss of flow is based on ATR CORE-I tests which demonstrated that the ATR can be cooled by natural convection for more than one hour when depressurized and operating at 2.5 MW (Atkinson 1976b). CORE-I tests also demonstrated that at 1.4 MW, flow reversal from forced downflow to natural convection upflow occurs within ATR fuel element channels without a heat transfer crisis (Hanson, Griebenow, and Richert 1970). The maximum allowed core power during depressurized operation is 500 kW. The core power will be further reduced to less than 25 kW (< 5% initial power) within 1 minute following reactor scram (Sharp 2004). Adequate cooling would be ensured for several hours at these conditions (if not longer) through natural circulation and the large upper vessel volume of water available as a heat sink. Forced flow from at least one ECP would be restored within this time period.

The accident consequences are not sensitive to the low flow scram setpoints. For convenience, the setpoint is conservatively based on pressurized operating conditions (discussed below). Previous analysis of emergency coolant pump coastdown during open head operation was performed with the reactor operating at 2.5 MW (Varacalle 1975, 1976; Merrill 1977, 1980). The results show that three standard deviations from FI are reached approximately 38 seconds from initiation of the transient. The number of standard deviations for CHF were greater than 6.0 at 38 seconds. At 38 seconds, the emergency coolant flow had dropped to 302 gpm. This analysis bounds the conditions for the updated safety basis for depressurized operation wherein the reactor power level is limited to 500 kW.

ATR CORE-1 heat transfer tests determined that the emergency coolant flow needed to prevent FI for decay heat removal after a loss of primary coolant flow is no more than 2,000 gpm (Atkinson 1976b). This value was used to determine the worst-case setpoint for the PPS emergency coolant pump recirculation flow subsystem for pressurized operation. The flow in the recirculation line during pressurized operation (normal PCS flow provided by the PCP with one emergency coolant pump operating) was scaled to the value that would correspond to 2000 gpm passing through the core if the PCP failed. This flow corresponds to 90 gpm or less of recirculation flow during normal, pressurized operation of the ATR (Reilly 1977). With this worst-case setpoint, and with a 10 sec response time to allow for startup of the standby emergency coolant pump to full speed before a reactor trip (in the pressurized mode), the reactor would be tripped (by a release of the safety rods), or the standby emergency coolant pump would be operating by the time emergency coolant flow from the failed pump had decreased to 805 gpm.

The 2000 gpm value was used as the worst-case setpoint for the PPS emergency coolant pump discharge flow subsystem which is operable during depressurized operation. The PPS emergency coolant pump recirculation flow subsystem is also operable during depressurized operation. The 90 gpm (determined as the worst-case setpoint for pressurized operation) is conservative for depressurized operation (McCracken 1992). With these setpoints, the reactor will be tripped significantly before three standard deviations from FI would be approached for an emergency coolant pump coastdown during depressurized operation. (The logic on low recirculation flow does not include the 10-sec delay in the depressurized mode. The scram is diverse to the low flow scram.) The subsystem trip responses are:

TRIP INITIATOR	FLOW AFTER 10-SEC RESPONSE TIME
2,000-gpm emergency coolant flow trip	805 gpm
90-gpm recirculation flow trip	1,065 gpm

15.9.1.2 Extended Loss of Flow Due to Extended Failure of Both Emergency Coolant Pumps

15.9.1.2.1 Identification of Causes and Accident Description—Extended loss of flow due to failure of both emergency coolant pumps is considered a Condition 4 loss of flow event. This postulated event differs from the momentary loss of flow event discussed above in that the operators are not able to reestablish primary coolant flow from at least one emergency coolant pump.

15.9.1.2.2 Analysis of Effects and Consequences—Core heat generation following a complete loss of flow event can be successfully transferred by natural convection for at least several hours as concluded for the momentary loss of flow event in Section 15.9.1.1. For an extended loss of flow, operators would establish core cooling (if needed) by manually actuating the emergency firewater injection system (EFIS).

15.9.1.3 Butterfly Valve Closes When Shaft Fails or Valve is Closed and Stop Fails

15.9.1.3.1 Identification of Causes and Accident Description—Rapid closing of the butterfly valve can occur by failure of the valve shaft or by the valve stop failing. These events are conservatively considered to be a Condition 4 loss of flow event (Sharp 2004). Torque on the disk is expected to be substantially lower in depressurized operation than in pressurized operation, and it is very conservative to consider this event.

15.9.1.3.2 Analysis of Effects and Consequences—The butterfly valve disk is smaller than the valve throat area, therefore, a complete flow stoppage cannot occur (Sharp 2004). Flow to the reactor core is expected to remain greater than the low flow scram setpoint. For this condition, the continued flow will be sufficient to remove the maximum allowed core power level of 500 kW during depressurized operation. Should the flow to the reactor core be reduced to less than the low flow scram setpoint, resulting in a reactor scram, the event would be bounded by the evaluation for an extended loss of flow event discussed above.

15.9.1.4 Core Flow Bypass

15.9.1.4.1 Identification of Causes and Accident Description—Core flow bypass due to a hole in an outlet pipe inside the vessel is discussed as a Condition 4 loss of flow event even though the estimated probability is below the Condition 4 bounds (i.e., a beyond design basis event). This event, however, bounds all credible core bypass events that are due to holes in a 16-inch outlet pipe in the reactor vessel.

15.9.1.4.2 Analysis of Effects and Consequences—An analysis of core flow bypass (Merrill and Haroldsen 1976), due to holes in a 16-inch outlet pipe in the ATR vessel, determined that even with a 16-inch hole, 2,162 gpm of emergency coolant flow still passes through the core (with operation of one emergency coolant pump), and that all quadrant flows are nearly the same. Since this amount of emergency coolant flow is sufficient, no PPS action is required.

15.9.2 Loss of Coolant Accidents

Operation of the ATR in a depressurized or open head condition is infrequent. Additionally, piping stress is less severe relative to pressurized operation because of reduced pressures and temperature conditions. Therefore, the occurrence of a loss of coolant accident (LOCA) during depressurized operation has a greatly reduced frequency of occurrence.

Sharp (2004) identifies the postulated LOCA events that may occur during depressurized operation.

A. Opening valve outside radiographic limits or at radiographic boundary

B. Piping rupture outside radiographic limits

C. Multiple drain valves opened

D. Pressure tube penetration or bottom head closure plug failure.

Sharp (2004) categorizes each of these events with respect to their expected frequency of occurrence. Event A is categorized as Condition 3 and Events B, C, and D are categorized as Condition 4. Each event is discussed below.

15.9.2.1 Opening Valve Outside Radiographic Limits or at Radiographic Boundary

15.9.2.1.1 Identification of Causes and Accident Description—Opening a valve outside the radiographic limits or at the radiographic boundary is considered a Condition 3 event (Sharp 2004) that could cause a LOCA during open head operation. Outside of radiographic limits refers to piping outside of the PCS radiographic boundary. The PCS radiographic boundary is defined in Chapter 5 (Primary Coolant System).

15.9.2.1.2 Analysis of Effects and Consequences—Opening a valve outside the radiographic limits or at the radiographic boundary would result in a slow system draining which could be easily terminated by operator action (e.g., closing the valve). If the loss of coolant were not mitigated, the vessel level subsystem would initiate release of the safety rods to terminate depressurized reactor operation. Following reactor scram, decay heat must be removed from the core in order to ensure no fuel damage occurs. With irradiated fuel in the reactor vessel, the EFIS would be in manual mode and reactor operators would take action to initiate emergency firewater injection. No credit for operator action to initiate emergency firewater injection is assumed in the evaluation of this event unless irradiated fuel, as defined by the Technical Safety Requirements, is present in the reactor core.

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Depressurized operation is performed with fresh or nearly fresh fuel in the core that would not be categorized as irradiated by the definition provided in the ATR Technical Safety Requirements. Furthermore, depressurized operation normally occurs at a core power level less than 20 kW. For these conditions, the maximum core decay heat generation 10 minutes following reactor scram is less than 300 W and much less than 30 W in the limiting fuel element (Sharp 2004). The decay heat generation at the time of possible uncovering of the core is low and will be adequately removed without fuel damage occurring (Sharp 2004). Therefore, if power is limited to ≤ 20 kW during depressurized operation, EFIS operability and actuation are not required in response to a LOCA event. This condition assumes that fresh or nearly fresh fuel was initially loaded into the reactor core for the intended operation at depressurized conditions.

15.9.2.2 Piping Rupture Outside Radiographic Limits

15.9.2.2.1 Identification of Causes and Accident Description—PCS piping rupture outside the radiographic limits is conservatively considered a Condition 4 event that could cause a LOCA during depressurized reactor operation.

15.9.2.2.2 Analysis of Effect and Consequences—The break size and drain rate for this event are bounded by the evaluation for the Condition 3 valve opening event discussed above. Therefore, the consequences for a piping rupture outside the radiographic limits are bounded by the consequences determined for the opening of a valve outside the radiographic limits.

15.9.2.3 Multiple Drain Valves Opened

15.9.2.3.1 Identification of Causes and Accident Description—The unintentional opening of several drain valves (all of which are outside the radiographic limits), or opening a normally inaccessible or locked valve, is considered a Condition 4 event that could cause a LOCA during depressurized operation.

15.9.2.3.2 Analysis of Effects and Consequences—Tests at the ATR, and an analysis at Idaho State University for ATR siphon breaker performance, have shown that the vessel will not drain if multiple drain valves are opened (Gose and Neill 1975). The potential consequences of opening several drain valves are therefore the same as for the discussion in Section 15.9.2.1.

15.9.2.4 Pressure Tube Penetration or Bottom Head Closure Plug Failure

15.9.2.4.1 Identification of Causes and Accident Description—Failure of a bottom head closure plate penetration (closure plug), resulting in a LOCA, is conservatively considered a Condition 4 event during depressurized operation. As discussed in Section 15.6.2, this event is not considered credible during normal (pressurized) reactor operation. The event is more likely to occur during reactor shutdown due to an error occurring during maintenance activities.

15.9.2.4.2 Analysis of Effects and Consequences—The break size and drain rate for this bottom head LOCA event are bounded by the evaluation for the Condition 3 valve opening event. Therefore, no additional analysis or evaluation is required. Because the event consequences are bounded by the evaluation of the valve opening event, no controls or commitment are required to prevent maintenance activities on systems associated with this portion of the primary coolant system. However, maintenance activities on associated systems are not expected to occur during depressurized operation.

15.9.3 Reactivity Accidents

Polkinghorne (2003b) calculated a conservative overpower limit for application to depressurized operation using the ATR version of RELAP5/MOD2.5 (Davis 1999) and ATR-SINDA Version B Revision 1 and SINDA-SAMPLE Version B Revision 1 (Polkinghorne 2002). The thermal analysis for steady-state depressurized operation with 3,600 gpm of emergency coolant flow and 125°F at the vessel inlet determined that depressurized operation could occur up to a power of 7.3 MW in an extreme 70/20 lobe power split while maintaining margin from CHF and FI of at least three standard deviations (Polkinghorne 2003b). Therefore, a large margin exists for depressurized operation at a power level of 500 kW before approaching core damage thresholds. The core power limit (7.3 MW) is applied in each of the reactivity insertion accident analyses for depressurized operation listed below.

For ATR depressurized operation, two PPS trips are available to provide reactor protection on high neutron level. A conservative scram setpoint of 1.15 MW (2.3 × 500 kW) was modeled in the reactivity insertion accident analysis. This bounds the worst case actuation setpoint of 600 kW (1.2 × 500 kW), considering potential measurement and instrumentation uncertainties of both the neutron level and WRS neutron level functions. During depressurized operation, both trips are operable at all times. A 35-msec delay time was assumed in the reactivity insertion accident analyses, which also bounds the performance for each of these functions.

The period protection function of the WRS is also available during depressurized operation. The period function provides reactor protection for a reactivity insertion event initiated from power levels significantly less than 500 kW. The period ranges assumed for the analyses are shown in Table 15.9-1 (Sharp 2004).

A ramp reactivity insertion event (e.g., $/sec) initiated from a relatively low power level will result in a scram actuated by the period function of the WRS. The limiting case for a specific reactivity insertion rate, resulting in the peak core power level attained, occurs when the event is initiated from the minimum power level which fails to exceed the period setpoint prior to leaving the highest power interval (150 to 500 kW). In this case, a minimum reactor power period occurs in combination with the reactor scram actuation being delayed until actuated by either neutron level or WRS neutron level functions. Table 15.9-2 provides various ramp reactivity insertion rates and the corresponding limiting initial and maximum core power levels (Polkinghorne 2003a). The maximum core power level as a function of the reactivity insertion ramp rate is also presented in Figure 15.9-1 (Polkinghorne 2003a).

A number of postulated events that would result in a reactivity insertion and/or power excursion have been analyzed:

A. Withdrawal of all outer shims from 10^{-5} N_F

B. Rapid reg-rod-withdrawal

C. Withdrawal of all neck shims and outer shims from 10^{-5} N_F

D. Withdrawal of all outer shims from 10^{-10} N_F

E. Cold loop draining due to open and accessible valve

F. Loop heaters accidentally on

G. Slow power increase or lobe power balance shift

H. Perched fuel drop from within measurement accuracy (one in.)

I. Cold water addition due to inadvertent startup of a PCP or inadvertent actuation of emergency firewater injection

J. Withdrawal of all outer shims and neck shims from 10^{-10} N_F

K. Cold experiment loop draining due to an open, normally inaccessible valve

L. Step Reactivity Insertions

Sharp (2004) categorizes each of these occurrences with respect to their expected frequency. Event A is categorized as Condition 2, Events B through I are categorized as Condition 3 and Events J through L are categorized as Condition 4. The events and analyses are discussed below.

15.9.3.1 Withdrawal of All Outer Shims from 10^{-5} N_F

15.9.3.1.1 Identification of Causes and Accident Description—The continuous withdrawal of all outer shims from 10^{-5} N_F is categorized as a Condition 2 event. Since protection from the ATR intermediate timer is terminated above 10^{-5} N_F, there is no restriction on withdrawal of the outer shims. In certain circumstances, withdrawal of all outer shims is, in fact, a normal operating procedure during both pressurized and depressurized operation. However for this analysis, it is postulated that operator inattention while pulling the outer shims could result in an excessive reactivity insertion.

15.9.3.1.2 Analysis of Effects and Consequences—As discussed in Section 15.4.2, the minimum reactivity insertion rate for withdrawal of all OSCCs is 0.053$/sec. Initiated from a relatively low power level, a 0.053$/sec reactivity insertion will result in a scram actuation by the period function of the WRS. The limiting case, resulting in the peak core power level, results when the insertion is initiated from the minimum power level which fails to exceed the WRS period setpoint prior to leaving the highest power interval. In this case a minimum period is achieved in combination with a reactor scram actuated by either the neutron level or WRS neutron level functions.

As shown in Figure 15.9-1 (Polkinghorne 2003a), the maximum power level predicted for a 0.053$/sec ramp reactivity insertion accident is less than 1.4 MW. Thermal analysis for the ATR hot channel, using models for steady-state depressurized operation with 3600 gpm emergency coolant flow at 125°F at the inlet, determined that depressurized operation could occur up to a power of 7.3 MW in an extreme 70/20 lobe power split while maintaining margin from CHF and FI of at least three standard deviations (Polkinghorne 2003b). Therefore, fuel integrity is maintained and the event consequences will not challenge the plant protection criteria for a Condition 2 event.

15.9.3.2 Rapid Reg-Rod-Withdrawal

15.9.3.2.1 Identification of Causes and Accident Description—A component failure can cause the regulating rod to drive out, thus inserting positive reactivity. The postulated reg-rod-withdrawal accident results in a 1.05$/sec insertion up to a maximum (total) insertion of 0.57$ (Section 5.4.7.1). Rapid reg-rod-withdrawal is categorized as a Condition 3 event during depressurized operation (Sharp 2004). This is a conservative event category since reg-rod movement is manually controlled during depressurized operation.

15.9.3.2.2 Analysis of Effects and Consequences—As discussed above, the maximum reactivity insertion rate for a rapid reg-rod-withdrawal is 1.05$/sec. The limiting event during depressurized operation, resulting in the peak core power level, results when the reactivity insertion is initiated from 500 kW. Figure 15.9-2 presents the core power transient and Figure 15.9-3 presents the net core reactivity for the rapid reg-rod-withdrawal event (Polkinghorne 2003a). A reactor scram is initiated when the power level reaches 1.15 MW at approximately 0.5 seconds. The safety rods are released following a 0.035 second delay. The maximum core power of 1.32 MW is attained at approximately 0.6 seconds. Thermal analysis for the ATR hot channel determined that depressurized operation could occur up to a power of 7.3 MW in an extreme 70/20 lobe power split while maintaining margin from CHF and FI of at least three standard deviations (Polkinghorne 2003b). Therefore, fuel integrity is maintained and the event consequences will not challenge the plant protection criteria for a Condition 3 event.

15.9.3.3 Withdrawal of All Neck Shims and Outer Shims from 10^{-5} N_F

15.9.3.3.1 Identification of Causes and Accident Description—Normally one neck shim can be selected for withdrawal when the ATR power level is above 10^{-5} N_F, the upper level of the power range where the control rod withdrawal timer interlock is turned off. Additionally, procedures do not support multiple neck shim withdrawal. Therefore, withdrawal of all neck shims and outer shims from 10^{-5} N_F is categorized as a Condition 3 event during depressurized operation.

15.9.3.3.2 Analysis of Effects and Consequences—The reactivity insertion of this postulated control rod withdrawal accident is 0.079$/sec (Section 5.4.2). Initiated from a relatively low power level, a 0.079$/sec reactivity insertion will result in a scram actuation by the period function of the WRS. The limiting case, resulting in the peak core power level, occurs when the insertion is initiated from the minimum power level which fails to exceed the period setpoint prior to leaving the highest power interval. In this case a minimum period is achieved in combination with a reactor scram actuated by either neutron level or WRS neutron level functions.

From Figure 15.9-1, withdrawal of all neck shims and outer shims from 10^{-5} N_F (worth 0.079$/sec) results in less than 1.5 MW peak power. Thermal analysis for the ATR hot channel determined that depressurized operation could occur up to a power of 7.3 MW in an extreme 70/20 lobe power split while maintaining margin from CHF and FI of at least three standard deviations (Polkinghorne 2003b). Therefore, fuel integrity is maintained and the event consequences will not challenge the plant protection criteria for a Condition 3 event.

15.9.3.4 Withdrawal of All Outer Shims from 10^{-10} N_F.

15.9.3.4.1 Identification of Causes and Accident Description—This occurrence requires a timer interlock failure, which can occur undetected due to a single failure in the Log-N system. This is categorized as a Condition 3 event during depressurized reactor operation (Sharp 2004). As discussed in Section 15.9.3.1, the maximum reactivity addition from simultaneous withdrawal of all outer shims is 0.053$/sec.

15.9.3.4.2 Analysis of Effects and Consequences—Withdrawal of all outer shims from 10^{-10} N_F (worth 0.053$/sec) is bounded by the Condition 2 case that considered withdrawal of all outer shims from 10^{-5} N_F. The peak power is less than 1.4 MW. Therefore, fuel integrity is maintained and the event consequences will not challenge the plant protection criteria for a Condition 3 event as defined in Section 15.0.4

15.9.3.5 Cold Loop Draining Due to an Open and Accessible Valve.

15.9.3.5.1 Identification of Causes and Accident Description—Cold loop draining due to an open and accessible valve is considered a Condition 3 event. The reactivity insertion rate for the cold loop draining event is 0.06$/sec (Durney 1980b) and the total reactivity insertion is significantly less then the negative reactivity provided by the safety rods following reactor scram

15.9.3.5.2 Analysis of Effects and Consequences—.

From Figure 15.9-1, a 0.06$/sec reactivity insertion results in less than 1.5 MW peak power. Thermal analysis for the ATR hot channel determined that depressurized operation could occur up to a power of 7.3 MW in an extreme 70/20 lobe power split while maintaining margin from CHF and FI of at least three standard deviations (Polkinghorne 2003b). Therefore, fuel integrity is maintained and the event consequences will not challenge the plant protection criteria for a Condition 3 event as defined in Section 15.0.4.

15.9.3.6 Loop Heaters Accidentally On

15.9.3.6.1 Identification of Causes and Accident Description—Loop heaters accidentally left on in a single loop during depressurized reactor operation, resulting in loop voiding, is categorized as a Condition 3 event (Sharp 2004). This is a very conservative event frequency category for this accident since loop heaters are not used during depressurized operation. The maximum worth of this event would be 0.22$/sec for the AHTL (Durney 1980a, 1983) or 0.10$/sec for other loops (Atkinson 1976a). The event is enveloped by the AHTL response.

15.9.3.6.2 Analysis of Effects and Consequences—From Figure 15.9-1, a 0.22$/sec reactivity insertion results in a peak power less than 1.9 MW. Thermal analysis for the ATR hot channel determined that depressurized operation could occur up to a power of 7.3 MW in an extreme 70/20 lobe power spit while maintaining margin from CHF and FI of at least three standard deviations (Polkinghorne 2003b). Therefore, fuel integrity is maintained and the event consequences will not challenge the plant protection criteria for a Condition 3 event.

15.9.3.7 Slow Power Increase or Lobe Power Balance Shift

15.9.3.7.1 Identification of Causes and Accident Description—Shim failure or operator error could cause a slow power increase or a lobe power balance shift. The slow power increase or lobe power balance shift is considered a Condition 3 event during depressurized reactor operation. Depressurized operations are required to be performed with reactivity control in manual (i.e., the regulating rod control system is not used). Thus the frequency of occurrence and the postulated effects and consequences for this event are very conservative.

15.9.3.7.2 Analysis of Effects and Consequences—The lobe power unbalance accident causes a different type of power increase; the total reactor power will be held nearly constant by the operator while the power increases in one lobe. If one of the neutron level subsystem detectors is assumed to be failed (the detector facing the lobe in which the power is rising), then the neutron level PPS subsystem will not be effective in stopping the power rise in the affected lobe until total reactor power reaches the neutron level or WRS trip point. The accident is essentially the same as the lobe power shift accident discussed for pressurized operation in SAR Section 15.4.3. The analysis performed for pressurized conditions concluded that a reactor scram setpoint of 1.24 N_F would result in a maximum lobe power of 134% of the initial lobe power. This corresponds to the affected lobe power leading the reactor power by up to a maximum of 10% (Atkinson 1976b). Consistent with the other reactivity accidents discussed in Section 15.9, a conservative scram setpoint of 1.15 MW (2.3 × 500 kW) is assumed for this event. The corresponding maximum lobe power will be 2.4 times the initial lobe power level (10% greater than the assumed scram setpoint). Conservatively assuming that the total core power is 2.4 times the initial core power (i.e., all lobes 10% above the safety analysis scram setpoints) results in a maximum core power of 1.2 MW (2.4 × 500 kW). Since the core power could increase to 7.3 MW before a margin of three standard deviations from CHF or FI is reached (Polkinghorne 2003b), a large protective margin exists for this accident.

15.9.3.8 Perched Fuel Drop from Within Measurement Accuracy (One Inch)

15.9.3.8.1 Identification of Causes and Accident Description—A perched fuel element drop from within the measurement accuracy (one in.) occurring during depressurized operation is categorized as a Condition 3 event. The driving mechanisms for inducing an element drop are coolant flow and vibration. Core coolant flow is established before reactor criticality is achieved. All essential vibration-inducing mechanisms such as flow, pumps, and diesels are also in operation prior to criticality. The vibration induced by these mechanisms is also less severe compared to pressurized operation primarily due to the reduced flow requirements at depressurized conditions.

15.9.3.8.2 Analysis of Effects and Consequences—ATRC measurements for conservative conditions indicate a maximum event insertion of 0.062$ for a 1-in. insertion of the highest worth element. This estimate of the reactivity insertion for the fuel element movement is very conservative due to the assumption that a 1-in. movement is worth 1/48 of the total worth. A more exact treatment using the accepted S-curve reactivity dependence results in only 0.03$ added for a 1-in. movement of the 7NB fuel element (Durney and Hallinan 1978). (The 1-in. movement of the 7F element would be worth somewhat less due to the boron content). However, the reactivity insertion assumed for the 7F element in this accident (0.062$) envelops the insertion from the 7NB element.

The limiting case, resulting in the peak core power level, results when the reactivity insertion is initiated from 500 kW. Figure 15.9-4 presents the core power transient and Figure 15.9-5 presents the net core reactivity for a 0.10$ step reactivity insertion, which bounds a 0.062$ step insertion (Polkinghorne 2003a). Following a prompt jump to 556 kW, the core power slowly increased to the reactor scram setpoint of 1.15 MW at approximately 53 seconds. The safety rods were released following a 0.035 second delay. The maximum core power calculated was 1.15 MW. Thermal analysis for the ATR hot channel determined that depressurized operation could occur up to a power of 7.3 MW in an extreme 70/20 lobe power split while maintaining margin from CHF and FI of at least three standard deviations (Polkinghorne 2003b). Therefore, fuel integrity is maintained and the event consequences will not challenge the plant criteria for a Condition 3 event.

15.9.3.9 Cold Water Addition Due to Inadvertent Startup of a Primary Coolant Pump or Inadvertent Actuation of Emergency Firewater Injection

15.9.3.9.1 Identification of Causes and Accident Description—Cold water addition due to the inadvertent startup of a PCP or inadvertent actuation of emergency firewater injection are considered Condition 3 events during depressurized reactor operation. This type of occurrence results in a positive reactivity insertion in the core. The cold water due to the inadvertent PCP startup is transported to the core at greater than 10,000 gpm. For inadvertent actuation of the EFIS, only 4000 gpm of cold water will be transported into and mixed with the water in the reactor vessel. Therefore, the PCP startup event is more limiting.

15.9.3.9.2 Analysis of Effects and Consequences—Startup of a PCP when the ATR is operating at depressurized conditions can result in injection of cold water into the core resulting in an increase in reactivity (assumes the isothermal temperature coefficient is negative). The reactivity insertion can be modeled as a ramp increase over a period of 10 or 20 sec, which is the approximate time for the pump to reach full speed. A conservative core reactivity void (or density change) coefficient of 0.253$/% is assumed for a 70/20 core power split (Carpenter 1974). Assuming the core coolant is initially at 125°F (conservatively high), and the coolant temperature at the primary pump suction is 40°F (conservatively low), then the reactivity worth of the change in core temperature (neglecting the effect of the regulating rod) would be 0.433$ (Sharp 2004).

The resulting reactivity insertion rate is estimated to be 0.04$/sec. From Figure 15.9-1, a 0.04$/sec reactivity insertion results in a peak power less than 1.4 MW. Thermal analysis for the ATR hot channel determined that depressurized operation could occur up to a power of 7.3 MW in an extreme 70/20 lobe power split while maintaining margin from CHF and FI of at least three standard deviations (Polkinghorne 2003b). Therefore, fuel integrity is maintained and the event consequences will not challenge the plant protection criteria for a Condition 3 event.

15.9.3.10 Withdrawal of All Outer Shims and Neck Shims from 10^{-10} N_F

15.9.3.10.1 Identification of Causes and Accident Description—As discussed in Section 15.9.3.3 withdrawal of all outer shims and neck shims from 10^{-5} N_F is categorized as a Condition 3 event during depressurized operation. The additional failure of the control rod withdrawal timer interlock is also required for this event to occur from 10^{-10} N_F, hence, this occurrence is categorized as a Condition 4 event. The reactivity insertion of this postulated control rod withdrawal accident is 0.079$/sec (Section 15.4.2).

15.9.3.10.2 Analysis of Effects and Consequences—If initiated from a relatively low power level, a 0.079$/sec reactivity insertion will result in a scram actuation by the period function of the WRS. As a result, the analysis of this event is equivalent to withdrawal of the outer shims and neck shims from 10^{-5} N_F discussed in Section 15.9.3.3. That is, the peak core power level occurs when the insertion is initiated from the minimum power level which fails to exceed the period setpoint prior to leaving the highest power interval. In this case a minimum period is achieved in combination with a reactor scram actuated by either neutron level or WRS neutron level functions.

From Figure 15.9-1, withdrawal of all neck shims and outer shims (worth 0.079$/sec) results in a peak power less than 1.5 MW. Thermal analysis for the ATR hot channel determined that depressurized operation could occur up to a power of 7.3 MW in an extreme 70/20 lobe power split while maintaining margin from CHF and FI of at least three standard deviations (Polkinghorne 2003b). Therefore, fuel integrity is maintained and the event consequences will not challenge the plant protection criteria for a Condition 4 event.

15.9.3.11 Cold Experiment Loop Draining Due to an Open Normally Inaccessible

15.9.3.11.1 ***Identification of Causes and Accident Description***—A cold experiment loop draining event due to an open, but normally inaccessible, valve is considered a Condition 4 event (Sharp 2004). This event is equivalent to a cold loop draining event due to an open accessible valve (Section 15.9.3.5).

15.9.3.11.2 Analysis of Effects and Consequences—The accident conditions and consequences of this event are bounded by the evaluation in Section 15.9.3.5.

15.9.3.12 Step Reactivity Insertions

15.9.3.12.1 Identification of Causes and Accident Description—Perched fuel drop of two fuel elements from within measurement accuracy (one in.) is considered a Condition 4 event during depressurized operation. The driving mechanisms for inducing subsequent element release are gravity, flow and vibration. Before reactor criticality is achieved, core coolant flow is established and essentially all vibration inducing mechanisms (e.g., flow, pumps, and diesels) are in operation. The reactivity worth of two fuel elements falling into place is approximately 0.06$ (Sharp 2004).

Movement of an experiment (or equivalent) may also cause a step reactivity insertion during depressurized operation that is categorized as a Condition 4 event. The reactivity insertion for this occurrence is limited by restricting the reactivity worth due to movement of an experiment (or equivalent) to $\leq$ 0.10$. Driven experiments could have a worth greater than 0.10$ and, therefore, are not loaded into the reactor during depressurized operation. Therefore, movement of a driven experiment does not represent a credible reactivity insertion event.

15.9.3.12.2 Analysis of Effects and Consequences—Drop of two perched fuel elements from within measurement accuracy or movement of an experiment (or equivalent) may result in a maximum step reactivity insertion of 0.10$. The consequences of this occurrence are bounded by the analysis performed for the Condition 3 drop of a single perched element (assumed to be worth 0.10$) discussed in Section 15.9.3.8.

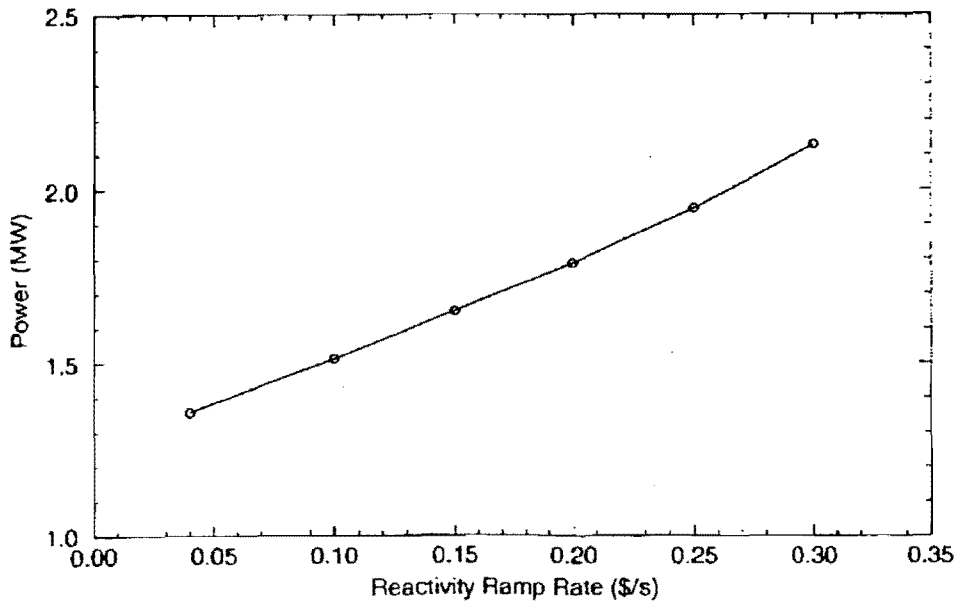

Figure 15.9-1. Maximum expected power versus constant ramp reactivity insertion rate for depressurized (open head) operation with 70/20 core power distribution.

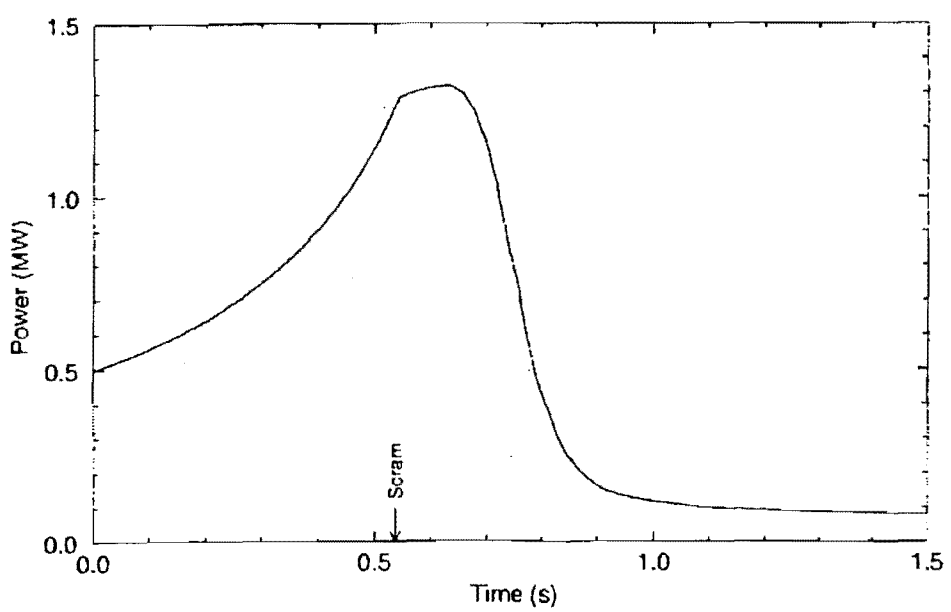

Figure 15.9-2. Reactor power transient for a rapid reg-rod withdrawal accident during depressurized (open head) operation.

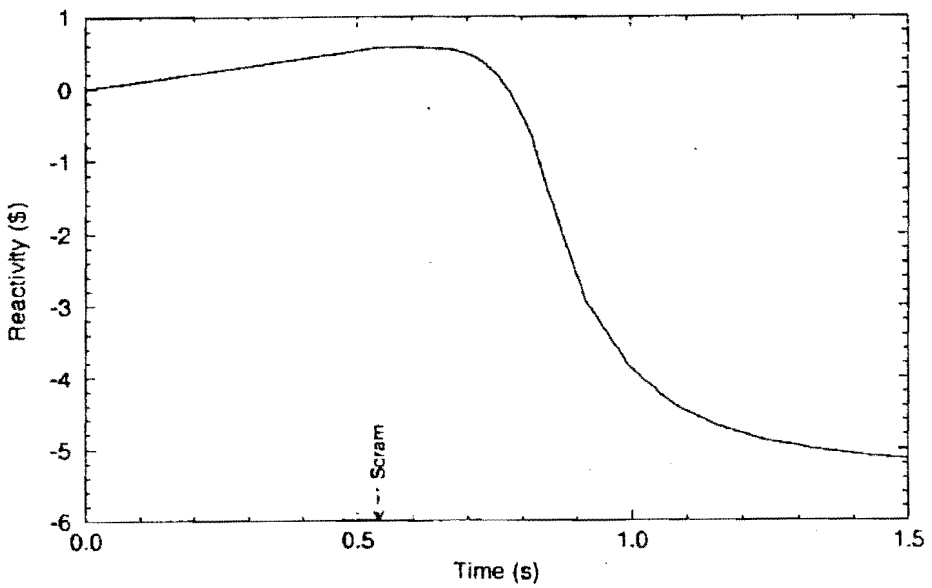

Figure 15.9-3. Core reactivity transient for a rapid reg-rod withdrawal accident during depressurized (open head) operation.

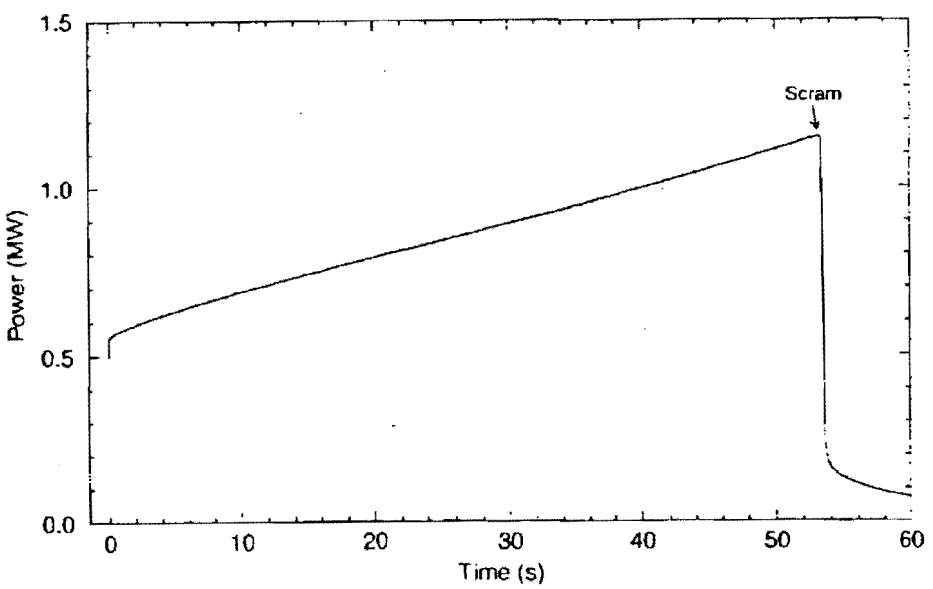

Figure 15.9-4. Reactor power transient for a 0.10$ reactivity insertion accident during depressurized (open head) operation.

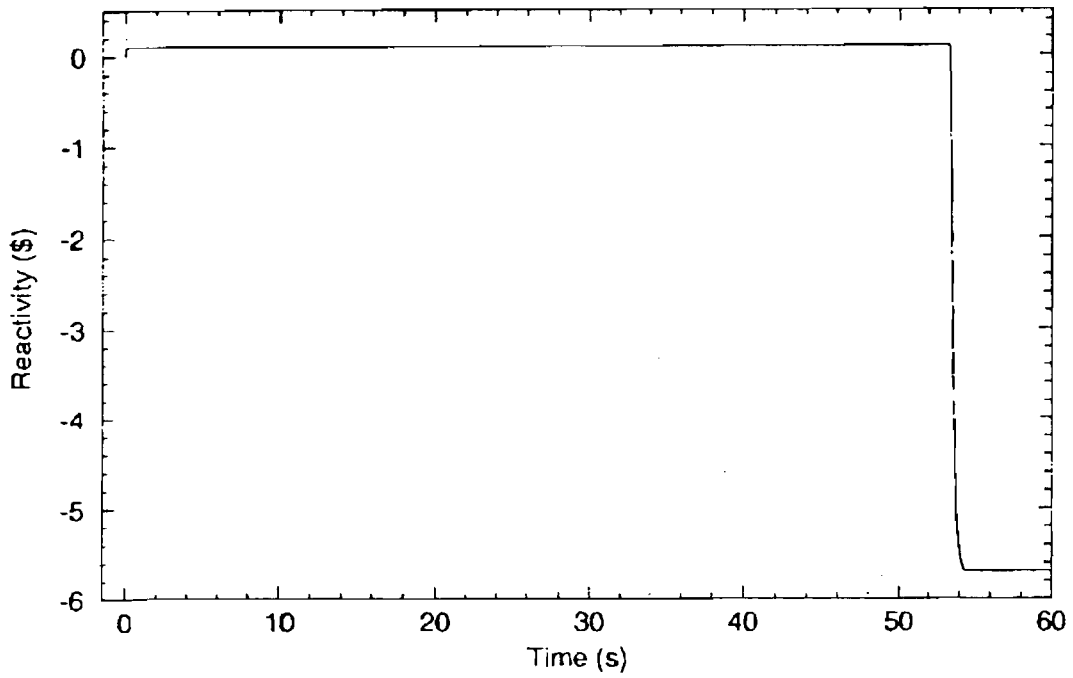

Figure 15.9-5. Core reactivity transient for a 0.10$ step reactivity insertion accident during depressurized (open head) operation.

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15.10 Fuel Failures

The ATR fuel element cladding is the first barrier to radioisotope release. If the fuel element cladding does fail, radioisotopes will be released into, but still contained by, the PCS. Radioisotopes released into the PCS will be detected approximately 6 sec after release by gamma detectors on the outlet piping, a part of the PPS fission break subsystem (FBS). The fission break subsystem will terminate reactor operation by initiating a release of the safety rods. The only fuel failures of concern to the safety analysis are those that fail the fuel cladding and release radioisotopes.

A number of fuel failures without the occurrence of reactor or PCS accidents have been postulated and analyzed:

A. Partial fuel channel blockage from small debris in the PCS.

B. Unexpected high peaking in the fuel due to an analysis or loading error.

C. Small defects in fuel elements or minor handling damage to fuel.

D. Fuel channel blockage from a sight glass or other inventoried items left in the reactor system after a shutdown.

E. Fuel channel blockage due to failure of small structural elements in the ATR vessel above the core or due to failure of the inlet screen.

F. Structural fuel element defects affecting a few fuel elements in the core due to design or fabrication errors or handling damage.

G. Fuel plate buckling due to an excessive buildup of oxide on fuel plate surfaces.

H. Gas leakage into the fuel element coolant channels due to an insulating jacket failure during normal pressurized operation.

I. Fuel channel blockage or fuel damage due to the failure of large structural elements above or within the core.

J. Structural fuel element defects affecting many fuel elements in the core due to design or fabrication errors or severe handling damage.

Based on evaluations contained in the DBR (EG&G Idaho 1994), events A, B, and C are considered Condition 2; events D, E, F, G, and H are considered Condition 3; and events I and J are considered Condition 4. The events and analyses are discussed in the following sections.

15.10.1 Partial Fuel Channel Blockage from Small Debris in the Primary Coolant System

15.10.1.1 Identification of Causes and Accident Description. Small debris in the PCS may partially block the fuel element coolant channel. Fuel channel blockage could cause localized fuel failure without the occurrence of reactor or PCS accidents. This event is considered a Condition 2 event.

15.10.1.2 Analysis of Effects and Consequences. Analysis and testing have shown that because of the ATR fuel element vents, ATR fuel channels need to be blocked more than 80% before potential fuel damage is likely (Neder 1971). Therefore, small amounts of blockage by anticipated debris in the PCS will not cause fuel element failure due to hydraulic forces. Analyses of heat transfer indicate adequate cooling is maintained (Buescher 1993). Routine inspections of irradiated fuel elements would help detect any possible structural damage to the fuel elements.

15.10.2 Fuel Channel Blockage from a Sight Glass or Other Inventoried Items Left in the Reactor System After a Shutdown or Due to Failure of Small Structural Elements in the ATR Vessel above the Core or Due to Failure of the Inlet Screen

15.10.2.1 Identification of Causes and Accident Description. Sight glass or other inventoried items left in the reactor system after shutdown, the failure of a small structural element in the ATR vessel above the core, or the failure of the inlet screen could result in blockages in the fuel coolant channel. (Screen failure could allow debris to reach the fuel element and could generate debris by failure.) Blockage of the fuel element coolant channel could cause fuel failure without the occurrence of reactor or PCS accidents. Blockage by inventoried items and by the failure of small structural elements in the vessel is considered a Condition 3 event.

15.10.2.2 Analysis of Effects and Consequences. Fuel channel blockage greater than 80% will cause some fuel cladding failure, as has been experienced at the Engineering Test Reactor (ETR) (Keller 1962; Smith 1967). The ETR used aluminum plate-type fuel elements that were similar in design heat fluxes, power densities, materials, and fabrication processes to ATR fuel elements. The significant differences between the ETR and the ATR for flow blockages are the fuel element side plate vents in the ATR design and the fuel element cross section (ETR square vs ATR cylindrical segment). Due to hydraulic communication through the fuel element vents, any damage in the ATR is expected to be less extensive than for similar-sized blockages in the ETR.

Two ETR fuel element blockage incidents were studied to scope the damage and possible PPS protective action for credible fuel channel blockage accidents (Atkinson 1976b). These two incidents were fuel blockage due to a sight glass which was left inside the ETR vessel after a shutdown on December 12, 1961 (Keller 1962), and fuel channel blockage due to metallic colored tape that had been dropped onto the core on February 20, 1967 (Smith 1967). The worst damage from these two incidents was from the sight glass incident which resulted in partial to complete melting in 18 fuel plates in 6 different fuel elements. An estimated 12.4 gm of U-235 with 134 gm of aluminum was dispersed in the PCS, and about 6.4 curies of radioactivity was released to the atmosphere. These consequences should bound the worst expected fuel melting for a relatively probable fuel channel blockage in the ATR.

The study of these blockage incidents found that the ETR fast fission break (FFB) subsystem (a subsystem comparable to the ATR FBS at that time called the N-16 system in ETR and later changed to the ETR FFB subsystem) was the first of any subsystems to detect the flow blockages, and the first to initiate any protective action (by automatically lowering the reactor power). The study also indicated that during the sight glass incident, the ETR would have been shut down significantly earlier with possibly less fuel damage and radiation release if there were a ETR FFB setpoint for an intermediate power level. Fuel damage during the sight glass incident occurred at an ETR power of 90 MW (0.51 N_F), and a post accident data study indicates that the ETR operated at this power for over 40 minutes with flow blockage and fuel melting occurring (Hanson 1971). The ETR FFB trip was set for at least 20% above normal-full power PCS radiation levels (ETR full power was approximately 175 MW). Thus, when the ETR FFB did eventually trip, the trip level corresponded to 233% of the operating level at 90 MW.

Studies for fuel failure accidents, particularly from ETR experience with fuel failure accidents, have concluded that an ATR FBS can adequately respond to detect fuel cladding failures and trip the safety rods to terminate power operation. The study of ETR experience also showed that fuel failures, which would be most probable during a reactor startup, may occur at an intermediate power level (at about 0.5 N_F) and that an intermediate power level trip point is desirable for a FBS for normal startups. Although the study could not determine a firm quantitative basis for an ATR FBS trip point, it was concluded that a nominal trip point (or a limiting control setting) of 150% or less of the normal operating radiation level readings of the ATR FBS detectors for intermediate trip level should be applied in the range of 0.5 to 0.75 of full power levels. However, the subsystem design conservatively includes two ranges, one with a trip level at 150% of the normal coolant radiation level at $N_F/3$ to detect and limit any very large blockage early in the startup sequence and a second with a trip level at 150% of the normal coolant radiation level at N_F. The coolant transport time from the core to the fast fission break detector locations is assumed to be approximately 6 sec.

Since the non-gaseous fission products are retained in the PCS for these types of fuel-failure accidents (except for some small PCS leakage), and since the radiation releases during the ETR incidents were not excessive (with use of proper operational controls for system cleanup and effluent release), then the consequences of this type of accident are acceptable for a Condition 3 fault.

15.10.3 Fuel Channel Blockage or Fuel Damage Due to the Failure of Large Structural Elements Above or Within the Core

15.10.3.1 Identification of Causes and Accident Description. Failure of large structural elements in the reactor vessel above or in the core could block the fuel element coolant channel or severely damage fuel elements. Both events could cause fuel failure without the occurrence of reactor or PCS accidents. Both events are considered Condition 4 events and envelop those events that result in large blockages.

15.10.3.2 Analysis of Effects and Consequences. Gross fuel element damage may result in the release of a significant amount of fission products into the PCS. However, without a PCS break, most fission products will be retained in the PCS, except for gaseous fission products and those escaping due to PCS leakage. Because of the serpentine geometry of the ATR core, it is improbable that a substantial portion of the core would be involved for a major component failure. Based on engineering judgment, it is assumed that no more than two complete fuel elements would be affected. This judgment is based on the serpentine core design, which makes it difficult to block multiple elements and historical experience with material found in the vessel. The dose at the nearest site boundary and the LPZ for extensive damage of two fuel elements would be less than 5% of the dose for the full core melt resulting from a large break LOCA. The radiological results of the large break LOCA are discussed in Section 15.12.10. The dose at the LPZ from two fuel elements damaged without compromise to the PCS boundary would be less than 9.3 rem thyroid and 0.7 rem EDE (1/20 of the large break LOCA results of Section 15.12.10).

15.10.4 Gas Leakage into the Fuel Element Coolant Channels Due to an Insulating Jacket Failure During Normal Pressurized Operation

15.10.4.1 Identification of Causes and Accident Description. Helium gas is used in the annulus of the IPT to provide insulation. In the LIPT and the AHTL, the gas pressure exceeds the PCS pressure. Failure of the insulating jacket normally would result in water entering the annulus of the standard tubes; for the failure in the LIPT or AHTL, helium would enter the PCS. Helium gas leakage into the fuel element coolant channel could cause fuel failure due to degraded heat transfer. Rupture of the insulating jacket of the IPT is assumed to be a Condition 3 event since the jacket is not inspected for flaws (ultrasonic exams are not performed during service). The gas delivery system is regulated and has a pressure relief system; consequently, a failure of the insulating jacket with higher than normal gas pressure is not considered credible (Atkinson 1994). The reactivity effect of gas released into the core is considered in Section 15.4.

15.10.4.2 Analysis of Effects and Consequences. An evaluation of the gas flow from the AHTL (represents both AHTL and LIPT) at normal gas pressures indicates a gas flow rate of 0.136 ft^3/s (8.16 ft^3/min) at core outlet pressure (Ambrosek 1993). However, the evaluation of the code used in this analysis against measured data indicates that the code underpredicts the gas flow rate. Based on the comparisons, the results obtained with the code were scaled upward accordingly. The assumed flow rate for insulating jacket failure based on this analysis is 12.1 ft^3/min.

The study of gas leak flow distribution tests (Bulmahn 1975a) from a height of approximately 3 ft (0.91 m) above the fuel channels (the location of the first IPT weld above the ATR core, considered the closest probable leak location above the core) showed that the gas leak flow distributed fairly evenly into all channels in the test section. Although the channels closest to the IPT would receive somewhat more gas flow, the hot channels in ATR fuel are those closest to the reflector, away from the IPT. Therefore, for a thermal analysis with gas leak flow, uniform distribution was assumed. Uniform gas flow distribution is the type of flow which may be expected with jetting of gas leak flow from an IPT gas jacket flaw.

The gas leak flow thermal analysis, discussed in Varacalle (1976), used a gas leak flow distribution based on the gas leak flow distribution tests reported in Bulmahn (1975a) and assumed uniform distribution. The analysis used two-phase two-component hydraulics and heat transfer equations and analysis methods, and included slug flow to annular flow boundary data for downflow from Golan and Stenning (1969). Channel 19 which was the hot channel as well as the largest channel had the most gas flow (4.23 ft^3/min = 0.0020 m^3/sec). The thermal analysis for 75.5 ft^3/min of total gas flow [this extreme gas flow was a conservative envelope for assumed failure of pressure regulation (Bulmahn 1975b)], with 4.23 ft^3/min down channel 19 for the hot fuel element in a 70 MW lobe did not predict either FI or CHF for this large gas flow. The modest gas flow for the Condition 3 event (12.1 ft^3/min) is about 16 percent of the flow considered by Varacalle (1976) and therefore will not result in an approach to either FI or CHF. The insulating jacket failure meets Condition 2 criteria and is acceptable as a Condition 3 event.

15.10.5 Structural Fuel Element Defects Affecting a Few Fuel Elements in the Core Due to Design or Fabrication Errors or Handling Damage

15.10.5.1 Identification of Causes and Accident Description. Localized fuel element structural defects due to design, fabrication errors, or handling damage could cause fuel failure without the occurrence of reactor or PCS accidents. Fuel failures due to localized defects or damage are considered Condition 3 events.

15.10.5.2 Analysis of Effects and Consequences. Local fuel element defects or damage can result in some cladding failure, but the fission products released would be small and significantly less than for fuel-channel-blockage accidents. The effect may not be significant relative to the setting for the ATR FBS (Section 15.10.2.2). If the fission product concentration does not reach the setting for the FBS, routine monitoring and controls ensure releases from the PCS are limited.

15.10.6 Structural Fuel Element Defects Affecting Many Fuel Elements in the Core Due to Design or Fabrication Errors or Severe Handling Damage

15.10.6.1 Identification of Causes and Accident Description. Gross fuel element structural defects due to design, fabrication errors, or severe handling damage could cause fuel failures without the occurrence of reactor or PCS accidents. Gross fuel element defects or damage are considered Condition 4 events.

15.10.6.2 Analysis of Effects and Consequences. The effects of gross fuel failure due to structural or design defects or handling damage are on the order of the large channel blockage events in Section 15.10.3.2 above. The consequences are assumed to be enveloped by the dose estimated for the large blockage (two fuel elements or equivalent are fully involved).

15.10.7 Fuel Plate Buckling Due to an Excessive Buildup of Oxide on Fuel Plate Surface

15.10.7.1 Identification of Causes and Accident Description. Fuel plate buckling due to an excessive buildup of oxide on fuel plate surfaces could cause fuel failure without the occurrence of reactor or PCS accidents. Excessive oxide buildup is considered a Condition 3 event because of the meticulous chemistry control of the primary coolant and routine oxide surveillance measurements at the end of each cycle.

15.10.7.2 Analysis of Effects and Consequences. Fuel plate buckling due to high thermal stresses when an excessive oxide film builds up on the fuel plate surface will not necessarily lead to cladding failure. Tests reported in Griebenow (1971) have demonstrated that severe fuel channel distortion will need to occur before adjacent plates contact and plate collapse and failure will result. Therefore, the amount of cladding failure which might occur during this accident is small. A release of the safety rods by action of the FBS would terminate reactor operation before significantly more fuel failures occurred. The amount of melting involved is estimated to be similar to the blockage considered in Section 15.10.2. The failure may result in fission product concentrations that approach the FBS setting. If not, routine monitoring and controls provide assurance that releases from the PCS are limited.

15.10.8 Unexpected High Peaking in the Fuel Due to an Analysis or Loading Error

15.10.8.1 Identification of Causes and Accident Description. An analysis error or loading error may cause unexpected high power peaking in the fuel. Severe, high power peaking could cause localized fuel element failure without the occurrence of reactor or PCS accidents. Unexpected high power peaking in the fuel in considered a Condition 2 event.

15.10.8.2 Analysis of Effects and Consequences. Higher peak powers in the fuel than expected from analysis due to analytic or loading errors can result in peak powers that are above normal. The worst-case error involving increased peaking in a 7F element results in a 15% increase in power peaking due to replacing a depleted element with a new 7F element (Atkinson 1976a). The increase could be expected to be greater than 15% if the new element were non-borated. Peak power in an element is increased by not more than 9% by boron removal (Kim and McClure 1993). Therefore, a loading error involving replacement of a depleted element with a new 7NB element is estimated to cause not more than a 26% increase in peaking (product of a 9% and 15% increase). The increased peaking would correspond to a lobe power of 76 MW. As shown in Polkinghorne (1994), Condition 2 margins to CHF and FI are maintained at lobe powers of 88.5 MW and 96.0 MW for two-pump and three-pump operation, respectively. Therefore, no fuel damage will occur due to fuel element analysis or loading errors.

15.10.9 Small Defects in Fuel Element or Minor Handling Damage to Fuel

15.10.9.1 Identification of Causes and Accident Description. Small defects in the fuel element or minor handling damage to the fuel could cause fuel failure without the occurrence of reactor or PCS accidents. Small defects in and minor damage to the fuel element are considered Condition 2 events.

15.10.9.2 Analysis of Effects and Consequences. ATR experience for operation with fuel element cladding defects has shown that small localized clad failures will not result in excessive radiation levels in the ATR PCS, and that operation of the bypass demineralizer system can maintain low radiation levels in the primary coolant for minor fission product leakage. These types of failures only raise the PCS fission product activity levels by about 0.2 μCi/mL; and the activity in the PCS remains well below the 20 μCi/mL limit for normal PCS activity (Dawson 1970).

15.10.10 References

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