15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

The results of event analyses for the current cycles are in Appendix 15C for SSES Unit 1 and Appendix 15D for SSES Unit 2. Appendix 15E contains information and analytical results that are for non-limiting events for the initial cycles for Units 1 and 2. Note that since the data in Appendix 15E is for the initial cycles for Units 1 and 2, the values for key parameters/variables do not represent the actual values if these events were to occur for the current cycles for Units 1 and 2. However the data and figures in Appendix 15E do show qualitative behavior of the non-limiting events.

15.4.1 ROD WITHDRAWAL ERROR - LOW POWER

This event is non-limiting, and therefore, it is not explicitly analyzed each cycle.

15.4.1.1 Control Rod Removal Error During Refueling

15.4.1.1.1 Identification of Causes and Frequency Classification

The event considered here is inadvertent criticality due to the complete withdrawal or removal of the second most reactive rod (assuming most reactive rod is fully withdrawn) during refueling. The probability of the initial causes alone is considered low enough to warrant its being categorized as an infrequent incident, since refueling system interlocks and administrative controls will prevent an inadvertent Rod Withdrawal Error (RWE) while in the REFUEL mode.

15.4.1.1.2 Sequence of Events and Systems Operation

15.4.1.1.2.1 Initial Control Rod Removal

During refueling operations both refueling system interlocks and administrative controls which ensure control rods are latched provide assurance that inadvertent criticality does not occur because a control rod has been removed or is withdrawn in coincidence with another control rod.

15.4.1.1.2.2 Fuel Movement With Control Rod Removed

Fuel movement and other core alterations with control rods removed will be controlled by the Technical Specifications. The Technical Specification requirements, the associated refueling interlocks, and administrative controls which ensure control rods are latched sufficiently minimize the possibility of loading fuel into a cell containing no control rod, moving the refueling platform over the core, and withdrawing additional control rods when there is uncontrolled fuel in the core.

15.4.1.1.2.3 Control Rod Removal Without Fuel Removal

The design of the control rod, incorporating the velocity limiter, does not physically permit upward removal of the control rod without simultaneous or prior removal of the four adjacent fuel bundles. This precludes any hazardous condition.

15.4.1.1.2.4 Identification of Operator Actions

As discussed above, refueling system interlocks and administrative controls are required to prevent this event. The administrative controls require that the operator verifies that all fully inserted control rods are latched.

15.4.1.1.2.5 Effect of Single Failure and Operator Errors

If any one of the operations involved in initial failure or error is followed by any other Single Equipment Failure (SEF) or Single Operator Error (SOE), the necessary safety actions are taken (e.g., rod block or scram) automatically prior to limit violation.

15.4.1.1.3 Core and System Performances

Since the probability of inadvertent criticality during refueling is prevented by a combination of system design and administrative controls, the core and system performances were not analyzed. SDM calculations and tests ensure that the core remains subcritical with the highest worth control rod fully withdrawn. (See Subsection 4.3.2 for a description of the methods and results of the shutdown margin analysis.) Additional reactivity insertion due to control rod withdrawal is prevented by interlocks (See Subsection 7.6.1a.1) and administrative controls. As a result, no radioactive material is ever released from the fuel, making it unnecessary to assess any radiological consequences.

No mathematical models are involved in this event. The need for input parameters or initial conditions is not required as there are no results to report. Consideration of uncertainties is not appropriate.

15.4.1.1.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since it is a highly localized event and does not result in any change in the core pressure or temperature.

15.4.1.1.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event since no radioactive material is released from the fuel.

15.4.1.2 Continuous Rod Withdrawal During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

Note: Unit 2 RSCS removed.

The event is defined as: while operating below the low power setpoint and coincident with a failure or bypass of the RWM the operator makes a procedural error and withdraws an out of sequence control rod of maximum worth. The probability of initial causes or errors of this event alone is considered low enough to warrant its being categorized as an infrequent incident. The probability of further development of this event is extremely low because it is contingent upon the failure of the RWM system, concurrent with a high worth rod, out-of-sequence rod selection contrary to procedures, plus operator ignorance of any alarm annunciations prior to safety system actuation. Whenever the RWM is inoperable or bypassed, there is a Technical Specification requirement that a second operator verify that the correct control rod withdrawal sequence is followed.

15.4.1.2.2 Sequence of Events and Systems Operation

15.4.1.2.2.1 Sequence of Events

Control rod withdrawal errors are not considered credible in the startup and low power ranges. The RWM plus procedural requirements prevent the operator from selecting and withdrawing an out-of-sequence control rod.

The purpose of the RWM is to control rod patterns during startup, such that only specified rod sequences and relative positions are allowed over the operating range from all control rods inserted to approximately 10% of rated core power. The sequences effectively limit the potential amount and rate of reactivity increase during a Control Rod Drop Accident. The RWM is designed to act as a backup to operator control of the rod sequences. Therefore if the RWM is inoperable or bypassed the Technical Specifications require that a second operator verify that any subsequent rod selection and withdrawal is in accordance with the specified rod sequence.

In the unlikely event that the RWM fails to prevent an out-of-sequence control rod from being withdrawn in the reactor startup range, fuel failure will not occur as shown by generic analyses performed by General Electric in Reference 15.4-10. Protection is provided by the IRM upscale scram function and/or APRM scram which are both single failure proof designed systems.

15.4.1.2.2.2 Identification of Operator Actions

No operator actions are required to preclude this event since the plant design as discussed above prevents its occurrence.

15.4.1.2.2.3 Effects of Single Failure and Operator Errors

If any one of the operations involved the initial failure or error followed by another SEF or SOE, the necessary safety actions are taken (e.g., rod blocks) prior to any limit violation.

15.4.1.2.3 Core and System Performance

The performance of the RWM and procedural requirements prevent erroneous selection and withdrawal of an out-of-sequence control rod. The core and system performance is not affected by such an operator error.

No mathematical models are involved in this event. The need for input parameters or initial conditions is not required as there are no results to report. Consideration of uncertainties is not appropriate.

15.4.1.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since this is a localized event with very little change in the gross core characteristics of temperature and pressure.

15.4.1.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.2 ROD WITHDRAWAL ERROR - AT POWER

This event has been identified as a limiting event, and therefore, it has been analyzed for the current cycles for Units 1 and 2.

15.4.2.1 Identification of Causes and Frequency Classifications

15.4.2.1.1 Identification of Causes

While operating in the power range in a normal mode of operation the reactor operator makes a procedural error and withdraws the maximum worth control rod until the Rod Block Monitor (RBM) System inhibits further withdrawal or the control rod is fully withdrawn.

15.4.2.1.2 Frequency Classification

The probability of this event is considered low enough to warrant its being categorized as an infrequent incident. However, because of the lack of a sufficient frequency data base, this event is classified and analyzed as an incident of moderate frequency. Starting with U2C14 and U1C16 ARTS has been implemented and credit for the RBM is taken.

15.4.2.2 Sequence of Events and Systems Operation

15.4.2.2.1 Sequence of Events

The sequence of events for the Rod Withdrawal Error (RWE) transient, as analyzed with conservative RWE assumptions, is presented in Tables 15C.4.2-1 and 15D.4.2-1. It is assumed that the operator takes no mitigating actions during the course of this event. Following the event, the operator will re-insert the control rod to reduce core power to rated conditions.

15.4.2.2.2 System Operations

The focal point of this event is localized to a small portion of the core. A discussion of the event follows below.

While operating in the power range in a normal mode of operation (except as noted in Subsection 15.4.2.3.2), the reactor operator makes a procedural error and withdraws the maximum worth control rod until either the RBM system inhibits further withdrawal or the rod is completely withdrawn.

Under most normal operating conditions no operator action is required since the transient which would occur would be very mild.

If the rod withdrawal error is severe enough, the rod block monitor (RBM) system will sound alarms, and block further withdrawal of the control rod.

The ARTS RBM is designed to block control rod withdrawal if localized thermal power in the vicinity of the control rod exceeds a predetermined trip set point. There are three reactor power ranges (low, intermediate and high) that have set points established that define their range. Each thermal power range has a localized thermal power trip set point to block control rod movement. The trip set points for each thermal power range are set by analysis to assure that the SLMCPR (point of boiling transition) and LHGR limit (centerline melt and 1% plastic strain limit) are not reached before the control rod movement is blocked. The power ranges and trip set points are specified in the COLR.

15.4.2.2.3 Effect of Single Failure and Operator Errors

The effect of operator errors has been discussed above. Termination of this event is assured by the RBM system or complete withdrawal of the control rod.

15.4.2.3 Core and System Performance

15.4.2.3.1 Mathematical Model

For this transient the reactivity insertion rate is very slow; therefore, it is adequate to assume that the core has sufficient time to equilibrate (i.e., that both the neutron flux and heat flux are in phase). Making use of the above assumption, this transient is calculated using a steady-state three-dimensional coupled nuclear-thermal-hydraulics computer program. All spatial effects are included in the calculation.

Commencing with Unit 2 Cycle 13 and Unit 1 Cycle 15, the methods for modeling and analyzing this event are described in References 15.4-14, 15.4-17, and 15.4-18.

The primary output from this code, in addition to the basic nuclear parameters, is: the variation of the linear heat generation rate (LHGR); the variation of the minimum critical power ratio (MCPR); the total reactor power; and the variation of the in-core instrumentation output during the transient.

The analytical methods and assumptions which are used in evaluating the consequences of this accident are considered to provide a realistic, yet conservative assessment of the consequences.

15.4.2.3.2 Input Parameters and Initial Conditions

The number of possible RWE transients is extremely large due to the number of control rods and the wide range of exposures and power levels. In order to encompass all of the possible RWEs which could conceivably occur, a limiting analysis is defined such that a conservative assessment of the consequences is provided.

The conservative assumptions are:

(1) The assumed rod withdrawal error is a continuous withdrawal of the maximum worth rod.

(2) The core is assumed to be operating at rated conditions. (See Tables 15C.0-2 and 15D.0-2 for Units 1 and 2).

(3) The reactor is presumed to be devoid of all xenon. This insures that the amount of excess reactivity which must be controlled by the movable control rods is maximized.

(4) Furthermore, it is assumed that the operator has fully inserted the maximum worth rod prior to its removal and selected the remaining control rod pattern in such a way as to approach thermal limits, (MCPR limit and LHGR limit), in the fuel bundles in the vicinity of the rod to be withdrawn. It should be emphasized that this control rod configuration would be highly abnormal and could only be achieved by deliberate operator action or by numerous operator errors.

(5) It is assumed that the operator makes a procedural error and withdraws the maximum worth control rod until either the RBM system inhibits further withdrawal or the rod is completely withdrawn.

(6) In addition to the above conditions, it is possible that as the reactor power increases there could be a loss of pressure control which would result in a higher power level if the steam bypass is inoperable versus having it operable. Therefore, this event is analyzed for the conditions of the steam bypass operable and inoperable. Loss of pressure control is assumed to occur for steam flows greater than steam passing capability of the Turbine Control Valves.

The conservative assumptions indicated above provide a high degree of assurance that the transient as analyzed bounds all RWEs which could possibly occur.

With the implementation of ARTS the functioning of the RBM is now credited in the RWE analysis.

<u>15.4.2.3.2.1 RBM System Operation</u>

With ARTS implementation the RBM has three power dependent trip levels (rod withdrawal permissives removed). The trip levels are automatically varied with reactor power to protect against fuel damage. The initial RBM signal is normalized to a fixed (constant) reference signal. The trip levels are set at a fixed level above the reference and will vary as step functions of core power. This will allow longer rod withdrawals at low powers where thermal margins are higher and allow only short rod withdrawals at high power.

The ARTS based RBM uses an improved LPRM assignment. As opposed to the flow biased RBM system, this improved LPRM assignment provides readily predictable behavior and will limit thermal margin reduction during rod withdrawals without restricting rod withdrawals on the basis of core power level.

For each power range, (low, intermediate, and high) Allowable Values and Nominal trip set points are established by analysis to assure that the SLMCPR and LHGR limit are not reached before the control rod movement is blocked. The analysis for the rod withdrawal error also establishes the MCPR limits when the RBM is inoperable. These set points and limits are included in the COLR.

<u>15.4.2.3.3 Results</u>

The ΔCPRs determined for this event for steam bypass operable are given in Tables 15C.0-1 and 15D.0-1 for Units 1 and 2, respectively. The increase in the LHGR during the event was also determined for the condition of the steam bypass operable and inoperable. In general, for this event, the increase in the LHGR is less than the PAPT (Protection Against Power Transient) Limit for both Units 1 and 2. If, for a particular condition (e.g. bypass inoperable), the analytical results were to indicate that the PAPT Limit would be exceeded, a reduction in the normal steady state

LHGR Limit would be established for that condition and would be recorded in the COLR. Maintaining the LHGR less than the PAPT Limit assures that the plastic strain limit of 1% for the cladding is not exceeded.

15.4.2.3.4 Considerations of Uncertainties

The conservative assumptions which assure that this event has been conservatively analyzed have been previously discussed in Subsection 15.4.2.3.2.

15.4.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since this is a localized event with very little change in the gross core characteristics. Typically, an increase in total core power is less than 7% and the changes in pressure are negligible. If there is a loss of pressure control and the steam bypass system is inoperable the increase in core power is less than 14%. The increase in system pressure is small, (i.e., considerably less than that of the limiting transient for system overpressure evaluations) for this event, and therefore represents no threat to the Reactor Coolant Pressure Boundary.

15.4.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.3 CONTROL ROD MALOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

This event is covered with evaluation cited in Subsections 15.4.1 and 15.4.2.

15.4.4 ABNORMAL STARTUP OF IDLE RECIRCULATION PUMP

This event is non-limiting, and therefore, it is not explicitly analyzed each cycle. The analysis described below was performed for the initial cycles for Units 1 and 2.

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

This action results directly from the operator's manual action to initiate pump operation. It assumes that the remaining loop is already operating.

15.4.4.1.1.1 Normal Restart of Recirculation Pump at Power

This transient is categorized as an incident of moderate frequency.

15.4.4.1.1.2 Abnormal Startup of Idle Recirculation Pump

This transient is categorized as an incident of moderate frequency.

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

Table 15E.4.4-1 lists the sequence of events for Figure 15E.4.4-1.

15.4.4.2.1.1 Operator Actions

The normal sequence of operator actions expected in starting the idle loop is as follows. The operator should:

(1) Adjust rod pattern as necessary for new power level following idle loop start.

(2) Determine that the idle recirculation pump suction and discharge bypass valves are open, the discharge valve is closed, and the scoop tube in the idle loop is in the starting position, if not, place them in this configuration.

(3) Readjust flow of the running loop downward to less than half of rated flow.

(4) Determine that the temperature difference between the two loops is no more than 50°F apart.

(5) Start the idle loop pump and allow pump speed and drive flow to reach a settled state.

(6) Open the discharge valve and slowly adjust pump speed.

(7) Readjust power, as necessary, to satisfy plant requirements per standard procedure.

Note: The time to do above work is approximately 1/2 hour.

15.4.4.2.2 Systems Operation

This event assumes and takes credit for normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. In particular, credit is taken for high flux scram to terminate the transient. No ESF action occurs as a result of the transient.

15.4.4.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a quick rise in reactor power level. Corrective action first occurs in the high flux trip and being part of the reactor protection system, it is designed to single failure criteria. Therefore, shutdown is assured. Operator errors are not of concern here in view of the fact that automatic shutdown events follow so quickly after the postulated failure.

15.4.4.3 Core and System Performance

15.4.4.3.1 Mathematical Model

The nonlinear dynamic model described in Reference 15.4-15 is used to simulate this event.

15.4.4.3.2 Input Parameters and Initial Conditions

This analysis has been performed unless otherwise noted with plant conditions tabulated in Table 15E.0-2.
One recirculation loop is idle and filled with cold water (100°F). Normal procedure when starting an idle loop with one pump already running requires heating the idle recirculation loop to within 50°F of core inlet temperature prior to loop startup.

The active recirculation loop is operating with about 80% of normal rated diffuser flow going across the active jet pumps.

The core is receiving 38% of its normal rated flow. The remainder of the coolant flows in the reverse direction through the inactive jet pumps.

Reactor power is 55% of rated (prior to power uprate). Normal procedures require startup of an idle loop at a lower power.

The idle recirculation pump suction valve is open, but the pump discharge valve is closed.

The idle pump fluid coupler is at a setting which approximates 50% generator speed demand.

15.4.4.3.3 Results

The transient response to the incorrect startup of a cold, idle recirculation loop is shown in Figure 15E.4.4-1. Shortly after the pump begins to move, a surge in flow from the standard jet pump diffusers causes the core inlet flow to rise sharply.

A short-duration neutron flux peak reaches the flow referenced APRM flux set point at 10 seconds and reactor scram is initiated. The neutron flux peak is given in Table 15E.0-1. Surface heat flux follows the slower response of the fuel and its peak value is also given in Table 15E.0-1. Nuclear system pressures do not increase significantly above initial. The water level does not reach the high set point.

15.4.4.3.4 Consideration of Uncertainties

This particular transient is analyzed for an initial power level that is much higher than that expected for the actual event. The much slower thermal response of the fuel mitigates the effects of the rather sharp neutron flux spike and even in this high range of power, no threat to thermal limits is possible.

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event since no significant pressure increases are incurred during this transient. See Figure 15E.4.4-1.

15.4.4.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.5 RECIRCULATION FLOW CONTROL FAILURE WITH INCREASING FLOW

This event is a limiting event, and therefore, it has been analyzed for the current cycles for Units 1 and 2.

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

An unlikely Integrated Control System failure that results in the occurrence of a common failure of both speed control loops may in turn result in a speed increase of both recirculation pumps. The ramp rate of increase in the speed of the recirculation pumps is assumed to be slow.

15.4.5.1.2 Frequency Classification

This transient disturbance is classified as an incident of moderate frequency.

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

Tables 15C.4.5-1 and 15D.4.5-1 list the sequence of events for the recirculation flow controller failure for Units 1 and 2.

15.4.5.2.1.1 Identification of Operator Actions

Initial action by the operator will include:

(1) Transfers flow control to manual and reduces flow to minimum.

(2) Identify cause of failure.

Reactor pressure will be controlled as required, depending on whether a restart or cooldown is planned. In general, the corrective action would be to hold reactor pressure and condenser vacuum for restart after the malfunctioning flow controller has been repaired. The operator should perform those activities listed in Table 15E.1.1-1.

15.4.5.2.2 Systems Operation

The analysis of this transient assumes and takes credit for normal functioning of plant instrumentation and controls, and the reactor protection system, except for the APRM flow biased scram. The MG set electrical and mechanical stop design feature is also not credited. Operation of engineered safeguards is not expected.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a gradual rise in reactor power level. Corrective action occurs from either the high flux trip or the high pressure trip and, being part of the reactor protection system, these trips are designed to meet the single failure criteria. Therefore, shutdown is assured. Operator errors are not of concern here in view of the fact that automatic shutdown events follow soon after the postulated failure.

15.4.5.3 Core and System Performance

15.4.5.3.1 Mathematical Model

Since the transient is gradual, it has been demonstrated that a quasi-steady state analysis will yield acceptable results. A quasi-steady state analysis was performed for Unit 1 and Unit 2.

15.4.5.3.2 Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Tables 15C.0-2 and 15D.0-2 for Units 1 and 2.

For this event a number of different Power/Flow conditions are analyzed. Commencing with Unit 2 Cycle 13, and Unit 1 Cycle 15, the methods for modeling and analyzing this event are described in References 15.4-14, 15.4-17, 15.4-18 and 15.4-19.

The Unit 1 and Unit 2 quasi – steady state analyses were performed for steam bypass operable and for it inoperable. The initial MCPR and LHGR were determined, using the approved three dimensional nodal simulation methods. The maximum core flow is determined based on the change in recirculation pump speed at different power levels.

For steam bypass operable, a flow excursion is assumed to occur that is equal to or greater than a normal rate of increase in reactor coolant flow and that the flow increases to that corresponding to the flow at maximum pump runout. It is also assumed that this flow excursion passes through the 100% power/ 100% flow statepoint. The fuel assembly power distributions are determined at a number of flow/power statepoints along the flow excursion path, including the condition of maximum core flow, using the three dimensional simulator. Using these power distributions, the MCPR for each statepoint is determined. The change in the MCPR along the flow path is used to establish the flow dependent CPR operating limits for Units 1 and 2. The fuel assembly power distributions are also used to determine whether or not the LHGR needs to be limited at lower powers and flows to prevent the LHGR limits from being exceeded at some point during the excursion.

For steam by-pass inoperable, steam flow will exceed the capacity of the turbine control valves and the power and pressure will rise until the reactor protection system trips on either high power or high steam dome pressure. The three dimensional simulator is run with the setpoints power and pressure set at their reactor protection system trip setpoints. The resulting MCPR and LHGRs are used as before to determine the ΔCPR and whether or not the LHGR limit is exceeded.

15.4.5.3.3 Results

The nuclear system pressure increase is limited by the high pressure analytical trip setpoint and operation of the safety/relief valves which are set to open at the nominal relief valve setpoints.

The peak neutron flux rise approaches the high neutron flux analytical trip setpoint. Since the transient is relatively slow, the change in heat flux is essentially the same as the change in neutron flux.

The ΔCPRs for this event are given in Tables 15C.0-1 and 15D.0-1 for Units 1 and 2. The change in the LHGR was determined for this event and it was less than the Protection Against Power Transients (PAPT) Limit.

15.4.5.3.4 Considerations of Uncertainties

The analysis addresses uncertainties by conservatively setting the rate of increase in recirculation pump speed, using the maximum allowable Technical Specification scram insertion time, and using analytical set points for the high neutron flux trip and the high pressure trip. The conservative assumptions used in the quasi-steady state methodology account for any uncertainties.

15.4.5.4 Barrier Performance

This transient results in an increase in reactor vessel pressure slightly above the high pressure analytical trip setpoint and therefore represents no threat to the RCPB.

15.4.5.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event since no radioactive material is released from the fuel.

15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTIONS

Not applicable to BWRs.

15.4.7 MISPLACED BUNDLE ACCIDENT

References 15.4-14, 15.4-17, and 15.4-18 describe the methodology used for cycles starting with U2C13 and U1C15. Bounding analyses have been performed for the ATRIUM-10 fuel for this event using this methodology. Cycle specific analyses were performed for cycles containing ATRIUM-11 fuel commencing with Unit 2 Cycle 21. These analyses determined that for this event, for all expected operational conditions, less than 0.1% of the fuel rods will fail. Since less than 0.1% of the fuel rods in the core will fail, the radiological release will be less than a small fraction of that permitted by 10CFR Part 50.67. Also Design Bases analysis shows that the control rod drop accident analyzed in Subsection 15.4.9 is bounding.

15.4.7.1 Identification of Causes and Frequency Classification

15.4.7.1.1 Identification of Causes

One of the events discussed in this section is the improper loading of a fuel bundle and subsequent operation of the core. Three errors must occur for this event to take place in the initial core loading. First, a bundle must be misloaded into a wrong position in the core. Second, the bundle which was supposed to be loaded where the mislocation occurred would have to be overlooked and also put in an incorrect location. Third, the misplaced bundles would have to be overlooked during the core verification performed following core loading.

Another possible event is to misload the fuel assembly by rotating it either 90° or 180° from its proper orientation. For this event to occur, two operator errors will have to occur. The assembly will have to have been placed in its proper location but oriented incorrectly. The subsequent verification of the core loading will have to have overlooked the incorrect orientation of the fuel assembly handle.

15.4.7.1.2 Frequency of Occurrence

It is assumed the bundle is misplaced to the worst possible location, and the plant is operated with the mislocated bundle.

Similarly the placement of an assembly with the incorrect orientation is assumed to be in a location(s) that yield the largest change in LHGR and ΔCPR.

Neither of these events have occurred at the SSES Units. These events are categorized as infrequent incidents.

15.4.7.2 Sequence of Events and Systems Operation

The postulated sequence of events for the misplaced bundle accident (MBA) and the Rotated Bundle Accident (RBA) are presented in Table 15C.4.7-1 and Table 15D.4.7-1 for Units 1 and 2, respectively.

Fuel loading errors, undetected by in-core instrumentation following fueling operations, may result in undetected reductions in thermal margins during power operations. No detection is assumed, and therefore, no corrective operator action or automatic protection system functioning occurs.

15.4.7.2.1 Effect of Single Failure and Operator Errors

This analysis already represents the worst case (i.e., operation of a misplaced bundle with three Single Equipment Failure (SEF) or Single Operator Error (SOE) or two SEF or SOE for the Rotated Bundle Accident) and there are no further operator errors which can make the event results any worse.

15.4.7.3 Core and System Performance

15.4.7.3.1 Mathematical Model

The methodology used for the bounding analyses applied to U2C13 U1C15 and subsequent cycles is described in References 15.4-14, 15.4-17, and 15.4-18.

A three-dimensional steady-state BWR simulator model is used to calculate the core performance resulting from the misplaced bundle accident. For the analysis of the rotated bundle accident, the change in local peaking in the bundle due to the change in water gap width is modeled using a two dimensional lattice code in conjunction with a three-dimensional steady-state BWR simulator.

15.4.7.3.2 Input Parameters and Initial Conditions

Misloaded Fuel Bundle

By placing a misloaded fresh fuel assembly face adjacent to other fresh assemblies, excessive power peaking occurs in the mislocated and surrounding bundles near the middle and through to the end of cycle. At the beginning of a cycle the worst mislocation will be caused by placing a misloaded once burned assembly, which is at or near peak reactivity, adjacent to as many other once burned assemblies as possible. In addition, the limiting once burned assembly is assumed to be misloaded adjacent to the limiting MCPR assembly at Beginning of Cycle (BOC). The results of both instances are lower CPRs and higher LHGRs for the mislocated and immediately surrounding bundles.

The analysis is performed by examining the cycle at separate exposure steps throughout the cycle to determine the most severe consequences of either a misloaded fresh fuel assembly or a misloaded once burned assembly. The analysis also considers the effect of fuel designs with axially varying enrichments and/or gadolinia loadings.

At each exposure step examined, the core MCPR and the limiting LHGR with the mislocated bundle is compared to the MCPR and the LHGR of the properly loaded core to determine the ΔCPR and whether or not the LHGR limit is violated. This determination is made with nominal operating control rod patterns. No credit is taken for the ability of the core monitoring system to detect a power distribution anomaly due to a mislocated bundle.

Rotated Bundle

Another possibility for misloading a fuel bundle is to load the bundle rotated by 90° or 180° from its correctly loaded orientation.

Due to the location of the channel spacer buttons, a rotated bundle will be tilted towards the adjacent fuel bundles in the same control cell. When the bundle is rotated and inserted into the core, these buttons contact the top guide and push the top of the bundle toward the center of the control cell. The tilt increases the size of the inter-assembly water gaps along the sides of the bundle adjacent to the core top guide and decreases the size of the water gaps along the sides of the fuel bundle adjacent to the control rod. This change in water gap size changes both the reactivity of the fuel bundle and the local pin power distribution within the bundle.

The change in pin power distribution will affect the MCPR and the LHGR for this assembly and its neighbors.

The bounding analyses performed for the Rotated Bundle took into consideration variations of pin power distributions as a function of water gap size and variations in fuel assembly lattice design including enrichment and gadolinia loadings.

To analyze this event the change in pin power distributions are determined assuming a conservatively wide water gap exists for the rotated fuel bundle. For fuel with axially varying enrichments and/or gadolinia loadings, the individual lattices are analyzed. The change in pin power distribution is evaluated at separate exposure steps throughout the cycle. The effect of change in pin power distribution on MCPR and LHGR are evaluated using the 3D core simulation program.

Since the Rotated Bundle is considered an infrequent incident, some fuel damage is permitted provided that the resulting radiological consequences will be a small fraction of the 10FR100 limits. To ensure that these dose limits are met, the analysis methodology established the criteria that at least 99.9% of the fuel rods in the core will avoid boiling transition and the cladding of these fuel rods will not exceed the 1% plastic strain criterion.

15.4.7.3.3 Results

Misloaded Fuel Bundle and Rotated Bundle

The Mislocated Bundle and the Rotated Bundle bounding analyses confirmed that for all normal operational conditions with ATRIUM-10 and ATRIUM-11 fuel, 99.9% of the fuel rods in the core avoid boiling transition and the fuel rods will not exceed the 1% plastic strain criteria.

15.4.7.3.4 Considerations of Uncertainties

In order to assure the conservatism of the analysis of the mislocated bundle, major input parameters are taken as a worst case, i.e., the bundle is placed in the location with the highest bundle power in the core and the bundle is operating on design thermal limits.

For the analysis of the rotated bundle it is assumed that the bundle is placed in the location with the highest LHGR and/or the lowest CPR in the core and the bundle is operating on design thermal limits

This assures that the ΔCPR and the LHGR are the upper bounds for the mislocated and the rotated bundle errors.

15.4.7.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since it is a very mild and highly localized event. No perceptible change in the core pressure would be observed.

15.4.7.5 Radiological Consequences

Analyses confirmed that for all normal operating conditions with ATRIUM-10 and ATRIUM-11 fuel, 99.9% of the fuel rods in the core avoid boiling transition and the fuel rods will not exceed the 1% plastic strain criteria for both the Misloaded Bundle and the Rotated Bundle events. Any radiological doses resulting from these events would be a small fraction of the 10CFR50.67 regulatory dose limits. The regulatory dose consequences from these events are bounded by the control rod drop event given in Subsection 15.4.9.

15.4.8 SPECTRUM OF ROD EJECTION ASSEMBLIES

Not applicable to BWRs.

The BWR has precluded this event by incorporating into its design mechanical equipment which restricts any movement of the control rod drive system assemblies. The control rod drive housing support assemblies are described in Chapter 4.

15.4.9 CONTROL ROD DROP ACCIDENT (CRDA)

15.4.9.1 Identification of Causes and Frequency Classification

15.4.9.1.1 Identification of Causes

The control rod drop accident is the result of a postulated event in which a high worth control rod is inserted in-sequence into the core. Subsequently, it becomes decoupled from its drive mechanism. The mechanism is withdrawn but the decoupled control rod is assumed to be stuck in place. At a later optimum moment, the control rod suddenly falls free and drops out of the core.

This results in the insertion of large positive reactivity to the core and causes a localized power excursion.

A more detailed discussion is given in Reference 15.4-1.

15.4.9.1.2 Frequency Classification

The CRDA is categorized as a limiting fault because it is not expected to occur during the lifetime of the plant; but, if postulated to occur, it has consequences that include the potential for the release of radioactive material from the fuel.

15.4.9.2 Sequence of Events and System Operation

15.4.9.2.1 Sequence of Events

Before the control rod drop accident (CRDA) is possible, the sequence of events presented in Tables 15.4-1 must occur. No operator actions are required to terminate this transient.

15.4.9.2.2 Systems Operation

Note: Unit 2 RSCS removed.

The unlikely set of circumstances, referenced above, makes possible the rapid removal of a control rod. The dropping of the rod results in high reactivity in a small region of the core. For large, loosely coupled cores, this would result in a highly peaked power distribution and subsequent operation of shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion.

The RWM restricts control rod patterns during startup, such that only specified rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% of rated core power. The sequences effectively limit the potential amount and rate of reactivity increase during a Control Rod Drop Accident. The RWM is designed to act as a backup to operator control of the rod sequences. Therefore if the RWM is inoperable or bypassed, the Technical Specifications require that a second operator verify that any subsequent rod selection and withdrawal is in accordance with the specified rod sequence.

The termination of this excursion is accomplished by automatic safety features of inherent shutdown mechanisms. Therefore, no operator action during the excursion is required. Other plant instrumentation and controls are assumed to function normally. The main condenser is assumed to isolate on main steam line high radiation (i.e., isolation of mechanical vacuum pump and steam jet air ejectors occurs). No credit is taken for other plant instrumentation and controls.

15.4.9.2.3 Effect of Single Failures and Operator Errors

Systems mitigating the consequences of this event are the RWM and APRM scram. The RWM in conjunction with the Technical Specification requirement that a second operator verify the control rod sequence and withdrawal if the RWM is bypassed essentially provides a redundant system and therefore together provide single failure protection. The APRM scram system is designed to single failure criteria. Therefore, termination of this transient within the limiting results discussed below is assured.

No operator error (in addition to the one that initiates this event) can result in a more limiting case since the reactor protection system will automatically terminate the transient.

15.4.9.3 Core and System Performance

15.4.9.3.1 Mathematical Model

A method for determining the energy deposition in the UO_2 fuel in the fuel bundles surrounding the dropped control rod is described in References 15.4-14 and 15.4-16. The necessary input data for this method are:

Delayed Neutron Fraction
Doppler Coefficient
Bundle Peaking Factors
Worth of the Dropped Control Rod.

The method allows one to determine the number of fuel bundles that have an energy deposition of 170 cal/gm or more. Assemblies which have this amount or more of energy deposited in the fuel are assumed to fail. The method also allows one to determine the maximum energy deposition in any fuel bundle. This value must be less than 280 cal/gm.

The methods for modeling and analyzing this event for Unit 1 are described in References 15.4-14, 15.4-17, and 15.4-18. The methods for modeling and analyzing this event for Unit 2 are described in References 15.4-14, 15.4-17, 15.4-20, and 15.4-21. These methods and models were used to determine the maximum control rod worth, Doppler coefficient, delayed neutron fraction, and fuel bundle peaking factors that are needed to assess the effects of the control rod drop accident.

15.4.9.3.2 Input Parameters and Initial Conditions

The core at the time of rod drop accident is assumed to be at the point in cycle which results in the highest control rod worth, to contain no xenon, and to be in a hot startup condition. The highest control rod worth is assumed to occur at one of the following core conditions for Unit 1 and Reference 15.4-18 methodology:

a. Peak hot zero power reactivity
b. Peak hot excess reactivity
c. Maximum control rod density

Unit 2 evaluations are performed at the core conditions specified in Reference 15.4-20.

The control rod sequence and rods withdrawn are shown in Tables 15C.4.9-2 and 15D.4.9-2 for the current cycles for Units 1 and 2. Removing xenon, which competes well for neutron absorptions, increases the fractional absorptions, or worth, of the control rods. Control rod density of up to 50%("black and white" rod pattern), which nominally occurs at the hot-startup condition, is also assumed. This assumption ensures that withdrawal on the next rod results in the maximum increment of reactivity.

Reference 15.4-6 limits the maximum enthalpy for the control rod drop accident to 280 calories per gram. Therefore, the maximum incremental rod worths are maintained at very low values so that a postulated CRDA cannot result in peak enthalpies in excess of 280 calories per gram for any plant condition.

15.4.9.3.3 Results

The radiological evaluations are conservatively based on the assumed failure of 2000 fuel rods. The number of rods which exceed the damage threshold is less than 2000 for all plant operating conditions.

The results of the compliance-check calculations, are shown in the Tables 15C.4.9-2 and 15D.4.9-2 for the current cycles for Units 1 and 2. These tables show the maximum incremental rod worth, the number of fuel rods that exceed an energy deposition of 170 cal/gm, (Unit 1), the number of fuel rods that are assumed to have failed (Unit 2) and the maximum energy deposition in the fuel bundles surrounding the dropped control rod. The conclusion is that the 280 cal/gm design limit is not exceeded and the assumed failure of 2000 pins for the radiological evaluation is conservative.

15.4.9.4 Barrier Performance

An evaluation of the barrier performance was not made for this accident since this is a highly localized event with no significant change in the gross core temperature or pressure.

15.4.9.5 Radiological Consequences

ATRIUM-11 has a higher assembly uranium mass and higher core average burnup than ATRIUM-10, resulting in a conservative core source term for Susquehanna core designs. Two cases are analyzed for the CRDA. The first case follows the guidance in Regulatory Guide 1.183 and is applicable when the plant is at power. Rods with 170 cal/gm or more energy deposited in the fuel are assumed to fail. This analysis conservatively assumes that 0.77% of the fuel within a failed fuel rod melts. This fuel melt assumption is intended to ensure compatibility with the same assumption made in GE's Topical Report NEDO-31400A, which evaluated the elimination of certain main steam radiation monitor safety functions (Reference 15.4-7). The second case potentially occurs during low power operation with the mechanical vacuum pump (MVP) running. At low power with fewer than 35 rods failing, main steam line dose rates may be too low to be reliably sensed by the Main Steam Line Radiation Monitors to generate a trip signal for the mechanical vacuum pump. Failure to trip the vacuum pump would result in an unfiltered release of fission products to the environment from the turbine building vent stack.

Two separate radiological analyses are provided for each of these cases:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10CFR50.67 guidelines. This analysis is referred to as the "Design Basis Analysis."

(2) The second analysis is based on assumptions considered to provide a realistic conservative estimate of radiological consequences. This analysis is referred to as the "Realistic Analysis."

A schematic of the leakage path is shown in Figure 15.4-1.

Specific parametric values used for the design basis and the realistic analyses are presented in Table 15.4.-11.

15.4.9.5.1 Design Basis Analysis

The design basis analysis is based on Regulatory Guide 1.183 (Reference 15.4-6). The RADTRAD Computer Program (Reference 15.4-8) is used to evaluate the radiological consequences for the design basis analysis.

It is assumed that 10 percent of the iodines and noble gases and 12 percent of the Cs and Rb contained in the gap of the fuel rods that experience cladding damage are released from the fuel. The release from the fuel melting is assumed to be 100% of the noble gases and 50% of the iodines within the region that melts per Regulatory Guide 1.183, Appendix C.1. Solids released from the melted fuel are in accordance with the fractions shown in Regulatory Guide 1.183, Table 1.

Activity released from the fuel is assumed to instantaneously mix in the reactor coolant. Of this activity, 100% of the noble gases, 10% of the radioiodine, and 1% of the remaining nuclides are assumed to reach the condenser and turbine. Of the activity reaching the condenser and turbine, 100% of the noble gases, 10% of the iodine, and 1% of the particulate nuclides are available for release to the environment. The turbine and condenser leak to the atmosphere at 1% per day for 24 hours, after which time, the leakage is assumed to terminate. No credit is taken for holdup in the turbine building.

If the mechanical vacuum pump is running, it removes activity from the main condenser at the rate of 1212%/day. The rate is based on a realistic condenser volume of 195,000 ft^3 and extraction of non-condensable gases at 75 cfm by the mechanical vacuum pump. Radioiodine chemical species released from the main condenser/turbine is 97% elemental and 3% organic iodine.

The activity airborne in the condenser is presented in Tables 15.4-2 and 15.4-3 for the cases with 2000 failed rods and 35 failed rods, respectively. The activity releases to the environment are presented in Tables 15.4-4 and 15.4-5 for 2000 failed rods and 35 failed rods, respectively.

15.4.9.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The RADTRAD Computer Program (Reference 15.4-8) is used to evaluate the radiological consequences for the realistic analysis.

The following assumptions are used in calculating fission product activity release from the fuel:

a) The reactor has been operating at design power for 3 years until 30 minutes prior to the accident. When translated into actual plant operation, this assumption means that the reactor was shut down from design power, taken critical, and brought to the initial temperature conditions within 30 minutes of the departure from design power. The 30 minute time represents a conservative estimate of the shortest period in which the required plant changes could be accomplished and defines the decay time to be applied to the fission product inventory calculations. 2000 fuel rods are conservatively assumed to be damaged.

b) An average of 1.8 percent of the noble gas, cesium and rubidium activity and 0.32 percent of the halogen activity in a failed fuel rod is assumed to be released. These percentages are consistent with actual measurements made during defective fuel experiments (Reference 15.4-9).

c) The fraction of other solid fission product activity available for release from the fuel is negligible.

d) The fission products produced during the nuclear excursion are neglected.

The following assumptions are used in calculating the amount of fission product activity transported from the reactor vessel to the main condenser:

a) All of the noble gas activity and 2% of the iodine, cesium and rubidium activity released from the damaged fuel rods is conservatively assumed to be immediately available for release from the condenser.

b) The reactor water sample line is isolated by a Main Steam Line Radiation Monitor high high radiation signal prior to the release of any activity to the reactor building.

Of those fission products released from the fuel and transferred to the condenser, it is assumed that 100 percent of the noble gases are airborne in the condenser. The iodine activity airborne in the condenser is a function of the partition factor, volume of air, and volume of water. A partition factor of 140 is assumed in condenser for iodine, cesium and rubidium activity. By using the above conditions, the activity airborne in the condenser is presented in Tables 15.4-6 and 15.4-7.

The following assumptions and conditions are used to evaluate the activity released to the environment:

a) The leak rate out of the condenser is 0.5 percent per day of the combined condenser and turbine free volume.

b) The activity released from the condenser becomes airborne in the turbine building. The turbine building ventilation rate is seven air changes per day.

c) No filtration or plateout of iodines occurs in the building prior to release to the atmosphere.

Based on the above assumptions, the fission product release to the environment is presented in Tables 15.4-8 and 15.4-9.

15.4.9.5.3 Results

Offsite

The calculated radiological doses at the site boundary and low population zone for the design basis and realistic cases with 2000 failed fuels rods and 35 failed fuel rods are presented in Table 15.4-10. The doses are well within the 10CFR50.67 dose limits and the Regulatory Guide 1.183 acceptance criteria.

Control Room

A detailed description of the control room model can be found in Appendix 15B. The parameters used in the analysis are provided in Table 15.4-11. The radiological exposure to the control room personnel for the design basis case is given in Table 15.4-10. The doses are well within the 10CFR50.67 dose limits.

15.4.10 REFERENCES

15.4-1 R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs," March 1972 (NEDO-10527).

15.4-2 C. J. Paone, "Bank Position Withdrawal Sequence," September 1976 (NEDO-21231).

15.4-3 R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs," July 1972 Supplement 1 (NEDO-10527).

15.4-4 R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs," January 1973 Supplement 2 (NEDO-10527).

15.4-5 "GE BWR Generic Reload Application for 8x8 Fuel" (NEDO-20360).

15.4-6 USNRC Regulatory Guide 1.183, Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors, July 2000.

15.4-7 General Electric Topical Report, NEDO-31400A, Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and Reactor Scram Function of the Main Steam Line Radiation Monitor", October 1992.

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15.4.12 Deleted

15.4-13 Deleted

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15.4-17 EMF-2158(P)(A) Rev. 0, “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/Microburn-B2,” Siemens Power Corporation, October 1999.

15.4-18 XN-NF-80-19(P)(A) Volume 4 Revision 1, “Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads,” Exxon Nuclear company, June 1986.

15.4-19 EMF-CC-043(P) Rev. 9, “XCOBRA Code Theory and User’s Manual,” Framatome ANP, Inc., November 2003.

15.4-20 ANP-10333P-A Revision 0, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Control Rod Drop Accident (CRDA)”, Framatome Inc., March 2018.

15.4-21 ANP-3771P, “Susquehanna Atrium-11 Control Rod Drop Accident Analyses with the AURORA-B CRDA Methodology”, Framatome, Inc., May 2019.

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TABLE 15.4-2

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
AIRBORNE ACTIVITY IN CONDENSER, Ci
(2000 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2 HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
Co-58	3.44E-05	3.43E-05	3.43E-05	3.42E-05	3.37E-05	3.27E-05	2.54E-05
Co-60	1.41E-06	1.41E-06	1.41E-06	1.41E-06	1.40E-06	1.39E-06	1.38E-06
Kr-85	7.47E+03	7.47E+03	7.46E+03	7.45E+03	7.40E+03	7.39E+03	7.36E+03
Kr-85m	1.25E+05	1.07E+05	9.16E+04	3.61E+04	3.02E+03	4.38E-02	5.16E-44
Kr-87	2.37E+05	1.38E+05	7.98E+04	3.02E+03	4.90E-01	4.43E-18	0.00E+00
Kr-88	3.20E+05	2.51E+05	1.96E+05	4.53E+04	9.07E+02	2.12E-05	1.53E-71
Rb-86	1.24E-01	1.24E-01	1.24E-01	1.22E-01	1.19E-01	1.06E-01	4.04E-02
Sr-89	6.37E-02	6.36E-02	6.36E-02	6.32E-02	6.22E-02	5.97E-02	4.18E-02
Sr-90	8.50E-03	8.50E-03	8.49E-03	8.47E-03	8.41E-03	8.41E-03	8.40E-03
Sr-91	8.01E-02	7.44E-02	6.92E-02	4.45E-02	1.38E-02	7.20E-05	1.21E-24
Sr-92	8.65E-02	6.70E-02	5.18E-02	1.11E-02	1.85E-04	1.86E-12	9.00E-82
Y-90	8.97E-05	1.80E-04	2.70E-04	7.85E-04	1.99E-03	5.47E-03	8.40E-03
Y-91	8.29E-04	8.45E-04	8.59E-04	9.23E-04	9.96E-04	9.99E-04	7.35E-04
Y-92	8.88E-04	1.43E-02	2.22E-02	2.26E-02	1.95E-03	1.92E-09	1.67E-62
Y-93	9.94E-04	9.27E-04	8.65E-04	5.72E-04	1.89E-04	1.35E-06	3.41E-25
Zr-95	1.20E-03	1.20E-03	1.20E-03	1.19E-03	1.18E-03	1.14E-03	8.60E-04
Zr-97	1.22E-03	1.17E-03	1.12E-03	8.76E-04	4.51E-04	2.35E-05	1.81E-16
Nb-95	1.20E-03	1.20E-03	1.20E-03	1.20E-03	1.19E-03	1.19E-03	1.10E-03
Mo-99	1.60E-02	1.58E-02	1.57E-02	1.47E-02	1.23E-02	5.78E-03	8.24E-06
Tc-99m	1.40E-02	1.40E-02	1.40E-02	1.36E-02	1.18E-02	5.60E-03	7.98E-06
Ru-103	1.38E-02	1.38E-02	1.38E-02	1.37E-02	1.34E-02	1.27E-02	8.06E-03
Ru-105	9.60E-03	8.21E-03	7.02E-03	2.75E-03	2.24E-04	2.95E-09	1.45E-51
Ru-106	5.37E-03	5.36E-03	5.36E-03	5.34E-03	5.30E-03	5.27E-03	5.02E-03
Rh-105	8.93E-03	8.93E-03	8.90E-03	8.39E-03	6.35E-03	1.55E-03	7.58E-09
Sb-127	1.64E-02	1.63E-02	1.62E-02	1.54E-02	1.36E-02	7.93E-03	7.35E-05
Sb-129	4.90E-02	4.17E-02	3.55E-02	1.35E-02	1.03E-03	9.92E-09	3.27E-52
Te-127	1.48E-02	1.48E-02	1.47E-02	1.45E-02	1.33E-02	8.38E-03	1.09E-03
Te-127m	1.16E-03	1.16E-03	1.16E-03	1.16E-03	1.16E-03	1.17E-03	1.04E-03
Te-129	4.68E-02	4.37E-02	3.97E-02	1.95E-02	6.22E-03	4.82E-03	2.82E-03
Te-129m	8.03E-03	8.03E-03	8.03E-03	7.99E-03	7.84E-03	7.38E-03	4.31E-03
Te-131m	3.36E-02	3.28E-02	3.21E-02	2.78E-02	1.91E-02	3.62E-03	1.98E-09
Te-132	2.45E-01	2.43E-01	2.41E-01	2.28E-01	1.96E-01	1.04E-01	4.11E-04
I-131	4.67E+03	4.65E+03	4.63E+03	4.52E+03	4.24E+03	3.27E+03	3.48E+02
I-132	6.80E+03	5.03E+03	3.72E+03	6.09E+02	5.07E+00	1.07E-01	4.25E-04
I-133	9.53E+03	9.21E+03	8.91E+03	7.27E+03	4.24E+03	3.85E+02	3.58E-07
I-134	1.07E+04	4.84E+03	2.19E+03	1.90E+01	6.06E-05	1.15E-29	0.00E+00
I-135	9.09E+03	8.18E+03	7.37E+03	3.92E+03	7.27E+02	3.82E-01	1.46E-29
Xe-133	9.68E+05	9.63E+05	9.57E+05	9.25E+05	8.44E+05	5.73E+05	1.87E+04
Xe-135	2.61E+05	2.48E+05	2.30E+05	1.47E+05	4.44E+04	1.90E+02	4.14E-19

TABLE 15.4-2 (CONTINUED)

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
AIRBORNE ACTIVITY IN CONDENSER, Ci
(2000 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2 HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
Cs-134	1.19E+01	1.19E+01	1.19E+01	1.18E+01	1.17E+01	1.17E+01	1.14E+01
Cs-136	3.01E+00	3.00E+00	2.99E+00	2.95E+00	2.82E+00	2.41E+00	6.09E-01
Cs-137	9.10E+00	9.10E+00	9.10E+00	9.07E+00	9.01E+00	9.01E+00	9.00E+00
Ba-139	1.24E-01	7.53E-02	4.55E-02	2.22E-03	7.07E-07	1.33E-22	0.00E+00
Ba-140	1.20E-01	1.20E-01	1.20E-01	1.18E-01	1.13E-01	9.59E-02	2.33E-02
La-140	1.26E-03	3.29E-03	5.28E-03	1.64E-02	4.00E-02	8.44E-02	2.68E-02
La-141	1.13E-03	9.45E-04	7.91E-04	2.74E-04	1.62E-05	4.95E-11	7.89E-59
La-142	1.09E-03	6.93E-04	4.42E-04	2.97E-05	2.21E-08	1.93E-22	0.00E+00
Ce-141	2.84E-03	2.84E-03	2.83E-03	2.81E-03	2.76E-03	2.59E-03	1.49E-03
Ce-143	2.63E-03	2.58E-03	2.52E-03	2.22E-03	1.58E-03	3.47E-04	7.05E-10
Ce-144	2.36E-03	2.36E-03	2.36E-03	2.35E-03	2.34E-03	2.32E-03	2.18E-03
Pr-143	1.03E-03	1.03E-03	1.03E-03	1.05E-03	1.07E-03	1.03E-03	2.83E-04
Nd-147	4.49E-04	4.48E-04	4.47E-04	4.39E-04	4.18E-04	3.46E-04	6.69E-05
Np-239	3.25E-02	3.21E-02	3.17E-02	2.94E-02	2.40E-02	9.92E-03	4.71E-06
Pu-238	7.00E-06	7.00E-06	7.00E-06	6.98E-06	6.93E-06	6.94E-06	6.96E-06
Pu-239	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.57E-07
Pu-240	1.17E-06	1.17E-06	1.17E-06	1.17E-06	1.16E-06	1.16E-06	1.16E-06
Pu-241	2.79E-04	2.79E-04	2.79E-04	2.78E-04	2.76E-04	2.76E-04	2.75E-04
Am-241	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.73E-07
Cm-242	4.88E-05	4.88E-05	4.87E-05	4.86E-05	4.81E-05	4.75E-05	4.25E-05
Cm-244	2.49E-06	2.49E-06	2.49E-06	2.48E-06	2.47E-06	2.46E-06	2.46E-06
Kr-83m	5.69E+04	3.95E+04	2.74E+04	3.06E+03	8.87E+00	3.47E-11	0.00E+00
Xe-133m	3.06E+04	3.02E+04	2.98E+04	2.76E+04	2.25E+04	9.12E+03	3.61E+00
Xe-135m	2.12E+05	1.60E+04	2.20E+03	6.25E+02	1.16E+02	6.10E-02	2.33E-30

TABLE 15.4-3

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
AIRBORNE ACTIVITY IN CONDENSER, Ci
(35 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2 HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
Kr-85	1.31E+02	7.89E+01	4.76E+01	2.30E+00	7.13E-04	7.12E-04	7.09E-04
Kr-85m	2.19E+03	1.13E+03	5.84E+02	1.12E+01	2.91E-04	4.22E-09	4.97E-51
Kr-87	4.15E+03	1.46E+03	5.09E+02	9.35E-01	4.72E-08	4.27E-25	0.00E+00
Kr-88	5.60E+03	2.65E+03	1.25E+03	1.40E+01	8.73E-05	2.04E-12	1.47E-78
Rb-86	2.18E-03	1.31E-03	7.90E-04	3.78E-05	1.14E-08	1.02E-08	3.89E-09
Sr-89	1.11E-03	6.72E-04	4.06E-04	1.95E-05	5.99E-09	5.75E-09	4.02E-09
Sr-90	1.49E-04	8.98E-05	5.42E-05	2.62E-06	8.11E-10	8.10E-10	8.09E-10
Sr-91	1.40E-03	7.87E-04	4.41E-04	1.38E-05	1.33E-09	6.94E-12	1.17E-31
Sr-92	1.51E-03	7.08E-04	3.31E-04	3.44E-06	1.78E-11	1.79E-19	8.67E-89
Y-90	1.57E-06	1.90E-06	1.72E-06	2.43E-07	1.92E-10	5.27E-10	8.09E-10
Y-91	1.45E-05	8.93E-06	5.48E-06	2.85E-07	9.60E-11	9.63E-11	7.08E-11
Y-92	1.55E-05	1.51E-04	1.42E-04	6.99E-06	1.88E-10	1.85E-16	1.61E-69
Y-93	1.74E-05	9.80E-06	5.52E-06	1.77E-07	1.83E-11	1.30E-13	3.29E-32
Zr-95	2.10E-05	1.27E-05	7.65E-06	3.69E-07	1.13E-10	1.10E-10	8.28E-11
Zr-97	2.13E-05	1.24E-05	7.16E-06	2.71E-07	4.35E-11	2.27E-12	1.74E-23
Nb-95	2.11E-05	1.27E-05	7.68E-06	3.71E-07	1.15E-10	1.15E-10	1.06E-10
Mo-99	2.80E-04	1.67E-04	9.99E-05	4.53E-06	1.19E-09	5.57E-10	7.94E-13
Tc-99m	2.46E-04	1.48E-04	8.94E-05	4.21E-06	1.14E-09	5.40E-10	7.69E-13
Ru-103	2.42E-04	1.46E-04	8.79E-05	4.23E-06	1.29E-09	1.23E-09	7.76E-10
Ru-105	1.68E-04	8.68E-05	4.48E-05	8.49E-07	2.16E-11	2.84E-16	1.40E-58
Ru-106	9.39E-05	5.67E-05	3.42E-05	1.65E-06	5.11E-10	5.08E-10	4.84E-10
Rh-105	1.56E-04	9.43E-05	5.68E-05	2.59E-06	6.11E-10	1.50E-10	7.30E-16
Sb-127	2.88E-04	1.72E-04	1.03E-04	4.77E-06	1.31E-09	7.64E-10	7.08E-12
Sb-129	8.58E-04	4.41E-04	2.27E-04	4.18E-06	9.94E-11	9.56E-16	3.15E-59
Te-127	2.59E-04	1.56E-04	9.40E-05	4.47E-06	1.28E-09	8.08E-10	1.05E-10
Te-127m	2.03E-05	1.23E-05	7.42E-06	3.59E-07	1.12E-10	1.13E-10	1.00E-10
Te-129	8.18E-04	4.62E-04	2.53E-04	6.01E-06	6.00E-10	4.65E-10	2.72E-10
Te-129m	1.41E-04	8.49E-05	5.12E-05	2.47E-06	7.56E-10	7.11E-10	4.16E-10
Te-131m	5.88E-04	3.47E-04	2.05E-04	8.60E-06	1.84E-09	3.49E-10	1.91E-16
Te-132	4.29E-03	2.57E-03	1.54E-03	7.04E-05	1.89E-08	1.00E-08	3.96E-11
I-131	8.17E+01	4.91E+01	2.95E+01	1.40E+00	4.09E-04	3.15E-04	3.35E-05
I-132	1.19E+02	5.32E+01	2.37E+01	1.88E-01	4.88E-07	1.03E-08	4.09E-11
I-133	1.67E+02	9.73E+01	5.68E+01	2.25E+00	4.08E-04	3.71E-05	3.45E-14
I-134	1.87E+02	5.11E+01	1.40E+01	5.89E-03	5.84E-12	1.10E-36	0.00E+00
I-135	1.59E+02	8.65E+01	4.70E+01	1.21E+00	7.00E-05	3.68E-08	1.41E-36
Xe-133	1.69E+04	1.02E+04	6.11E+03	2.86E+02	8.13E-02	5.52E-02	1.80E-03
Xe-135	4.57E+03	2.62E+03	1.47E+03	4.56E+01	4.27E-03	1.83E-05	3.98E-26
Cs-134	2.08E-01	1.25E-01	7.57E-02	3.65E-03	1.13E-06	1.13E-06	1.10E-06
Cs-136	5.26E-02	3.17E-02	1.91E-02	9.10E-04	2.72E-07	2.32E-07	5.87E-08
Cs-137	1.59E-01	9.62E-02	5.80E-02	2.80E-03	8.68E-07	8.68E-07	8.67E-07

TABLE 15.4-3 (CONTINUED)

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
AIRBORNE ACTIVITY IN CONDENSER, Ci
(35 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2 HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
Ba-139	2.18E-03	7.96E-04	2.90E-04	6.87E-07	6.81E-14	1.28E-29	0.00E+00
Ba-140	2.11E-03	1.27E-03	7.64E-04	3.64E-05	1.09E-08	9.24E-09	2.25E-09
La-140	2.21E-05	3.48E-05	3.37E-05	5.07E-06	3.86E-09	8.13E-09	2.59E-09
La-141	1.97E-05	9.98E-06	5.05E-06	8.47E-08	1.56E-12	4.77E-18	7.60E-66
La-142	1.90E-05	7.32E-06	2.82E-06	9.17E-09	2.13E-15	1.86E-29	0.00E+00
Ce-141	4.97E-05	3.00E-05	1.81E-05	8.70E-07	2.66E-10	2.49E-10	1.43E-10
Ce-143	4.61E-05	2.73E-05	1.61E-05	6.86E-07	1.52E-10	3.35E-11	6.80E-17
Ce-144	4.14E-05	2.50E-05	1.51E-05	7.28E-07	2.25E-10	2.23E-10	2.10E-10
Pr-143	1.80E-05	1.09E-05	6.59E-06	3.24E-07	1.03E-10	9.92E-11	2.73E-11
Nd-147	7.86E-06	4.73E-06	2.85E-06	1.36E-07	4.02E-11	3.33E-11	6.45E-12
Np-239	5.69E-04	3.39E-04	2.02E-04	9.08E-06	2.31E-09	9.56E-10	4.54E-13
Pu-241	4.88E-06	2.95E-06	1.78E-06	8.59E-08	2.66E-11	2.66E-11	2.65E-11
Kr-83m	9.95E+02	4.17E+02	1.75E+02	9.46E-01	8.55E-07	3.35E-18	0.00E+00
Xe-133m	5.35E+02	3.19E+02	1.90E+02	8.53E+00	2.16E-03	8.78E-04	3.47E-07
Xe-135m	3.71E+03	1.69E+02	1.40E+01	1.93E-01	1.12E-05	5.88E-09	2.25E-37

TABLE 15.4-4

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
ACTIVITY RELEASED TO ENVIRONS, Ci
(2000 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2 HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
Kr-85	1.56E-03	3.11E+00	6.22E+00	2.49E+01	7.43E+01	7.43E+01	7.43E+01
Kr-85m	2.60E-02	4.84E+01	8.98E+01	2.39E+02	3.30E+02	3.30E+02	3.30E+02
Kr-87	4.95E-02	7.73E+01	1.22E+02	1.82E+02	1.84E+02	1.84E+02	1.84E+02
Kr-88	6.67E-02	1.19E+02	2.12E+02	4.72E+02	5.49E+02	5.49E+02	5.49E+02
Rb-86	0.00E+00	5.17E-05	1.03E-04	4.11E-04	1.22E-03	1.22E-03	1.22E-03
Sr-89	0.00E+00	2.65E-05	5.30E-05	2.11E-04	6.29E-04	6.29E-04	6.29E-04
Sr-90	0.00E+00	3.54E-06	7.08E-06	2.83E-05	8.46E-05	8.46E-05	8.46E-05
Sr-91	0.00E+00	3.22E-05	6.22E-05	2.02E-04	3.79E-04	3.79E-04	3.79E-04
Sr-92	0.00E+00	3.20E-05	5.68E-05	1.23E-04	1.42E-04	1.42E-04	1.42E-04
Y-90	0.00E+00	0.00E+00	0.00E+00	1.47E-06	1.08E-05	1.08E-05	1.08E-05
Y-91	0.00E+00	0.00E+00	0.00E+00	2.94E-06	9.39E-06	9.39E-06	9.39E-06
Y-92	0.00E+00	3.24E-06	1.09E-05	7.61E-05	1.39E-04	1.39E-04	1.39E-04
Y-93	0.00E+00	0.00E+00	0.00E+00	2.55E-06	4.88E-06	4.88E-06	4.88E-06
Zr-95	0.00E+00	0.00E+00	1.00E-06	3.99E-06	1.19E-05	1.19E-05	1.19E-05
Zr-97	0.00E+00	0.00E+00	0.00E+00	3.46E-06	7.76E-06	7.76E-06	7.76E-06
Nb-95	0.00E+00	0.00E+00	1.00E-06	4.01E-06	1.20E-05	1.20E-05	1.20E-05
Mo-99	0.00E+00	6.63E-06	1.32E-05	5.11E-05	1.41E-04	1.41E-04	1.41E-04
Tc-99m	0.00E+00	5.85E-06	1.17E-05	4.63E-05	1.32E-04	1.32E-04	1.32E-04
Ru-103	0.00E+00	5.75E-06	1.15E-05	4.58E-05	1.36E-04	1.36E-04	1.36E-04
Ru-105	0.00E+00	3.72E-06	6.90E-06	1.83E-05	2.52E-05	2.52E-05	2.52E-05
Ru-106	0.00E+00	2.23E-06	4.47E-06	1.78E-05	5.33E-05	5.33E-05	5.33E-05
Rh-105	0.00E+00	3.72E-06	7.43E-06	2.91E-05	7.84E-05	7.84E-05	7.84E-05
Sb-127	0.00E+00	6.82E-06	1.36E-05	5.31E-05	1.50E-04	1.50E-04	1.50E-04
Sb-129	0.00E+00	1.89E-05	3.51E-05	9.23E-05	1.25E-04	1.25E-04	1.25E-04
Te-127	0.00E+00	6.15E-06	1.23E-05	4.88E-05	1.42E-04	1.42E-04	1.42E-04
Te-127m	0.00E+00	0.00E+00	0.00E+00	3.88E-06	1.16E-05	1.16E-05	1.16E-05
Te-129	0.00E+00	1.89E-05	3.64E-05	1.08E-04	1.77E-04	1.77E-04	1.77E-04
Te-129m	0.00E+00	3.34E-06	6.69E-06	2.67E-05	7.95E-05	7.95E-05	7.95E-05
Te-131m	0.00E+00	1.38E-05	2.74E-05	1.02E-04	2.57E-04	2.57E-04	2.57E-04
Te-132	0.00E+00	1.02E-04	2.03E-04	7.89E-04	2.20E-03	2.20E-03	2.20E-03
I-131	9.73E-04	1.94E+00	3.87E+00	1.53E+01	4.45E+01	4.45E+01	4.45E+01
I-132	1.42E-03	2.46E+00	4.29E+00	8.62E+00	9.47E+00	9.47E+00	9.47E+00
I-133	1.98E-03	3.90E+00	7.68E+00	2.79E+01	6.55E+01	6.55E+01	6.55E+01
I-134	2.22E-03	3.13E+00	4.55E+00	5.72E+00	5.73E+00	5.73E+00	5.73E+00
I-135	1.89E-03	3.60E+00	6.85E+00	2.05E+01	3.33E+01	3.33E+01	3.33E+01
Xe-133	2.02E-01	4.02E+02	8.02E+02	3.16E+03	9.05E+03	9.05E+03	9.06E+03
Xe-135	5.44E-02	1.06E+02	2.06E+02	6.72E+02	1.25E+03	1.25E+03	1.26E+03
Cs-134	2.47E-06	4.94E-03	9.88E-03	3.95E-02	1.18E-01	1.18E-01	1.18E-01
Cs-136	0.00E+00	1.25E-03	2.50E-03	9.92E-03	2.92E-02	2.92E-02	2.92E-02
Cs-137	1.90E-06	3.79E-03	7.58E-03	3.03E-02	9.06E-02	9.06E-02	9.06E-02
Ba-139	0.00E+00	4.13E-05	6.62E-05	1.03E-04	1.04E-04	1.04E-04	1.04E-04
Ba-140	0.00E+00	5.01E-05	1.00E-04	3.97E-04	1.17E-03	1.17E-03	1.17E-03

TABLE 15.4-4 (CONTINUED)

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
ACTIVITY RELEASED TO ENVIRONS, Ci
(2000 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2 HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
La-140	0.00E+00	0.00E+00	2.69E-06	3.00E-05	2.21E-04	2.21E-04	2.21E-04
La-141	0.00E+00	0.00E+00	0.00E+00	2.02E-06	2.64E-06	2.64E-06	2.64E-06
La-142	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.02E-06	1.02E-06	1.02E-06
Ce-141	0.00E+00	1.18E-06	2.36E-06	9.43E-06	2.80E-05	2.80E-05	2.80E-05
Ce-143	0.00E+00	1.09E-06	2.15E-06	8.07E-06	2.06E-05	2.06E-05	2.06E-05
Ce-144	0.00E+00	0.00E+00	1.97E-06	7.87E-06	2.35E-05	2.35E-05	2.35E-05
Pr-143	0.00E+00	0.00E+00	0.00E+00	3.46E-06	1.05E-05	1.05E-05	1.05E-05
Nd-147	0.00E+00	0.00E+00	0.00E+00	1.48E-06	4.33E-06	4.33E-06	4.33E-06
Np-239	0.00E+00	1.35E-05	2.68E-05	1.03E-04	2.81E-04	2.81E-04	2.81E-04
Pu-241	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.77E-06	2.77E-06	2.77E-06
Kr-83m	1.18E-02	2.00E+01	3.39E+01	6.19E+01	6.55E+01	6.55E+01	6.55E+01
Xe-133m	6.37E-03	1.27E+01	2.52E+01	9.69E+01	2.64E+02	2.64E+02	2.64E+02
Xe-135m	4.42E-02	3.32E+01	3.60E+01	3.85E+01	4.09E+01	4.30E+01	6.04E+01

TABLE 15.4-5

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
ACTIVITY RELEASED TO ENVIRONS, Ci
(35 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2 HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
Kr-85	3.30E-02	5.18E+01	8.31E+01	1.28E+02	1.31E+02	1.31E+02	1.31E+02
Kr-85m	5.52E-01	8.11E+02	1.23E+03	1.67E+03	1.68E+03	1.68E+03	1.68E+03
Kr-87	1.05E+00	1.32E+03	1.78E+03	2.03E+03	2.03E+03	2.03E+03	2.03E+03
Kr-88	1.42E+00	2.00E+03	2.95E+03	3.79E+03	3.80E+03	3.80E+03	3.80E+03
Rb-86	0.00E+00	8.62E-04	1.38E-03	2.13E-03	2.17E-03	2.17E-03	2.17E-03
Sr-89	0.00E+00	4.42E-04	7.08E-04	1.09E-03	1.11E-03	1.11E-03	1.11E-03
Sr-90	0.00E+00	5.89E-05	9.45E-05	1.46E-04	1.49E-04	1.49E-04	1.49E-04
Sr-91	0.00E+00	5.38E-04	8.41E-04	1.22E-03	1.23E-03	1.23E-03	1.23E-03
Sr-92	0.00E+00	5.39E-04	7.90E-04	1.01E-03	1.01E-03	1.01E-03	1.01E-03
Y-90	0.00E+00	0.00E+00	1.81E-06	4.33E-06	4.62E-06	4.62E-06	4.62E-06
Y-91	0.00E+00	5.80E-06	9.37E-06	1.47E-05	1.50E-05	1.50E-05	1.50E-05
Y-92	0.00E+00	5.00E-05	1.26E-04	2.83E-04	2.89E-04	2.89E-04	2.89E-04
Y-93	0.00E+00	6.69E-06	1.05E-05	1.52E-05	1.53E-05	1.53E-05	1.53E-05
Zr-95	0.00E+00	8.33E-06	1.34E-05	2.06E-05	2.10E-05	2.10E-05	2.10E-05
Zr-97	0.00E+00	8.31E-06	1.31E-05	1.95E-05	1.98E-05	1.98E-05	1.98E-05
Nb-95	0.00E+00	8.35E-06	1.34E-05	2.07E-05	2.11E-05	2.11E-05	2.11E-05
Mo-99	0.00E+00	1.10E-04	1.76E-04	2.70E-04	2.74E-04	2.74E-04	2.74E-04
Tc-99m	0.00E+00	9.74E-05	1.56E-04	2.41E-04	2.45E-04	2.45E-04	2.45E-04
Ru-103	0.00E+00	9.58E-05	1.54E-04	2.37E-04	2.41E-04	2.41E-04	2.41E-04
Ru-105	0.00E+00	6.23E-05	9.45E-05	1.28E-04	1.29E-04	1.29E-04	1.29E-04
Ru-106	0.00E+00	3.72E-05	5.97E-05	9.22E-05	9.39E-05	9.39E-05	9.39E-05
Rh-105	0.00E+00	6.20E-05	9.93E-05	1.53E-04	1.55E-04	1.55E-04	1.55E-04
Sb-127	0.00E+00	1.14E-04	1.82E-04	2.79E-04	2.84E-04	2.84E-04	2.84E-04
Sb-129	0.00E+00	3.18E-04	4.81E-04	6.50E-04	6.54E-04	6.54E-04	6.54E-04
Te-127	0.00E+00	1.02E-04	1.64E-04	2.53E-04	2.58E-04	2.58E-04	2.58E-04
Te-127m	0.00E+00	8.07E-06	1.29E-05	2.00E-05	2.04E-05	2.04E-05	2.04E-05
Te-129	0.00E+00	3.16E-04	4.92E-04	6.94E-04	6.99E-04	6.99E-04	6.99E-04
Te-129m	0.00E+00	5.57E-05	8.94E-05	1.38E-04	1.41E-04	1.41E-04	1.41E-04
Te-131m	0.00E+00	2.31E-04	3.67E-04	5.54E-04	5.63E-04	5.63E-04	5.63E-04
Te-132	1.08E-06	1.70E-03	2.71E-03	4.15E-03	4.22E-03	4.22E-03	4.22E-03
I-131	2.06E-02	3.23E+01	5.18E+01	7.97E+01	8.11E+01	8.11E+01	8.11E+01
I-132	3.01E-02	4.16E+01	6.01E+01	7.50E+01	7.51E+01	7.51E+01	7.51E+01
I-133	4.21E-02	6.51E+01	1.03E+02	1.54E+02	1.57E+02	1.57E+02	1.57E+02
I-134	4.71E-02	5.39E+01	6.86E+01	7.42E+01	7.42E+01	7.42E+01	7.42E+01
I-135	4.02E-02	6.03E+01	9.31E+01	1.31E+02	1.32E+02	1.32E+02	1.32E+02
Xe-133	4.28E+00	6.70E+03	1.07E+04	1.65E+04	1.68E+04	1.68E+04	1.68E+04
Xe-135	1.15E+00	1.78E+03	2.78E+03	4.03E+03	4.07E+03	4.08E+03	4.12E+03
Cs-134	5.24E-05	8.23E-02	1.32E-01	2.04E-01	2.08E-01	2.08E-01	2.08E-01
Cs-136	1.33E-05	2.08E-02	3.34E-02	5.15E-02	5.24E-02	5.24E-02	5.24E-02
Cs-137	4.02E-05	6.31E-02	1.01E-01	1.57E-01	1.59E-01	1.59E-01	1.59E-01

TABLE 15.4-5 (CONTINUED)

CONTROL ROD DROP ACCIDENT - DESIGN BASIS ANALYSIS
ACTIVITY RELEASED TO ENVIRONS, Ci
(35 FAILED RODS)

ISOTOPE	1 MIN	1 HOUR	2 HOUR	8 HOUR	1 DAY	4 DAY	30 DAY
Ba-139	0.00E+00	7.02E-04	9.58E-04	1.10E-03	1.11E-03	1.11E-03	1.11E-03
Ba-140	0.00E+00	8.34E-04	1.34E-03	2.06E-03	2.10E-03	2.10E-03	2.10E-03
La-140	0.00E+00	1.49E-05	3.23E-05	8.36E-05	8.96E-05	8.96E-05	8.96E-05
La-141	0.00E+00	7.25E-06	1.09E-05	1.46E-05	1.47E-05	1.47E-05	1.47E-05
La-142	0.00E+00	6.25E-06	8.66E-06	1.02E-05	1.02E-05	1.02E-05	1.02E-05
Ce-141	0.00E+00	1.97E-05	3.16E-05	4.88E-05	4.96E-05	4.96E-05	4.96E-05
Ce-143	0.00E+00	1.81E-05	2.88E-05	4.36E-05	4.43E-05	4.43E-05	4.43E-05
Ce-144	0.00E+00	1.64E-05	2.63E-05	4.06E-05	4.14E-05	4.14E-05	4.14E-05
Pr-143	0.00E+00	7.13E-06	1.15E-05	1.78E-05	1.81E-05	1.81E-05	1.81E-05
Nd-147	0.00E+00	3.11E-06	4.99E-06	7.69E-06	7.82E-06	7.82E-06	7.82E-06
Np-239	0.00E+00	2.24E-04	3.58E-04	5.47E-04	5.56E-04	5.56E-04	5.56E-04
Pu-241	0.00E+00	1.93E-06	3.10E-06	4.79E-06	4.88E-06	4.88E-06	4.88E-06
Kr-83m	2.51E-01	3.39E+02	4.81E+02	5.82E+02	5.83E+02	5.83E+02	5.83E+02
Xe-133m	1.35E-01	2.11E+02	3.37E+02	5.14E+02	5.23E+02	5.23E+02	5.23E+02
Xe-135m	9.38E-01	6.09E+02	6.40E+02	6.48E+02	6.51E+02	6.63E+02	7.68E+02

Table 15.4-6

CONTROL ROD DROP ACCIDENT
ACTIVITY AIRBORNE IN CONDENSER (curies)
(Realistic Analysis)
(2000 Rods)

Condenser Airborne Activity As a
Function of Time Post-Accident
(curies)

ISOTOPE	2 Hr	8 Hr	24 Hr	96 Hr	720 Hr
Kr-85	1.25E+03	1.25E+03	1.24E+03	1.22E+03	1.07E+03
Kr-85m	1.53E+04	6.04E+03	5.06E+02	7.25E-03	7.49E-45
Kr-87	1.33E+04	5.06E+02	8.23E-02	7.32E-19	0.00E+00
Kr-88	3.28E+04	7.58E+03	1.52E+02	3.50E-06	2.22E-72
Rb-86	2.57E-02	2.54E-02	2.47E-02	2.18E-02	7.28E-03
I-131	2.00E+00	1.95E+00	1.84E+00	1.40E+00	1.30E-01
I-132	1.61E+00	2.63E-01	2.11E-03	7.83E-13	1.47E-94
I-133	3.84E+00	3.14E+00	1.84E+00	1.64E-01	1.34E-10
I-134	9.46E-01	8.23E-03	2.63E-08	4.90E-33	0.00E+00
I-135	3.18E+00	1.69E+00	3.15E-01	1.63E-04	5.47E-33
Xe-133	1.60E+05	1.55E+05	1.42E+05	9.46E+04	2.71E+03
Xe-135	3.83E+04	2.42E+04	7.13E+03	2.90E+01	5.50E-20
Cs-134	2.46E+00	2.46E+00	2.45E+00	2.40E+00	2.06E+00
Cs-136	6.20E-01	6.12E-01	5.88E-01	4.95E-01	1.10E-01
Cs-137	1.89E+00	1.88E+00	1.88E+00	1.85E+00	1.62E+00
Kr-83m	4.58E+03	5.12E+02	1.49E+00	5.74E-12	0.00E+00
Xe-133m	4.98E+03	4.61E+03	3.76E+03	1.50E+03	5.21E-01
Xe-135m	1.72E+02	2.70E-01	5.03E-02	2.61E-05	8.74E-34

Table 15.4-7

CONTROL ROD DROP ACCIDENT
ACTIVITY AIRBORNE IN CONDENSER (curies)
(Realistic Analysis)
(35 Rods)

ISOTOPE	Condenser Airborne Activity As a Function of Time Post-Accident (curies)				
	2 Hr	8 Hr	24 Hr	96 Hr	720 Hr
Kr-85	7.95E+00	3.84E-01	1.19E-04	1.19E-04	1.18E-04
Kr-85m	9.76E+01	1.86E+00	4.86E-05	7.05E-10	8.30E-52
Kr-87	8.51E+01	1.56E-01	7.89E-09	7.13E-26	0.00E+00
Kr-88	2.09E+02	2.34E+00	1.46E-05	3.41E-13	2.46E-79
Rb-86	1.64E-04	7.84E-06	2.37E-09	2.12E-09	8.07E-10
I-131	1.27E-02	6.03E-04	1.76E-07	1.36E-07	1.45E-08
I-132	1.02E-02	8.11E-05	2.02E-10	7.62E-20	0.00E+00
I-133	2.45E-02	9.69E-04	1.76E-07	1.60E-08	1.49E-17
I-134	6.04E-03	2.54E-06	2.52E-15	4.76E-40	0.00E+00
I-135	2.03E-02	5.22E-04	3.02E-08	1.59E-11	6.07E-40
Xe-133	1.02E+03	4.77E+01	1.36E-02	9.20E-03	3.00E-04
Xe-135	2.44E+02	7.47E+00	6.83E-04	2.82E-06	6.10E-27
Cs-134	1.57E-02	7.58E-04	2.34E-07	2.34E-07	2.28E-07
Cs-136	3.96E-03	1.89E-04	5.64E-08	4.81E-08	1.22E-08
Cs-137	1.20E-02	5.81E-04	1.80E-07	1.80E-07	1.80E-07
Kr-83m	2.92E+01	1.58E-01	1.43E-07	5.59E-19	0.00E+00
Xe-133m	3.18E+01	1.42E+00	3.61E-04	1.46E-04	5.77E-08
Xe-135m	1.10E+00	8.33E-05	4.82E-09	2.54E-12	9.69E-41

Table 15.4-8

CONTROL ROD DROP ACCIDENT
ACTIVITY RELEASED TO ENVIRONS (curies)
(Realistic Analysis)
(2000 Rods)

ISOTOPE	Activity Released to Environs As a Function of Time Post-Accident (curies)				
	2 Hr	8 Hr	24 Hr	96 Hr	720 Hr
Kr-85	1.26E-01	1.27E+00	5.34E+00	2.38E+01	1.73E+02
Kr-85m	1.73E+00	1.05E+01	1.78E+01	1.85E+01	1.85E+01
Kr-87	2.07E+00	5.17E+00	5.36E+00	5.36E+00	5.36E+00
Kr-88	3.98E+00	1.88E+01	2.50E+01	2.51E+01	2.51E+01
Rb-86	2.60E-06	2.61E-05	1.08E-04	4.57E-04	2.18E-03
I-131	2.02E-04	2.02E-03	8.21E-03	3.24E-02	1.02E-01
I-132	2.04E-04	8.30E-04	1.01E-03	1.01E-03	1.01E-03
I-133	3.97E-04	3.57E-03	1.15E-02	2.20E-02	2.30E-02
I-134	1.83E-04	3.31E-04	3.33E-04	3.33E-04	3.33E-04
I-135	3.47E-04	2.45E-03	5.13E-03	5.77E-03	5.77E-03
Xe-133	1.62E+01	1.61E+02	6.45E+02	2.40E+03	5.78E+03
Xe-135	4.08E+00	3.18E+01	7.75E+01	9.71E+01	9.72E+01
Cs-134	2.48E-04	2.51E-03	1.05E-02	4.69E-02	3.37E-01
Cs-136	6.27E-05	6.30E-04	2.59E-03	1.07E-02	4.40E-02
Cs-137	1.90E-04	1.93E-03	8.07E-03	3.60E-02	2.61E-01
Kr-83m	6.11E-01	2.15E+00	2.43E+00	2.43E+00	2.43E+00
Xe-133m	5.08E-01	4.91E+00	1.86E+01	5.56E+01	8.01E+01
Xe-135m	2.85E-01	2.93E-01	2.93E-01	2.94E-01	2.96E-01

Table 15.4-9

CONTROL ROD DROP ACCIDENT
ACTIVITY RELEASED TO ENVIRONS (curies)
(Realistic Analysis)
(35 Rods)

ISOTOPE	Activity Released to Environs As a Function of Time Post-Accident (curies)				
	2 Hr	8 Hr	24 Hr	96 Hr	720 Hr
Kr-85	3.86E+00	1.73E+01	2.18E+01	2.18E+01	2.18E+01
Kr-85m	5.37E+01	1.69E+02	1.83E+02	1.83E+02	1.83E+02
Kr-87	6.57E+01	1.17E+02	1.18E+02	1.18E+02	1.18E+02
Kr-88	1.24E+02	3.31E+02	3.46E+02	3.46E+02	3.46E+02
Rb-86	7.97E-05	3.56E-04	4.47E-04	4.47E-04	4.47E-04
I-131	6.21E-03	2.76E-02	3.45E-02	3.46E-02	3.46E-02
I-132	6.37E-03	1.55E-02	1.59E-02	1.59E-02	1.59E-02
I-133	1.22E-02	5.05E-02	6.05E-02	6.06E-02	6.06E-02
I-134	5.89E-03	8.62E-03	8.63E-03	8.63E-03	8.63E-03
I-135	1.07E-02	3.75E-02	4.19E-02	4.19E-02	4.19E-02
Xe-133	4.98E+02	2.20E+03	2.75E+03	2.75E+03	2.75E+03
Xe-135	1.26E+02	4.70E+02	5.39E+02	5.39E+02	5.39E+02
Cs-134	7.62E-03	3.42E-02	4.30E-02	4.30E-02	4.30E-02
Cs-136	1.93E-03	8.60E-03	1.08E-02	1.08E-02	1.08E-02
Cs-137	5.84E-03	2.62E-02	3.30E-02	3.30E-02	3.30E-02
Kr-83m	1.92E+01	4.25E+01	4.33E+01	4.33E+01	4.33E+01
Xe-133m	1.56E+01	6.78E+01	8.36E+01	8.37E+01	8.37E+01
Xe-135m	1.03E+01	1.04E+01	1.04E+01	1.04E+01	1.05E+01

Table 15.4-10

CONTROL ROD DROP ACCIDENT RADIOLOGICAL EFFECTS
Design Basis Case

	2000 Failed Rods with Condenser Leakage REM TEDE	35 Failed Rods with MVP Running REM TEDE
Acceptance Criterion - Offsite	6.3	6.3
EAB	0.16	2.2
LPZ	0.04	0.17
Acceptance Criterion - CRHE	5.0	5.0
CRHE	0.38	1.5

CONTROL ROD DROP ACCIDENT RADIOLOGICAL EFFECTS
Realistic Case

Acceptance Criterion - Offsite	6.3	6.3
EAB	0.00043	0.010
LPZ	0.00014	0.00093
Acceptance Criterion - CRHE	5.0	5.0
CRHE	0.00090	0.012

TABLE 15.4-11

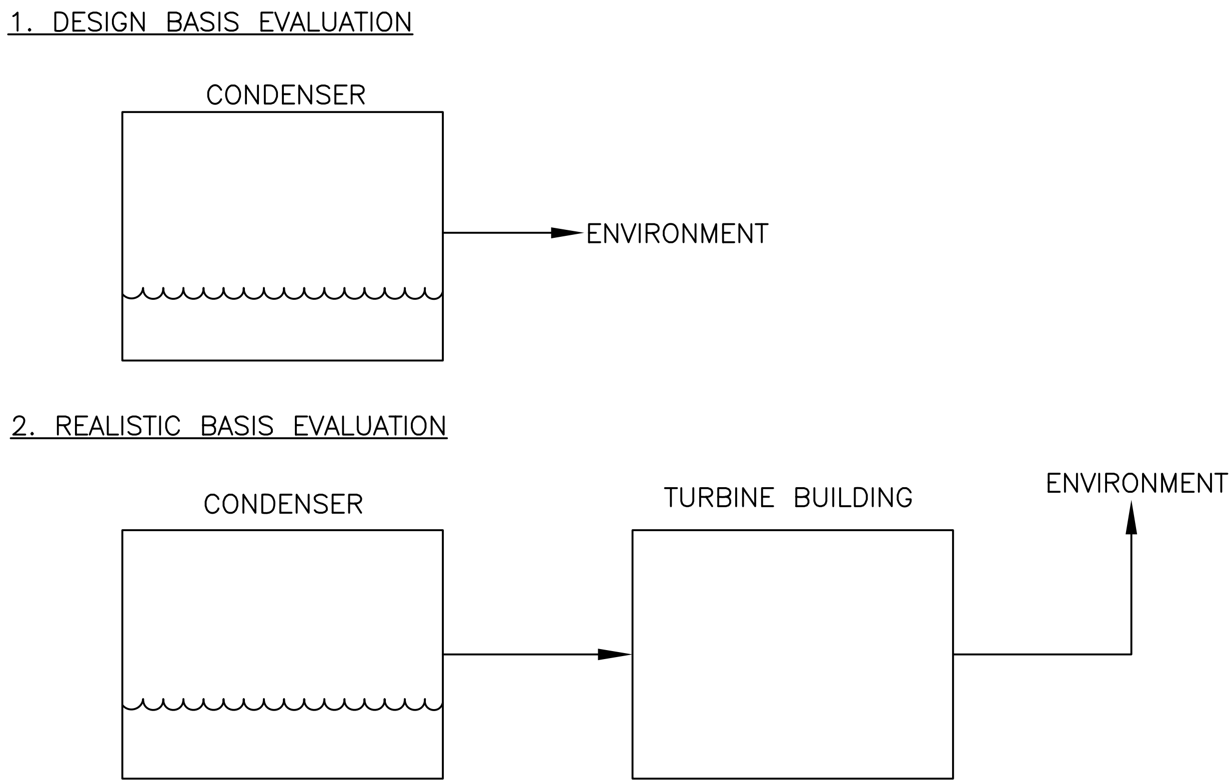
CONTROL ROD DROP ACCIDENT – PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Assumptions
I. Data And Assumptions Used To Estimate Radioactive Source Term From Postulated Accidents		
A. Reactor power level(MWt)	4032	4032
B. Number of fuel bundles in core	764	764
C. Number of ATRIUM-11 rods per fuel bundle	101.2	101.2
D. Number of fuel rods damaged by CRDA (rods)	2000	2000
E. Fuel melting in damaged rod (percent)	0.77	0.0
F. Core radial peaking factor	1.6	1.6
G. Gap activity release to reactor coolant from damaged rods	10 percent noble gases and iodines 12 percent Cs and Rb	1.8 percent noble gases and 0.32 percent iodines 1.8 percent Cs and Rb
H. Fuel activity release to reactor coolant from melted regions	100 percent noble gases 50 percent iodines 25 percent alkalis 5 percent tellurium 2 percent barium and strontium 0.25 percent noble metals 0.02 percent lanthanides 0.05 percent ceriums	NA
II. Data And Assumptions Used To Estimate Activity Released		
A. Activity in Reactor Coolant Transported to Condenser	100 percent noble gases 10 percent iodine 1 percent others	100 percent noble gases 2 percent iodine 2 percent Cs and Rb
B. Activity in the Condenser Available for Release to Environment	100 percent noble gases 10 percent iodine 1 percent others	100 percent noble gases 0.7 percent iodine 0.7 percent others
C. Leak Rate from Condenser to Environment (percent/day)	1	0.5
D. Removal Rate from Condenser to Environment with MVP running (percent/day)	1212	1212
E. Number of Rods Damaged by CRDA Needed to Cause MVP Trip and Isolation	30	30
F. Radioiodine Species Released from Condenser	97 percent Elemental 3 percent Organic	97 percent Elemental 3 percent Organic
G. Turbine Building Release Rate	No holdup credited	700 %/day
H. Accident Duration (hr)	24	24

TABLE 15.4-11

CONTROL ROD DROP ACCIDENT – PARAMETERS TO BE TABULATED
FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Assumptions
III. Data And Assumptions Used To Evaluate Control Room Doses		
A. Control structure habitability envelope free volume(ft^3)	518,000	518,000
B. Control room free volume(ft^3)	110,000	110,000
C. Control structure air intake flow(cfm)	6391	6391
D. Control structure unfiltered outside air infiltration rate – ingress/egress (cfm)	10	10
E. Control structure unidentified unfiltered outside air infiltration rate (cfm)	500	500
F. Control structure filter efficiency (percent)	0	0
IV. Dispersion Data		
A. Site Boundary/Low Population Zone distance(meters)	549/4827	549/4827
B. X/Q's for Site Boundary	Table 2.3-92 (0.5 percentile)	Table 2.3-92 (50 percentile)
C. X/Q's for LPZ	Table 2.3-105 (0.5 percentile)	Table 2.3-105(50 percentile)
D. X/Q's for CRHE	Appendix 15B	Appendix 15B
V. Dose Data		
A. Method of calculation	Appendix 15B	Appendix 15B
B. Isotopic data and dose conversion factors	Appendix 15B	Appendix 15B
C. Activity in condenser	Tables 15.4-2 & 15.4-3	Tables 15.4-6 & 15.4-7
D. Activity released to environment	Tables 15.4-4 & 15.4-5	Tables 15.4-8 & 15.4-9
E. Offsite and control room doses	Table 15.4-10	Table 15.4-10



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LEAK PATH FOR CONTROL ROD
DROP ACCIDENT

FIGURE 15.4-1, Rev 55

AutoCAD: Figure Fsar 15_4_1.dwg