15.6 DECREASE IN REACTOR COOLANT INVENTORY

The analyses described in this subsection of Chapter 15 are bounding analyses and are applicable to both Unit 1 and Unit 2. The tables and figures for this subsection are immediately following the text of the subsection and are not in Appendices 15C, 15D, or 15E.

For the Instrument Line Break and the Steamline Break outside Containment, there is no fuel damage. Design basis radiological release and dose consequences are based on Technical Specification limits of iodine concentration in the reactor water. Therefore, the resulting consequences are independent of the design of the fuel assemblies that comprise the core.

For the Loss of Coolant Accidents inside of Containment a conservative radiological design basis analysis is performed in accordance with NRC guidelines. This analysis bounds the current core designs for Units 1 and 2. A second radiological analysis that is based on realistic assumptions is also performed. The realistic LOCA analysis shows that no fuel failures occur and the radiological release is dependent on the various activation and corrosion products contained in the reactor coolant. As was the case for the Steamline Break, the concentration of these radionuclides in the reactor coolant is limited by the Technical Specifications and the resulting dose consequences are independent of the core design.

15.6.1 INADVERTENT SAFETY RELIEF VALVE OPENING

This event is discussed and analyzed in Subsection 15.1.4.

15.6.2 INSTRUMENT LINE BREAK

This accident is less severe than the event analyzed in Subsection 15.6.5. Quantitative results for the spectrum of LOCA events which bound this event may be found in Section 6.3.

15.6.2.1 Identification of Causes and Frequency Classification

15.6.2.1.1 Identification of Causes

There is no specific event or circumstance identified which results in the failure of an instrument line. However, for the purpose of evaluating the consequences of a small line rupture, the failure of an instrument line is assumed to occur.

15.6.2.1.1.1 Event Description

A circumferential rupture of an instrument line which is connected to the primary coolant system is postulated to occur outside the primary containment but inside the secondary containment. This failure results in the release of primary system coolant to the secondary containment, until the reactor is depressurized. This event could be postulated to occur in the drywell; however, the effects would not be as significant as those from a failure in the secondary containment.

15.6.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.2.2 Sequence of Events and Systems Operation

15.6.2.2.1 Sequence of Events

The sequence of events for this accident is shown in Table 15.6-1.

15.6.2.2.1.1 Identification of Operator Actions

This instrument line should be automatically isolated by the excess flow check valves. The operator shall, if necessary, attempt to isolate the affected instrument line. Note, failure of certain instrument lines may result in a full scram. The following assumes there is no automatic scram. Depending on which line is broken, the operator shall determine whether to continue plant operation until a scheduled shutdown can be made or to proceed with an immediate, orderly plant shutdown, and may initiate SGTS or other ventilation effluent treatment systems when directed by the appropriate Emergency Operating Procedure.

Operator action can be initiated by any one or any combination of the following:

(1) Operator comparing radiation, temperature, humidity, fluid and noise readings with several instruments monitoring the same process variable such as reactor level, jet pump flow, steam flow, and steam pressure.

(2) By annunciation of the control function, either high or low in the main control room.

(3) By a half-channel scram if rupture occurred on a reactor protection system instrument line.

(4) By a general increase in the area radiation monitor readings.

(5) By an increase in the ventilation process radiation monitor readings.

(6) By increases in area temperature monitor readings in the containment.

(7) Leak detection system actuations.

Upon receiving one or more of the above signals and having made the decision to shut down the plant, the operator should proceed to shut down the reactor in an orderly manner.

15.6.2.2.2 System Operation

Normal plant instrumentation and controls are assumed to be fully operational during the entire plant transient to ensure positive identification of the break and safe shutdown of the plant. Minimum reactor and plant protection system operations are assumed for the analysis, e.g., minimum ECCS flow, and suppression pool cooling capability. As a consequence of the accident, the reactor is manually scrammed and the reactor vessel cooled and depressurized over approximately a 5-hour period.

15.6.2.2.3 The Effect of Single Failures and Operator Errors

The initiating event is handled by a protection sequence which can accommodate additional SCF or SOE occurrences. See Appendix 15A for further discussion.

15.6.2.3 Core and System Performance

15.6.2.3.1 Qualitative Summary - Results

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Subsections 15.6.4, 15.6.5, and 15.6.6. Consequently, instrument line breaks are considered to be bounded by the steamline break, Subsection 15.6.4. Details of this calculation, including those pertinent to core and system performance are discussed in detail in Subsection 15.6.4.3.

Instrument line breaks result in a slower rate of coolant loss and are bounded by the calculations referenced above. Since the rate of coolant loss is slow, an orderly reactor system depressurization follows reactor scram and the primary system is cooled down and maintained without ECCS actuation. No fuel damage or core uncovery occurs as a result of this accident.

15.6.2.3.2 Quantitative Results

Instrument line breaks, because of their small size, are substantially less limiting from a core and system performance standpoint than the steamline break outside containment. Similarly, instrument line breaks are considered within the spectrum considered in ECCS performance calculations discussed in detail in Subsection 6.3.3

Therefore, all information concerning ECCS models employed, input parameters, and detailed results for a more limiting (steamline break) event may be found in Reference 15.6-13.

15.6.2.3.3 Considerations of Uncertainties

The approach toward conservatively analyzing this event is discussed in detail for a more limiting case (steamline break) in Reference 15.6-13.

15.6.2.4 Barrier Performance

The release of primary coolant through the orificed instrument line could result in an increase in secondary containment compartment pressure and the potential of isolation of the normal ventilation system.

The following assumptions and conditions are the basis for the mass loss during the 5 hour reactor shutdown period of this event:

(1) Shutdown and depressurization initiated at 10 min. after break occurs and continues for 5 hours.

(2) Normal depressurization and cooldown of reactor pressure vessel.

(3) The break is postulated in a location where, due to a cross-tie between instruments, both liquid and steam is released from the reactor pressure vessel. The primary containment penetrations/instrument lines considered are those carrying reactor coolant and listed in FSAR Table 6.2-12a. Each of these lines is provided with an excess flow check valve and there is a flow-restricting orifice installed upstream of the check valve to limit blow-down in the event of a break outside primary containment coupled with an excess flow check valve failure. The limiting configuration was identified as a line off of the RPV level instrumentation condensing chambers that has both a liquid and steam reactor nozzle feeding it. The RPV nozzles are cross connected by a ½" diameter line. The line configuration allows the steam path to bypass the flow restricting orifice. For this case, a break outside of the primary containment penetration is supplied by both a liquid source which will be restricted by a 3/8" orifice, and by a steam source which will be restricted by the ½" diameter line.

(4) Moody critical blowdown flow model (Reference 15.6-1) is applicable and flow is critical at the orifice.

The total integrated mass of fluid released into the secondary containment via the break during the blowdown is 73,713 pounds. Of this total, 23,706 pounds is steam.

Release of this mass coolant results in a secondary containment pressure which is well below the design pressure.

15.6.2.5 Radiological Consequences

Design Basis analysis shows that the event analyzed in Subsection 15.6.5 is bounding. The following describes a realistic analysis of the event. The dose consequences of the Instrument Line Break are determined using the calculated mass of coolant released over approximately a 5 hour period. The reactor was assumed to be at full power prior to the break. The calculated total mass of coolant released is, 73,713 pounds. Of this 50,007 pounds is liquid and 23,706 pounds is steam. Table 15.6-2 presents the mass released as a function of time.

The reactor water iodine concentration existing at the time of the break was assumed to be equal to 0.2 micro-curies/gram dose equivalent I-131. This is the maximum equilibrium concentration for continued full power operation allowed by the SSES Technical Specifications. All of the iodine activity in the steam from the flashed liquid, steam from the steam dome, and 10 percent from the remaining liquid released from the break is assumed to become airborne inside secondary containment. No credit is taken for holdup in the secondary containment. Although there will be some activation and corrosion products released, the isotopes of primary importance are the iodine isotopes. The iodine isotopes and noble gas activity released from the break to the environment are presented in Table 15.6-3. The dose consequences for a realistic analysis of the instrument line break are presented in Table 15.6-4. Specific values of parameters used in the analysis are presented in Table 15.6-5. The leakage path used in the evaluation is shown in Figure 15.6-1. The radiological consequences are well within 10CRF50.67 dose acceptance criteria.

15.6.3 STEAM GENERATOR TUBE FAILURE

This section is not applicable to the direct cycle BWR.

15.6.4 STEAM SYSTEM PIPING BREAK OUTSIDE CONTAINMENT

This event involves the postulation of a large steamline pipe break outside containment. It is assumed that the largest steamline, instantaneously and circumferentially breaks at a location downstream of the outermost isolation valve. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line, and actuate the necessary protective features. This postulated event represents the envelope evaluation of steamline failures outside containment.

15.6.4.1 Identification of Causes and Frequency Classification

15.6.4.1.1 Identification of Causes

A main steamline break is postulated without the cause being identified. These lines are designed to high quality engineering codes and standards and to restrictive seismic and environmental requirements. However, for the purpose of evaluating the consequences of a postulated large steam line rupture, the failure of a main steamline is assumed to occur.

15.6.4.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.4.2 Sequence of Events and Systems Operation

15.6.4.2.1 Sequence of Events

Accidents that result in the release of radioactive materials directly outside the containment are the results of postulated breaches in the reactor coolant pressure boundary or the steam power conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the four main steam lines. The sequence of events is given in Table 15.6-6.

15.6.4.2.1.1 Identification of Operator Actions

Normally, the reactor operator will maintain reactor vessel water inventory and core cooling with the HPCI and/or RCIC system. Without operator action, HPCI and RCIC would initiate automatically on low water level (L2) following isolation of the main steam supply system (i.e., MSIV closure). The core would be covered throughout the accident and there would be no fuel damage. Without taking credit for the RCIC water makeup capability and assuming HPCI failure, the operator must initiate the ADS or manual relief valve system to ensure termination of the accident without fuel damage.

15.6.4.2.2 Systems Operation

A postulated guillotine break of one of the four main steamlines outside the containment results in mass loss from both ends of the break. The flow from the upstream side is initially limited by the flow restrictor upstream of the inboard isolation valve. Flow from the downstream side is initially limited by the total area of the flow restrictors in the three unbroken lines. Subsequent closure of the MSIVs further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when the full closure is reached.

A discussion of plant and reactor protection system action and ESF action is given in Sections 6.3, 7.3, and 7.6.

15.6.4.2.3 The Effect of Single Failures and Operator Errors

The effect of single failures has been considered in analyzing this event. The ECCS aspects are covered in Section 6.3. The break detection and isolation considerations are defined in Sections 7.3 and 7.6. All of the protective sequences for this event are capable of SCF and SOE accommodation and completion of the necessary safety action. Refer to Appendix 15A for further details.

15.6.4.3 Core and System Performance

Quantitative results (including mathematical models, input parameters, and consideration of uncertainties) for this event are given in References 15.6-11 and 15.6-12. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause fuel damage.

15.6.4.3.1 Input Parameters and Initial Conditions

Refer to References 15.6-11 and 15.6-12 for initial conditions.

15.6.4.3.2 Results

There is no fuel damage as a consequence of this accident.

Refer to References 15.6-11 and 15.6-12 for the results of this analysis.

15.6.4.3.3 Considerations of Uncertainties

Sections 6.3 and 7.3 contain discussions of the uncertainties associated with the ECCS performance and the containment isolation systems, respectively.

15.6.4.4 Barrier Performance

Since this break occurs outside the containment, barrier performance within the containment envelope is not applicable. Details of the barrier performance outside primary containment can be found in Subsection 6.2.3.

The following assumptions and conditions are used in determining the mass loss from the primary system from the inception of the break to full closure of the MSIVs:

(1) The reactor is operating at the power level associated with maximum mass release.

(2) Nuclear system pressure is 1050 psia and remains constant during closure.

(3) An instantaneous circumferential break of the main steamline occurs.

(4) Isolation valves start to close at 0.5 sec on high flow signal and are fully closed at 5.5 sec.

(5) The Moody critical flow model (Reference 15.6-1) is applicable.

Initially only steam will issue from the broken end of the steamline. The flow in each line is limited by critical flow at the limiter to a maximum of 200% of rated flow for each line. Rapid depressurization of the RPV causes the water level to rise resulting in a steam-water mixture flowing from the break until the valves are closed.

15.6.4.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(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 10 CFR Part 50.67 guidelines. This analysis is referred to as the "design basis analysis."

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

A schematic of the release path is shown in Figure 15.6-2.

15.6.4.5.1 Design Basis Analysis

The Design Basis Analysis is based on NRC Regulatory Guide 1.183. The dose consequences of the Main Steamline Break are determined using the calculated mass of coolant released in the time required for the MSIVs to fully close after the break occurs. The reactor was assumed to be at a hot standby condition prior to the break in order to maximize the calculated liquid release. The calculated total mass of coolant released is 97,970 pounds. Of this, 84,840 pounds is liquid and 13,130 pounds is steam. These values were increased by 20% for dose analyses. There is no fuel damage as a result of this accident.

Consistent with Regulatory Guide 1.183, two cases were analyzed. In the first case, the reactor water iodine concentration existing at the time of the break was assumed to be equal to 0.2 micro-curies/gram dose equivalent I-131. This is the maximum equilibrium concentration for continued full power operation allowed by the SSES Technical Specifications. In the second case, the iodine concentration prior to the break was assumed to be 4.0 micro-curies/gram dose equivalent I-131, which is the maximum short-term concentration permitted by the Technical Specifications. This concentration corresponds to an assumed pre-existing iodine spike. In each case, all iodine activity in the coolant which is released from the break is assumed to become airborne. Although there will be some activitation and corrosion products released, the isotopes of primary importance are the iodine isotopes. The iodine isotopes and noble gas activity released from the break and to the environment prior to isolation valve closure are presented in Table 15.6-7.

The RADTRAD Computer Program (Reference 15.6-9) is used to evaluate the radiological consequences for the design basis analysis. For the model, the mass releases are increased by a 20% margin to add conservatism.

The activity release is modeled as an instantaneous puff release per Regulatory Guide 1.194 (Reference 15.6-6). This is justified since all of the activity's is assumed to be released in the

5.5 seconds it takes for the MSIVs to close. The activity is assumed to be released as a ground level release with no holdup in the turbine building.

The specific models, assumptions and parameters used in the analyses are presented in Table 15.6-10.

15.6.4.5.2 Realistic Analysis

For the realistic analysis, it is assumed that the reactor is operating at full power prior to the break. The dose consequences are determined based on the calculated mass of coolant released in the time required for the MSIVs to fully close after the break occurs. The total integrated mass of coolant leaving the break is 40,316 pounds. (This value is increased by 20% for dose analyses). Of this quantity, 19,994 pounds is liquid and 20,322 pounds is steam. Of the 19,994 pounds of liquid released, 8,000 pounds is flashed to steam.

The activity released from the hypothetical steamline break accident is a function of the coolant activity, valve closure time, and mass of coolant released. A portion of the released coolant exists as steam prior to the blowdown, and as such does not contain the same concentration per unit of mass as does the steam generated as a consequence of the blowdown. Therefore, it is necessary to subtract the initial steam mass from the total mass released and assign to it only 8 percent of the iodine activity contained by an equivalent mass of primary coolant. The isotopic activity released to the environment is shown in Table 15.6-8.

The radiological dose consequences are calculated using the RADTRAD computer code. For the model, the mass releases are increased by a 20% margin to add conservatism,

The activity release is modeled as an instantaneous puff release based on guidance from Regulatory Guide 1.194. This is reasonable since the activity is released over a very short period of time. The release is assumed to be a ground level release with no holdup in the turbine building.

The specific models, assumptions, and parameters used in the analyses are presented in Table 15.6-10.

15.6.4.5.3 Results

OFFSITE

The calculated exposures at the site boundary and low population zone for the design basis and realistic analyses are presented in Table 15.6-9. The dose consequences are well within 10CFR 50.67 guidelines.

CONTROL ROOM

A detailed description of the control room model can be found in Appendix 15B. The radiological exposure to the control room personnel for the design basis case is given in Table 15.6-9. The dose consequences satisfy the 10CFR50.67 acceptance criterion.

15.6.5 LOSS-OF-COOLANT ACCIDENTS (RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY) - INSIDE CONTAINMENT

This event involves the postulation of a spectrum of piping breaks inside containment varying in size, type, and location. The break type includes steam and/or liquid process system lines. This event is also coupled with severe natural environmental conditions including earthquake coincidence.

The event has been analyzed quantitatively in Sections 6.2, 6.3, 7.1 and 8.3. Therefore, the following discussion provides only new information not presented in the subject sections. All other information is covered by cross-referencing.

The postulated event represents the envelope evaluation for liquid or steam line failures inside containment.

15.6.5.1 Identification of Causes and Frequency Classification

15.6.5.1.1 Identification of Causes

There are no realistic, identifiable events which would result in a pipe break inside the containment of the magnitude required to cause a loss-of-coolant accident coincident with safe shutdown earthquake plus SACF criteria requirements. The subject piping is designed to strict emergency code and standard criteria, and for severe seismic and environmental conditions. However, since such an accident provides an upper limit estimate to the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.

15.6.5.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.5.2 Sequence of Events and Systems Operation

15.6.5.2.1 Sequence of Events

Representative sequences of events associated with this accident are shown in Table 6.3-1B-2 for Unit 1 and Unit 2 for ATRIUM-10 and ATRIUM-11 fuel for core system performance.

Following the pipe break and scram, the MSIV will begin closing due to the loss of offsite power. The low-low water level or high drywell pressure signal will initiate HPCI, CS and LPCI systems.

15.6.5.2.1.1 Identification of Operator Actions

Since automatic actuation and operation of the ECCS is a system design basis, no operator actions are required for the accident. However, by procedural requirement, the operator will perform the following described actions.

The operator will, after checking that all rods are inserted at time 0 plus approximately 10 seconds, determine plant condition by observing the annunciators. After observing that the Emergency Core Cooling Systems are initiated on low water level or high drywell pressure and low RPV pressure, the operator will check that the diesel generators have started and are in standby condition. The

operator will also ensure primary containment isolations and ensure that reactor water level is properly maintained. Within approximately 20 minutes, the operator aligns the RHR heat exchangers for long term containment cooling. After the RHR system and other auxiliary systems are in proper operation, the operator will monitor suppression pool temperature, drywell temperature and pressure, and the hydrogen concentration in the drywell for proper activation of the containment air purge system if necessary.

15.6.5.2.2 Systems Operations

Accidents that could result in the release of radioactive fission products directly into the containment are the results of postulated nuclear system primary coolant pressure boundary pipe breaks. Possibilities for all pipe break sizes and locations are examined in Sections 6.2 and 6.3, including the severance of small process system lines, the main steamlines upstream of the flow restrictors, and the recirculation loop pipelines. The greatest release of radioactive material to the containment result from a complete circumferential break of one of the two recirculation loop pipelines. The minimum required functions of any Reactor and Plant Protection System are discussed in Sections 6.2, 6.3, 7.3, 7.6, and 8.3, and Appendix 15A.

15.6.5.2.3 The Effect of Single Failures and Operator Errors

Single failures and operator errors have been considered in the analysis of the entire spectrum of primary system breaks. The consequences of a LOCA with considerations for SCF and SOE occurrence are shown to be fully accommodated without the loss of any required safety function. See Appendix 15A for further details.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The analytical methods and associated assumptions which are used in evaluating the consequences of this accident are considered to provide a conservative assessment of the expected consequences of this very improbable event.

The details of these calculations, their justification, and bases for the models are developed in Sections 6.2, 6.3, 7.3, 7.6, 8.3 and Appendix 15A.

15.6.5.3.2 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are given in Table 6.3-1B for Unit 1 and for Unit 2.

15.6.5.3.3 Results

Results of this event are given in detail in Sections 6.2 and 6.3.

The conservative, design basis analyses of the ATRIUM-10 and ATRIUM-11 fuel LOCA, are described in Section 6.3.3.

15.6.5.3.4 Consideration of Uncertainties

This event was conservatively analyzed; see Sections 6.2, 6.3, 7.3, 7.6, 8.3 and Appendix 15A for details.

15.6.5.4 Barrier Performance

The design basis for the containment is to maintain its integrity and experience normal stresses after the instantaneous rupture of the largest single primary system piping within the structure while also accommodating the dynamic effects of the pipe break at the same time an SSE is also occurring. Therefore, any postulated loss-of-coolant accident does not result in exceeding the containment design limit. For details and results of the analyses, see Sections 3.8, 3.9, and 6.2. This conclusion is valid for both Units.

15.6.5.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(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 10 CFR Part 50.67 guidelines. This analysis is referred to as the "design basis analysis."

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

15.6.5.5.1 Design Basis Analysis

The methods, assumptions, and conditions used to evaluate this accident are in accordance with the requirements of Regulatory Guide 1.183. The RADTRAD computer code (Reference 15.6-9) is used to evaluate offsite and unprotected control room doses for this event. Specific values of parameters used in this evaluation are presented in Table 15.6-22.

15.6.5.5.1.1 Fission Product Release from Fuel

The assumptions related to the release of radioactive material from the fuel and containment as stated in Regulatory Guide (1.183) were used in this analysis.

The core inventory release fractions airborne into the primary containment, by radionuclide groups, for the gap release and early in-vessel damage phases are listed as follows.

Group	Gap Release Phase	Early In-vessel Phase	Total
Noble Gases	0.05	0.95	1
Halogens	0.05	0.25	0.3
Alkali Metals	0.05	0.2	0.25
Tellurium Metals	0	0.05	0.05
Ba, Sr	0	0.02	0.02
Noble Metals	0	0.0025	0.0025
Cerium Group	0	0.0005	0.0005
Lanthanides	0	0.0002	0.0002

The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase is modeled as increasing in a linear fashion over the duration of the phase. The values used in this analysis for the release to the primary containment are provided as follows.

Phase	Onset	Duration
Gap Release	2 min	0.5 hr
Early In-Vessel Release	0.5 hr	1.5 hr

The primary containment free volume consists of the drywell free volume and the wetwell free volume.

Drywell free volume	= 239600 ft^3
Wetwell free volume	= 148590 ft^3
Total Primary containment free volume	= 388190 ft^3

For the first two hours following the event, the airborne activity released to the primary containment is assumed to only be mixed in the drywell free volume of 239600 ft^3.

The core inventory release fractions and release timing for the activity released directly into the suppression pool volume are the same as above except for noble gases which are assumed to only be present in the primary containment air. The volume of the suppression pool is 132,000 ft^3.

The activity airborne in the primary containment for the design basis case is presented in Table 15.6-11.

15.6.5.5.1.2 Fission Product Transport to the Environment

The transport pathway to the environment is by several different mechanisms discussed below.

a) Containment leakage: Leakage from the primary containment shell and its penetrations (excluding the main steam lines) is mixed with the air in the secondary containment, and discharged from there to the environment via the SGTS. Also, a small fraction of primary containment leakage can bypass secondary containment and be discharged to the environment without being processed by SGTS.

   Per Technical Specifications, the leakage rate for the primary containment is defined as 1% by weight of containment air per 24 hours. In accordance with Regulatory Guide 1.183, the primary containment is assumed to leak at this rate for the first 24 hours and reduced after the first 24 hours to 50% of the leak rate based on the significant reduction of the calculated internal pressure of primary containment at 24 hours. Of the 1% primary containment leakage, 15 scfh (0.0223% per 24 hours) is assumed to bypass the secondary containment and be released directly to the environment. Similar to the above the bypass leakage is reduced by 50% after 24 hours. The fraction of the primary containment leakrate which enters the secondary containment and is processed by the SCGT is the difference between the total leakrate and the secondary containment bypass leakrate or 0.9777% for the first 24 hours and 0.4889%/day thereafter.

b) Water leakage to secondary containment from Engineered Safety Feature (ESF) components outside the primary containment, such as the Emergency Core Cooling System (ECCS):

It is expected that during the postulated post-accident operation of the emergency systems, the total liquid leakage to the reactor building from pumps, seals, and valves will be small, i.e., on the order of gallons per hour. This leakage will be minimized by normal system tests and maintenance operations. Those ESF systems which contribute to this leakage are identified in Section 18.1.69, and the Leakage Rate Test program. The total leakage from these systems is maintained $\leq$2.5 gpm, in accordance with Technical Specification 5.5.2, and Section 18.1.69. Therefore, in order to conservatively bound the total post-accident liquid leakage for dose analysis purposes, it is assumed that 5 gpm of liquid leakage from all potential sources contribute to the radioactive releases to the reactor building. The leakage is assumed to begin at time equal 0 and continue for the 30 day duration of the LOCA. In accordance with Regulatory Guide 1.183 with the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase. An analysis of the response of the suppression pool under post LOCA-accident conditions determined that the maximum bulk suppression pool water temperature does not exceed 212 °F. Therefore, per Regulatory Guide 1.183, "If the temperature of the leakage is less than 212 °F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates." A flash fraction of 10% is conservatively used herein. The iodine species assumed available for release to the environment from this leakage are to be 97% elemental and 3% organic. Secondary containment airborne activity is processed by the SGTS prior to release to the environment.

c) Leakage from the Main Steam Isolation Valves (MSIV): The Isolated Condenser Treatment Method (ICTM) routes leakage past MSIVs to the main condenser utilizing the main steam drain lines as a pathway. In the condenser, volumetric dilution and plate-out hold up fission products until eventual release to the environment through the low pressure turbine seals.

The activity available for release from this leakage path is conservatively taken to be instantaneously released from the core and mixed in the drywell free volume (239,600 ft^3) for the first two hours. After two hours the activity is assumed to be further diluted in the drywell plus wetwell free volume (388,190 ft^3).

The MSIV leakage test limits provided in the plant Technical Specification Surveillance Requirements are ≤ 100 scfh from any one valve or ≤ 300 scfh total from the four valves. This analysis assumes one main steam line is faulted and has the 100 scfh flow. The remaining leakage is evenly split between the three non-faulted lines. The leakages are reduced by 50% after the first 24 hours based on the reduction in the drywell accident pressure.

Per a Letter from Fermi 2 to USNRC (Reference 15.6-14), the NRC states that an acceptable method for modeling the removal of aerosols, elemental and organic iodine in the main steam line piping is provided in Appendix A to AEB-98-03 (Reference 15.6-10). Credit is taken for aerosol and elemental iodine plateout in the piping based this information. Only the horizontal runs of piping are considered in determining plateout. Additionally, since aerosol plateout is a mechanistic settling process only the bottom one half of the inside surface area of the lines is applicable for plateout. Since the bottom half of a circular pipe has sides which are essentially vertical or inclined, the area for aerosol plateout is modeled as the projected area of the diameter of the pipe. To account for the phenomenon that steam condensation in the piping could potentially wash out and re-evolve some of the settled aerosols an additional factor of 2 reduction in

the conservatively calculated projected area is used. Therefore, the aerosol settling area is defined as one half the projected area of the diameter of the pipe.

The key parameter in the removal equations is the settling velocity of the material of interest. Reference 15.6-10, provides values for aerosol settling velocities. This analysis conservatively uses an aerosol settling velocity equal to ¼ of 10th percentile value from Reference 15.6-10.

The elemental removal is based on the same methodology except that the elemental iodine deposition velocity based on information provided by J. E. Cline, in Reference 15.6-15. Re-suspension of elemental iodine is also included in the analysis.

Figure 15.6-3 shows the various leakage pathways for the LOCA.

The fission product activity in the secondary containment at any time (t) is a function of the leakage rate from the primary containment and the volumetric discharge rate from the secondary containment. Upon receipt of appropriate signals, the reactor building ventilation isolation valves isolate the reactor building atmosphere in 10 seconds. This rapid closure time prevents possible uncontrolled escape of radioactivity. Upon reactor building isolation, the recirculation system is designed to circulate the reactor building air to provide a delay mechanism whereby radioisotopes are retained in the reactor building and undergo radioactive decay rather than direct escape through the SGTS. A further function of the recirculation system is to provide thorough mixing of the recirculated flow to ensure that the SGTS cannot extract an unmixed quantity of radioactivity. Any fission product removal effects in the secondary containment such as plateout, are neglected; however, the effects of decay are considered. A mixing efficiency of 50 percent has conservatively been assumed in the analysis although a higher efficiency is expected.

The performance characteristic of the SGTS will be verified by periodic tests. The system removal efficiency is designed to be in excess of 99 percent removal of all forms of iodine and 0.3 micron or larger particulates. The SGTS has an exhaust flow of 225 percent per day of the secondary containment free air volume.

The activity buildup in the secondary containment and activity release to environment are presented in Tables 15.6-13 and 15.6-14 respectively.

15.6.5.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The RADTRAD computer code (Reference 15.6-9) is used to evaluate the radiological consequences for this event. Specific values of parameters used in the evaluation are presented in Table 15.6-22.

15.6.5.5.2.1 Fission Product Release from Fuel

GE LOCA evaluations have determined that 10x10 fuel is expected to produce lower peak cladding temperature than 9x9 fuel in a given reactor. This is reasonable since the smaller rod diameter and larger surface area decreases stored energy. Like the 10x10 fuel results, the 11x11 fuel has an even smaller rod diameter and larger surface area having less stored energy relative to the 10x10 fuel. Since Susquehanna specific GE analyses of the 9x9-2 fuel (using the best estimate SAFER/GESTR-LOCA methodology) showed no fuel failures, and for EPU conditions the LOCA analysis results are comparable to those determined for recent pre-EPU LOCA analyses, it is

expected that a realistic analysis at EPU conditions would result in no fuel failures (Reference 15.6-13). No ATRIUM-10 or ATRIUM-11 fuel failures would be expected in a realistic analysis.

Thus, for the realistic analysis of dose consequences, no failures are assumed. Since this accident does not result in any fuel damage, the only activity released to the drywell is that activity contained in the reactor coolant plus any additional activity which may be released as a consequence of reactor scram and vessel depressurization.

The design bases coolant iodine concentrations:

I-131	0.013 μCi/gm
I-132	0.12 μCi/gm
I-133	0.089 μ Ci/gm
I-134	0.24 μCi/gm
I-135	0.13 μCi/gm

For iodine, the pre LOCA initial reactor coolant iodine concentrations are based on Improved Technical Specification equilibrium operating I-131 Dose Equivalent (DE) TEDE reactor coolant system specific activity limit of 0.2 μCi/gm. The equivalent reactor coolant iodine activity concentrations are then given as follows:

I-131	0.0479 μCi/gm
I-132	0.442 μCi/gm
I-133	0.328 μCi/gm
I-134	0.884 μCi/gm
I-135	0.479 μCi/gm

As a consequence of reactor scram and depressurization, additional iodine activity is released from those rods which experienced cladding perforation during normal operation. Measurements performed (Reference 15.6-4) at operating BWRs during reactor shutdown have been used to develop an analytical model for the prediction of iodine and noble gas spiking as a consequence of reactor scram and vessel depressurization. The spiking for the non coolant activation and other fission products isotopes was conservatively determined by using an activity spike model which increases the equilibrium activity release rates for the pertinent isotopes from the fuel by a factor of 500.

Considering that approximately 40% of the released liquid flashes to steam, it is conservatively assumed that 50% of the released iodine activity is airborne initially. The total activity airborne in the containment is presented in Table 15.6-15.

15.6.5.5.2.2 Fission Product Transport to the Environment

The leak rate from the primary containment to the secondary containment is 1.0%/day where 50% mixing is assumed to occur. The transport pathways for the released activity to reach the environment are the same as described in section 15.6.5.5.1.2. The activity buildup in the secondary containment is presented in Table 15.6-16. The integrated isotopic activity released to the environment is presented in Table 15.6-17.

15.6.5.5.3 Results

15.6.5.5.3.1 Offsite Exposure

The radiological exposures resulting from the activity released to the environment as a consequence of the LOCA have been determined for the realistic and design basis cases. The design basis doses use the 0.5 percent direction dependent X/Q's and the analytical model as described in Appendix 15B. The realistic doses use the 50 percent direction independent X/Q's and the same dose model. The design basis and realistic LOCA doses are presented in Table 15.6-18.

15.6.5.5.3.2 Control Room Doses

Control room X/Q values were calculated using the ARCON 96 computer code in accordance with Regulatory Guide 1.194 (Reference 15.6-5 and 15.6-6). NRC occupancy factors were assumed. A detailed description of the control room dose model can be found in Appendix 15B. The radiological exposure of the control room personnel for the design and realistic basis case is given in Table 15.6-21.

15.6.6 FEEDWATER LINE BREAK-OUTSIDE CONTAINMENT

In order to evaluate the plant response to large liquid process line pipe break outside containment, the failure of a feedwater line is assumed. The postulated break of the feedwater line, representing the largest liquid line outside the containment, provides the envelope evaluation relative to this type of occurrence. The break is assumed to be instantaneous, circumferential, and in the 30” diameter feedwater header, just downstream of the reactor feedwater pumps.

A more limiting event from a core performance evaluation standpoint (Feedwater Line Break Inside Containment) has been qualitatively analyzed in Section 6.3. Therefore, the following discussion provides only new information not presented in Section 6.3.

It is assumed that the reactor is operating at an initial power level of 4032 MWt for this analysis.

15.6.6.1 Identification of Causes and Frequency Classification

15.6.6.1.1 Identification of Causes

A feedwater line break is assumed without the cause being identified. The subject piping is designed, to strict emergency codes and standards, and to severe seismic environmental requirements.

15.6.6.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.6.2 Sequence of Events and Systems Operation

15.6.6.2.1 Sequence of Events

The sequence of events is shown in Table 15.6-23.

15.6.6.2.1.1 Identification of Operator Actions

Since automatic actuation and operation of the ECCS is a system design basis, no operator actions are required for this accident. However, by procedural requirements the operator will perform the following actions which are shown below for informational purposes:

(1) The operator determines that line break has occurred and evacuates the area of the turbine building.

(2) The operator is not required to take any action to prevent primary reactor system mass loss, but should ensure that the reactor is shut down and that RCIC and/or HPCI are operating normally.

(3) The operator will implement site radiation incident procedures.

(4) If possible, the operator will shutdown the feedwater system and will deenergize any electrical equipment which may be damaged by the feedwater system in the turbine building.

(5) The operator will continue to monitor reactor water level and the performance of the ECCS systems while the radiation incident procedure is being implemented and begins normal reactor cooldown measures.

(6) When the reactor pressure has decreased below 100 psi, the operator will initiate RHR in the shutdown cooling mode to continue cooling down the reactor.

The above operator procedures occur over an elapsed time of 3-4 hours.

15.6.6.2.2 Systems Operations

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the reactor isolation system and ECCS system. The reactor protection system (safety relief valves, ECCS, and control rod drive) and plant protection system (RHR heat exchangers) are assumed to function properly to assure a safe shutdown.

The ESF systems and RCIC/HPCI systems are assumed to operate normally.

15.6.6.2.3 The Effect of Single Failures and Operator Errors

The feedwater line outside the containment is a special case of the general loss-of-coolant accident break spectrum considered in Section 6.3. The general single-failure analysis for loss-of-coolant accidents is discussed in detail in Subsection 6.3.3.3. For the feedwater line break outside the containment which can be isolated, either the RCIC or the HPCI can provide adequate flow to the vessel to maintain core cooling and prevent fuel rod clad failure. A single failure of either the HPCI or the RCIC would still provide sufficient flow to keep the core covered with water. See Appendix 15A for further description of the analysis.

15.6.6.3 Core and System Performance

15.6.6.3.1 Qualitative Summary

The accident evaluation qualitatively considered in this subsection is considered to be a conservative assessment of the consequences of the postulated failure (i.e., severance) of one of the feedwater piping lines external to the containment.

15.6.6.3.2 Qualitative Results

The feedwater line break outside the containment is less limiting than either the steamline breaks outside the containment (analysis presented in References 15.6-11 and 15.6-12 and Subsection 15.6.4) or the feedwater line break inside the containment. It is qualitatively evaluated as less limiting than the design basis accident (the recirculation line break analysis presented in Subsections 6.3.3 and 15.6.5).

The RCIC and HPCI initiate on low low-water level and together restore the reactor water level to the normal elevation. The low-low-low water level for reactor isolation is not expected to be reached. The fuel is covered throughout the transient and there are no pressure or temperature transients sufficient to cause fuel damage.

15.6.6.3.3 Consideration of Uncertainties

Sections 6.3 and 7.3 contain discussions of uncertainties associated with ECCS Performance and Containment Isolation Systems, respectively.

15.6.6.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the results of postulated breaches in the reactor coolant pressure boundary or the steam power-conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steamlines as described in Subsection 15.6.4. The feedwater system piping break is less severe than the main steamline break. Results of the main steamline break analysis can be found in Subsection 15.6.4.3.2.

15.6.6.5 Radiological Consequences

Two separate radiological analyses are provided for this accident.

(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 10 CFR Part 50.67 guidelines. This analysis is referred to as the “design basis analysis.”

(2) The second is based on assumptions considered to provide a realistic conservative estimate of the radiological consequences. This analysis is referred to as the “realistic analysis” and should not exceed a small fraction (I,e., 10 percent) of 10CFR50.57 guidelines.

The dose consequences of the Feedwater Line Break represent an upper bound on feedwater release and provide the maximum steam release for this event. The feedwater line break is

postulated in the 30" diameter feedwater header, just downstream of the reactor feed pumps, to maximize the discharge enthalpy. The reactor scrams on low level immediately following the break. The entire hotwell inventory is pumped through the feedwater heater strings and out of the break.

All of the feedwater released through the break is processed through the condensate demineralizer with an assumed iodine decontamination factor of ten. The activity released from the hypothetical feedwater line break accident is a function of the coolant activity, accident duration, and mass of coolant released. The calculated total mass of coolant released for the realistic and design basis accidents are 2.50 X 10^6 pounds. Of this, 2.24 X 10^6 pounds is liquid and 2.60 X 10^5 pounds is steam.

For all analyses, ten percent of the iodine activity in the coolant and 100 percent of the iodine activity in the flashed steam is assumed to become airborne. Although there will be some activation and corrosion products released, the isotopes of primary importance are the iodine isotopes. The iodine activity released from the break and to the environment are presented in Table 15.6-25.

The specific models, assumptions, and parameters used in the analysis are presented in Table 15.6-24.

A schematic of the release path is shown in Figure 15.6-4.

15.6.6.5.1 Design Basis Analysis

The NRC provides no specific regulatory guidelines for the evaluation of this accident.

Since this event is bounded by the Main Steam Line Break, the guidelines and dose acceptance criteria of Standard Review Plan 15.6.4 are applied to this analysis. Consistent with this guidance two cases are analyzed. In the first case, the reactor water iodine concentration existing at the time of the break is assumed to be equal to 0.2 micro-curies/gram dose equivalent I-131. This is the maximum equilibrium concentration for continued full power operation allowed by the SSES Technical Specifications. In the second case, the iodine concentration prior to the break is assumed to be 4.0 micro-curies/gram dose equivalent I-131, which is the maximum short-term concentration permitted by the Technical Specifications. This concentration corresponds to an assumed pre-existing iodine spike.

15.6.6.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The reactor water iodine concentration existing at the time of the break is assumed to be equal to the design basis reactor coolant values given in Table 11.1-2. The specific models, assumptions and parameters used in the evaluation are presented in Table 15.6-24. A schematic diagram of the leakage path for this accident is shown in Figure 15.6-4.

15.6.6.5.3 Fission Product Release

There is no fuel damage as a consequence of this accident. In addition, an insignificant quantity of activity (compared to that existing in the main condenser hotwell prior to occurrence of the break) is released from the contained piping system prior to isolation closure.

Noble gas activity in the condensate is negligible since the air ejectors remove practically all noble gas from the condenser.

15.6.6.5.4 Fission Product Transport to the Environment

The transport pathway consists of liquid release from the break, carryover to the turbine building atmosphere due to flashing and partitioning and unfiltered release to the environment through the turbine building ventilation system. The release of activity to the environment is presented in Table 15.6-25. The release is assumed to take place within 2 hours of the occurrence of the break.

15.6.6.5.5 Results

Offsite

The calculated exposures at the site boundary and low population zone for the design basis and realistic analyses are presented in Table 15.6-26. For the design basis analysis, the consequences of Case 1 and case 2 are less than 10% of the 10CFR100 limits. The realistic analysis dose consequences are well within 10CFR50.67 guidelines.

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 15.6-4. The radiological exposure to the control room personnel for the design basis case is given in Table 15.6-26. The calculated dose meets the 10CFR50.67 control room dose acceptance criterion.

15.6.7 REFERENCES

15.6-1 F. J. Moody, "Maximum Two-Phase Vessel Blowdown From Pipes," ASME Paper Number 65-WA/HT-1, March 15, 1965.

15.6-2 NEDO-21143-1, "Conservative Radiological Accident Evaluation - The CØNACO3 Code."

15.6-3 Nguyen, D., "Realistic Accident Analysis for General Electric Boiling Water Reactor - The RELAC Code and User's Guide."

15.6-4 Brutschy, F. J., G. R. Hills, N. R. Horton, A. J. Levine, "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup," NEDO-10585, August, 1972.

15.6-5 NUREG/CR-6331, ARCON96, Atmospheric Relative Concentrations in Building Wakes, Revision 1, May, 1997.

15.6-6 USNRC Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, June 2003.

15.6-7 Not Used

15.6-8 Not Used

15.6-9 NUREG/CR-6604, RADTRAD A Simplified Model for RADionuclide Transport and Removal And Dose Estimation, and Supplement 1, 6/8/99.

15.6-10 AEB-98-03, Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NEREG-1465) Source Term, 12/9/98.

15.6-11 NE-092-001A, Rev. 1, "SSES Power Uprate Licensing Topical Report", and NRC letter dated November 30, 1993, from Thomas E. Murley to Robert G. Byram (PP&L), Subject: Licensing Topical Report for Power Uprate with Increased Core Flow, Rev 0, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788) (TAC NOS. M83426 and M83427) with enclosed Safety Evaluation Report.

15.6-12 NEDC-32161P, December 1993, (General Electric Report)," Power Uprate Engineering Report for Susquehanna Steam Electric Station Units 1 and 2.

15.6-13 "Safter/GESTR-LOCA Analysis Basis Documentation for Susquehanna Steam Electric Station Units 1 and 2," NEDC-32281P, September 1993.

15.6-14 Letter from Fermi 2 to USNRC, NRC-03-0095, Response to NRC Request for Additional Information Regarding the Implementation of Alternative Source Term, NRC Docket No. 50-341, 12/12/03.

15.6-15 J.E. Cline, MSIV Leakage Iodine Transport Analysis, Letter Report, 3/26/1991 (ADAMS Accession Number ML003683718).

TABLE 15.6-1

SEQUENCE OF EVENTS FOR INSTRUMENT LINE BREAK

Time	Event
0	Instrument line fails.
0 - 10 min.	Identification of break attempted.
10 min.	Manual Scram
310 minutes	Reactor Vessel depressurized and break flow terminated.

TABLE 15.6-2

MASS RELEASES - INSTRUMENT LINE BREAK

Time (minutes)	Liquid Fraction		Mass Steam From Dome (lbm)
	Mass Steam (lbm)	Mass Liquid (lbm)	
0	0	0	0
10	2,036	4,195	926
70	8,736	17,190	5,046
130	12,592	27,963	6,801
190	14,532	36,452	7,532
250	15,368	43,066	7,971
310	15,589	50,007	8,117

TABLE 15.6-3

INSTRUMENT LINE BREAK
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	Realistic Analysis[1]	
	EAB[2]	LPZ[3]
I-131	3.39E-01	4.52E-01
I-132	3.13E+00	4.18E+00
I-133	2.32E+00	3.10E+00
I-134	6.25E+00	8.35E+00
I-135	3.39E+00	4.52E+00
Co-58	3.49E-02	4.68E-02
Co-60	3.49E-03	4.68E-03
Sr-89	2.17E-02	2.90E-02
Sr-90	1.61E-03	2.15E-03
Sr-91	4.82E-01	6.45E-01
Sr-92	7.69E-01	1.03E+00
Zr-95	2.80E-04	3.74E-04
Zr-97	2.24E-04	2.99E-04
Nb-95	2.94E-04	3.93E-04
Mo-99	1.54E-01	2.06E-01
Tc-99m	1.96E+00	2.62E+00
Ru-103	1.33E-04	1.78E-04
Ru-106	1.82E-05	2.43E-05
Te-129m	2.80E-04	3.74E-04
Te-132	3.42E-01	4.58E-01
Cs-134	1.12E-03	1.50E-03
Cs-136	7.69E-04	1.03E-03
Cs-137	1.68E-03	2.24E-03
Ba-139	1.12E+00	1.50E+00
Ba-140	6.29E-02	8.42E-02
Ce-141	2.73E-04	3.65E-04
Ce-143	2.45E-04	3.27E-04
Ce-144	2.45E-04	3.27E-04
Pr-143	2.66E-04	3.55E-04
Nd-147	9.79E-05	1.31E-04
Np-239	1.68E+00	2.24E+00
Kr-83m	4.38E-01	4.68E-02
Kr-85m	8.46E-01	4.68E-03
Kr-85	3.02E-03	2.90E-02
Kr-87	2.27E+00	2.15E-03
Kr-88	2.72E+00	6.45E-01
Kr-89	2.72E-02	1.03E+00
Xe-131m	2.27E-03	3.74E-04
Xe-133m	4.23E-02	2.99E-04
Xe-133	1.24E+00	3.93E-04
Xe-135m	1.04E+00	2.06E-01
Xe-135	3.32E+00	2.62E+00
Xe-137	1.01E-01	1.78E-04
Xe-138	3.17E+00	2.43E-05

(1) Based on 0.2 μCi/gram dose equivalent I-131.
(2) 2 hour release.
(3) Duration of accident release.

TABLE 15.6-4

INSTRUMENT LINE BREAK INSIDE SECONDARY CONTAINMENT RADIOLOGICAL CONSEQUENCES (REALISTIC ANALYSIS)

Exclusion Area Boundary (2 hr)	Low Population Zone (duration)	Control Structure Habitability Envelope (duration)
Rem TEDE	Rem TEDE	Rem TEDE
8.38E-04 (1)	9.36E-05	3.36E-02

(1) 8.38E-04 = 8.38×10^{-04}

TABLE 15.6-5

INSTRUMENT LINE BREAK ACCIDENT
PARAMETERS FOR POSTULATED ACCIDENT ANALYSIS

I.	Data and Assumptions Used to Estimate Radioactive Sources from Postulated Accidents	Realistic Analysis
A.	Reactor power (MWt)	4032
B.	Fuel damage	None
C.	Reactor coolant activity before the accident	
	1. Iodine concentration in coolant (μCi/gm I-131 dose equivalent)	0.2
	2. Noble gas release rate for duration of the accident (μCi/sec at 30 minutes decay)	100,000
D.	Iodine carry over fraction reactor water to steam (percent)	8
II.	Data and Assumptions Used to Estimate Activity Released to the Environment	
A.	Mass releases:	Table 15.6-2
B.	Fraction of noble gases airborne from reactor steam release	100 percent
C.	Fraction of iodines airborne from reactor steam release and from coolant liquid release that flashes to steam	100 percent
D.	Fraction of iodines airborne from reactor coolant liquid release	10 percent
E.	Credit taken for holdup in secondary containment	None
F.	Iodine removal efficiency of the Standup Gas Treatment System	0 percent
G.	Plate-out inside Secondary Containment	0 percent
III.	Data and Assumptions Used To Evaluate Control Room Doses	
A.	Control structure habitability envelope free volume (ft^3)	518,000
B.	Control Room free volume (ft^3)	110,000
C.	Control Structure filtered air intake flow (cfm)	5229 – 6391
D.	Control structure unfiltered outside air filtration rate – ingress/egress (cfm)	10
E.	Control structure unidentified unfiltered outside air infiltration rate (cfm)	500
F.	Control structure filter efficiency for iodine (percent)	99
IV.	Dispersion Data	
A.	EAB and LPZ distance (meters)	549/4827
B.	X/Qs for time intervals	Table 2.3-92
C.	X/Qs for LPZ	Table 2.3-105
D.	X/Qs for Control Structure Habitability Envelope	Appendix 15B
V.	Dose Data	
A.	Method of dose calculations	Appendix 15B
B.	Dose conversion assumptions	Appendix 15B
C.	Activity released to environment	Table 15.6-3
D.	Doses	Table 15.6-4

TABLE 15.6-6

SEQUENCE OF EVENTS FOR STEAM LINE BREAK OUTSIDE CONTAINMENT

Time-sec	Event
0	Guillotine break of one main steam line outside primary containment
~0.5	High steamline flow signal initiates closure of main steam line isolation valve.
<1.25	Reactor begins scram.
6.0	Main steam isolation valves fully closed.
~131	Safety relief valves open on high vessel pressure. The valves open and close to maintain vessel pressure at approximately 1200 psig.
330	Reactor water level above core begins to drop slowly due to loss of steam through the safety valves. Reactor pressure still at approximately 1200 psig.
2109	Operator initiates ADS. Vessel depressurizes rapidly.
~2300	Low pressure ECCS systems initiated. Reactor fuel uncovered partially.
2365	Core effectively reflooded and clad temperature heatup terminated. No fuel rod failure.

TABLE 15.6-7

STEAM LINE BREAK ACCIDENT
ACTIVITY RELEASED TO THE ENVIRONMENT (curies)
(Design Basis Analysis)

Isotope	Case 1 Activity (1)	Case 2 Activity (2)
I-131	2.40E+00	4.79E+01
I-132	2.21E+01	4.42E+02
I-133	1.64E+01	3.28E+02
I-134	4.42E+01	8.85E+02
I-135	2.40E+01	4.79E+02
Kr-83m	4.19E-01	4.19E-01
Kr-85m	7.29E-01	7.29E-01
Kr-85	2.41E-03	2.41E-03
Kr-87	2.36E+00	2.36E+00
Kr-88	2.45E+00	2.45E+00
Kr-89	1.56E+01	1.56E+01
Xe-131m	1.81E-03	1.81E-03
Xe-133m	3.39E-02	3.39E-02
Xe-133	9.89E-01	9.89E-01
Xe-135m	3.15E+00	3.15E+00
Xe-135	2.75E+00	2.75E+00
Xe-137	1.84E+01	1.84E+01
Xe-138	1.10E+01	1.10E+01

Notes:

1. Iodine concentration in coolant = 0.2 μCi/g dose equivalent I-131
2. Iodine concentration in coolant = 4.0 μCi/g dose equivalent I-131

TABLE 15.6-8 STEAM LINE BREAK ACCIDENT ACTIVITY RELEASED TO THE ENVIRONMENT (curies) (Realistic Analysis)	
Isotope	**Activity**
I-131	1.45E-01
I-132	1.35E+00
I-133	9.96E-01
I-134	2.72E+00
I-135	1.46E+00
Kr-83m	4.27E-02
Kr-85m	7.45E-02
Kr-85	2.46E-04
Kr-87	2.41E-01
Kr-88	2.51E-01
Xe-131m	1.84E-04
Xe-133m	3.46E-03
Xe-133	1.01E-01
Xe-135m	3.21E-01
Xe-135	2.81E-01
Xe-138	1.12E+00

Table 15.6-9

STEAM LINE BREAK OUTSIDE CONTAINMENT
RADIOLOGICAL DOSES (rem TEDE)
Design Basis Case

Location / Dose Type	Design Basis Analysis		Realistic Analysis
	Case 1 [(1)]	Case 2 [(2)]	
Acceptance Criteria - Offsite	2.5	25	2.5
EAB	0.10	2.0	9.80E-04
LPZ	0.006	0.12	3.60E-05
Acceptance Criteria - CRHE	5.0	5.0	5
CRHE	0.05	0.93	7.20E-03

Notes:

1. Case 1: Iodine concentration in coolant = 0.2 μCi/g dose equivalent I-131
2. Case 2: Iodine concentration in coolant = 4.0 μCi/g dose equivalent I-131

TABLE 15.6-10

STEAM LINE BREAK ACCIDENT - PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Engineering Assumptions
I. Data and Assumptions Used to Estimate Radioactive Source from Postulated Accidents		
A. Reactor power (MWt)	4032	4032
B. Fuel damaged	None	None
C. Reactor coolant activity before the accident		
1. Iodine concentration in coolant		
Realistic (μCi/gm)	-	Table 11.1-2
Conservative case 1(μCi/gm dose-equivalent I-131)	0.2	N/A
Conservative case 2(μCi/gm dose-equivalent I-131)	4.0	N/A
2. Noble gas release rate prior to MSIV closure (μCi/sec at 30 min decay)	403,000	100,000
D. Iodine carryover fraction reactor water to steam (percent)	2	2
II. Data and assumption used to estimate activity released		
A. Isolation valve closure time (sec)	5.5	5.5
B. Coolant mass releases from break (lb_m)		
Total	97,970(2)	40,316(1)
Liquid	84,840	19,994
Steam from flashed liquid	6,480	8,000
Steam from steam dome	6,650	20,322
C. Fraction of iodine in:		
Released coolant assumed airborne (%)	100	100
Steam from flashed liquid (%)	100	100
Steam from steam dome (%)	8	8
D. Holdup in Turbine Building	No	No
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 filtered air intake flow (cfm)	5229 – 6391	5229 – 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 for iodine (percent)	99	99
IV. Disposition Data		
A. Release height (m)	0	0
B. Boundary for SB/LPZ distance (meters)	549/4827	549/4827
C. X/Q's for SB (0-2 hrs) sec/m^3	Table 2.3-92 (0.5 percentile)	Table 2.3-119 (50 percentile)
D. X/Q's for LPZ(0-8 hrs) sec/m^3	Table 2.3-105 (0.5 percentile)	Table 2.3-105 (50 percentile)
E. XQ's for Control Room sec/m^3(puff)	6.3E-04	1.6E-03
V. Dose Data		
A. Method of dose calculation	Appendix 15B	Appendix 15B
B. Dose conversion assumptions	Appendix 15B	Appendix 15B
C. Doses	Table 15.6-9	Table 15.6-9

(1) The 8000 lbm of steam from flashed liquid is included in the 19,994 lbm liquid release. The total release is the sum of the mass of liquid plus the mass of steam from the steam dome.

(2) Values as shown are increased by 20% in dose analysis.

TABLE 15.6-11
LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN PRIMARY CONTAINMENT (curies)
(Design Basis Accident)

Isotope	Primary Containment Airborne Activity As a Function of Time Post-Accident (curies)									
	0.1667 hr	0.5 hr	1 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Co-58			3.23E+02	8.36E+02	5.00E+00	9.47E-04	9.06E-04	8.30E-04	7.16E-04	6.19E-04
Co-60			1.33E+01	3.43E+01	2.06E-01	3.92E-05	3.86E-05	3.74E-05	3.55E-05	3.37E-05
Kr-85	2.79E+04	8.37E+04	6.13E+05	1.67E+06	1.67E+06	1.66E+06	1.63E+06	1.58E+06	1.51E+06	1.43E+06
Kr-85m	4.52E+05	1.29E+06	8.72E+06	2.04E+07	8.03E+06	6.71E+05	9.61E+00	1.97E-09	1.40E-25	9.96E-42
Kr-87	7.93E+05	1.98E+06	1.10E+07	1.74E+07	6.60E+05	1.07E+02	9.52E-16	7.56E-50	0.00E+00	0.00E+00
Kr-88	1.14E+06	3.14E+06	2.04E+07	4.35E+07	1.00E+07	2.01E+05	4.62E-03	2.45E-18	8.47E-44	2.94E-69
Rb-86	3.86E+03	1.03E+04	2.29E+04	4.39E+04	2.60E+02	4.85E-02	4.27E-02	3.32E-02	2.18E-02	1.43E-02
Sr-89			5.99E+05	1.55E+06	9.25E+03	1.75E+00	1.65E+00	1.48E+00	1.23E+00	1.02E+00
Sr-90			8.01E+04	2.07E+05	1.24E+03	2.37E-01	2.33E-01	2.26E-01	2.15E-01	2.05E-01
Sr-91			6.99E+05	1.68E+06	6.50E+03	3.86E-01	1.99E-03	5.28E-08	1.25E-15	2.95E-23
Sr-92			6.23E+05	1.25E+06	1.61E+03	5.13E-03	5.08E-11	4.99E-27	1.04E-53	2.17E-80
Y-90			1.07E+03	3.74E+03	9.90E+01	5.35E-02	1.50E-01	2.09E-01	2.14E-01	2.05E-01
Y-91			7.85E+03	2.04E+04	1.32E+02	2.74E-02	2.71E-02	2.46E-02	2.08E-02	1.76E-02
Y-92			4.05E+04	1.96E+05	2.64E+03	4.90E-02	4.87E-08	2.70E-20	1.00E-40	3.73E-61
Y-93			8.71E+03	2.10E+04	8.35E+01	5.31E-03	3.74E-05	1.85E-09	1.24E-16	8.30E-24
Zr-95			1.13E+04	2.93E+04	1.75E+02	3.31E-02	3.16E-02	2.87E-02	2.45E-02	2.10E-02
Zr-97			1.10E+04	2.73E+04	1.28E+02	1.27E-02	6.51E-04	1.72E-06	8.70E-11	4.40E-15
Nb-95			1.13E+04	2.93E+04	1.76E+02	3.35E-02	3.30E-02	3.17E-02	2.94E-02	2.69E-02
Mo-99			1.49E+05	3.81E+05	2.14E+03	3.46E-01	1.60E-01	3.42E-02	2.62E-03	2.01E-04
Tc-99m			1.32E+05	3.41E+05	1.99E+03	3.32E-01	1.55E-01	3.32E-02	2.54E-03	1.95E-04
Ru-103			1.30E+05	3.36E+05	2.00E+03	3.77E-01	3.53E-01	3.08E-01	2.46E-01	1.96E-01
Ru-105			7.67E+04	1.70E+05	3.99E+02	6.26E-03	8.10E-08	1.36E-17	6.91E-34	3.52E-50
Ru-106			5.05E+04	1.31E+05	7.82E+02	1.49E-01	1.46E-01	1.40E-01	1.31E-01	1.22E-01
Rh-105			8.40E+04	2.17E+05	1.23E+03	1.78E-01	4.30E-02	2.48E-03	2.14E-05	1.84E-07
Sb-127			1.54E+05	3.94E+05	2.26E+03	3.82E-01	2.19E-01	7.23E-02	1.14E-02	1.79E-03
Sb-129			3.90E+05	8.59E+05	1.96E+03	2.88E-02	2.72E-07	2.44E-17	4.40E-34	7.91E-51
Te-127			1.39E+05	3.59E+05	2.12E+03	3.74E-01	2.32E-01	9.67E-02	3.82E-02	2.66E-02
Te-127m			1.09E+04	2.83E+04	1.70E+02	3.26E-02	3.25E-02	3.13E-02	2.83E-02	2.53E-02
Te-129			4.11E+05	9.66E+05	2.83E+03	1.75E-01	1.34E-01	1.15E-01	8.88E-02	6.88E-02
Te-129m			7.56E+04	1.96E+05	1.17E+03	2.20E-01	2.04E-01	1.75E-01	1.36E-01	1.05E-01
Te-131m			3.09E+05	7.80E+05	4.07E+03	5.36E-01	1.00E-01	3.49E-03	1.30E-05	4.83E-08
Te-132			2.29E+06	5.87E+06	3.33E+04	5.52E+00	2.87E+00	7.78E-01	8.83E-02	1.00E-02
I-131	1.70E+06	4.55E+06	1.18E+07	2.38E+07	1.71E+06	1.48E+06	1.13E+06	6.52E+05	2.62E+05	1.05E+05
I-132	2.41E+06	6.14E+06	1.54E+07	2.85E+07	4.58E+05	3.66E+03	2.97E+00	8.04E-01	9.12E-02	1.03E-02
I-133	3.45E+06	9.13E+06	2.33E+07	4.58E+07	2.74E+06	1.48E+06	1.32E+05	1.06E+03	3.38E-01	1.08E-04
I-134	3.31E+06	6.77E+06	1.18E+07	1.09E+07	6.91E+03	2.03E-02	3.79E-27	1.32E-76	0.00E+00	0.00E+00
I-135	3.25E+06	8.38E+06	2.07E+07	3.77E+07	1.47E+06	2.52E+05	1.31E+02	3.51E-05	3.93E-16	4.40E-27
Xe-133	3.62E+06	1.08E+07	7.94E+07	2.16E+08	2.10E+08	1.95E+08	1.32E+08	5.83E+07	1.48E+07	3.76E+06
Xe-135	1.01E+06	3.16E+06	2.38E+07	6.72E+07	5.40E+07	2.25E+07	1.32E+05	2.29E+00	2.46E-08	2.64E-16
Cs-134	3.69E+05	9.79E+05	2.19E+06	4.20E+06	2.51E+04	4.79E+00	4.71E+00	4.55E+00	4.29E+00	4.05E+00
Cs-136	9.34E+04	2.48E+05	5.53E+05	1.06E+06	6.26E+03	1.15E+00	9.70E-01	6.86E-01	3.85E-01	2.16E-01
Cs-137	2.83E+05	7.51E+05	1.68E+06	3.22E+06	1.93E+04	3.68E+00	3.63E+00	3.52E+00	3.35E+00	3.19E+00

TABLE 15.6-11
LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN PRIMARY CONTAINMENT (curies)
(Design Basis Accident)

Isotope	Primary Containment Airborne Activity As a Function of Time Post-Accident (curies)									
	0.1667 hr	0.5 hr	1 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Ba-139			6.92E+05	1.08E+06	3.17E+02	1.94E-05	3.60E-21	1.24E-52	0.00E+00	0.00E+00
Ba-140			1.13E+06	2.92E+06	1.73E+04	3.17E+00	2.66E+00	1.86E+00	1.03E+00	5.69E-01
La-140			1.70E+04	6.55E+04	2.06E+03	1.08E+00	2.32E+00	2.08E+00	1.18E+00	6.55E-01
La-141			8.82E+03	1.91E+04	3.98E+01	4.51E-04	1.36E-09	1.23E-20	4.85E-39	1.91E-57
La-142			6.38E+03	1.05E+04	4.25E+00	6.09E-07	5.24E-21	3.88E-49	5.06E-96	0.00E+00
Ce-141			2.67E+04	6.91E+04	4.12E+02	7.75E-02	7.17E-02	6.12E-02	4.71E-02	3.62E-02
Ce-143			2.42E+04	6.14E+04	3.24E+02	4.42E-02	9.60E-03	4.53E-04	2.79E-06	1.72E-08
Ce-144			2.22E+04	5.75E+04	3.44E+02	6.56E-02	6.42E-02	6.14E-02	5.71E-02	5.30E-02
Pr-143			9.68E+03	2.51E+04	1.53E+02	2.99E-02	2.84E-02	2.10E-02	1.20E-02	6.88E-03
Nd-147			4.22E+03	1.09E+04	6.42E+01	1.17E-02	9.58E-03	6.37E-03	3.22E-03	1.63E-03
Np-239			3.02E+05	7.72E+05	4.29E+03	6.73E-01	2.74E-01	4.56E-02	2.29E-03	1.15E-04
Pu-238			6.59E+01	1.70E+02	1.02E+00	1.95E-04	1.92E-04	1.87E-04	1.78E-04	1.70E-04
Pu-239			6.17E+00	1.60E+01	9.57E-02	1.83E-05	1.81E-05	1.77E-05	1.68E-05	1.60E-05
Pu-240			1.10E+01	2.85E+01	1.70E-01	3.25E-05	3.21E-05	3.11E-05	2.96E-05	2.82E-05
Pu-241			2.62E+03	6.79E+03	4.07E+01	7.75E-03	7.64E-03	7.41E-03	7.05E-03	6.70E-03
Am-241			1.30E+00	3.35E+00	2.01E-02	3.86E-06	3.91E-06	3.99E-06	4.11E-06	4.21E-06
Cm-242			4.59E+02	1.19E+03	7.11E+00	1.35E-03	1.32E-03	1.25E-03	1.14E-03	1.04E-03
Cm-244			2.35E+01	6.07E+01	3.63E-01	6.93E-05	6.83E-05	6.63E-05	6.31E-05	6.00E-05
SUM	2.19E+07	5.75E+07	2.40E+08	5.43E+08	2.91E+08	2.23E+08	1.35E+08	6.06E+07	1.66E+07	5.30E+06

TABLE 15.6-13
LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN REACTOR BUILDING (curies)
(Design Basis Accident)

Isotope	Reactor Building Airborne Activity As a Function of Time Post-Accident (curies)									
	0.1667 hr	0.5 hr	1 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Co-58			3.29E-02	2.50E-01	2.10E-01	1.07E-02	1.73E-05	1.57E-05	1.32E-05	1.12E-05
Co-60			1.35E-03	1.03E-02	8.66E-03	4.42E-04	7.39E-07	7.06E-07	6.55E-07	6.07E-07
Kr-85	9.30E-01	8.25E+00	7.61E+01	4.94E+02	2.61E+03	3.56E+03	1.78E+03	1.72E+03	1.64E+03	1.56E+03
Kr-85m	1.51E+01	1.27E+02	1.08E+03	6.01E+03	1.26E+04	1.44E+03	1.04E-02	2.14E-12	1.52E-28	1.08E-44
Kr-87	2.65E+01	1.95E+02	1.37E+03	5.14E+03	1.03E+03	2.30E-01	1.04E-18	8.22E-53	0.00E+00	0.00E+00
Kr-88	3.80E+01	3.10E+02	2.53E+03	1.28E+04	1.57E+04	4.31E+02	5.02E-06	2.66E-21	9.21E-47	3.19E-72
Rb-86	1.32E-01	1.07E+00	4.24E+00	1.62E+01	1.19E+01	5.95E-01	8.85E-04	6.78E-04	4.35E-04	2.79E-04
Sr-89			6.10E+01	4.64E+02	3.89E+02	1.97E+01	3.17E-02	2.79E-02	2.26E-02	1.84E-02
Sr-90			8.15E+00	6.20E+01	5.22E+01	2.67E+00	4.46E-03	4.27E-03	3.97E-03	3.70E-03
Sr-91			7.11E+01	5.03E+02	2.74E+02	4.35E+00	3.81E-05	9.96E-10	2.30E-17	5.33E-25
Sr-92			6.34E+01	3.73E+02	6.78E+01	5.79E-02	9.73E-13	9.41E-29	1.92E-55	3.91E-82
Y-90			1.17E-01	1.30E+00	4.23E+00	6.06E-01	2.88E-03	3.95E-03	3.95E-03	3.70E-03
Y-91			8.00E-01	6.15E+00	5.58E+00	3.10E-01	5.20E-04	4.64E-04	3.83E-04	3.17E-04
Y-92			5.21E+00	7.98E+01	1.13E+02	5.58E-01	9.36E-10	5.11E-22	1.86E-42	6.74E-63
Y-93			8.86E-01	6.29E+00	3.51E+00	5.99E-02	7.16E-07	3.50E-11	2.29E-18	1.50E-25
Zr-95			1.15E+00	8.75E+00	7.36E+00	3.73E-01	6.04E-04	5.42E-04	4.53E-04	3.78E-04
Zr-97			1.12E+00	8.17E+00	5.39E+00	1.43E-01	1.25E-05	3.25E-08	1.61E-12	7.94E-17
Nb-95			1.15E+00	8.78E+00	7.40E+00	3.78E-01	6.31E-04	5.98E-04	5.42E-04	4.85E-04
Mo-99			1.52E+01	1.14E+02	9.02E+01	3.90E+00	3.06E-03	6.46E-04	4.83E-05	3.62E-06
Tc-99m			1.34E+01	1.02E+02	8.39E+01	3.75E+00	2.97E-03	6.26E-04	4.69E-05	3.51E-06
Ru-103			1.32E+01	1.00E+02	8.43E+01	4.26E+00	6.75E-03	5.81E-03	4.54E-03	3.54E-03
Ru-105			7.81E+00	5.08E+01	1.68E+01	7.06E-02	1.55E-09	2.56E-19	1.27E-35	6.34E-52
Ru-106			5.14E+00	3.91E+01	3.29E+01	1.68E+00	2.79E-03	2.64E-03	2.42E-03	2.21E-03
Rh-105			8.55E+00	6.48E+01	5.16E+01	2.01E+00	8.23E-04	4.68E-05	3.95E-07	3.33E-09
Sb-127			1.56E+01	1.18E+02	9.50E+01	4.31E+00	4.20E-03	1.36E-03	2.10E-04	3.23E-05
Sb-129			3.97E+01	2.57E+02	8.27E+01	3.24E-01	5.22E-09	4.61E-19	8.11E-36	1.43E-52
Te-127			1.41E+01	1.07E+02	8.91E+01	4.22E+00	4.44E-03	1.83E-03	7.05E-04	4.79E-04
Te-127m			1.11E+00	8.48E+00	7.16E+00	3.68E-01	6.23E-04	5.90E-04	5.22E-04	4.57E-04
Te-129			4.18E+01	2.89E+02	1.19E+02	1.97E+00	2.56E-03	2.16E-03	1.64E-03	1.24E-03
Te-129m			7.69E+00	5.85E+01	4.92E+01	2.49E+00	3.91E-03	3.31E-03	2.50E-03	1.90E-03
Te-131m			3.14E+01	2.33E+02	1.71E+02	6.05E+00	1.92E-03	6.58E-05	2.39E-07	8.71E-10
Te-132			2.33E+02	1.76E+03	1.40E+03	6.22E+01	5.50E-02	1.47E-02	1.63E-03	1.81E-04
I-131	5.81E+01	4.77E+02	2.05E+03	8.51E+03	8.57E+03	3.70E+03	1.41E+03	8.13E+02	3.26E+02	1.31E+02
I-132	8.15E+01	6.27E+02	2.52E+03	9.21E+03	2.95E+03	9.13E+01	9.71E+00	2.59E+00	2.87E-01	3.18E-02
I-133	1.18E+02	9.58E+02	4.06E+03	1.63E+04	1.37E+04	3.69E+03	1.65E+02	1.32E+00	4.20E-04	1.34E-07
I-134	1.13E+02	7.11E+02	2.06E+03	3.88E+03	3.47E+01	5.08E-05	4.73E-30	1.64E-79	0.00E+00	0.00E+00
I-135	1.11E+02	8.79E+02	3.60E+03	1.35E+04	7.38E+03	6.30E+02	1.63E-01	4.38E-08	4.88E-19	5.45E-30
Xe-133	1.21E+02	1.07E+03	9.86E+03	6.39E+04	3.31E+05	4.28E+05	1.57E+05	6.85E+04	1.73E+04	4.40E+03
Xe-135	3.42E+01	3.28E+02	3.07E+03	2.04E+04	9.80E+04	6.93E+04	3.07E+02	4.86E-03	5.14E-11	5.46E-19
Cs-134	1.26E+01	1.03E+02	4.05E+02	1.55E+03	1.15E+03	5.88E+01	9.76E-02	9.29E-02	8.57E-02	7.91E-02
Cs-136	3.18E+00	2.60E+01	1.02E+02	3.91E+02	2.87E+02	1.42E+01	2.01E-02	1.40E-02	7.68E-03	4.21E-03

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TABLE 15.6-13
LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN REACTOR BUILDING (curies)
(Design Basis Accident)

Isotope	Reactor Building Airborne Activity As a Function of Time Post-Accident (curies)									
	0.1667 hr	0.5 hr	1 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Cs-137	9.63E+00	7.87E+01	3.11E+02	1.19E+03	8.86E+02	4.52E+01	7.51E-02	7.19E-02	6.69E-02	6.22E-02
Ba-139			7.04E+01	3.24E+02	1.34E+01	2.19E-04	6.89E-23	2.34E-54	0.00E+00	0.00E+00
Ba-140			1.15E+02	8.74E+02	7.26E+02	3.58E+01	5.09E-02	3.51E-02	1.90E-02	1.03E-02
La-140			1.90E+00	2.37E+01	8.79E+01	1.22E+01	4.45E-02	3.93E-02	2.18E-02	1.18E-02
La-141			8.98E-01	5.72E+00	1.67E+00	5.09E-03	2.60E-11	2.32E-22	8.94E-41	3.44E-59
La-142			6.50E-01	3.15E+00	1.79E-01	6.87E-06	1.00E-22	7.32E-51	9.34E-98	0.00E+00
Ce-141			2.72E+00	2.07E+01	1.73E+01	8.74E-01	1.37E-03	1.16E-03	8.69E-04	6.54E-04
Ce-143			2.47E+00	1.84E+01	1.37E+01	4.99E-01	1.84E-04	8.55E-06	5.15E-08	3.10E-10
Ce-144			2.26E+00	1.72E+01	1.45E+01	7.40E-01	1.23E-03	1.16E-03	1.05E-03	9.57E-0
Pr-143			9.86E-01	7.51E+00	6.43E+00	3.38E-01	5.44E-04	3.97E-04	2.22E-04	1.24E-04
Nd-147			4.30E-01	3.26E+00	2.70E+00	1.32E-01	1.83E-04	1.20E-04	5.95E-05	2.95E-05
Np-239			3.07E+01	2.31E+02	1.81E+02	7.59E+00	5.25E-03	8.60E-04	4.22E-05	2.07E-06
Pu-238			6.71E-03	5.10E-02	4.30E-02	2.20E-03	3.68E-06	3.52E-06	3.28E-06	3.06E-06
Pu-239			6.28E-04	4.77E-03	4.03E-03	2.06E-04	3.47E-07	3.33E-07	3.10E-07	2.89E-07
Pu-240			1.12E-03	8.52E-03	7.18E-03	3.67E-04	6.14E-07	5.87E-07	5.47E-07	5.09E-07
Pu-241			2.67E-01	2.03E+00	1.71E+00	8.75E-02	1.46E-04	1.40E-04	1.30E-04	1.21E-04
Am-241			1.32E-04	1.00E-03	8.48E-04	4.35E-05	7.48E-08	7.53E-08	7.58E-08	7.59E-08
Cm-242			4.68E-02	3.56E-01	2.99E-01	1.53E-02	2.52E-05	2.35E-05	2.10E-05	1.87E-05
Cm-244			2.39E-03	1.82E-02	1.53E-02	7.82E-04	1.31E-06	1.25E-06	1.16E-06	1.08E-06
SUM	7.43E+02	5.90E+03	3.40E+04	1.70E+05	5.01E+05	5.11E+05	1.60E+05	7.10E+04	1.93E+04	6.09E+03

TABLE 15.6-14
LOSS OF COOLANT ACCIDENT
ACTIVITY RELEASED TO ENVIRONMENT (curies)
(Design Basis Accident)

Isotope	Total Activity Released To The Environment As a Function of Time Post-Accident (curies)							
	0.5 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Co-58	0.00E+00	6.53E-03	1.76E-02	1.97E-02	1.98E-02	1.98E-02	1.98E-02	1.98E-02
Co-60	0.00E+00	2.68E-04	7.21E-04	8.08E-04	8.13E-04	8.13E-04	8.13E-04	8.13E-04
Kr-85	9.55E-01	7.52E+01	2.28E+03	1.37E+04	4.76E+04	1.26E+05	2.85E+05	4.62E+05
Kr-85m	1.52E+01	9.91E+02	1.70E+04	3.65E+04	3.80E+04	3.80E+04	3.80E+04	3.80E+04
Kr-87	2.53E+01	1.05E+03	5.97E+03	6.42E+03	6.42E+03	6.42E+03	6.42E+03	6.42E+03
Kr-88	3.77E+01	2.22E+03	2.83E+04	4.44E+04	4.47E+04	4.47E+04	4.47E+04	4.47E+04
Rb-86	2.95E-02	4.43E-01	1.05E+00	1.18E+00	1.21E+00	1.25E+00	1.28E+00	1.29E+00
Sr-89	0.00E+00	1.21E+01	3.26E+01	3.65E+01	3.67E+01	3.67E+01	3.67E+01	3.67E+01
Sr-90	0.00E+00	1.62E+00	4.35E+00	4.88E+00	4.90E+00	4.90E+00	4.90E+00	4.90E+00
Sr-91	0.00E+00	1.37E+01	3.36E+01	3.57E+01	3.57E+01	3.57E+01	3.57E+01	3.57E+01
Sr-92	0.00E+00	1.12E+01	2.31E+01	2.34E+01	2.34E+01	2.34E+01	2.34E+01	2.34E+01
Y-90	0.00E+00	2.50E-02	1.20E-01	1.84E-01	1.91E-01	1.91E-01	1.92E-01	1.92E-01
Y-91	0.00E+00	1.59E-01	4.35E-01	4.93E-01	4.96E-01	4.96E-01	4.96E-01	4.97E-01
Y-92	0.00E+00	1.16E+00	5.95E+00	6.70E+00	6.71E+00	6.71E+00	6.71E+00	6.71E+00
Y-93	0.00E+00	1.70E-01	4.22E-01	4.49E-01	4.49E-01	4.49E-01	4.49E-01	4.49E-01
Zr-95	0.00E+00	2.28E-01	6.15E-01	6.88E-01	6.92E-01	6.92E-01	6.92E-01	6.92E-01
Zr-97	0.00E+00	2.18E-01	5.57E-01	6.03E-01	6.04E-01	6.04E-01	6.04E-01	6.04E-01
Nb-95	0.00E+00	2.29E-01	6.17E-01	6.91E-01	6.95E-01	6.95E-01	6.95E-01	6.95E-01
Mo-99	0.00E+00	2.99E+00	7.95E+00	8.81E+00	8.85E+00	8.85E+00	8.85E+00	8.85E+00
Tc-99m	0.00E+00	2.67E+00	7.15E+00	7.97E+00	8.01E+00	8.01E+00	8.01E+00	8.01E+00
Ru-103	0.00E+00	2.62E+00	7.05E+00	7.90E+00	7.94E+00	7.94E+00	7.94E+00	7.94E+00
Ru-105	0.00E+00	1.44E+00	3.25E+00	3.35E+00	3.35E+00	3.35E+00	3.35E+00	3.35E+00
Ru-106	0.00E+00	1.02E+00	2.74E+00	3.07E+00	3.09E+00	3.09E+00	3.09E+00	3.09E+00
Rh-105	0.00E+00	1.69E+00	4.53E+00	5.01E+00	5.03E+00	5.03E+00	5.03E+00	5.03E+00
Sb-127	0.00E+00	3.09E+00	8.24E+00	9.16E+00	9.21E+00	9.21E+00	9.21E+00	9.21E+00
Sb-129	0.00E+00	7.32E+00	1.64E+01	1.69E+01	1.69E+01	1.69E+01	1.69E+01	1.69E+01
Te-127	0.00E+00	2.80E+00	7.53E+00	8.41E+00	8.45E+00	8.45E+00	8.45E+00	8.45E+00
Te-127m	0.00E+00	2.21E-01	5.96E-01	6.68E-01	6.71E-01	6.71E-01	6.71E-01	6.71E-01
Te-129	0.00E+00	7.95E+00	1.88E+01	1.96E+01	1.96E+01	1.96E+01	1.96E+01	1.96E+01
Te-129m	0.00E+00	1.53E+00	4.11E+00	4.60E+00	4.63E+00	4.63E+00	4.63E+00	4.63E+00
Te-131m	0.00E+00	6.17E+00	1.61E+01	1.77E+01	1.77E+01	1.77E+01	1.77E+01	1.77E+01
Te-132	0.00E+00	4.60E+01	1.22E+02	1.36E+02	1.37E+02	1.37E+02	1.37E+02	1.37E+02
I-131	1.32E+01	2.32E+02	6.76E+02	1.24E+03	2.65E+03	4.80E+03	6.87E+03	7.81E+03
I-132	1.84E+01	2.93E+02	6.35E+02	6.75E+02	6.76E+02	6.76E+02	6.76E+02	6.76E+02
I-133	2.66E+01	4.54E+02	1.26E+03	1.98E+03	2.59E+03	2.66E+03	2.66E+03	2.66E+03
I-134	2.35E+01	1.92E+02	2.68E+02	2.68E+02	2.68E+02	2.68E+02	2.68E+02	2.68E+02
I-135	2.49E+01	3.93E+02	9.71E+02	1.22E+03	1.26E+03	1.26E+03	1.26E+03	1.26E+03
Xe-133	1.24E+02	9.73E+03	2.92E+05	1.70E+06	5.24E+06	9.78E+06	1.31E+07	1.41E+07
Xe-135	3.50E+01	3.05E+03	9.24E+04	4.09E+05	5.86E+05	5.87E+05	5.87E+05	5.87E+05
Cs-134	2.81E+00	4.24E+01	1.00E+02	1.13E+02	1.16E+02	1.21E+02	1.26E+02	1.29E+02
Cs-136	7.13E-01	1.07E+01	2.53E+01	2.84E+01	2.92E+01	2.99E+01	3.06E+01	3.08E+01
Cs-137	2.16E+00	3.25E+01	7.69E+01	8.68E+01	8.93E+01	9.26E+01	9.66E+01	9.92E+01

TABLE 15.6-14
LOSS OF COOLANT ACCIDENT
ACTIVITY RELEASED TO ENVIRONMENT (curies)
(Design Basis Accident)

Isotope	Total Activity Released To The Environment As a Function of Time Post-Accident (curies)							
	0.5 hr	2 hr	8 hr	24 hr	96 hr	240 hr	480 hr	720 hr
Ba-139	0.00E+00	1.13E+01	1.94E+01	1.94E+01	1.94E+01	1.94E+01	1.94E+01	1.94E+01
Ba-140	0.00E+00	2.28E+01	6.13E+01	6.85E+01	6.89E+01	6.89E+01	6.89E+01	6.89E+01
La-140	0.00E+00	4.19E-01	2.29E+00	3.61E+00	3.75E+00	3.75E+00	3.76E+00	3.76E+00
La-141	0.00E+00	1.65E-01	3.63E-01	3.73E-01	3.73E-01	3.73E-01	3.73E-01	3.73E-01
La-142	0.00E+00	1.06E-01	1.89E-01	1.89E-01	1.89E-01	1.89E-01	1.89E-01	1.89E-01
Ce-141	0.00E+00	5.39E-01	1.45E+00	1.62E+00	1.63E+00	1.63E+00	1.63E+00	1.63E+00
Ce-143	0.00E+00	4.85E-01	1.27E+00	1.40E+00	1.40E+00	1.40E+00	1.40E+00	1.40E+00
Ce-144	0.00E+00	4.49E-01	1.21E+00	1.35E+00	1.36E+00	1.36E+00	1.36E+00	1.36E+00
Pr-143	0.00E+00	1.96E-01	5.28E-01	5.94E-01	5.97E-01	5.97E-01	5.97E-01	5.97E-01
Nd-147	0.00E+00	8.51E-02	2.28E-01	2.55E-01	2.57E-01	2.57E-01	2.57E-01	2.57E-01
Np-239	0.00E+00	6.06E+00	1.61E+01	1.78E+01	1.79E+01	1.79E+01	1.79E+01	1.79E+01
Pu-238	0.00E+00	1.33E-03	3.58E-03	4.01E-03	4.04E-03	4.04E-03	4.04E-03	4.04E-03
Pu-239	0.00E+00	1.25E-04	3.35E-04	3.76E-04	3.78E-04	3.78E-04	3.78E-04	3.78E-04
Pu-240	0.00E+00	2.22E-04	5.98E-04	6.70E-04	6.74E-04	6.74E-04	6.74E-04	6.74E-04
Pu-241	0.00E+00	5.30E-02	1.43E-01	1.60E-01	1.61E-01	1.61E-01	1.61E-01	1.61E-01
Am-241	0.00E+00	2.62E-05	7.05E-05	7.91E-05	7.95E-05	7.95E-05	7.95E-05	7.95E-05
Cm-242	0.00E+00	9.28E-03	2.50E-02	2.80E-02	2.81E-02	2.81E-02	2.81E-02	2.81E-02
Cm-244	0.00E+00	4.74E-04	1.28E-03	1.43E-03	1.44E-03	1.44E-03	1.44E-03	1.44E-03
SUM	3.50E+02	1.89E+04	4.42E+05	2.22E+06	5.97E+06	1.06E+07	1.41E+07	1.52E+07

TABLE 15.6-15
LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN PRIMARY CONTAINMENT (curies)
(Realistic Analysis)

Isotope	Primary Containment Airborne Activity As a Function of Time Post-Accident (curies)				
	2 hr	8 hr	24 hr	96 hr	720 hr
I-131	8.18E+02	8.01E+02	7.56E+02	2.96E+01	3.08E+00
I-132	1.45E+03	1.57E+03	1.59E+03	8.99E-01	0.00E+00
I-133	1.84E+03	1.50E+03	8.82E+02	4.06E+00	3.71E-09
I-134	4.51E+02	3.93E+00	1.26E-05	0.00E+00	0.00E+00
I-135	1.53E+03	8.18E+02	1.53E+02	4.07E-03	0.00E+00
Xe-131m	4.98E+01	4.92E+01	4.76E+01	4.02E+01	8.87E+00
Xe-133m	2.47E+02	2.32E+02	1.94E+02	7.62E+01	2.01E-02
Xe-133	8.79E+03	8.56E+03	7.95E+03	5.41E+03	1.75E+02
Xe-135m	2.78E+02	1.46E+02	7.09E+01	1.89E-03	0.00E+00
Xe-135	7.48E+03	5.15E+03	1.76E+03	7.71E+00	0.00E+00
Xe-138	2.25E+01	4.81E-07	0.00E+00	0.00E+00	0.00E+00
Kr-83m	3.23E+02	3.32E+01	7.75E-02	0.00E+00	0.00E+00
Kr-85m	1.25E+03	4.94E+02	4.16E+01	6.03E-04	0.00E+00
Kr-87	1.11E+03	4.22E+01	6.89E-03	0.00E+00	0.00E+00
Kr-88	2.87E+03	6.64E+02	1.34E+01	3.12E-07	0.00E+00
Kr-85	3.74E+02	3.74E+02	3.74E+02	3.73E+02	3.70E+02
Co-58	6.49E-01	6.48E-01	6.43E-01	4.66E-04	0.00E+00
Co-60	6.00E-02	6.00E-02	6.00E-02	4.47E-05	0.00E+00
Sr-89	2.08E+02	2.07E+02	2.05E+02	1.47E-01	0.00E+00
Sr-90	7.78E+00	7.78E+00	7.78E+00	5.80E-03	0.00E+00
Sr-91	2.33E+03	1.50E+03	4.68E+02	1.83E-03	0.00E+00
Sr-92	2.24E+03	4.83E+02	8.07E+00	6.05E-11	0.00E+00
Zr-95	1.40E+00	1.39E+00	1.38E+00	1.00E-03	0.00E+00
Zr-97	1.44E+00	1.12E+00	5.83E-01	2.27E-05	0.00E+00
Nb-95	1.42E+00	1.42E+00	1.42E+00	1.06E-03	0.00E+00
Mo-99	7.28E+02	6.83E+02	5.77E+02	2.02E-01	0.00E+00
Tc-99m	7.67E+03	4.15E+03	1.13E+03	2.07E-01	0.00E+00
Ru-103	6.59E-01	6.56E-01	6.48E-01	4.59E-04	0.00E+00
Ru-106	9.00E-02	8.99E-02	8.98E-02	6.66E-05	0.00E+00
Te-129m	1.36E+00	1.35E+00	1.33E+00	9.34E-04	0.00E+00
Te-132	1.62E+03	1.54E+03	1.34E+03	5.27E-01	0.00E+00
Cs-134	1.57E+01	1.57E+01	1.57E+01	1.17E-02	0.00E+00
Cs-136	3.73E+00	3.68E+00	3.56E+00	2.26E-03	0.00E+00
Cs-137	8.13E+00	8.13E+00	8.13E+00	6.06E-03	0.00E+00
Ba-139	2.13E+03	1.04E+02	3.34E-02	0.00E+00	0.00E+00
Ba-140	3.03E+02	2.99E+02	2.88E+02	1.82E-01	0.00E+00
Ce-141	1.33E+00	1.32E+00	1.30E+00	9.11E-04	0.00E+00
Ce-143	1.14E+00	1.01E+00	7.19E-01	1.18E-04	0.00E+00
Ce-144	1.24E+00	1.24E+00	1.24E+00	9.16E-04	0.00E+00
Pr-143	1.28E+00	1.28E+00	1.26E+00	8.47E-04	0.00E+00
Nd-147	4.68E-01	4.60E-01	4.41E-01	2.72E-04	0.00E+00
Np-239	7.91E+03	7.34E+03	6.03E+03	1.86E+00	0.00E+00

TABLE 15.6-16

LOSS OF COOLANT ACCIDENT
ACTIVITY AIRBORNE IN REACTOR BUILDING (curies)
(Realistic Analysis)

Isotope	Reactor Building Airborne Activity As a Function of Time Post-Accident (curies)				
	2 hr	8 hr	24 hr	96 hr	720 hr
I-131	5.96E-01	1.45E+00	1.73E+00	7.11E-01	6.61E-02
I-132	1.03E+00	2.71E+00	3.04E+00	8.00E-01	2.79E-03
I-133	1.34E+00	2.72E+00	2.02E+00	9.77E-02	7.95E-11
I-134	3.29E-01	7.10E-03	2.88E-08	2.90E-33	0.00E+00
I-135	1.12E+00	1.48E+00	3.50E-01	9.79E-05	0.00E+00
Xe-131m	3.38E-02	8.31E-02	1.03E-01	4.82E-02	1.40E-02
Xe-133m	1.68E-01	3.98E-01	4.38E-01	1.03E-01	2.38E-05
Xe-133	5.97E+00	1.45E+01	1.73E+01	6.22E+00	1.77E-01
Xe-135m	3.08E-01	5.04E-01	2.08E-01	3.35E-04	0.00E+00
Xe-135	5.20E+00	9.51E+00	4.66E+00	1.51E-02	0.00E+00
Xe-138	5.61E-01	1.65E+00	1.75E+00	5.30E-01	2.20E-02
Kr-83m	2.19E-01	5.60E-02	1.65E-04	1.18E-16	0.00E+00
Kr-85m	8.48E-01	8.32E-01	8.86E-02	6.41E-07	0.00E+00
Kr-87	7.54E-01	7.11E-02	1.47E-05	0.00E+00	0.00E+00
Kr-88	1.95E+00	1.12E+00	2.85E-02	3.31E-10	0.00E+00
Kr-85	2.54E-01	6.30E-01	7.97E-01	3.97E-01	3.48E-01
Co-58	4.40E-04	1.09E-03	1.37E-03	6.63E-04	4.53E-04
Co-60	4.07E-05	1.01E-04	1.28E-04	6.36E-05	5.55E-05
Sr-89	7.11E-02	1.76E-01	2.21E-01	1.05E-01	6.50E-02
Sr-90	5.26E-03	1.31E-02	1.65E-02	8.23E-03	7.24E-03
Sr-91	1.74E+00	2.79E+00	1.10E+00	2.86E-03	0.00E+00
Sr-92	3.00E+00	1.61E+00	3.39E-02	1.70E-10	0.00E+00
Zr-95	9.14E-04	2.27E-03	2.85E-03	1.37E-03	9.12E-04
Zr-97	7.81E-04	1.52E-03	9.96E-04	2.59E-05	1.75E-16
Nb-95	9.69E-04	2.41E-03	3.04E-03	1.51E-03	1.21E-03
Mo-99	5.12E-01	1.20E+00	1.28E+00	2.99E-01	3.75E-04
Tc-99m	7.43E+00	9.79E+00	2.97E+00	2.90E-01	3.64E-04
Ru-103	4.33E-04	1.07E-03	1.34E-03	6.33E-04	3.52E-04
Ru-106	6.10E-05	1.51E-04	1.91E-04	9.48E-05	7.95E-05
Te-129m	9.14E-04	2.26E-03	2.82E-03	1.32E-03	6.79E-04
Te-132	1.14E+00	2.68E+00	2.95E+00	7.75E-01	2.70E-03
Cs-134	3.66E-03	9.09E-03	1.15E-02	5.71E-03	4.91E-03
Cs-136	2.52E-03	6.19E-03	7.56E-03	3.21E-03	7.15E-04
Cs-137	5.49E-03	1.36E-02	1.73E-02	8.59E-03	7.56E-03
Ba-139	3.90E+00	4.74E-01	1.92E-04	0.00E+00	0.00E+00
Ba-140	2.06E-01	5.06E-01	6.18E-01	2.61E-01	5.59E-02
Ce-141	8.93E-04	2.21E-03	2.75E-03	1.29E-03	6.51E-04
Ce-143	8.32E-04	1.82E-03	1.65E-03	1.81E-04	3.24E-10
Ce-144	8.00E-04	1.99E-03	2.51E-03	1.24E-03	1.03E-03
Pr-143	8.74E-04	2.17E-03	2.72E-03	1.22E-03	2.89E-04
Nd-147	3.24E-04	7.91E-04	9.61E-04	3.96E-04	6.75E-05
Np-239	5.61E+00	1.30E+01	1.35E+01	2.77E+00	1.16E-03

TABLE 15.6-17
LOSS OF COOLANT ACCIDENT
ACTIVITY RELEASED TO ENVIRONMENT (curies)
(Realistic Analysis)

Isotope	Total Activity Released To The Environment As a Function of Time Post-Accident (curies)				
	2 hr	8 hr	24 hr	96 hr	720 hr
I-131	2.45E+00	9.66E+00	2.79E+01	6.17E+01	1.48E+02
I-132	4.14E+00	1.81E+01	5.38E+01	1.08E+02	1.61E+02
I-133	5.68E+00	2.05E+01	4.80E+01	6.52E+01	6.68E+01
I-134	3.36E+00	4.21E+00	4.22E+00	4.22E+00	4.22E+00
I-135	5.11E+00	1.53E+01	2.47E+01	2.58E+01	2.58E+01
Xe-131m	1.56E-01	6.67E-01	2.09E+00	5.09E+00	1.69E+01
Xe-133m	7.78E-01	3.25E+00	9.58E+00	1.87E+01	2.42E+01
Xe-133	2.75E+01	1.17E+02	3.61E+02	8.11E+02	1.61E+03
Xe-135m	1.58E+00	5.77E+00	1.86E+01	2.59E+01	2.72E+01
Xe-135	2.48E+01	9.02E+01	1.89E+02	2.17E+02	2.19E+02
Xe-138	4.47E+00	4.48E+00	4.48E+00	4.48E+00	4.48E+00
Kr-83m	1.51E+00	2.81E+00	2.98E+00	2.98E+00	2.98E+00
Kr-85m	4.58E+00	1.30E+01	1.84E+01	1.87E+01	1.87E+01
Kr-87	6.32E+00	9.66E+00	9.80E+00	9.80E+00	9.80E+00
Kr-88	1.16E+01	2.70E+01	3.20E+01	3.20E+01	3.20E+01
Kr-85	1.17E+00	5.03E+00	1.60E+01	4.07E+01	2.12E+02
Co-58	1.94E-03	7.72E-03	2.28E-02	5.50E-02	2.47E-01
Co-60	1.79E-04	7.13E-04	2.12E-03	5.16E-03	2.57E-02
Sr-89	3.13E-01	1.25E+00	3.68E+00	8.82E+00	3.80E+01
Sr-90	2.32E-02	9.23E-02	2.74E-01	6.67E-01	3.33E+00
Sr-91	8.26E+00	2.68E+01	4.99E+01	5.50E+01	5.50E+01
Sr-92	1.73E+01	3.76E+01	4.31E+01	4.32E+01	4.32E+01
Zr-95	4.03E-03	1.60E-02	4.74E-02	1.14E-01	5.06E-01
Zr-97	3.59E-03	1.27E-02	2.82E-02	3.60E-02	3.64E-02
Nb-95	4.27E-03	1.70E-02	5.04E-02	1.23E-01	5.94E-01
Mo-99	2.28E+00	8.81E+00	2.41E+01	4.56E+01	6.28E+01
Tc-99m	3.66E+01	1.09E+02	1.79E+02	2.05E+02	2.22E+02
Ru-103	1.91E-03	7.59E-03	2.24E-02	5.34E-02	2.21E-01
Ru-106	2.68E-04	1.07E-03	3.17E-03	7.71E-03	3.77E-02
Te-129m	4.03E-03	1.60E-02	4.71E-02	1.12E-01	4.50E-01
Te-132	5.06E+00	1.96E+01	5.44E+01	1.06E+02	1.59E+02
Cs-134	1.61E-02	6.42E-02	1.90E-01	4.64E-01	2.29E+00
Cs-136	1.11E-02	4.41E-02	1.29E-01	2.95E-01	8.89E-01
Cs-137	2.42E-02	9.63E-02	2.86E-01	6.96E-01	3.48E+00
Ba-139	3.00E+01	4.64E+01	4.72E+01	4.72E+01	4.72E+01
Ba-140	9.11E-01	3.61E+00	1.05E+01	2.41E+01	7.18E+01
Ce-141	3.94E-03	1.56E-02	4.61E-02	1.10E-01	4.36E-01
Ce-143	3.74E-03	1.40E-02	3.56E-02	5.60E-02	6.14E-02
Ce-144	3.52E-03	1.40E-02	4.16E-02	1.01E-01	4.91E-01
Pr-143	3.85E-03	1.53E-02	4.53E-02	1.07E-01	3.40E-01
Nd-147	1.43E-03	5.65E-03	1.64E-02	3.73E-02	1.04E-01
Np-239	2.50E+01	9.61E+01	2.60E+02	4.74E+02	6.12E+02

TABLE 15.6-18

LOSS-OF-COOLANT ACCIDENT
SUMMARY OF OFFSITE DOSES

Dose Location	Dose (Rem / TEDE)		
	Regulatory Limit	Design Basis Analysis	Realistic Analysis
THYROID			
2 Hour Site Boundary	25	1.32E+01[(1)]	1.7E-02
30 Day Low Population Zone	25	4.2E+00	7.9E-03

1. 1.07E+01 = 1.07 X 10^{+01}

TABLE 15.6-21

LOSS-OF-COOLANT ACCIDENT
SUMMARY OF CONTROL ROOM OPERATOR DOSES
(Design Basis Analysis)

Dose	Control Room Operator Dose (Rem TEDE)		
	Regulatory Limit	Design Basis Analysis	Realistic Basis Analysis
30 Day Operator Dose	5	4.28	0.11

TABLE 15.6-22

LOSS-OF-COOLANT ACCIDENT
PARAMETERS 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. Fuel damaged (percent)	100	0
C. Activity released to primary containment atmosphere	Subsection 15.6.5.5.1.1	Subsection 15.6.5.5.2.1
D. Activity released to suppression pool water	Subsection 15.6.5.5.1.1	Subsection 15.6.5.5.2.1
E. Iodine form fractions (percent)		
1. Aerosol	95	95
2. Elemental	4.85	4.85
3. Organic	0.15	0.15
II. Data And Assumptions Used To Estimate Activity Released		
A. Primary containment leak rate (percent/day)	1 (0-24 hr) 0.5 (1-30 d)	1 (0-24 hr) 0.5 (1-30 d)
B. Reactor building leak rate(percent/day)	225 (0-10 min) 0 > 10 min	225 (0-10 min) 0 > 10 min
C. Secondary containment bypass leak rate (SCFH)	15	15
D. MSIV Leakage (SCFH for 4 steam lines)	300	300
E. ESF leak rate		
1. Leakage rate inside reactor building (gpm)	5	5
2. Flashing fraction for iodine (percent)	10	10
Primary containment free volume (ft^3)		
Drywell	239600	239600
Wetwell	148590	148590
TOTAL	388190	388190
G. Reactor building free volume (ft^3)		
1. Zone 1	1,488,600	1,488,600
2. Zone 2	1,598,600	1,598,600
3. Zone 3	2,668,000	2,668,000
TOTAL Used in Analysis (Zone 1 + Zone 3)	4,156,600	4,156,600
H. Suppression pool water volume (ft^3)	132,000	132,000
I. Standby Gas Treatment System Parameters		
1. SGTS flow during drawdown (cfm)	11,110	11,110

TABLE 15.6-22

LOSS-OF-COOLANT ACCIDENT
PARAMETERS FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Assumptions
2. Drawdown time to reach 0.25 inch of vacuum water gage in reactor building (minutes)	10	10
3. SGTS flow following drawdown (cfm)	6495	6495
4. SGTS filter efficiencies (percent)		
Iodine (All species)	99	99
J. Reactor Building Recirculation System Parameters		
1. Flow rate (cfm)	83,000	83,000
2. Mixing efficiency (percent)	50	50
3. Filter efficiency	0	0
4. Recirculation system actuation (seconds)	10 to 30	10 to 30
K. Post-LOCA activity concentrations in primary containment and reactor building	Tables 15.6-11, 15.6-13	Tables 15.6-15, 15.6-16
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 filtered air intake flow (cfm)	5229 - 6391	5229 - 6391
D. Control structure unfiltered outside air infiltration rate – ingress/egress (cfm)	10	10
E. Control structure unidentified unfiltered outside air infiltration rate (cfm)	600	600
F. Control structure filter efficiency for iodine (percent)	99	99
IV. Dispersion Data		
A. Site Boundary/Low Population Zone distance (meters)	549/4827	549/4827
B. Site Boundary atmospheric dispersion factors	Table 2.3-92 (0.5 percentile)	Table 2.3-92 (0.5 percentile)
C. Low Population Zone atmospheric dispersion factors	Table 2.3-105 (0.5 percentile)	Table 2.3-105 (0.5 percentile)
D. Control room atmospheric dispersion factors	Appendix 15B	Appendix 15B
V. Dose Data		
A. Method of calculation	App 15B	Appendix 15B
B. Isotopic data and dose conversion factors	App 15B	Appendix 15B
C. Activity released to environment	Table 15.6-14	Table 15.6-17
D. Offsite doses	Table 15.6-18	Table 15.6-18
E. Control room doses	Table 15.6-21	Table 15.6-21

TABLE 15.6-23

SEQUENCE OF EVENTS FOR FEEDWATER LINE BREAK OUTSIDE CONTAINMENT

Time-sec	Event
0	One feedwater line breaks.
0+	Feedwater line check valves isolate the reactor from the break.
5	Reactor scram on low water level.
<30	A low-low water reactor level RCIC and HPCI would initiate and are expected to maintain the water level above low-low-low level trip and eventually restore it to the normal elevation.
1 to 2 hours	Normal reactor cooldown procedure established.

TABLE 15.6-24

FEEDWATER LINE BREAK ACCIDENT - PARAMETERS
FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Assumptions
I. Data and Assumptions Used to Estimate Radioactive Source from Postulated Accidents		
A. Reactor power (MWt)	4032	4032
B. Fuel damaged	None	None
C. Reactor coolant iodine activity before the accident		
Realistic (µCi/gm)	N/A	Table 11.1-2
Conservative case 1 (µCi/gm dose-equivalent I-131)	0.2	N/A
Conservative case 2 (µCi/gm dose-equivalent I-131)	4.0	N/A
II. Data and Assumption Used to Estimate Activity Released		
A. Coolant mass releases from break (lb_m)		
Total	2.5×10^6	2.5×10^6
Liquid	2.24×10^6	2.24×10^6
Steam from flashed liquid	2.6×10^5	2.6×10^5
B. Fraction of iodine released from:		
Coolant assumed airborne (%)	100	100
Steam (%)	100	100
C. Holdup in Turbine Building	No	No
D. Iodine decontamination factor for the condensate demineralizer	10	10
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 filtered air intake flow (cfm)	5229 – 6391	5229 – 6391
D. Control structure unfiltered outside air infiltration rate – ingress/egress (cfm)	10	10
E. Control structure unidentified outside air infiltration rate (cfm)\	500	500
F. Control structure filter efficiency for iodine (percent)	99	99
IV. Disposition Data		
A. Boundary for SB/LPZ distance (meters)	549/4827	549/4827
B. Site Boundary X/Q	Table 2.3-92 (0.5 percentile)	Table 2.3-92 (50 percentile)
C. LPZ X/Q	Table 2.3-105 (0.5 percentile)	Table 2.3-105 (50 percentile)
D. CRHE X/Q	Appendix 15B	Appendix 15B
V. Dose Data		
A. Method of dose calculation	Appendix 15B	Appendix 15B
B. Dose conversion assumptions	Appendix 15B	Appendix 15B
C. Doses	Table 15.6-26	Table 15.6-26

TABLE 15.6-25

FEEDWATER LINE BREAK ACCIDENT
ACTIVITY RELEASED TO THE ENVIRONMENT (curies)

Isotope	Design Basis Analysis		Realistic Analysis
	Case 1[(1)]	Case 2[(2)]	
I-131	8.42E-02	1.68E+00	2.29E-02
I-132	7.78E+01	1.56E+01	2.11E-01
I-133	5.77E-01	1.15E+01	1.56E-01
I-134	1.56E+00	3.11E+01	4.22E-01
I-135	8.42E-01	1.68E+01	2.29E-01

Notes:
1. Iodine concentration in coolant = 0.2 µCi/gm dose equivalent I-131.
2. Iodine concentration in coolant = 4.0 µCi/gm dose equivalent I-131.

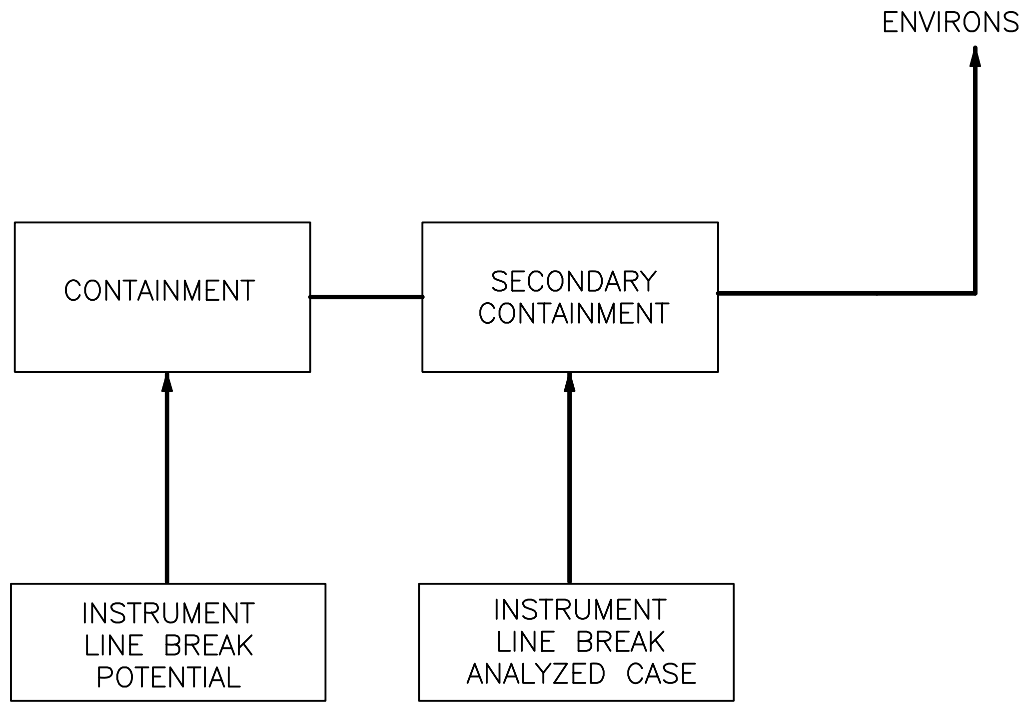
TABLE 15.6-26

FEEDWATER LINE BREAK OUTSIDE CONTAINMENT
RADIOLOGICAL DOSES (REM TEDE)

Location/ Dose Type	Design Basis Analysis		Realistic Analysis
	Case 1[1]	Case 2[2]	
Site Boundary (2 Hr)	3.40E-03	6.77E-02	1.44E-04
Low Population Zone (Duration)	2.01E-04	4.00E-03	5.32E-06
CRHE (Duration)	9.52E-04	1.90E-02	2.58E-04

Notes:

1. Case 1: iodine concentration in coolant = 0.2 μCi/gm dose equivalent I-131.
2. Case 2: iodine concentration in coolant = 4.0 μCi/gm dose equivalent I-131.



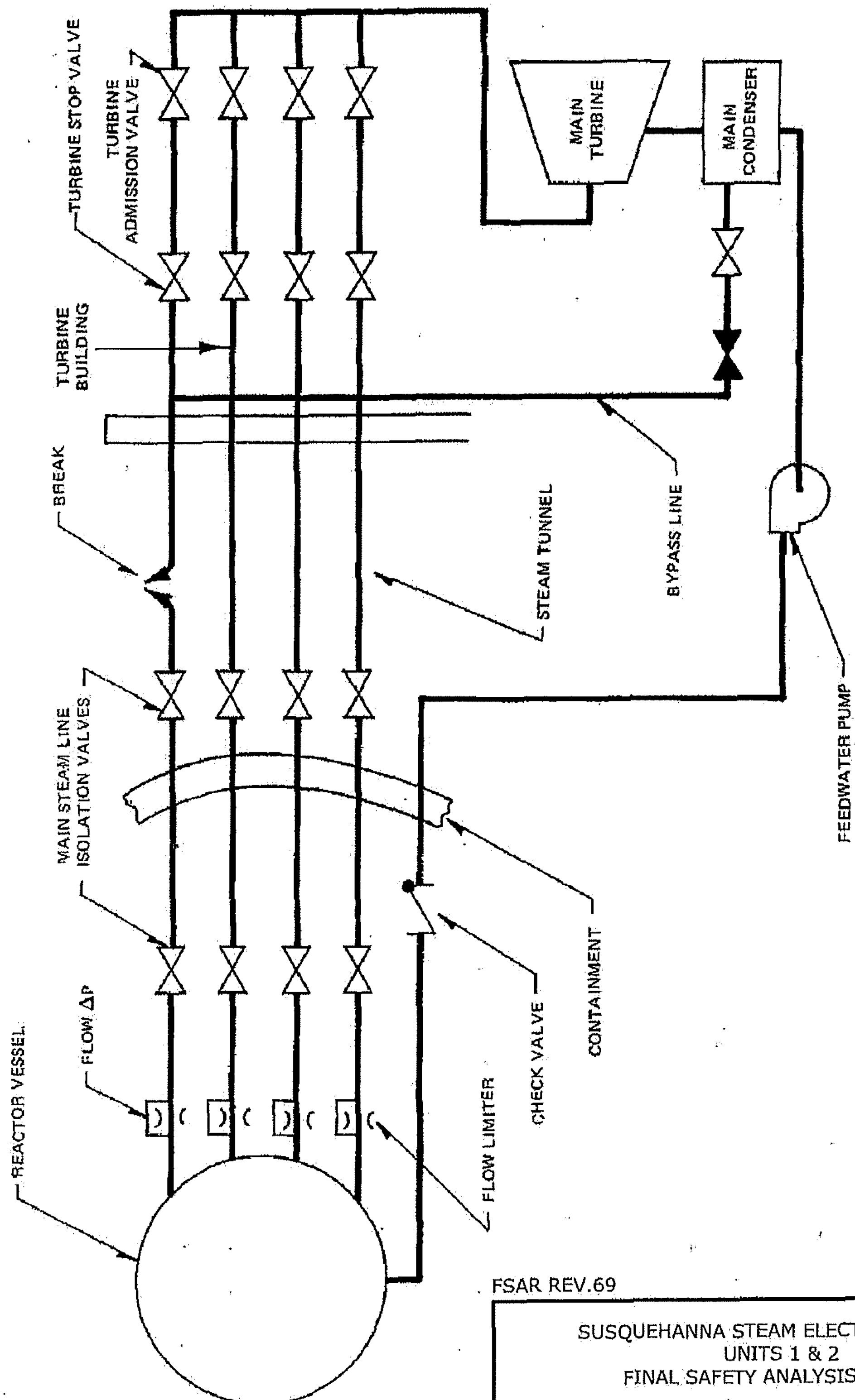
FSAR REV.65

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

LEAKAGE PATH
FOR
INSTRUMENT LINE BREAK

FIGURE 15.6-1, Rev 51

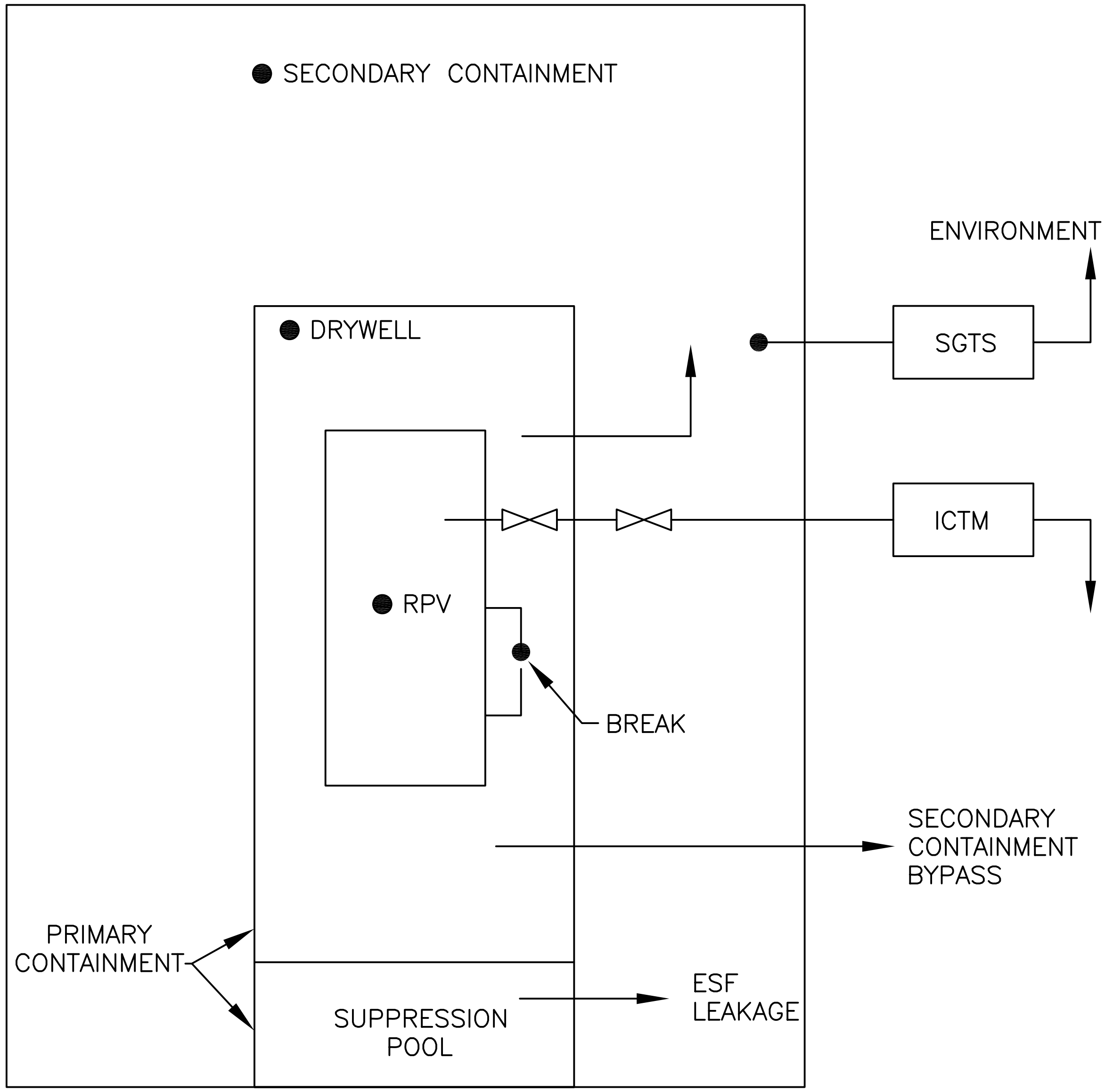
AutoCAD: Figure Fsar 15_6_1.dwg



FSAR REV.69

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
STEAM FLOW SCHEMATIC FOR STEAM BREAK OUTSIDE CONTAINMENT
FIGURE 15.6-2, Rev 50

AutoCAD: Figure Fsar 15_6_2.dwg



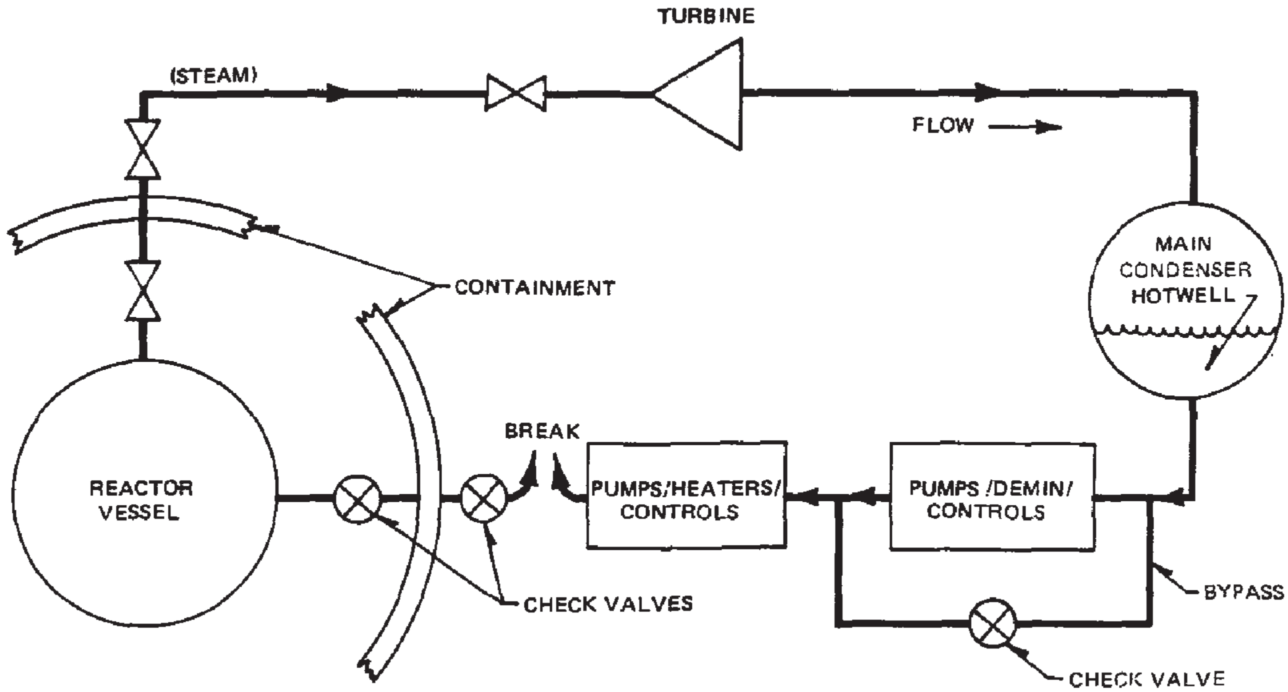
FSAR REV.65

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

LEAKAGE FLOW FOR LOCA

FIGURE 15.6-3, Rev 53

AutoCAD: Figure Fsar 15_6_3.dwg



FSAR REV.65

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

LEAKAGE PATH FOR
FEEDWATER LINE BREAK
OUTSIDE CONTAINMENT

FIGURE 15.6-4, Rev 49

AutoCAD: Figure Fsar 15_6_4.dwg