14.15 STEAM GENERATOR TUBE RUPTURE EVENT

14.15.1 IDENTIFICATION OF EVENT AND CAUSES

The SG is the interface heat exchanger between the RCS (primary) and the main steam system (secondary). The reactor coolant flows through tubes in the SG and transfers its heat to the feedwater on the shell side, thereby generating saturated steam. There are two SGs per reactor unit.

The Steam Generator Tube Rupture (SGTR) event is a penetration of the barrier between the RCS and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety, in that a leaking SG tube allows the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant would then mix with water in the secondary side of the affected SG. This radioactivity would be transported by steam to the turbine and then to the condenser, or directly to the condenser via the turbine bypass valves, or directly to the atmosphere via the MSSVs or the ADVs. Any noncondensible radioactive gases entering the condenser are removed by the condenser priming and air removal system and discharged to the plant vent.

Experience with nuclear SGs indicates that the probability of complete severance (double-ended break) of a tube is remote. The more probable modes of failure, which result in smaller penetrations, are those involving the occurrence of pinholes or small cracks in the tubes, and of cracks in the seal welds between the tubes and tube sheet. In the event of a SG tube leakage or rupture, the reactor coolant leaks into the secondary side of the SG. The reactor coolant transfer causes the level in the affected SG to increase and the pressurizer level to decrease, provided that the tube leak rate exceeds the capacity of the charging pumps. In the case of a double-ended tube rupture (design basis SGTR event), the leak rate far exceeds the charging pump capacities and, consequently, the pressurizer level will decrease. The decrease in the pressurizer level and the inability of the heaters to maintain pressurizer pressure causes the RCS pressure to decrease. The rate of RCS depressurization is determined by the leak rate, the charging flow rate and the pressurizer heater capacity. Furthermore, when the pressurizer level decreases to the point where the heaters would be uncovered, this mitigation for the pressure decrease is lost.

The drop in the RCS pressure will also initiate a reactor trip on the TM/LP pressure limit, ensuring that the SAFDLs are not exceeded. Following sufficient time for trip signal processing delays and decay of the CEA holding coil flux, the CEAs enter the core and add negative reactivity, which rapidly reduces the core fission power and heat generation rate, and causes the reactor coolant temperature to decrease.

At approximately the time of reactor trip, the pressurizer empties and the RCS pressure rapidly decreases to the hot leg saturation pressure. The decrease in RCS pressure will also initiate a SIAS. As the pressure drops below the HPSI pump shut-off head, SI flow is delivered to the core. The RCS pressure gradually increases following the initiation of the SIAS and the SI flow, and stabilizes at a pressure near that of the HPSI pump head. The SI flow offsets the coolant mass loss due to the ruptured tube, and results in slowing the depressurization of the RCS. Note that the larger the pressure difference between the primary and secondary, the larger the leak rate.

The SG pressure remains constant until the reactor trip on low pressurizer pressure occurs. The rapid closure of the turbine control and stop valves following turbine trip sharply reduces the secondary steam flow and causes a secondary pressure "spike" to occur. The quick opening of the steam dump and bypass control system (not credited in the safety analysis) following turbine trip, however, limits the magnitude of the secondary pressure spike, and gradually reduces the secondary pressure as the RCS residual heat reaches decay levels.

Based on available indications (i.e., reactor trip, pressurizer level indicators, SG level indicators, condenser off-gas radiation monitor, radiation monitors in the SG blowdown sample lines, SG level indicators, etc.), the operator can identify the nature of the event and manually isolate the SG with the ruptured tube. Once the isolation has occurred, the operator can initiate cooldown per the Emergency Operating Procedures (EOPs). During the cooldown period, the operator may steam the affected SG in order to prevent it from overfilling. The analysis credits backflow from the SG to the primary.

The objective of this analysis is to determine the maximum 0-2 hour EAB TEDE, the 30 day LPZ TEDE, and the 30 day Control Room TEDE which would result from a design basis SGTR event. Doses from this event must meet 10 CFR 50.67 and Reference 1 limits: a) below 10% of the 10 CFR 50.67 limits for the EAB and LPZ limits due to Concurrent Iodine Spike (CIS), b) below the 10 CFR 50.67 EAB and LPZ limits for the fuel damage and Preaccident Iodine Spike (PIS), and c) below the 10 CFR 50.67 Control Room limit for all SGTR events.

The SGTR event analysis accounts for SG tube plugging. Tube plugging reduces the heat transfer surface area and the flow area in the SG, which reduces RCS flow rate and lowers SG pressure. Tube plugging increases the activity release due to increased SG DP.

Isolation of an ADV may occur when an ADV begins to leak at an excessive rate. The ADV is isolated to prevent further leakage and damage to the valve. The SGTR event assumes that the ADV of the unaffected SG is isolated at the onset of the event. Thus, the initial plant cooldown is accomplished using the ADV of the affected SG only. The operator will be required to identify the blocked ADV, initiate actions to unblock the ADV of the intact SG, and isolate the affected SG to mitigate the release of radioactivity to the environment. After the operator isolates the affected SG, the operator will continue cooling down the RCS using the intact SG. The affected SG level will be maintained by using backflow to the RCS. The operator continues the cooldown until the shutdown entry conditions are reached.

The use of the affected ADV in this analysis is for the purpose of maximizing the radiological releases during the event.

14.15.2 SEQUENCE OF EVENTS AND SYSTEMS OPERATION

The sequence of events for a typical limiting case is presented in Table 14.15-2. Several cases were analyzed to examine the effect of time of reactor trip, initial SG pressure, AFW actuation and flow, subcooling, plugged tubes, and cooldown rate on radiological dose consequences. The results, in most cases, did not differ significantly and the sequence of events for the presented case utilizes several assumptions regarding system operation that are chosen to maximize the radiological doses. The operator actions assumed in the analysis are consistent with EOPs.

The analysis assumed a loss of forced circulation following the reactor trip, which results in higher hot leg temperature, higher fraction of the leak flow flashing into the affected SG, slower cooldown and RCS depressurization, and reduces the capability to cool down the plant via the unaffected SG. All of these effects result in higher doses.

No credit was taken in the analysis for operation of the turbine bypass valves to the condenser. All of the steam releases are assumed to be directly to the atmosphere via the MSSVs or the ADVs.

The SG blowdown is assumed to be unavailable for level control.

The analysis assumed the lowest allowed opening setpoint for the MSSVs to maximize their releases to the atmosphere. Furthermore, minimum AFW flow was assumed based on the automatic action of the AFAS, which maximizes SG pressures and ADV releases to the atmosphere during the post-trip period prior to operator action.

The ADV of the unaffected or intact SG is isolated at the onset of the event. Therefore, initially, all of the heat removal is through the ADV of the affected SG. Also, the unblocking of the isolated ADV may comprise up to a 2 hour delay as personnel need to access the manual control station which is outside the Control Room or manually operate the ADV using the handweel.

The operator actions assumed in this analysis are consistent with the Calvert Cliffs EOPs. The first operator action is assumed at 15 minutes following the reactor trip. Subsequently, a time delay of two minutes between each discrete operator action is assumed. The major post-trip EOP analysis assumptions regarding operator actions are:

1. Operate the ADV on the affected SG: 15 minutes after reactor trip, the operator takes manual control of the ADV on the affected SG to prevent further cycling of the MSSVs.
2. Take manual control of the AFW to the SGs: Two minutes after opening the affected SG ADV, the operator takes manual control of the AFW flow to each SG, with flow initially delivered to both SGs.
3. Stabilize the plant and maintain cold leg temperature: The operator quickly diagnoses the event and stabilizies the RCS to a temperature which precludes a challenge to the MSSVs using the SG ADVs and AFW. The length of the stabilization period is assumed to be no more than 10 minutes from the time that the operator takes manual control of the ADVs. As a result of this diagnosis, the operator initiates action to unisolate the ADV of the intact SG, which is assumed to be isolated at this time. The actions may take up to 1 hour after taking control.
4. Cool the RCS before isolating the Affected SG: After the stabilization period, the operator begin to cool the RCS at a rate of up to 100°F/hr to maximum steam releases.
5. Isolate the Affected SG: The operator isolates the affected SG when T_{HOT} is less than 515°F (including uncertainties). The analysis assumes no opening of the ADV or MSSVs of the affected SG after 2 hours. However, the ADV of the affected SG may be opened 24 hours into the accident to hasten shutdown.
6. Plant cooldown after isolation of the affected SG: Following the isolation of the affected SG, the operator cools down the plant using the ADV on the intact SG at a maximum of 35°F/hr to maximize steam releases.
7. Maintain SG pressure and level: The pressure and level of the affected SG will initially be controlled by steaming to atmosphere for up to 2 hours. In addition, the RCS will be aggressively cooled down to achieve backflow from the affected SG as early in the event as possible.
8. Maintain subcooling margin during the event: A target subcooling margin of 50°F is maintained by the operator. This value consists of 25°F required by the EOPs and 25°F of core exit thermocouple uncertainty.
9. Maintain pressurizer level: The pressurizer level is maintained by controlling safety injection flow. In addition, the RCS is aggressively cooled down to achieve backflow from the affected SG as early in the event as possible.
10. Pressurizer control actions and control systems: The operator uses the HPSI system and the pressurizer vent (or auxiliary spray) to control RCS inventory and subcooling.

The combination of the assumed cooldown rate and the high subcooling margin including instrument uncertainties result in a conservatively slow depressurization of the RCS, which maximizes the tube leakage. The increased leak rate raises the final activity level released through the affected SG. It also leads to a high liquid level in the SG early in the event resulting in the opening of the affected SG ADV and more frequent releases to the environment. However, at 2 hours into the event, the affected SG is completely isolated. Thus, the affected SG level is maintained by using backflow to the RCS. The ADV steaming is increased by the assumption of a lower actual SG level to accommodate instrument uncertainties.

Together, these assumptions, in combination with the radiological assumptions presented in Section 14.15.3.2, assure that the radiological dose results from the analysis conservatively bound the expected doses for this event.

14.15.3 ANALYSIS OF EFFECTS AND CONSEQUENCES

14.15.3.1 Core and System Performance

A. Mathematical Models

The thermal hydraulic response of the NSSS to the SGTR was simulated using the Reference 6 computer program up to the time the operator takes control of the plant (15 minutes after trip). Operator actions to mitigate the effects of the SGTR event and bring the plant to shutdown cooling entry conditions were simulated using a CESEC-based cooldown algorithm, referred to as the COOL code.

B. Input Parameters and Initial Conditions

The input parameters and initial conditions used in the analysis are listed in Table 14.15-1 for the present cycles of Unit 1 and Unit 2. The selected values of these inputs maximize the radiological releases to the atmosphere during the transient.

The maximum allowed Technical Specification core inlet temperature, including instrument uncertainties, results in a correspondingly high initial SG pressure. This increases the steam released through the MSSVs and the ADVs throughout the event.

The minimum core flow results in higher than average coolant temperature and higher enthalpy fluid entering the SG, a resultant increase in flashing fraction, and higher activity releases through the MSSVs and ADVs.

A maximum initial pressure and a maximum initial pressurizer liquid volume, delay the reactor trip. Delaying reactor trip is conservative because it increases the amount of heat to be removed and increases steam releases.

The SG level is maintained within a small range during operation, the limits of which would have no effect on the trip time and insignificant effect on the AFW actuation time.

The analysis assumed the lowest allowed opening setpoint for the MSSVs to maximize their releases to the atmosphere.

The selection of fuel and moderator temperature coefficients are not significant, as there is no change in the core power or temperature prior to reactor trip. The TM/LP trip uncertainty is applied to lower the setpoint to delay trip. Three HPSI pumps are assumed to be started on SIAS, thus maximizing the flow delivered to the RCS upon SIAS. These assumptions result in higher post-trip RCS pressures, and maximize the tube leakage.

The radiological consequences of the SGTR event are also dependent on the break size. As the break size is decreased from that of a double-ended rupture, the integral leak is reduced and the radiological consequences will be less severe. Therefore, the most adverse break size is the largest assumed break of a full double-ended rupture of a SG tube.

C. Results

Table 14.15-2 presents the sequence of events for the double-ended rupture of a SG tube event with the loss of forced circulation upon reactor trip. Figures 14.15-1 through 14.15-16 present the dynamic behavior of important NSSS parameters during this event. The only scenario presented is the one that assumes isolation of the affected SG 2 hours into the transient while maintaining the highest subcooling possible by accounting for core exit thermocouple uncertainty.

The sequence of events and NSSS response plots are based on the RSG configuration. Reference 7 assessed the impact of increased SI flow input to the Steam Generator Tube Rupture event. The increased SI flow case results in higher RCS Pressure. The higher RCS pressure case is less challenging than the base case presented in Table 14.15-2. Appendix C of Reference 7 provides figures for comparison to the base case.

The double-ended break of a SG tube results in a primary-to-secondary leak rate which exceeds the capacity of the charging pumps. As a result, pressurizer level and pressure gradually decrease from their initial values. For the case discussed here, maximum charging flow and zero letdown was assumed to delay the time of reactor trip. As the pressure decreases, the proportional heaters and then backup heaters are turned on to prevent further depressurization. All heaters are turned off automatically as the pressurizer level is decreasing to levels which result in uncovery of the heaters. The depressurization of the RCS and pressurizer level decrease continue, resulting in an approach to DNB SAFDL. The TM/LP trip is designed to trip the reactor before the DNB SAFDL is reached. The analysis of the SGTR event demonstrates that the action of the TM/LP trip prevents the DNB SAFDL from being exceeded, since the rate of depressurization for this event is less than the rate of depressurization for the RCS Depressurization event. The analysis credits a reactor trip only when the low pressurizer pressure floor of the TM/LP trip is reached. The loss of forced circulation (RCP pumps tripping) is assumed to occur 3 seconds after the trip breakers are opened, resulting in the initiation of the RCS flow coastdown.

The analysis also assumes the steam bypass system to the condenser will become unavailable and that the unaffected SG ADV is blocked for 60 minutes into cooldown. The affected SG ADV automatically opens at trip time and then modulates on a program based on RCS average temperature. The turbine valve closure due to the reactor trip causes the SG pressures to rise, and leads to the opening of the MSSVs. They reopen and close several times during the period until the operator takes action to cool the plant.

The loss of forced circulation and the RCS flow coastdown result in reduction of flow into the upper head region of the reactor vessel. This region becomes thermal-hydraulically decoupled from the rest of the RCS, and due to flashing caused by the depressurization and boiloff from the metal structure to coolant heat transfer, voids begin to form in this region.

The pressurizer empties due to the continued primary-to-secondary leak and the post-trip RCS liquid shrinkage. The continued RCS and pressurizer depressurization results in SIAS generation and delivery of the HPSI flow to the RCS when the RCS pressure decreases below the HPSI pump head.

The AFW actuation setpoint is reached in the unaffected SG and the AFW is delivered to both SGs following system and piping delays.

Fifteen minutes following the trip, the operator takes manual control of the plant, which consists of manual control of ADVs, AFW and HPSI. The analysis of the limiting case assumes that at this point the operator has diagnosed the event.

Following the diagnosis, the operators begin to cooldown the RCS at approximately 100°F/hr, using the ADV on the affected SG and the AFW system until the hot leg temperature of the affected loop reaches an isolation temperature of 493.21°F (515°F per EOPs minus 21.79°F uncertainty).

14.15.3.2 Radiological Consequences

The limiting SGTR event as re-analyzed by Reference 4 is considered to be a complete double-ended tube break. The SGTR event allows primary coolant to leak into the secondary side via the SG. In the case of the double-ended tube rupture, the leak rate far exceeds the charging pump capacities and, consequently, the pressurizer level decreases. The decrease in the pressurizer level and the inability of the heaters to maintain pressurizer pressure causes the RCS pressure to decrease. The drop in the pressure will cause a rector trip on TM/LP, ensuring that the DNB SAFDL is not exceeded. Peak linear heat rate is of no concern because there is no appreciable power increase during the transient. Thus, no fuel damage is postulated to occur during this event. The reactor trip also generates a turbine trip causing the secondary pressure to rapidly increase due to closure of the turbine valve. In the assumed evolution, the turbine bypass valves are not available to mitigate the rise in secondary pressure. The action of the ADVs and MSSVs will limit the secondary pressure until the operator is able to assume control. After the operator identifies the event, the operator initiates a cooldown of the RCS. In this analysis, the ADV of the intact SG is assumed to be isolated at the beginning of the event for up to 2 hours. Thus, this initial cooldown is carried out using the ADV of the affected SG only. After 2 hours, the operator isolates the affected SG and continues cooling down the RCS using the intact SG. The affected SG level will be maintained by using backflow to the RCS. The operator continues the cooldown via the ADV of the unaffected SG until the SDC entry conditions are reached. A 30 day cooldown via the ADV of the unaffected SG is conservatively assumed. Note that the operators can reopen the ADV of the affected SG for up to 8 hours after an initial cooldown of 24 hours post-accident to attain SDC in 32 hours post-accident.

Reference 7 assessed the impact of increased SI flow input to the Steam Generator Tube Rupture event. The analysis updates the mass and mass release associated with the RCS, Affected Steam Generator (ASG), and the Unaffected Steam Generator (USG) used in the SGTR dose analysis. The RCS, ASG, and USG masses are noted in Table 14.15-3.

The AST methodology of 10 CFR 50.67 and Reference 1 is used to calculate offsite and Control Room doses for a SGTR event. If no or minimal fuel damage is postulated, the activity is the maximum coolant activity allowed by the Technical Specifications, assuming 2 cases of iodine spiking. The PIS case assumes that a reactor transient has occurred prior to the postulated SGTR event and has raised the primary coolant iodine concentration to the maximum value permitted by the Technical Specifications, 30 μCi/gm. The CIS case assumes that the transient associated with the SGTR event causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes the iodine release rate from the fuel rods to the primary coolant increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value with an 8 hour duration.

A. Assumptions and Conditions

The assumptions and parameters employed for the evaluation of radiological releases are:

(1) CIS doses are calculated assuming that the iodine release rate from the fuel rods to the primary coolant increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (0.5 μCi/gm DEQ I-131 activity). The primary CIS activities are released homogeneously into the primary system over the 8 hour duration of the CIS spike.

(2) PIS doses are calculated assuming that a reactor transient has occurred prior to the postulated SGTR and has raised the primary coolant iodine concentration to the maximum value permitted by the Technical Specifications: 30 μCi/gm. The primary PIS activities are assumed to be homogeneously distributed throughout the primary system at the beginning of the accident.

(3) The specific activity of the primary coolant is assumed to be 100/E μCi/gm noble gas per Technical Specifications.

(4) An initial DEQ I-131 secondary activity of 0.1 μCi/gm is assumed (Technical Specification limit). The secondary activities are assumed to be homogeneously distributed throughout the secondary system at the beginning of the accident.

(5) The dose conversion factors were extracted from References 2 and 3.

(6) The iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic.

(7) The main Control Room inleakage points include the West Road inlets, the Turbine Building, and Access Control Units 11 and 13 on the Auxiliary Building roof. Installation of automatic isolation dampers and radiation monitors at Access Control Units 11 and 13 on the Auxiliary Building roof were credited.

- A Control Room inleakage rate of 3500 cfm was based on measured inleakage measurements.
- Control Room recirculation filtration is credited assuming 10,000 ± 10% cfm flow at 90% filter efficiency for elemental and organic iodine and 99% for particulates with a 20 minute delay time.
- 0-8, 8-24, and 24-720 hour breathing rates of 3.5E-04, 1.8E-04, and 2.3E-04 m^3/sec are assumed.
- 0-24, 24-96, and 96-720 hour Control Room occupancy factors of 1.0, 0.6, and 0.4 are assumed.

(8) The primary to secondary ruptured tube leakage and Technical Specification leakage of 200 gpd are assumed to continue until SDC conditions defined as 300°F and 270 psia are attained and releases from the SGs have been terminated. Per Reference 1, the Technical Specification leakage should be apportioned between affected and unaffected SGs in such a manner that the calculated dose is maximized. Thus, since the primary to secondary flow from the RCS to the affected SG was maximized in Reference 4 for the worst-case thermal-hydraulic conditions, all of the Technical Specification primary to secondary leakage is assumed to flow to the unaffected SG.

(9) The portion of the primary fluid leaking into the SG that flashes into steam is dependent on the enthalpy of the primary liquid and the saturation enthalpy of the SG. When there is a steam release to the atmosphere, the flashed portion is released before the steam in the SG. The flashing portion has a decontamination factor of 1.0. The non-flashing portion of the primary leak flow is assumed to mix uniformly with the liquid in the SG.

(10) The SG is assumed to have a decontamination factor of 100, so that the concentration of radioactivity in the steam phase is 1/100 of the concentration in the liquid phase.

(11) MOVs on the ASG drain lines are assumed to be open for the duration of the SGTR accident.

Additional inputs and assumptions are detailed in Table 14.15-3.

B. Mathematical Model

The behavior of the primary and secondary systems during and after a double-ended tube break SGTR event was modeled by Reference 4. The CESEC-III NSSS simulation code was used to model the SGTR for primary and secondary response during the initial portion of the event. However, CESEC-III does not have the capability to model the multiple operator actions credited in the SGTR event. Thus, the remainder of the event was simulated using the COOL-II code, which can model explicit operator actions. The COOL-II Code is a thermal-hydraulic code that simulates the plant cooldown by operator actions based upon the Calvert Cliffs EOPs. Because the COOL-II code does not have a kinetics model, CESEC-III is run to approximately 15 minutes past reactor trip to ensure all power being generated is from decay heat and a conservative decay heat curve is input to COOL-II. The system masses and mass releases were updated using the CESEC-III results from Reference 7. For the masses and mass releases specific to COOL, the existing COOL results were used or updated based on the CESEC results.

The SGTR occurs at a time t=0 with the PIS primary activity and the Technical Specification secondary activity uniformly distributed throughout their respective systems. The SGTR occurs at a time t=0 with the Technical Specification secondary activity uniformly distributed throughout the secondary system and with the CIS primary activity released homogeneously into the primary system over an 8 hour duration. The primary noble gases are released at a 200 gpd rate into the unaffected SG and at the time-dependent tube rupture leak rate into the affected SG and then directly through the ADVs and MSSVs into the environment, when the ADVs and MSSVs are in the open position. The primary iodines are released at a 200 gpd rate into the unaffected SG and at the time-dependent tube rupture leak rate into the affected SG, where a percentage is vented directly through the ADVs and MSSVs into the environment via flashing. The remaining iodines are added to the secondary system, which is released by steaming with a partition factor of 100 out of the ADVs, when the ADVs and MSSVs are in the open position. No cleanup mechanisms (spray, filtration, plateout) are assumed in the primary or secondary systems. The activity released to the environment is transported to the site boundary and to the Control Room via appropriate atmospheric dispersion coefficients. Control Room filtration is credited in this analysis. The Control Room and site boundary doses are calculated based on appropriate breathing rates and occupancy factors and on References 2 and 3 dose conversion factors.

The Control Room and offsite doses are calculated for the SGTR event based on the AST methodology of Reference 1. This was accomplished by utilizing the RADTRAD computer transport code. The RADTRAD computer code calculates TEDE and thyroid doses to personnel at the site boundary, low population zone, and Control Room per 10 CFR 50.67 resulting from any postulated accident which releases radioactivity within any primary or secondary system. RADTRAD models the transport of up to 63 radionuclides from the source region, through a secondary region, and then to the environment and to the Control Room. The code includes the capability to model time-dependent activity release; time-dependent spray/filtration/deposition removal processes, piping/filter/inleakage transfer mechanisms, atmospheric dispersion; and natural decay.

C. Results

The EAB, LPZ, and Control Room doses for the design-basis CIS and PIS SGTR event for the two cooldown modes described previously are detailed in the following table:

SGTR Event Results

	EAB Rem	**LPZ Rem**	**Control Room Rem**
CIS			
Unaffected ADV 0-30 days	0.2110	0.0505	1.7592
Affected ADV 0-2/24-32 hr	0.2107	0.0499	1.7796
Regulatory Limits	2.5	2.5	5
PIS			
Unaffected ADV 0-30 days	0.4806	0.1134	3.8747
Affected ADV 0-2/24-32 hr	0.4805	0.1136	3.9808
Regulatory Limits	25	25	5

Note that all values are below the regulatory limits.

14.15.4 CONCLUSION

The analysis of the SGTR event demonstrates that the action of the TM/LP trip prevents the DNB SAFDL from being exceeded. All doses are within 10 CFR 50.67 and Reference 1 limits, as approved by Reference 5 and updated in Reference 8.

This event is not affected by the transition to Advanced CE-14 HTP™ fuel because the key parameters for this event are plant related system responses which are unchanged from, or bounded by, the current analysis.

14.15.5 REFERENCES

1. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000
2. Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” September 1988
3. Federal Guidance Report (FGR) 12, “External Exposure to Radionuclides in Air, Water, and Soil,” September 1993
4. CA06595, Westinghouse Calculation CN-TAS-05-13, Revision 000, Calvert Cliffs Units 1 & 2 Steam Generator Tube Rupture Event
5. Letter from D. V. Pickett (NRC) to J. A. Spina (CCNPP), “Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Implementation of Alternative Radiological Source Term (TAC Nos. MC8845 and MC8846),” dated August 29, 2007
6. CESEC-III, Mod 5 computer program (ABB Topical Report “CESEC, Digital Simulation of a Combustion Engineering Nuclear Steam Supply System” Enclosure 1-P to LD-82-001, December, 1981
7. ECP-11-000465-CN-001, CAL ECN CA06595 CCN 1 INCREASESI FLOW FOR CA07350, REVISION 2 MAX HPSI INCREASE
8. CA09981 rev. 0, AST SGTR Analysis with MOV Leakage and Increased Safety Injection Flow

TABLE 14.15-1

INITIAL CONDITIONS AND INPUT PARAMETERS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

PARAMETER	UNITS	UNIT 1[a]	UNIT 2[a]
Core Power	MWt	2754	2754
T_{in}	°F	550	550
RCS Pressure	psia	2286	2286
SG Tubes Plugged		2500	2500
Core Mass Flow Rate	$x10^6$ lbm/hr	134.0	134.0
Secondary Pressure	psia	890.5	890.5
Tube ID	inches	0.654	0.654
Pressurizer Liquid Level at Full Power	ft^3	952	952
Low Pressurizer Pressure (TM/LP Floor) Setpoint	psia	1829	1829
Safety Injection Actuation (SIAS) Setpoint	psia	1765	1765

[a] These values represent inputs to the limiting transient scenario analyzed for each unit. In general, a range of initial conditions and input parameters, including uncertainties, were evaluated to determine the limiting case.

TABLE 14.15-2

SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT

TIME	EVENT	SETPOINT OR VALUE
0.0	Tube Rupture Occurs	---
8.4	Proportional Pressurizer Heaters Setpoint Reached, psia	2275
66.4	Backup Pressurizer Heaters Setpoint Reached, psia	2200
348.4	Pressurizer Heaters De-energize due to Low Pressurizer Level, ft^3	270
417.8	Low Pressurizer Pressure Trip Analysis Setpoint is Reached, psia	1829
418.7	Trip Breakers Open	---
	ADVs Open, °F	535
420.8	MSSVs Open, psia	935
421.7	Loss of Forced Circulation, RCPs Begin to Cost Down	
426.2	Maximum SG Pressure is Reached, psia	986
430.9	SIAS Setpoint is Reached, psia	1765
438.3	Pressurizer Empties	---
456.7	MSSVs Close, psia	878
	The MSSVs subsequently cycle repeatedly	
478.4	Safety Injection Flow Begins to Enter the RCS, psia	1351
749.9	AFW Actuation Setpoint is Reached Unaffected SG	204” BNL
1018.5	AFW is Initiated to Unaffected SG	100 gpm
1318.7	Operator Takes Manual Control of the Plant and Begins Cooldown at Rate of 100°F/hr by Adjusting the ADVs on the affected SG	---
1438.7	AFW Increase to Both SGs (2 minutes past takeover time)	200 gpm/SG
1800	Operator Opens the Pressurizer Vent	
2270	Hot Leg Reaches Isolation Temperature, °F	493.21
2280	Adequate Pressurizer Level, Inches (Operator Begins to Throttle HPSIs)	101
5040	Operator Unblocks ADV of Intact SG	

TABLE 14.15-3

ASSUMPTIONS FOR RADIOLOGICAL CONSEQUENCES OF THE STEAM GENERATOR TUBE RUPTURE EVENT

PARAMETER	DESIGN BASIS ASSUMPTION
RCS Mass, lbm	391900
ASG Mass, lbm	126700
USG Mass, lbm	60630
Primary system activity:	
Pre-existing iodine spike (PIS), μCi/gm	30
Event GIS, μCi/gm	0.5
Spiking factor	335
Secondary system activity, μCi/gm	0.1
Primary-to-secondary leak rate in the unaffected SG, gpd	200
EAB Atmospheric Dispersion factor (X/Q) sec/m^3, 0 - 2 hr	1.44×10^{-4}
LPZ Atmospheric Dispersion Factor (X/Q), sec/m^3	
0 - 2 hr	3.39×10^{-5}
2 - 24 hr	2.2×10^{-6}
24 - 720 hr	5.4×10^{-7}
Decontamination factor between the water and steam phases in the SGs	100
Breathing rate, m^3/sec	
0 - 8 hr	3.5×10^{-4}
8 - 24 hr	1.8×10^{-4}
24 - 720 hr	2.3×10^{-4}
Control Room Atmospheric Dispersion Factor (χ/Q), sec/m^3	
0 - 2 hr	3.83×10^{-3}
2 - 8 hr	3.25×10^{-3}
8 - 24 hr	1.32×10^{-3}
1 - 4 days	9.92×10^{-4}
4 - 30 days	7.92×10^{-4}