14.14 STEAM LINE BREAK EVENT

14.14.1 IDENTIFICATION OF EVENT AND CAUSE

The Main Steam System carries steam from the SGs to the turbine-generators and to other auxiliary equipment. An SLB may occur as a result of thermal stress or cracking in the main steam line. The guillotine-type break assumed in the Safety Analysis is the most adverse transient scenario and its probability of occurrence is extremely low.

A rupture in the Main Steam System increases the rate of heat extraction by the SGs and causes cooldown of the reactor coolant. With a negative moderator coefficient of reactivity, the cooldown will produce a positive reactivity addition. A severe decrease in main steam pressure will cause the MSIVs in the main steam lines to trip closed (Chapter 10). The MSIV is a gas hydraulic, bi-directional, balanced disk, Y-pattern, globe valve designed to hold pressure from either direction. If the steam line rupture occurs between the SG and the isolation valve, blowdown of the affected SG would continue. However, termination of flow from the intact SG occurs with closure of both isolation valves, either of which is capable of stopping flow. A detailed description of the MSIV verification program is provided in Section 14.14.2.

The SLB transient was analyzed in two distinct portions, referred to as pre-trip and post-trip. For the pre-trip portion of the event, the main concern is the power excursion seen due to the cooldown in combination with a negative MTC. A loss of power on reactor trip is also assumed. A parametric analysis of break size and MTC was performed to determine the limiting case with respect to the DNB SAFDL. Cases run for an inside Containment break credited high containment pressure. Cases run for an outside Containment break credited low SG pressure, high power-NI, and delta-T power trips.

For the post-trip portion of the event, the concern is a return to power in the vicinity of an assumed stuck rod. Full load and no load cases, with and without a loss of power, were analyzed to find the limiting cases with respect to the DNB and LHR SAFDL. A guillotine break of the main steam line was assumed, as this provides the largest cooldown of the RCS and the greatest potential for a post-trip return to power.

The one-loop full load and one-loop no load cases were not analyzed since Technical Specifications prohibit operation in these modes. Since the SLB event is classified as a postulated accident, site boundary doses must not exceed the 10 CFR 50.67 guidelines. Since the SGs are designed to withstand coolant system operating pressure on the tube side with atmospheric pressure on the shell side, the continued integrity of the RCS barrier is assured. The most limiting of the pre- and post-trip events are summarized in Section 14.14.4.3.

14.14.2 DISCUSSION OF MAIN STEAM ISOLATION VALVE TESTING

The MSIVs have been designed to close in less than six seconds under the pressure, temperature, and flow conditions applicable to the assumed accident. The design techniques used to ensure that the MSIVs will function reliably under accident conditions have been extensively verified by testing different valves that employ similar principles under conditions of pressure, temperature, and flow similar to those applicable to the assumed accident.

The testing under accident conditions has been performed on Y-type balanced disc valves. Construction of the MSIVs is described in UFSAR Chapter 10. The balanced disc valves which have been tested differ from the Calvert Cliffs MSIVs as follows:

1. The balanced disk valves are actuated by air and springs. They are only capable of closing when line pressure forces the valves shut because the actuators do not exert sufficient force to close the valves against full reverse pressure. The Calvert

Cliffs balanced disk MSIVs employ a hydraulic cylinder directly coupled to a nitrogen accumulator that is capable of closing the valves with pressure from either side. They are arranged in a similar manner to the tested balanced disk valves so that normal flow tends to close the valves. Testing has been performed on the Calvert Cliffs actuators showing they deliver the required force to close the valve under the maximum calculated (1085 psi) reverse flow force.

2. The balanced disk valves utilize hydraulic dash-pots to control the disk speed during the closure. Flow control valves throttle the flow of oil from one side of the piston to the other, effectively damping the motion and preventing steam flow from slamming the valves closed. The Calvert Cliffs MSIVs use automatic adjusting orifice and a fixed orifice to control valve speed during closure. The orifices provide a constant oil flow rate independent of upstream pressure. These orifices throttle the oil from below the piston to the oil reservoir.

3. The balanced disk valves have pilot assemblies in the disk which equalize pressure across the valves for opening. The Calvert Cliffs MSIVs have a check disk assembly in the center of the main valve disk. This equalizes steam pressures above and below the disk to balance the forces on the disk. This design reduces the forces necessary to close the valve during reverse flow conditions.

Testing of balanced disk valves under conditions similar to those existing in the assumed accident has verified that the Y-type valve configuration shuts reliably and does not bind.

The Calvert Cliffs MSIVs incorporate a bidirectional balanced disk design. This design balances the forces above and below the disk, which reduces the forces necessary to close the valve during reverse flow conditions. The maximum force required to close the valve under full reverse flow conditions has been calculated. The MSIV actuator was tested on a load cell showing that it can deliver the required force to close the valve in less than six seconds. Therefore, the actuator will function and close the valve quickly against reverse flow.

The manufacturer of the Calvert Cliffs MSIVs has tested the closure of his 16" American National Standards Institute Class 600 air-actuated balanced disk valve under rupture conditions using 1500 psig air as the flowing medium. Mass flow rates were typically 1800 pounds per second during closure. Closure occurred in as little as 2-1/2 seconds, and typically before the pressure had decayed below 1200 psig. Closure time under rupture conditions was within 10% of closure time under no-flow conditions using pressure compensated flow control valves for speed control. The Calvert Cliffs MSIVs are of 32" bore size and would experience mass flow rates of about 3330 pounds per second at the beginning of an accident. The variation in closing time between no-flow and rupture-flow conditions is less than 0.8 seconds.

This number has been verified by load testing of the actuators. This 0.8 seconds difference was applied to the accident analysis valve closure time (less than 6.0 seconds at that time), resulting in the Technical Specifications valve closure limit of less than 5.2 seconds. The accident analysis valve closure time has since been increased to 7.0 seconds to incorporate margin in the analysis. The manufacturer has also functionally tested balanced disk valves for 20,000 operations containing 575°F saturated steam under no flow conditions. In all cases the valves functioned reliably and the design techniques were verified.

In addition, a 20" balanced disk valve of similar construction, but made by a different manufacturer, has been tested for closure by General Electric Company and Commonwealth Edison Company (Reference 1). In this test, boiler steam of varying moisture contents was used.

After replacement of the MSIV actuator and modification of the valve internals from a stop valve to a bidirectional balanced disk, the valve was operated through at least four complete cycles with no flow. The correlation between test closure time and accident closure time is discussed in Chapter 10.

Periodic testing after installation is discussed in Chapter 10.

14.14.3 SEQUENCE OF EVENTS

The rupture of a main steam line will cause the affected SGs pressure and temperature to rapidly decrease. The steam released through the break in the affected SG extracts heat from the primary side which causes the primary coolant temperatures and pressure to rapidly decrease. The decrease in the primary coolant temperature, in combination with a negative MTC, results in positive reactivity addition which causes the core power level and core heat flux to increase.

The uncontrolled blowdown of the affected SG will initiate one of the following reactor trips: high containment pressure, high power, delta-T power, low SG pressure, or TM/LP. The power increase will be terminated after the CEAs are inserted. The drop in the SG pressure will also initiate a SGIS. Following appropriate delays, the MSIVs on both the affected and on the unaffected SGs will close and terminate the blowdown from the unaffected SG.

The blowdown from the affected SG will continue since the MSIV is downstream from the break area. The continued blowdown causes the RCS pressure to decrease until SIAS is initiated, which automatically starts the HPSI Pumps. Since the shutoff head of the HPSI Pumps is approximately 1200 psia, no safety injection (SI) flow is delivered immediately. Therefore, the pressure continues to decrease and the pressurizer empties.

The conservative assumption for the pre-trip portion of the event is that a LOOP after a turbine trip results in a coast down of the RCPs, which causes the core flow to decrease. For the post-trip portion of the event, both the LOOP and no LOOP cases are considered with respect to the return to power potential.

The cooldown of the RCS will insert positive reactivity from moderator and fuel temperature feedbacks. This positive reactivity addition will erode the negative reactivity added by the CEAs. The magnitude of core subcriticality depends on the SCRAM worth and the moderator and fuel temperature reactivity feedbacks. Since it is expected that all CEAs will be inserted after a reactor trip signal, the negative reactivity addition from the inserted CEAs will be sufficient to offset the positive reactivity inserted by the moderator and fuel temperature feedbacks. Consequently, the core will remain sufficiently subcritical to prevent any power increase over decay power levels if all CEAs are inserted.

The cooldown of the RCS is terminated when the affected SG blows dry and AFW flow is isolated to the ruptured SG. As the coolant temperature decreases, additional positive reactivity insertion occurs from the moderator reactivity feedback. However, the insertion of additional positive reactivity is not sufficient to significantly erode the negative reactivity inserted by the CEAs following the trip. In addition, the HPSI Pump flow will insert negative reactivity, which also ensures that the core remains subcritical during the event.

The HPSI pump flow will add sufficient coolant mass such that the pressurizer level will be reestablished and the RCS pressure will be maintained. The cooldown of the RCS is terminated once action is taken to terminate the AFW flow to the affected SG.

In summary, during this event various reactivity feedback mechanisms are observed. The various components of reactivity are summarized below.

1. Moderator Reactivity - The decrease in primary coolant temperatures, in combination with an effective negative MTC, inserts positive reactivity. The positive reactivity insertion continues until the coolant temperatures increase in response to the production of instantaneous fission power and the eventual dryout of the ruptured SG. The resulting increase in coolant temperature reduces the reactivity inserted during the moderator cooldown. This trend continues until the coolant temperatures once again begin to decrease due to the cooling action of AFW flow delivered to the ruptured SG. The decrease in coolant temperatures causes an increase in core reactivity.
2. Doppler Reactivity - The fuel temperatures rapidly decrease after reactor trip (i.e., from HFP to HZP temperatures) which leads to a significant amount of positive reactivity insertion. Thereafter, the fuel temperatures gradually decrease and the positive reactivity insertion continues although at a reduced rate.
3. Boron Reactivity - The decrease in pressure initiates SIAS, which causes the startup of the HPSI pumps. After the pumps reach full speed and the pressure drops below the shutoff head of the pumps, HPSI flow is delivered to the core. The boron injected via the HPSI pumps inserts negative reactivity.

   It is conservatively assumed that the RCS is initially assumed to be at zero ppm boron, for both the HFP and HZP post-trip SLB analyses. Assuming no boron in the RCS negates the negative reactivity insertion that would occur due to concentrating the boron during the cooldown. The charging pumps are not modeled.
4. SCRAM Worth - The reactor trip on low SG pressure causes the CEAs to drop into the core and insert negative reactivity. In the analysis, the temperature dependence of SCRAM worth has been included in the moderator temperature defect.
5. Total Reactivity - The total reactivity is the sum of each individual reactivity component. The total reactivity initially is very negative following reactor trip when the CEAs are fully inserted into the core. The insertion of positive reactivity from moderator and Doppler feedback causes the core to approach criticality. The total reactivity decreases as the moderator reactivity component decreases and boron enters the core.

The major difference between the description given above and the licensing analysis is the assumption on stuck CEA. For the licensing analysis, both the moderator cooldown curve and the SCRAM worth are calculated assuming a stuck CEA, which greatly reduces the magnitude of SCRAM worth and increases the magnitude of moderator reactivity feedback effects. The net effect causes the core reactivity to approach criticality during transient conditions. As the core approaches criticality, subcritical multiplication occurs and instantaneous fission power is produced. That is, a return to power causes temperatures to increase, which inserts negative reactivity from moderator and fuel temperature feedback mechanisms and limits the return to power.

The conservatisms and credits included in the licensing analysis are enumerated below.

1. The most reactive CEA is assumed to stick in the withdrawn position. A SCRAM worth calculated assuming no stuck CEAs would increase the available SCRAM worth.
2. The accident analysis credits the tripping of the main feedwater pumps upon SGIS. The inventory in the main feedwater system is allowed to blow down to the affected steam generator until closure of the MFIVs. The time delays for the

starting of the HPSI pumps and for the closure of the MFIVs are the most conservative values allowed by Technical Specifications. No credit is taken for closure, or rampdown, of the main feedwater regulation valves after reactor trip. Gross leakage to the affected steam generator through the MFIVs (400 gpm per valve) is assumed in the analysis.

3. The accident analysis models the delivery of AFW to the affected steam generator by assuming AFW is feeding the steam generators at the start of the transient for the breaks initiated from zero power conditions, and initiating the start sequence at the time of the break for the full power transient, thereby ignoring the time to deplete the steam generators to the level required for the AFAS signal initiation. After identification of the intact steam generator by the Auxiliary Feedwater Actuation System based upon differential SG pressure, the AFW flow is diverted to the intact steam generator within the 20 second response time (UFSAR Table 7-4). Gross leakage through the AFW block valves (40 gpm per valve) is assumed in the analysis. The licensing case thus has delivery of more cold AFW to the affected steam generator than expected.
4. Cases with and without LOOP on turbine trip are examined. With LOOP, the RCPs begin their coastdown. This results in negligible inlet plenum mixing which maximizes the RCS cooldown used in the moderator reactivity calculation. Furthermore, the elevated core outlet temperature results in slightly higher RCS pressures and consequently reduces safety injection flow.
5. Zero tube-plugging is assumed because this will maximize the RCS cooldown rate, total cooldown of the RCS, and moderator reactivity feedback.

14.14.4 CORE AND SYSTEM PERFORMANCE

14.14.4.1 Mathematical Models

The transient response of the RCS and steam systems to the SLB event was simulated using the S-RELAP5 thermal-hydraulic system code consistent with the methodology in Reference 14. The overall core conditions calculated by S-RELAP5 during the transient were used as the input to the XCOBRA-IIIC calculation for the pre-scram SLB. The XCOBRA-IIIC fuel assembly thermal-hydraulic code was used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly. The limiting assembly DNBR calculations were performed using an approved DNB correlation. The limiting design axial power profiles were used for this simulation. Both these computer codes are described in Section 14.1.4.1. A spectrum of SLB sizes, MTCs, and locations were analyzed for the pre-trip power excursion. Post-trip return to power analysis included full and no load, with and without LOOP, and single failure cases.

14.14.4.2 Input Parameters and Initial Conditions

The analysis is consistent with requirements for the analysis of Steam System Piping Failures as described in the Standard Review Plan, Section 15.1.5 (Reference 7). Assumptions made are consistent with SRP requirements:

1. Both cases with LOOP and without LOOP are examined.
2. The maximum worth CEA is assumed held fully withdrawn. Analyses have been performed assuming the most reactive CEA at the nominal zero power conditions as the stuck rod and analyses have been performed assuming the most reactive CEA at the minimum affected core sector temperature during the event as the stuck rod. The latter assumption produced the most limiting combination of scram worth and reactivity addition from moderator feedback.

3. The worst single active component failure was determined and assumed to occur.

Single failures considered in the analysis include, but are not limited to, the following:

1. A single failure within the SI system that would reduce the addition of the high concentration boric acid to the core.
2. A single failure within the feedwater system that leaves one condensate booster pump operating after a CSAS, resulting in further feedwater delivery prior to closure of the MFIVs.
3. A failure of a MSIV to close resulting in additional heat removal from the RCS.
4. A single failure of an MFIV to close upon SGIS resulting in further feedwater delivery.

For postulated failures 1, 2, 3, and 4, no credit is taken for the closure of the main feedwater regulating valves after trip.

In order to determine the worst single active component failure, the following scenarios were investigated for both full load and no load cases:

1. Post-trip SLB scenario with a stuck CEA, with and without LOOP, and a single active component failure resulting in the failure of one HPSI to start.
2. Post-trip SLB scenario with a stuck CEA, with LOOP, and a single active component failure resulting in the failure of an MFIV to close on the affected SG.

A conservatively high value of the AFW flow was assumed; limited only by the maximum possible flow rate through the suction line to the AFW pumps. An AFW flow value of 1550 gpm was used in the analysis. This flow is directed only to the unaffected SG, except for leakage past the AFW block valves after AFAS block occurs.

The limiting analyses assumed that the MFW pumps are tripped upon SGIS. The blow down of the inventory in the MFW system to the affected SG was assumed to continue until closure of the MFIVs. Averaged over this period, the rate of uncontrolled feedwater delivery to the affected SG was more than 50% of the full power feedwater flow rate to that generator.

The unrestrained delivery of MFW until closure of the MFIV bounds the postulated failure within the feedwater system that leaves a condensate booster pump operating.

The scenario of a single failure of the MSIV to close for the unaffected SG was considered, thus extending the blowdown from the unaffected SG until the MSIV for the affected SG closes. In this scenario credit for normal operation of the main turbine steam inlet valves, and the steam dump and bypass valves was taken as a backup to the failed MSIV. Although these valves are not safety-related, credit for operation of these valves is acceptable as a backup to the MSIV per SRP 15.1.5. The results of this scenario are less limiting than those for the HPSI pump failure condition.

The post-trip SLB initiated from both full and no load conditions uses the modified Barnett CHF Correlation (Reference 16) to calculate the minimum DNBR. In addition, the peak LHGR initiated from both full and no load conditions is determined via hand calculation. The three-dimensional power distribution peak (F_q) associated with the worst stuck CEA are used in the calculation.

a. <u>SLB – Post-Trip Return to Power</u>

The post-trip SLB assumes that the event is initiated by the double-ended rupture (guillotine break) downstream of steam outlet nozzle flow restrictor which has an area equivalent to 1.9 ft^2. The SLB event was initiated from an array of initial conditions. The most limiting conditions are listed in Table 14.14-1. The MTC of reactivity assumed in the analysis corresponds to EOC, since this MTC results in the greatest positive reactivity change during the RCS cooldown caused by the steam line rupture. Since the reactivity change associated with moderator feedback varies significantly over the moderator density covered in the analysis, a curve of reactivity insertion versus density, rather than a single value of MTC, is assumed in the analysis. The moderator cooldown curve assumed in the analysis is given in Figure 14.14-1. This moderator cooldown curve was conservatively calculated assuming that on reactor trip the CEA is stuck in the fully-withdrawn position that yields the most severe combination of SCRAM worth and reactivity insertion.

The charging system is not modeled. Since the initial boron concentration is very low, direct dilution from charging in the absence of letdown may help maintain pressurizer level, but has little impact on the results.

The reactivity defect associated with the fuel temperature decrease was also based on an EOC Doppler defect. The Doppler defect based on an EOC FTC, in conjunction with the decreasing fuel temperatures, causes the greatest positive reactivity insertion during the SLB event. The Doppler coefficient multiplier (uncertainty) on the FTC assumed in the analysis is a power based Doppler feedback model. The beta fraction assumed is the nominal for EOC conditions.

Conservatively, the boron concentration present in the RCS prior to the event is assumed to be 0 ppm. This negates the negative reactivity insertion that would occur due to concentrating the boron during the cooldown. No charging is assumed.

The minimum CEA worth assumed to be available for shutdown at the time of reactor trip was calculated for the stuck rod that produced the moderator cooldown curve in Figure 14.14-1.

During a return to power, negative reactivity credit was assumed in the analysis. This negative reactivity credit is due to the local heatup of the inlet fluid in the hot channel, which occurs near the location of the stuck CEA. The three-channel S-RELAP5 core model can only accommodate relatively simple radial and axial power distributions, associated reactivity feedback, and feedback weighting models. This tends to result in simple and conservative representation of highly complex neutronics and thermal-hydraulic phenomena. The inherent conservatisms are demonstrated by comparing the reactivity change calculated with S-RELAP5 against that calculated with the neutronics code.

A manual reactor trip at t=0 seconds is used to maximize the cooldown of the RCS.

The analysis assumptions regarding the AFW actuation analysis setpoint, the associated time delays, and the AFW flow and leakage through each leg are given below. They were conservatively chosen to deliver the maximum AFW flow that maximizes the primary cooldown and enhances the potential return to power.

Auxiliary feedwater was conservatively assumed to initiate at the start of the transient with credit taken for minimum AFW actuation delay time, which in all cases resulted in AFW initiation at a level far above the Technical Specification actuation setpoint plus uncertainties. This was done to ensure the analysis results would remain bounding in the event of any future revision of the Technical Specification or uncertainty and is consistent with the enveloping nature of the analysis.

Due to the early initiation of AFW, some flow reaches the affected SG prior to automatic isolation of that SG. The AFW block valve leakage is assumed to enter the damaged SG. Auxiliary feedwater pump flow is assumed to be at a runout value of 1550 gpm.

The analysis included isolation of the ruptured SG when the SG differential pressure reached the analysis setpoint. In addition, a time delay was assumed in the analysis to close the AFW isolation (i.e., block) valves.

One of the scenarios conservatively assumed that on a SIAS, only one HPSI pump starts. In addition, a maximum time delay for HPSI pumps to accelerate to full speed was assumed in the analysis.

The analysis assumes that a limited amount of fluid must be removed or "swept out" of the SI piping before the highly borated SI water reaches the RCS. A conservatively large sweep out volume of 180 ft^3 was assumed. This sweep out volume corresponds to the volume from the SI nozzle, through the check valve and to the first MOV.

b. <u>SLB – Pre-Trip Power Excursion</u>

The results of the post-trip analysis presented above are limiting with respect to return to power. For break sizes less that the maximum double-ended guillotine break, the potential for a pre-trip power excursion becomes the concern. Therefore, the assumptions for the pre-trip analysis listed below are designed to maximize the power excursion prior to reactor trip, rather than maximizing the post-trip return to power.

The limiting pre-trip SLB event was initiated from the conditions listed in Table 14.14-2. The assumptions for the pre-trip event that differ from the post-trip event are listed below.

The reactivity defect associated with the fuel temperature change was based on a nominal Doppler reactivity feedback consistent with the time in cycle (EOC). The pre-trip SLB analysis is performed over a range of MTC versus break size cases. Biasing the Doppler coefficient may result in a different limiting MTC value, but the total reactivity will not be altered. The Beta fraction assumed is the EOC nominal value. An EOC nominal Beta fraction without biasing is sufficient due to the power rise during the pre-trip SLB event being slow.

A spectrum of MTCs was employed to determine the effect of MTC on power range detector response during the pre-trip SLB.

Dose calculations are also performed to calculate the amount of dose as a result of a SLB.

The assumptions made to maximize the site boundary dose are listed in Table 14.14-3. During the event, two sources of radioactivity contribute to the site boundary dose; (1) the initial activity in the SG and (2) the activity associated with the primary-to-secondary leakage. The primary activity includes the maximum initial activity allowed by the Technical Specifications and any activity released to the coolant due to fuel failure. In calculating the site boundary dose, the analysis conservatively assumed that all activity is released to the atmosphere with a decontamination factor of 1.0.

The scenario that would result in the worse dose is an outside containment break. The dose analysis assumes that the SLB would result in 0.80% failed fuel.

14.14.4.3 Results

a. SLB – Post-Trip Return to Power

The post-trip SLB scenario with a stuck CEA and a single active component failure resulting in the failure of one HPSI pump to start is the limiting event. The limiting case among full load and no load, with and without LOOP, depends upon cycle-specific core physics parameters. In many instances, the limiting scenario for approach to the DNBR SAFDL is different from the limiting scenario for approach to the LHGR SAFDL. The post-trip SLB with a stuck CEA, a single active component failure resulting in a failure of one HPSI pump to start, LOOP, and initiated from a full load condition, while not always limiting, is presented here for example.

A maximum break size is limiting for the post-trip SLB scenario. This break size is limited to an area of 1.9 ft^2 by the SG steam nozzle flow restrictors.

Table 14.14-4 lists the sequence of events for the post-trip SLB with a stuck CEA, a single active component failure resulting in a failure of one HPSI pump to start, no-LOOP, and initiated from a full load condition. Moderator reactivity versus moderator density is presented in Figure 14.14-1. The NSSS responses during the transient are given in Figures 14.14-2 through 14.14-7.

The post-trip SLB results show that the minimum DNBR is above the modified Barnett SAFDL expressed as (2.33 + 0.000615 x core exit pressure) when applied to transients that result in core exit pressures that are ≤ 1047 psia. The modified Barnett SAFDL is a single value of 4.25 when the transient results in core exit pressures between 1047 and 1540 psia. The peak LHGR remains below the SAFDL of the more limiting of 21 kW/ft (per Reference 15) or the cycle specific limit.

b. SLB – Pre-Trip Power Excursion

The results of the parametric in break size and MTC demonstrated that an outside containment break size of 2.0 ft^2 resulted in a maximum consequence. This break size is limiting since it results in the most adverse conditions for DNBR and PLHR. Therefore, the results of the 2.0 ft^2 pre-trip SLB are presented here.

The sequence of events for the 2.0 ft^2 pre-trip SLB is given in Table 14.14-5. The reactivity insertion as a function of time is presented in Figure 14.14-12. The NSSS responses during the transient are given in Figures 14.14-8 through 14.14-13.

The limiting pre-trip SLB results presented use the average core power, as decalibrated by downcower density changes, to determine the time of trip for the VHPT. The use of the average core power does not account for change in the signal to the limiting excore detector due to change in the radial power distribution during the transient.

Cases were rerun to include consideration of the effects of the radial power change on the limiting excore detector. These cases resulted in the trip signal being reached sooner. The limiting pre-trip SLB results presented are based on the use of average core power.

The results of the analysis show that the SLB causes the core power to increase until a reactor trip signal is generated at the high power setpoint (NI power). The trip breakers open and the CEAs drop into the core, terminating the power and heat flux increases.

A LOOP on turbine trip is assumed to occur at the time of reactor trip.

When the SG Isolation Analysis Setpoint is reached, the MSIVs begin to close and are completely closed seconds later. The blowdown from the intact SG is terminated at this time.

The minimum DNBR during the transient was calculated based on the thermal-hydraulic code (XCOBRA-IIIC).

14.14.5 CONCLUSIONS

The post-trip SLB results show that the minimum DNBR is above the modified Barnett SAFDL and the peak LHGR remains below the SAFDL of the more limiting of 21 kW/ft (per Reference 15) or the cycle specific limit. The pre-trip SLB results show that the DNBR and FCM limits are not exceeded.

The radiological consequences of a power excursion in the pre-trip portion of an SLB are small when compared to the guidelines of 10 CFR 50.67:

EAB TEDE	0.2180 rem
LPZ TEDE	0.0577 rem
Control Room TEDE	4.6301 rem

These limiting doses are applicable to Calvert Cliffs Unit 1 and Unit 2 assuming a SLB resulting of 0.80% failed fuel (Reference 11).

14.14.6 REFERENCES

1. General Electric Company, "Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves," APED - 5750, March 1969. This report has been submitted to the Atomic Energy Commission (AEC) as a topical report.
2. Deleted
3. Deleted
4. Deleted

5. Deleted

6. Deleted

7. NUREG-0800, "Standard Review Plan," Section 15.1.5, Revision 2, July 1981 [Steam System Piping Failures Inside and Outside of Containment (PWR)] (Formerly NUREG-75/087)

8. Deleted

9. Deleted

10. Deleted

11. 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

12. Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” September 1988

13. Federal Guidance Report (FGR) 12, “External Exposure to Radionuclides in Air, Water, and Soil,” September 1993

14. EMF-2310(P)(A), Revision 1, “SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,” May 2004

15. Letter from Mr. D. V. Pickett (NRC) to Mr. G. H. Gellrich (CCNPP), “Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2 - Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel (TAC Nos. ME2831 and ME2832),” dated February 18, 2011

    2. IN-1412, TID-4500, “A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia,” Idaho Nuclear Corporation July 1970

TABLE 14.14-1

INITIAL CONDITIONS AND INPUT PARAMETERS ASSUMED FOR THE POST-TRIP SLB EVENT INITIATED FROM FULL POWER

PARAMETER	UNITS	VALUE
Initial Core Power	MWt	2737
Initial Core Inlet Temperature	°F	548
Initial RCS Pressure	psia	2250
Initial Vessel Flow Rate	gpm	370,000
SG Differential Pressure Analysis Setpoint	psid	250
SIAS Setpoint	psia	1640
Minimum CEA Worth Available at Trip	pcm	6026
Moderator Cooldown Curve	pcm vs. Density	Figure 14.14-1
Effective MTC	pcm/°F	-33
Beta Fraction (EOC nominal)	---	0.005215
Percentage of SG Plugged Tubes	%	0

TABLE 14.14-2

INITIAL CONDITIONS AND INPUT PARAMETERS ASSUMED FOR THE PRE-TRIP SLB EVENT INITIATED FROM FULL POWER

PARAMETER	UNITS	VALUE
Initial Core Power	MWt	2754
Initial Core Inlet Temperature	°F	548
Initial RCS Pressure	psia	2250
Initial Vessel Flow Rate	gpm	370,000
High Power Trip Setpoint	% RTP	112
High Power Trip Response	sec	0.4
TM/LP Trip Delay	sec	0.9
High Containment Pressure Trip Setpoint	psia	19.45
High Containment Pressure Trip Delay	sec	0.9
Low SG Pressure Trip Setpoint	psia	600 (harsh) 650 (non-harsh)
Low SG Pressure Trip Delay	sec	0.9
Minimum CEA Worth Available at Trip	pcm	5470.8
Doppler Reactivity Coefficient	pcm/°F	-1.48
Effective MTC	pcm/°F	-8 to -33
Kinetics, β_{eff}	---	0.005237
Radial Peak Temperature Dependence (Asymmetric case only)	---	Case Dependent
Temperature Shadowing or Decalibration Factor	%/°F	0.70
Resistance Temperature Detector Response, Hot/Cold	sec	0.0/25.0

TABLE 14.14-3

ASSUMPTIONS FOR THE RADIOLOGICAL EVALUATION FOR THE SLB EVENT

PARAMETER	UNITS	VALUE
RCS Maximum Allowable Concentration (DEQ I-131)[a]	μCi/gm	0.5
Secondary Maximum Allowable Concentration (DEQ I-131)[a]	μCi/gm	0.1
Partition Factor Assumed for All Doses	---	1.0
Atmospheric Dispersion Coefficient[b]	sec/m^3	1.44×10^{-4}
Breathing Rate	m^3/sec	3.5×10^{-4}
Dose Conversion Factor	REM/Ci	[c]

[a] Technical Specification limits.

[b] 0-2 hour accident condition.

[c] Dose conversion factors obtained from References 12 and 13.

TABLE 14.14-4

SEQUENCE OF EVENTS FOR THE POST-TRIP SLB EVENT

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	SLB Occurs, AFW startup process is initiated	---
0.0	Manual Trip Occurs	---
0.5	CEAs Begin to Drop into Core	---
10.8	SG Isolation Setpoint is Reached	600 psia
11.7	MSIV Begins to Close	---
15.9	SIAS Generated	1640 psia
33.6	SG Differential Pressure Analysis Value is Reached	250 psid
46.8	HPSI pump at full speed, but does not inject due to RCS cold leg pressure being higher than HPSI pump shutoff head	---
53.6	AFW Block Valve Closed on Affected SG	---
63.0	Lowest RCS cold leg pressure reaches HPSI pump shutoff head and HPSI flow begins filling SI lines with borated water	---
75.8	Affected SG MFW isolation valves fully closed (65.0 second delay)	---
209.0	Shutdown worth fully overcome by moderator and Doppler feedback	0.0 $
258.0	Maximum Post-Trip Reactivity	0.03160 $
426.4	HPSI pump delivers enough borated water to completely fill the SI lines and begins delivering boron	---
428.0	Peak Return to Power	258.76 MW

TABLE 14.14-5

SEQUENCE OF EVENTS FOR PRE-TRIP SLB EVENT WITH LOOP ON TURBINE TRIP INITIATED FROM FULL POWER

TIME (sec)	EVENT	ANALYSIS SETPOINT OR VALUE
0.0	SLB Occurs	2.0 ft^2
22.44	VHPT Setpoint is Reached	112% RTP
22.84	Trip Breakers Open	---
22.90	Maximum Core Average Heat Flux Occurs	3519.4 MW
23.34	CEA Begin to Drop into Core	---
24.40	Minimum DNBR Occurs	> MDNBR Limit