14.17 LOSS-OF-COOLANT ACCIDENT

14.17.1 INTRODUCTION AND SUMMARY

Title 10 CFR 50.46 (Reference 1) provides the acceptance criteria for Emergency Core Cooling Systems (ECCS) for light water nuclear power reactors. The ECCS performance analyses presented in this section demonstrate that the Calvert Cliffs Units 1 and 2 ECCS design satisfies these criteria.

Sections 14.17.2 and 14.17.3 describe the analyses for the large break LOCA and the small break LOCA, respectively. Sections 14.17.4.1 and 14.17.4.2 describe the ECCS performance of the current cycles for Units 1 and 2.

The ECCS performance analyses were performed for a spectrum of large and small break LOCA break sizes. The limiting break size, i.e., the break that results in the highest peak cladding temperature, was identified as the 0.34 ft^2 break in the cold leg pump discharge piping. The results of the analysis demonstrate that, for a PLHGR of 15.0 kW/ft the ECCS design meets the 10 CFR 50.46 Acceptance Criteria. Conformance is as follows:

Criterion (1) Peak Cladding Temperature. "The calculated maximum fuel element cladding temperature shall not exceed 2200°F."

The ECCS performance analysis yielded a peak cladding temperature of 1648°F for the 0.34 ft^2 break. The most recent 10 CFR 50.46 report contains all PCT penalties and benefits.

Criterion (2) Maximum Cladding Oxidation. "The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation."

The ECCS performance analysis yielded a maximum cladding oxidation of 0.0350 times the total thickness before oxidation for the 0.34 ft^2 break.

Criterion (3) Maximum Hydrogen Generation. "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

The ECCS performance analysis did not calculate the fraction of total hydrogen directly; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1% limit.

Criterion (4) Coolable Geometry. "Calculated changes in core geometry shall be such that the core remains amenable to cooling."

The ECCS performance analysis assures the core remains amenable to cooling from the effects of fuel cladding rupture and swelling, and the effects of LOCA. The analysis conservatively considers blockage effects due to clad swelling and rupture in the prediction of the hot fuel rod PCT. Since Criteria 1 and 2 are satisfied for the hot pin, it is clear that the hot pin remains amenable to cooling. It is therefore concluded that the remainder of the core also remains amenable to cooling. Therefore, the analysis demonstrates a coolable geometry.

Criterion (5) Long-Term Cooling. "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

The ECCS performance analysis showed that the rapid insertion of borated water from the safety injection tanks (SITs) and the SI pumps suitably limited the peak cladding temperature and cooled the core within a short period of time. Subsequently, the SI pumps will continue to supply cooling water from the refueling water tank or the containment sump.

14.17.2 LARGE BREAK LOCA ANALYSIS

The purpose of the large break LOCA analysis is to verify typical Technical Specification peaking factor limits and the adequacy of the ECCS by demonstrating that the following 10 CFR 50.46(b) criteria are met:

Criterion (1) Peak Cladding Temperature. "The calculated maximum fuel element cladding temperature shall not exceed 2200°F."

The ECCS performance analysis yielded a peak cladding temperature of 1620°F for the Double-Ended Guillotine break of 4.5832 ft^2/side. The most recent 10 CFR 50.46 report contains all PCT penalties and benefits.

Criterion (2) Maximum Cladding Oxidation. "The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation."

The ECCS performance analysis yielded a maximum cladding oxidation of 0.02460 times the total thickness before oxidation.

Criterion (3) Maximum Hydrogen Generation. "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

The ECCS performance analysis did not calculate the fraction of total hydrogen directly; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1% limit.

Criterion (4) Coolable Geometry. "Calculated changes in core geometry shall be such that the core remains amenable to cooling."

The ECCS performance analysis assures the core remains amenable to cooling despite the effects of fuel cladding rupture and swelling, and the effects of LOCA. The realistic large break LOCA analysis conservatively considers blockage effects due to clad swelling and rupture in the prediction of the hot fuel rod PCT. Therefore, the analysis demonstrates compliance with Criterion 4.

Criterion (5) Long-Term Cooling. "After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

The ECCS performance analysis showed that the rapid insertion of borated water from the SITs and the SI pumps suitably limited the peak cladding temperature and cooled the core within a short period of time.

Subsequently, the SI pumps will continue to supply cooling water from the refueling water tank or the containment sump.

14.17.2.1 <u>Event Description</u>

A large break LOCA is initiated by a postulated large rupture of the RCS primary piping. Based on deterministic studies, the worst break location is in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop containing the pressurizer. The break initiates a rapid depressurization of the RCS. A reactor trip signal is initiated when the low pressurizer pressure trip setpoint is reached; however, reactor trip is conservatively neglected in the large break LOCA analysis. The reactor is shutdown by coolant voiding in the core.

The plant is assumed to be operating normally at full power prior to the accident. The cold leg break is assumed to open instantaneously. For this break, a rapid depressurization occurs, along with a core flow stagnation and reversal. This causes the fuel rods to experience DNB. Subsequently, the limiting fuel rods are cooled by film convection to steam. The coolant voiding creates a strong negative reactivity effect and core criticality ends. As heat transfer from the fuel rods is reduced, the cladding temperature increases.

Coolant in all regions of the RCS begins to flash. At the break plane, the loss of subcooling in the coolant results in substantially reduced break flow. This reduces the depressurization rate and leads to a period of positive core flow or reduced downflow as the RCPs in the intact loops continue to supply water to the reactor vessel (in no-LOOP conditions). Cladding temperatures may be reduced and some portions of the core may rewet during this period. The positive core flow or reduced downflow period ends as two phase conditions occur in the RCPs, reducing their effectiveness. Once again, the core flow reverses as most of the vessel coolant inventory flows out through the broken cold leg.

Mitigation of the large break LOCA begins when the SIAS is initiated. This signal is initiated by either high containment pressure or low pressurizer pressure. Regulations require that a worst single failure be considered. This single failure has been determined to be the loss of one ECCS pumped injection train. The realistic large break LOCA methodology conservatively assumes an on-time start and normal lineup of the containment spray to conservatively reduce containment pressure and increase break flow. Hence, the analysis assumes the loss of a diesel generator, LPSI injection into the broken loop and one intact loop, HPSI injection into all four loops, and all containment spray pumps are operating.

When the RCS pressure falls below the SIT pressure, fluid from the SITs is injected into the cold legs. In the early delivery of SIT water, high pressure and high break flow will drive some of this fluid to bypass the core. During this bypass period, core heat transfer remains poor and fuel rod cladding temperatures increase. As RCS and containment pressures equilibrate, ECCS water begins to fill the lower plenum and eventually the lower portions of the core; thus, core heat transfer improves and cladding temperatures decrease.

Eventually, the relatively large volume of SIT water is exhausted and core recovery continues relying solely on pumped ECCS injection. As the SITs empty, the nitrogen gas used to pressurize the SITs exits through the break. This gas release may result in a short period of improved core heat transfer as the nitrogen gas displaces water in the downcomer. After the nitrogen gas has been expelled, the ECCS temporarily may not be able to sustain full core cooling because of the core decay heat and the higher steam temperature created by quenching in the lower portions of the core. Peak fuel rod cladding temperatures may increase for a short period until more energy is removed from the core by the HPSI and LPSI while the decay heat continues to fall. Steam generated from fuel rod rewet will entrain liquid and pass through the core, vessel upper plenum, the hot legs, the SGs, and the RCPs before it is vented out the break. Eventually (within a few minutes of the accident), the core reflood will progress sufficiently to ensure core-wide cooling. Full core quench occurs within a few minutes after core-wide cooling. Long-term cooling is then sustained with coolant provided by LPSI.

14.17.2.2 Evaluation Model

The realistic large break LOCA methodology is documented in Reference 2. The methodology follows the Code Scaling, Applicability, and Uncertainty evaluation approach (Reference 3). This method outlines an approach for defining and qualifying a best estimate thermal hydraulic code and quantifies the uncertainties in a LOCA analysis.

The realistic large break LOCA methodology consists of the following computer codes:

- RODEX3A for computation of the initial fuel stored energy, fission gas release, and fuel cladding gap conductance.
- S-RELAP5 for the system calculation (includes ICECON for containment response).

The modeling of plant components is performed by following guidelines developed to ensure accurate accounting for physical dimensions, and that the dominant phenomena expected during the large break LOCA event are captured. The basic building blocks for modeling are hydraulic volumes for fluid paths and heat structures for heat transfer. In addition, special purpose components exist to represent specific components such as the RCPs or the SG separators. All geometries are modeled at the resolution necessary to best resolve the flow field and the phenomena being modeled within practical computational limitations.

A typical calculation using S-RELAP5 begins with the establishment of a steady-state initial condition with all loops intact. The input parameters and initial conditions for this steady-state calculation are chosen to reflect plant Technical Specifications or to match measured data. Additionally, the RODEX3A code provides initial conditions for the S-RELAP5 fuel models. Specific parameters are discussed in Section 14.17.2.3.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break into the loop containing the pressurizer. The evolution of the transient through blowdown, refill, and reflood is computed continuously using S-RELAP5. Containment pressure is also calculated by S-RELAP5 and provides direct feedback for the pressure calculation using containment models derived from ICECON (Reference 4). The methods used in the application of S-RELAP5 to the large break LOCA are described in Reference 2.

The final step of the best estimate methodology is to combine all the uncertainties related to the code and plant parameters, and estimate the PCT at a high probability level. The steps taken to derive the PCT uncertainty estimate are summarized below:

1. Base Plant Input File Development

   First, base RODEX3A and S-RELAP5 input files for the plant (including the containment input file) are developed. Code input development guidelines are applied to ensure that model nodalization is consistent with the model nodalization used in the code validation.

2. Sampled Case Development

   The non-parametric statistical approach requires that many "sampled" cases be created and processed. For every set of input created, each "key LOCA parameter" is randomly sampled over a range established through code uncertainty assessment or expected operating limits (provided by plant Technical Specifications or data). Those parameters considered "key LOCA parameters" are listed in Table 14.17-1. This list includes both parameters related to LOCA phenomena (based on the Phenomena Identification and Ranking Table provided in Reference 2) and to plant operating parameters.

3. Determination of Adequacy of ECCS

   The realistic large break LOCA methodology uses a non-parametric statistical approach to determine values of PCT at the 95% probability level. Total oxidation and total hydrogen are based on the limiting PCT case. The adequacy of the ECCS is demonstrated when these results satisfy the criteria set forth in Section 14.17.2.

   The following are deviations from the approved realistic large break LOCA evaluation model (Reference 2) that were necessary to either correct or improve the calculation and/or to respond to additional information requested by the NRC. Each of these items has been approved for use at Calvert Cliffs until a revision to EMF-2103 is approved and implemented (Reference 8).

   - **Reactor Power** - The assumed reactor core power for the Calvert Cliffs realistic large break LOCA accident is 2754 MWt. This value represents the plant RTP (i.e., total reactor core heat transfer rate to the RCS) of 2737 MWt with a maximum power measurement uncertainty of 0.62% added to the RTP. The power was not sampled in the analysis.
   - **Rod Quench** - The realistic large break LOCA analysis was performed with a version of S-RELAP5 that requires both the void fraction to be less than 0.95 and the clad temperature to be less than 900°F before the rod is allowed to quench.
   - **Film Boiling Heat Transfer Limit** - The realistic large break LOCA analysis was performed with a version of S-RELAP5 that limits the contribution of the Forslund-Rohsenow model to no more than 15% of the total heat transfer at and above a void fraction of 0.9.
   - **Break Size** - The split versus double-ended break type is no longer related to break area. In concurrence with Regulatory Guide 1.157, both the split and the double-ended break will range in area between the minimum break area (A_{min}) and an area of twice the size of the broken pipe. The determination of break configuration, split versus double-ended, will be made after the break area is selected based on a uniform probability for each occurrence. A_{min} was calculated to be 28.7% of the DEG break area.
   - **10 CFR Part 50, Appendix A, General Design Criterion 35 (Emergency core cooling) - LOOP and No-LOOP Case Sets** - In concurrence with General Design Criterion 35, a set of 59 cases was run with a LOOP assumption and a second set with a no-LOOP assumption. The results from both case sets are shown in Figure 14.17-17.

- **Cold Leg Condensation Efficiency** - During recent realistic large break LOCA modeling studies, it was noted that cold leg condensation efficiency may be under-predicted. Water entering the downcomer post-SIT injection remained sufficiently subcooled to absorb the downcomer wall heat release without significant boiling. However, tests (Reference 5) indicate that the steam and water entering the downcomer from the cold leg, subsequent to the end of SIT injection, reach near saturation resulting from the condensation efficiency ranging between 80 to 100%. To assure that cold leg condensation would not be under-predicted, a realistic large break LOCA evaluation model update was made. Noting that saturated fluid entering the downcomer is the most conservative modeling scheme, steam and liquid multipliers were developed so as to approximately saturate the cold leg fluid at the cold leg pressure before it enters the downcomer. Providing saturated fluid conditions at the downcomer entrance conservatively reduces both the downcomer driving heat and the core flooding rate. The test results indicate that fluid conditions entering the downcomer range from saturated to slightly subcooled. Hence, it is conservative to force an approximation of saturated conditions for fluid entering the downcomer. The NRC stated in Reference 8 that it finds this departure from the previously approved realistic large break LOCA methodology acceptable because (1) the artificially saturated fluid conditions will conservatively reduce both the downcomer driving head and the core flooding rate, which becomes conducive to portions of the fuel remaining in a vapor-cooled environment, thus presenting a greater challenge to clad surface cooling, and (2) conditions in the downcomer following SIT discharge are expected to be slightly subcooled, meaning that assuming fully saturated conditions is conservative.
- **RODEX3A Temperature Compensation** - AREVA Inc. has acknowledged an issue concerning fuel thermal conductivity degradation as a function of burnup as raised by the NRC (Reference 6). In order to manage this issue, AREVA Inc. is modifying the way RODEX3A temperatures are compensated in the Reference 2 realistic large break LOCA methodology. In the current process, the realistic large break LOCA computes PCTs at many different times during an operating cycle. For each specific time in cycle, the fuel conditions are computed using RODEX3A prior to starting the S-RELAP5 portion of the analysis. A steady-state condition for the given time in cycle using S-RELAP5 is established. A base fuel centerline temperature is established in this process. Then a two-transformation adjustment to the base fuel centerline temperature is computed. The first transformation is a linear adjustment for an exposure of 10 MWd/MTU or higher. In the new process, a polynomial transformation is used in the first transformation instead of a linear transformation. The rest of the realistic large break LOCA process for initializing the S-RELAP5 fuel rod temperature should not be altered and the rest of LOCA transient should also continue in the original fashion. This approach was accepted by the NRC for first-cycle AREVA fuel only (Unit 1 Cycle 21 and Unit 2 Cycle 19) in Reference 8.

The NRC has concluded that a license condition is necessary to restrict plant operation to a single-cycle under the current large break LOCA analysis of record, and to obtain NRC review and approval of a generic disposition concerning the analysis of only first-cycle fuel in light of the fuel thermal conductivity degradation issue with the RODEX3A code (Reference 8).

In response to the license condition, the realistic large break LOCA analysis has been updated to specifically model both first and second cycle fuel rods. Third cycle fuel does not retain sufficient energy potential to achieve significant cladding temperatures nor cladding oxidation and is not included in the realistic large break LOCA individual pin calculations. The burnup for the individual first and second cycle rods analyzed is assigned according to the sampled time in cycle. The time in cycle is sampled once and is the same for both the fresh (first cycle) and once-burnt (second cycle) fuel. Burnup for the fresh and once-burnt rods is different in accordance with the cycle management. Likewise, pin pressure and thermal conductivity differ.

In addition to the thermal conductivity and fuel temperature adjustments for burnup, a burnup dependent reduction in allowed peaking is needed for the once-burnt fuel. For first cycle fuel, the realistic large break LOCA methodology increases the F_r to the Technical Specification maximum (including uncertainty) for the first cycle hot rods in the model. Shortly into the cycle, once-burnt fuel has insufficient energy potential to achieve this peaking. A burnup dependent reduction in allowed peaking is therefore applied through an adjustment in the second cycle F_r. This approach was accepted by the NRC in Reference 9 for both first-cycle and burned Advanced CE-14 HTP™ fuel.

14.17.2.3 Plant Description and Summary of Analysis Parameters

The analysis presented here is for a Combustion Engineering-designed PWR, which has 2x4 loop arrangement. There are two hot legs each with a U-tube SG and four cold legs each with a RCP. The RCS includes one pressurizer connected to a hot leg. The core contains 217 fuel assemblies. The Framatome fuel assemblies are modeled with 2, 4, 6, and 8 w/o Gadolinia pins. An evaluation for odd concentration Gadolinia rod power is performed each cycle to verify applicability of the AOR valves. The ECCS includes one HPSI, one LPSI, and one SIT injection path per RCS loop. The break is modeled in the same loop as the pressurizer as directed by the realistic large break LOCA methodology. The realistic large break LOCA transients are of sufficiently short duration that the switchover to sump cooling water (i.e., RAS) for ECCS pumped injection need not be considered.

The S-RELAP5 model explicitly describes the RCS, reactor vessel, pressurizer, and ECCS. The ECCS includes a SIT path and a LPSI/HPSI path per RCS loop. The HPSI and LPSI feed into a common header that connects to each cold leg pipe downstream of the RCP discharge. The ECCS pumped injection is modeled as a table of flow versus backpressure. This model also describes the secondary side SG that is instantaneously isolated (closed MSIV and feedwater trip) at the time of the break. A symmetric steam generator tube plugging level of 10% per SG was assumed.

As described in the realistic large break LOCA methodology, many parameters associated with large break LOCA phenomenological uncertainties and plant operation ranges are sampled. A summary of both phenomenological and plant parameters are given in Table 14.17-1. The large break LOCA phenomenological uncertainties are provided in Reference 2. Values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in the analysis are given in Table 14.17-2. Plant data are analyzed to develop uncertainties for the process parameters sampled in the analysis. Table 14.17-3 presents a summary of the uncertainties used in the analysis. Where applicable, the sampled parameter ranges are based on Technical Specification limits or supporting plant calculations that provide more bounding values.

For the realistic large break LOCA evaluation model, dominant containment parameters, as well as NSSS parameters, were established via a Phenomena Identification and Ranking Table process. Other model inputs are generally taken as nominal or conservatively biased. The Phenomena Identification and Ranking Table outcome yielded two important (relative to PCT) containment parameters - containment pressure and temperature. As noted in Table 14.17-3, containment temperature is a sampled parameter. Containment pressure response is indirectly ranged by sampling the containment volume (Table 14.17-3). Containment heat sink data and material thermal properties are given in Table 14.17-7. The heat transfer coefficients are calculated internally by S-RELAP5 during the transient and are variable based on the air/steam ratio in Containment. The Containment initial conditions and boundary conditions are given in Table 14.17-8. The containment spray is modeled at maximum heat removal capacity. All spray flow is delivered to the Containment.

14.17.2.4 <u>Analysis of Results</u>

Two case sets of 59 transient calculations were performed by sampling the parameters listed in Table 14.17-1. For each case set, a PCT was calculated for a UO_2 rod and for Gadolinia-bearing rods with concentrations of 2, 4, 6, and 8 w/o Gd_2O_3. The limiting case set containing the highest PCT corresponds to that with no offsite power available. A limiting PCT of 1620°F occurred in Case 47 for a fresh 8 w/o Gd_2O_3 rod. The most recent 10 CFR 50.46 report contains all PCT penalties and benefits. The major parameters for the limiting transient are presented in Table 14.17-4. Table 14.17-5 lists the results of the limiting case. The fraction of total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1% limit. The best-estimate PCT case is Case 1, which corresponded to the median case out of the 59-case set with no offsite power available. The nominal PCT was 1424°F for an 8 w/o Gd_2O_3 rod. This result can be used to quantify the relative conservatism in the limiting case result. In this analysis, it was 196°F.

The case results, event times, and analysis plots for the limiting PCT case are shown in Tables 14.17-5 and 14.17-6 and in Figures 14.17-6 through 14.17-16. Figure 14.17-1 shows linear scatter plots of the key parameters sampled for the 59 calculations. Parameter labels appear to the left of each individual plot. These figures show the parameter ranges used in the analysis. Figures 14.17-2 and 14.17-3 show the time of PCT and break size versus PCT scatter plots for the 59 calculations, respectively. Figures 14.17-4 and 14.17-5 show the maximum oxidation and total oxidation versus PCT scatter plots for the 59 calculations, respectively. Key parameters for the limiting PCT case are shown in Figures 14.17-6 through 14.17-16. Figure 14.17-6 is the plot of PCT independent of elevation; this figure clearly indicates that the transient exhibits a sustained and stable quench. A comparison of PCT results from both LOOP and no-LOOP case sets is shown in Figure 14.17-17. As seen in Figure 14.17-17 the peak PCT is from the LOOP case.

14.17.2.5 Conclusions

A realistic large break LOCA analysis was performed using NRC-approved realistic large break LOCA methods (Reference 2). Analysis results show that the limiting LOOP case has a PCT of 1620°F and a maximum oxidation thickness of 2.46% fall well within regulatory requirements. The most recent 10 CFR 50.46 report contains all PCT penalties and benefits. The total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1% limit.

The analysis supports operation at a nominal power level of 2754 MWt (including 0.62% uncertainty), a SG tube plugging level of up to 10% in all SGs, a LHGR of 15.0 kW/ft, a total peaking factor (F_q) up to a value of 2.37, and a nuclear enthalpy rise factor (F_r) up to a value of 1.81 (including 6% uncertainty) with no axial or burnup dependent power peaking limit and peak rod average exposures of up to 62,000 MWd/MTU. For large break LOCA, the 10 CFR 50.46(b) criteria presented in Section 14.17.2 are met and operation of Calvert Cliffs Units 1 and 2 with Advanced CE-14 HTP™ fuel is justified.

14.17.3 SMALL BREAK LOCA ANALYSIS

The purpose of the small break LOCA analysis is to verify typical Technical Specification peaking factor limits and the adequacy of the ECCS by demonstrating that the following 10 CFR 50.46(b) criteria are met:

Criterion (1) Peak Cladding Temperature. "The calculated maximum fuel element cladding temperature shall not exceed 2200°F."

The small break LOCA ECCS performance analysis yielded a peak cladding temperature of 1648°F.

Criterion (2) Maximum Cladding Oxidation. "The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation."

The small break LOCA ECCS performance analysis yielded a maximum cladding oxidation of 0.035 times the total thickness before oxidation.

Criterion (3) Maximum Hydrogen Generation. "The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

The small break LOCA ECCS performance analysis did not calculate the fraction of total hydrogen directly; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1% limit.

Criterion (4) Coolable Geometry. "Calculated changes in core geometry shall be such that the core remains amenable to cooling."

The small break LOCA ECCS performance analysis assures the core remains amenable to cooling despite the effects of fuel cladding rupture and swelling. The small break LOCA analysis conservatively considers blockage effects due to clad swelling and rupture in the prediction of the hot fuel rod PCT. Since Criteria 1 and 2 are satisfied for the hot pin, it is clear that the hot pin remains amenable to cooling. It is therefore concluded that the remainder of the core also remains amenable to cooling. Therefore, the analysis demonstrates compliance with Criterion 4.

14.17.3.1 Event Description

A postulated small break LOCA is defined as a break in the RCS pressure boundary which has an area of up to approximately 10% of a cold leg pipe area. The most limiting break location is in the cold leg pipe on the discharge side of the RCP (Reference 7). The break location results in the largest amount of inventory loss and the largest fraction of ECCS fluid being ejected out through the break. This produces the greatest degree of core uncovery, the longest fuel rod heatup time, and consequently, the greatest challenge to the 10 CFR 50.46(b) criteria (Reference 1).

The small break LOCA event is characterized by a slow depressurization of the primary system with a reactor trip occurring on a low pressurizer pressure signal. The SIAS occurs when the system has further depressurized. The capacity and shutoff head of the HPSI pumps are important parameters in the small break LOCA analysis. For the limiting break size, the rate of inventory loss from the primary system is large enough that the HPSI pumps cannot preclude significant core uncovery. The primary system depressurization rate is slow, extending the time required to reach the SIT pressure or to recover core liquid level on HPSI and LPSI flow. This tends to maximize the heat up time of the hot rod which produces the maximum PCT and local cladding oxidation. Core recovery for the limiting break begins when the SI flow that is retained in the RCS exceeds the mass flow rate out the break, followed by injection of SIT flow. For very small break sizes, the primary system pressure does not reach the SIT pressure.

14.17.3.2 Evaluation Model

The small break LOCA evaluation model for the event response of the primary and secondary systems and hot fuel rod used in this analysis (References 7 and 10) consists of two computer codes. The two computer codes used in this analysis are:

- The RODEX2-2A code was used to determine the burnup-dependent initial fuel rod conditions for the system calculations.
- The S-RELAP5 code was used to predict the thermal-hydraulic response of the primary and secondary sides of the reactor system and the hot fuel rod response.

The fuel-to-clad gap conditions used to initialize S-RELAP5 are taken at EOC, consistent with an EOC top-peaked axial power distribution.

This methodology has been reviewed and approved by the NRC to perform small break LOCA analyses for Calvert Cliffs in Reference 8 with the following restrictions and deviations from Reference 7:

1. Since the generic break spectrum model was shown to predict a non-conservative peak cladding temperature, the NRC staff concludes that a license condition is necessary to capture the more restrictive design criteria for Calvert Cliffs reload designs (Reference 8).

   The small break LOCA performed in accordance with the methodology of Technical Specification 5.6.5.b.9 shall be analyzed using a break spectrum with augmented detail related to break size. This revised methodology shall be applied to the Calvert Cliffs core reload designs starting with Unit 1 Cycle 21 and Unit 2 Cycle 19.

2. To support the acceptability of Calvert Cliffs operation at 2754 MWt and 15.0 kW/ft, the following NRC staff recommendations, modifications to the S-RELAP5 input modeling, and changes to the EOPs include the following:
   a. Modifications to the S-RELAP5 modeling to allow only one cold leg suction piping to clear of liquid for all small breaks with diameters of 4 inches or less.
   b. Removal of credit for the hot leg nozzle gaps and the upper core barrel flange.
   c. Leakage paths that represent communicate paths for fluid flow between the upper plenum and upper head directly into the upper downcomer region.
   d. Inclusion of a large reverse flow K-factor at the outlet of the core to prevent the downflow of liquid from above to cool the core hot bundle during periods of core uncovery.
   e. The HPSI and LPSI head-flow curve input to the S-RELAP5 code will be verified against the surveillance testing to be conducted prior to power operation with the AREVA/Framatome fuel loaded in the core. The head flow curve should include adjustments for all measurement uncertainties associated with the surveillance test.
   f. The simulator operator training and qualification should be conducted periodically to ensure the operators can trip the RCPs following the limiting small break LOCA within 4 minutes following loss of 20°F subcooling.

3. The following restriction is imposed on the S-RELAP5 small break LOCA methodology for Calvert Cliffs:

   Should the PCT increase above the current limiting break PCT of 1626°F in any subsequent evaluation, the licensee will be expected to correct the ability of the S-RELAP5 code to more accurately compute the two-phase level and resultant heat-up of the fuel cladding in the core.

An updated small break LOCA analysis was performed utilizing the supplemental small break LOCA methodology given in Reference 10. The small break LOCA methodology addresses items 1, 2 a through d, and 3 described above. For item 1, the supplemented methodology contains a break spectrum analysis which includes breaks of varying diameter up to 10% of the flow area for the cold leg. The spectrum includes a break size range from 1.0 to 9.49 inches in diameter, which is wide enough to establish a PCT trend. Additional break sizes are analyzed with a smaller break interval once the potential limiting break size is determined to confirm the limiting break size. For items 2 a through d, the supplemented methodology includes the changes to the S-RELAP5 modeling that addresses each of those items. For item 3, the supplemented methodology includes additional control volumes (cells) to represent the core which allows S-RELAP5 to more accurately compute the two-phase level and resultant heat-up of the fuel cladding in the core.

14.17.3.3 <u>Plant Description and Summary of Analysis Parameters</u>

Calvert Cliffs Units 1 and 2 are Combustion Engineering-designed 2x4 PWRs with two hot legs, four cold legs, and two vertical U-tube SGs. The reactor has a rated core power of 2754 MWt (including measurement uncertainty). The reactor vessel contains a downcomer, upper and lower plenums, and a reactor core containing 217 fuel assemblies. The hot legs connect the reactor vessel with the vertical U-tube

SGs. Main feedwater is injected into the downcomer of each SG. There are three AFW pumps, one motor-driven and two turbine-driven. The ECCS contains three HPSI pumps (a minimum of 2 OPERABLE per Technical Specifications), four SITs, and two LPSI pumps. Important system parameters and initial conditions used in the analysis are given in Table 14.17-9.

The RCS was nodalized in the S-RELAP5 model into control volumes interconnected by flow paths or "junctions." The model includes four SITs, a pressurizer, and two SGs with both primary and secondary sides modeled. All of the loops were modeled explicitly to provide an accurate representation of the plant. A SG tube plugging level of 10% in each SG was assumed. The HPSI system was modeled to deliver the minimum total flow asymmetrically to the broken loop and three intact loops in the S-RELAP5 model, with the highest individual loop HPSI flow and the flow from one SIT injected into the cold leg containing the break. The LPSI system was modeled to deliver SI flow to the loop containing the broken leg. The degraded HPSI flow used in the analysis is shown in Table 14.17-10. The degraded LPSI flow used in the analysis is shown in Table 14.17-11.

The heat generation rate in the S-RELAP5 reactor core model was determined from reactor kinetics equations with actinide and decay heating as prescribed in 10 CFR Part 50, Appendix K.

The input model included details of both main steam lines from the SGs to the turbine control valve, including the MSSV inlet piping connected to the main steam lines. The MSSVs were set to be fully open at their nominal (Technical Specification maximum) setpoints plus 3% tolerance.

The analysis assumed loss of offsite power concurrent with reactor scram on low pressurizer pressure. The single-failure criterion required by 10 CFR Part 50, Appendix K was satisfied by assuming the loss of one emergency diesel generator, which resulted in the disabling of one HPSI and one LPSI pump and the motor-driven AFW pump. Thus, a single HPSI pump was assumed to be operable. Charging pump flow was not credited in the analysis. Initiation of the HPSI and LPSI systems was delayed by 30 and 45 seconds, respectively, following SIAS activation. These delays represent the time required for diesel generator startup and switching. The disabling of the motor-driven AFW pump leaves two turbine-driven pumps available, only one of which automatically starts. The initiation of the turbine-driven pump was delayed 180 seconds beyond the time of the AFAS indicating low SG level. Operator startup of the second turbine-driven AFW pump was not credited in the analysis.

A spectrum of cold leg break sizes (0.0055 through 0.49 ft^2) was analyzed. The break spectrum calculations assumed RCP trip at reactor trip due to an assumed LOOP at reactor trip.

14.17.3.4 Results of the Small Break Analysis

The time sequence of events for the limiting break case is shown in Table 14.17-12. A 0.34 ft^2 break in the cold leg pump discharge piping with LOOP was determined to have the maximum PCT of 1648°F. The most recent 10 CFR 50.46 report contains all PCT penalties and benefits. The maximum local cladding oxidation was calculated to be 3.50% of the total cladding thickness before oxidation. The limiting core-wide metal reaction was calculated to be less than 0.01% of the maximum hypothetical amount for the active core as required by 10 CFR 50.46. Trend plots for parameters of interest are shown in Figures 14.17-18 through 14.17-23. These results indicate that a coolable geometry would be maintained during a small break LOCA event.

14.17.3.5 Conclusions

The results of the small break LOCA analysis conform to the 10 CFR 50.46 ECCS acceptance criteria of 2200°F, 17%, and 1% for peak cladding temperature, maximum cladding oxidation, and maximum core-wide oxidation.

14.17.4 CURRENT CYCLE ANALYSES

There exist changes or errors that affect the analysis of record PCT calculation. These changes or errors are documented via 10 CFR50.46 reporting. After accounting for all changes or errors, the licensing basis PCT remains below the 2,200 °F limit. Refer to the docketed annual or thirty-day 10 CFR 50.46(a)(3) report for details.

CCNPP License Amendment Request to Utilize Accident Tolerant Fuel (ATF) Lead Test Assembly (LTA) was approved in ML20363A242. The Enhanced Accident Tolerant Fuel (EATF) LTA is the Advanced CE14x14 HTP Framatome fuel design with Chromia-doped (Cr2O3/UO2) fuel pellets and Chromium-coated M5 clad features. A technical evaluation to assess the potential impacts of the EATF LTA design on the Calvert Cliffs LBLOCA and SBLOCA licensing bases was performed. The evaluation resulted in a bounding ΔPCT penalty applicable throughout the EATF LTA planned operating cycles.

14.17.4.1 Unit 1

The base large break LOCA and small break LOCA ECCS performance analyses presented in Sections 14.17.2 and 14.17.3, respectively, are applicable to the current cycle of Unit 1.

14.17.4.2 Unit 2

The base large break LOCA and small break LOCA ECCS performance analyses presented in Section 14.17.2 and 14.17.3, respectively, are applicable to the current cycle of Unit 2.

14.17.5 REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors"
2. EMF-2103(P)(A), Revision 0, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," Framatome ANP, Inc., April 2003
3. Technical Program Group, Quantifying Reactor Safety Margins, NUREG/CR-5249, EGG-2552, October 1989
4. XN-CC-39(A), Revision 1, "ICECON: A Computer Program to Calculate Containment Back Pressure for LOCA Analysis (Including Ice Condenser Plants)," Exxon Nuclear Company, October 1978

5. G.P. Liley and L.E. Hochreiter, “Mixing of Emergency Core Cooling Water with Steam: 1/3 - Scale Test and Summary,” EPRI Report EPRI-2, June 1975

6. U.S. Nuclear Regulatory Commission, Information Notice 2009-23, Accession Number ML091550527, “Nuclear Fuel Thermal Conductivity Degradation,” October 8, 2009

7. EMF-2328(P)(A), Revision 0, PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, March 2001

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

9. Letter from N. S. Morgan (NRC) to G. H. Gellrich (CCNPP), dated December 19, 2012, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Safety Evaluation of the Realistic Large-Break Loss-of-Coolant Accident Summary Report (TAC Nos. ME7672 and ME7673)

10. EMF-2328(P)(A) Revision 0, Supplement 1(P)(A), Revision 0, PWR Small Break LOCA Evaluation Model, S-REAP5 Based, December 2016

TABLE 14.17-1

SAMPLED LARGE BREAK LOCA PARAMETERS

Phenomenological

- Time in cycle (peaking factors, axial shape, rod properties, and burnup)
- Break type (guillotine versus split)
- Critical flow discharge coefficients (break)
- Decay heat[(a)]
- Critical flow discharge coefficients (surgeline)
- Initial upper head temperature
- Film boiling heat transfer
- Dispersed film boiling heat transfer
- Critical heat flux
- T_{min} (intersection of film and transition boiling)
- Initial stored energy
- Downcomer hot wall effects
- SG inlet plenum interfacial effects[(b)]
- Condensation interphase heat transfer coefficient[(b)]
- Metal-water reaction

Plant[(c)]

- Offsite power availability[(d)]
- Break size
- Pressurizer pressure
- Pressurizer liquid level
- SIT pressure
- SIT liquid level
- SIT temperature (based on containment temperature)
- Containment temperature
- Containment volume
- Initial RCS flow rate
- Initial operating RCS temperature
- Diesel start (for LOOP only)

(a) Not sampled in analysis, multiplier set to 1.0.

(b) Not sampled in analysis.

(c) Uncertainties for plant parameters are based on typical plant-specific data.

(d) This is no longer a sampled parameter. One set of 59 cases is run with LOOP and another set of 59 cases is run with no-LOOP.

TABLE 14.17-2

PLANT OPERATING RANGE SUPPORTED BY THE LOCA ANALYSIS

	EVENT	OPERATING RANGE
1.0	**Plant Physical Description**	
	1.1 Fuel	
	a) Cladding outside diameter	0.440”
	b) Cladding inside diameter	0.387”
	c) Cladding thickness	0.0265”
	d) Pellet outside diameter	0.3805”
	e) Pellet density	96% of theoretical
	f) Active fuel length	136.7”
	g) Gd_2O_3 concentrations	2 to 8 w/o
	1.2 RCS	
	a) Flow resistance	Analysis
	b) Pressure location	Analysis assumes location giving most limiting PCT (broken loop)
	c) Hot assembly location	Anywhere in core
	d) Hot assembly type	14x14 HTP™ fuel
	e) SG tube plugging	≤ 10% (a)
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Nominal reactor power	2754 MWt (b)
	b) LHR	15.0 kW/ft
	c) F_Q	2.37
	d) F_r	1.810 (c)
	2.2 Fluid Conditions	
	a) Loop flow	370,000 gpm ≤ M ≤ 422,250 gpm
	b) RCS cold leg temperature	546.0°F ≤ T ≤ 554.0°F
	c) Pressurizer pressure	2164 psia ≤ P ≤ 2336 psia
	d) Pressurizer level	32.2% ≤ L ≤ 67.2%
	e) SIT pressure	194.7 psia ≤ P ≤ 264.7 psia
	f) SIT liquid volume	1090 ft^3 ≤ V ≤ 1179 ft^3
	g) SIT temperature	60°F ≤ T ≤ 125°F (Coupled with Containment temperature)
	h) SIT resistance fL/D	As-built piping configuration: Line 11A: 5.80 Line 11B: 5.72 Line 12A: 5.19 Line 12B: 5.35
	i) Minimum ECCS boron	≥ 2300 ppm
3.0	**Accident Boundary Conditions**	
	a) Break location	Cold leg pump discharge piping
	b) Break type	Double-ended guillotine or split
	c) Break size (each side, relative to cold leg pipe area)	0.2876 ≤ A ≤ 1.0 full pipe area (split) 0.2876 ≤ A ≤ 1.0 full pipe area (guillotine)
	d) Worst single-failure	Loss of one emergency diesel generator
	e) Offsite power	On or Off

TABLE 14.17-2

PLANT OPERATING RANGE SUPPORTED BY THE LOCA ANALYSIS

EVENT	OPERATING RANGE		
f) ECCS pumped injection temperature	100°F		
g) HPSI pump delay	30.0 sec (w offsite power) 30.0 sec (w/o offsite power)		
h) LPSI pump delay	45.0 sec (w offsite power) 45.0 sec (w/o offsite power)		
i) Containment pressure	13.7[d] psia, nominal value		
j) Containment temperature	60°F ≤ T ≤ 125°F		
k) Containment spray delay	20 sec		
l) HPSI flow	RCS Cold Leg Pressure (psia)	Broken loop flow (gpm)	Intact loop flow (gpm)
	14.7	164.96	155.5
	215	164.96	155.5
	615	125.25	116.8
	900	87.04	79.5
	1015	70.15	63.0
	1100	50.87	44.2
	1150	35.31	29.1
	1180	21.81	15.9
	1195	10.66	5.9
	1195.1	0	0.0
m) LPSI flow	RCS Cold Leg Pressure (psia)	Broken loop flow (gpm)	Intact loop flow (gpm)
	14.7	1713.52	1659.62
	64.7	1422.6	1377.41
	114.7	1029.27	995.9
	149.7	604.27	583.83
	159.7	393.04	379.15
	169.7	206.06	198.2
	169.8	0	0

(a) In the realistic large break LOCA analysis, only the maximum 10% tube plugging in each SG was analyzed. By independently sampling the break loss discharge coefficients, any flow differences attributed to asymmetry in the SG tube plugging is covered by use of the realistic large break LOCA methodology.

(b) Includes 17 MWt uncertainties.

(c) The radial power peaking for the hot rod includes 6% measurement uncertainty and 3.5% allowance for control rod insertion effect.

$F_{r\ limit} = F_r$ *(1+uncert_F_r)*(1+uncert_cr_insertion)=1.65*(1.0+0.06)*(1+0.035)=1.810

(d) Nominal containment pressure range is -1.0 psi to +1.8 psi. For realistic large break LOCA, a reasonable value in this range is acceptable.

TABLE 14.17-3

STATISTICAL DISTRIBUTIONS USED FOR PROCESS PARAMETERS

PARAMETER	OPERATIONAL UNCERTAINTY DISTRIBUTION	PARAMETER RANGE
Pressurizer pressure (psia)	Uniform	2164 - 2336
Pressurizer liquid level (%)	Uniform	32.2 - 67.2
SIT liquid volume (ft^3)	Uniform	1090.0 - 1179.0
SIT pressure (psia)	Uniform	194.7 - 264.7
Containment temperature (°F)	Uniform	60 - 125
Containment volume (ft^3)	Uniform	1.989E+6 - 2.148E+6
Initial RCS flow rate (gpm)	Uniform	370,000 - 422,250
Initial RCS operating temperature (T_{cold}) (°F)	Uniform	546.0 - 554.0
RWT temperature for ECCS (°F)	Point	100
Offsite power availability [(a)]	Binary	0, 1
Delay for containment spray (sec)	Point	20
LPSI pump delay (sec)	Point	30.0 (w offsite power) 30.0 (w/o offsite power)
HPSI pump delay (sec)	Point	45.0 (w offsite power) 45.0 (w/o offsite power)

(a) This is no longer a sampled parameter. One set of 59 cases is run with LOOP and one set of 59 cases is run with No-LOOP.

TABLE 14.17-4

SUMMARY OF MAJOR PARAMETERS FOR THE LIMITING PCT CASE

	FRESH 8% Gd_2O_3 FUEL	ONCE-BURNT UO_2 FUEL
Core average burnup (EFPH)	8897	8923
Core power (MWt)	2754	
Hot rod LHGR (kW/ft) / Total Peaking (F_Q)	14.3618 / 2.26884	
Radial Peaking (F_r)	1.62	1.73
ASI	-0.0878	-0.0949
Break type	Guillotine	
Break size (ft^2/side)	4.5832	
Offsite power availability	Not available	
Decay heat multiplier	1.0	

TABLE 14.17-5

SUMMARY OF HOT ROD LIMITING PCT RESULTS

CASE #47 (Offsite Power Unavailable)	FRESH FUEL 8% Gd_2O_3 ROD	ONCE-BURNT UO_2 ROD
PCT		
Temperature	1620°F*	1545°F
Time	8.52 sec	8.36 sec
Elevation	7.859 ft	7.859 ft
Metal-Water Reaction		
Pre-transient local oxidation (%)	1.214	1.997
Transient local oxidation maximum (%)	0.543	0.463
Total local oxidation maximum (%)	1.757	2.460
Total core-wide oxidation (%)	0.0111	

* The most recent 10 CFR 50.46 report contains all PCT penalties and benefits.

TABLE 14.17-6

CALCULATED EVENT TIMES FOR THE LIMITING PCT CASE

TIME (sec)	EVENT
N/A	RCP trip
0.0	Break opened
0.6	SIAS initiated
8.5	PCT occurred
13.8	Start of broken loop SIT injection
16.2, 16.1, and 16.1	Start of intact loop SIT injection (Loops 2, 3, and 4, respectively)
27.1	Beginning of core recovery (beginning of reflood)
30.6	Broken loop HPSI delivery began
30.6, 30.6, and 30.6	Intact loop HPSI delivery began (Loops 2, 3, and 4, respectively)
45.6	Broken loop LPSI delivery began
45.6, N/A, and N/A	Intact loop LPSI delivery began (Loops 2, 3, and 4, respectively)
72.0, 68.2, and 68.5	Intact loop SITs emptied (Loops 2, 3, and 4, respectively)
72.6	Broken loop SIT emptied
340.0	Transient calculation terminated

TABLE 14.17-7

CONTAINMENT HEAT SINK DATA

DESCRIPTION	SLAB MATERIAL	MATERIAL THICK. (ft)	AREA (ft^2)
Shell and Dome	Paint	2.50E-04	73230
	Carbon Steel	2.08E-02	
	Concrete	3.00E+00	
Unlined Concrete	Concrete	4.00E+00	53000
Galvanized Steel	Zinc	3.17E-04	100800
	Carbon Steel	8.33E-03	
Painted Thin Steel	Paint	2.50E-04	70250
	Carbon Steel	2.07E-02	
Painted Steel	Paint	2.50E-04	55000
	Carbon Steel	5.25E-02	
Painted Thick Steel	Paint	2.50E-04	2966
	Carbon Steel	2.01E-01	
Containment Penetration Area	Paint	2.50E-04	3000
	Carbon Steel	6.25E-02	
	Concrete	3.75E+00	
Stainless Steel Lined Concrete	Stainless Steel	1.56E-02	7925
	Concrete	4.00E+00	
Containment Liner Plate Stiffeners	Paint	2.50E-04	4000
	Carbon Steel	6.67E-01	
	Concrete	2.00E+00	
Base Slab	Concrete	8.00E+00	13300
Sump Strainer 1	Stainless Steel	1.31E-02	308.774
Sump Strainer 2	Stainless Steel	1.97E-02	161.338
Sump Strainer 3	Stainless Steel	9.83E-03	3
Sump Strainer 4	Stainless Steel	4.08E-03	3433.5
Additional H/S 1	Carbon Steel	1.00E-02	193.05
Additional H/S 2	Paint	2.50E-04	42.79
	Carbon Steel	2.08E-02	
Additional H/S 3	Paint	2.50E-04	56.54
	Carbon Steel	4.17E-02	
Improvised H/S	Stainless Steel	8.33E-02	10000

THERMAL PROPERTIES

Material	Thermal Conductivity (Btu/hr-ft-°F)	Heat Capacity Btu/ft^3-°F
Concrete	2.5	35
Carbon steel	35	55
Stainless steel	10	62
Paint	1.5	32
Zinc	70	45

TABLE 14.17-8

CONTAINMENT INITIAL AND BOUNDARY CONDITIONS

Containment Volume	
Net free volume, ft^3	1,989,000 - 2,148,090
Initial Conditions	
Compartment pressure (nominal), psia	13.7
Compartment temperature, °F	$60 \leq T \leq 125$
Outside temperature, °F	10
Humidity,%	90
Containment spray	
Number of pumps operating	2
Spray flow rate (total, both pumps), gpm	4,600
Minimum spray temperature, °F	40
Fastest post-LOCA initiation of spray, sec	20
Initial Time, sec:	
Spray flow (minimum)	20
Fans (minimum)	0

TABLE 14.17-9

SMALL BREAK LOCA ANALYSIS ECCS PERFORMANCE

Key System Parameters and Initial Conditions

Reactor power, MWt	2754[a]
Peak LHR, kW/ft	15.0
Radial peaking factor (1.65 plus uncertainties)	1.81[b]
RCS flow rate, gpm	370000
Pressurizer pressure, psia	2250
Core inlet coolant temperature, °F	548
SIT pressure, psia	194.7
SIT fluid temperature, °F	125
AFW temperature, °F	112
Low SG level AFAS setpoint for harsh conditions,% wide range	29.26[a]
HPSI /LPSI fluid temperature, °F	100
Reactor scram low pressurizer pressure setpoint for harsh conditions, psia	1790[a]
Reactor scram delay time on low pressurizer pressure, sec	0.9
Scram CEA holding coil release delay time, sec	0.5
SIAS activation setpoint pressure for harsh conditions, psia	1640[a]
HPSI pump delay time on SIAS, sec	30
LPSI pump delay time on SIAS, sec	45
MSSV lift pressures	Nominal (Tech Spec Maximum) + 3% tolerance

[a] Includes uncertainty.

[b] Includes 1.06 F_r measurement uncertainty and 1.035 F_r rodded augmentation factor.

TABLE 14.17-10

SMALL BREAK LOCA ANALYSIS ECCS PERFORMANCE

HPSI FLOW RATE VERSUS COLD LEG PRESSURE

COLD LEG PRESSURE (psia)	FLOW RATE (gpm) Loop 1A, 1B, 2A	FLOW RATE (gpm) Loop 2B (broken)
15	155.51	165.0
215	155.51	165.0
615	116.77	125.3
900	79.51	87.0
1015	63.03	70.2
1100	44.23	50.9
1150	29.05	35.3
1180	15.89	21.8
1194	5.89	10.7
1195	0.0	0.0

TABLE 14.17-11

SMALL BREAK LOCA ANALYSIS ECCS PERFORMANCE

LPSI FLOW RATE VERSUS COLD LEG PRESSURE

COLD LEG PRESSURE (psia)	FLOW Rate (gpm)		
	Loop 1A, 1B	Loop 2A	Loop 2B (broken)
14.7	0.0	1659.62	1713.52
64.7	0.0	1377.41	1422.60
114.7	0.0	995.90	1029.27
149.7	0.0	583.83	604.27
159.7	0.0	379.15	393.04
169.7	0.0	198.20	206.06
169.8	0.0	0.0	0.0

TABLE 14.17-12

SMALL BREAK LOCA ECCS PERFORMANCE ANALYSIS

CALCULATED EVENT TIMES FOR LIMITING BREAK SPECTRUM CASE

Break Size (ft^2) – 0.34

	Time (sec)
Event initiation	0.0
Pressurizer pressure reaches low PZR pressure setpoint (1790 psia)	8
Reactor trip, offsite power lost, RCPs tripped, MFW terminated, and turbine tripped	9
Pressurizer pressure reaches SIAS setpoint (1640 psia)	10
HPSI flow begins	40
Loop seal 1A clears	92
Loop seal 2B clears (broken loop)	92
Loop seal 1B clears	94
Break uncovers	106
Hot rod rupture occurs	249
SIT flow begins	264
Minimum reactor vessel mass occurs	268
PCT occurs	269
Loop seal 2A clears	274
LPSI flow loop 2A and 2B (broken)	276
LPSI flow loop 1A and 1B	---
AFW Initiated	---