14.12 ANALYSIS OF THE PRIMARY CONTAINMENT RESPONSE

This section describes the analytical bases and results supporting BFN Units 1, 2, and 3 operation at the rated thermal power (RTP) level of 3952 MWt.

BFN Units 1, 2, and 3 use the Mark I primary containment design. The main function of the Mark I containment design is to accommodate pressure and temperature conditions within the drywell resulting from a LOCA or a reactor blowdown through the MSRV discharge piping and, thereby, to limit the release of fission products to values which will ensure off-site dose rates below the 10 CFR 50.67 limits. In the event of a pipe break in the drywell, water and/or steam from the RPV are discharged into the drywell. The resulting increase in the drywell pressure forces the water and steam, along with non- condensable gases initially existing in the drywell, through the vents which connect the drywell to the suppression pool. During a reactor blowdown through the MSRVs, the steam is directly discharged into the suppression pool. The reactor blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions, such as vessel dome pressure and the mass and energy of the fluid inventory in the RPV.

The long-term heatup of the suppression pool following a LOCA is governed by the capability of the RHR System to remove decay heat which is transferred from the RPV to the suppression pool.

The Primary Containment system requirements are:

Design Pressure	$\leq$ 56 psig
Design Temperature	$\leq$ 281°F

14.12.1 Methodology

The analyses of containment pressure and temperature responses for design basis accident were performed at a power level of 102% of RTP in accordance with NEDC-32424P-A, Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate (ELTR1) (Reference 1) using GE-Hitachi (GEH) codes and models. The M3CPT code was used to model the short-term containment pressure and temperature response. The modeling used in the M3CPT analyses is described in References 2 and 3. References 2 and 3 describe the basic containment analytical models used in GEH codes. Reference 4 describes the more detailed RPV model (LAMB) used for determining the vessel break flow in the containment analyses.

The LAMB code models the recirculation loop as a separate pressure node. It also allows for inclusion of flashing in the pipe and vessel during the blowdown and flow choking at the jet pump nozzles when the conditions warrant. The use of the LAMB blowdown flow in M3CPT was identified in ELTR1 by reference to the LAMB code qualification in Reference 4.

The SHEX code was used to model the long-term containment pressure and temperature response. The results from the SHEX long-term containment temperature response (and the results from the ODYN/STEMP containment temperature response for the limiting ATWS event) were passed as inputs to the ECCS pump NPSH evaluations for each event. These NPSH evaluations demonstrated that ECCS pump NPSHA was more than adequate and quantified the resulting NPSH margins without crediting Containment Accident Pressure (CAP). The key models in SHEX are based on models described in Reference 3. The GEH containment analysis methodologies have been applied to all BWR power uprate projects performed by GEH and accepted by the NRC.

Original long-term containment analyses did not credit passive heat sinks in the drywell, torus airspace, and suppression pool. This conservative assumption was identified to the NRC as Assumption 6 of Attachment 1 to the March 12, 1993 GE letter referenced in Reference 5. Long-term containment analyses performed for BFN Units 1, 2, and 3 now includes credit for these passive heat sinks. This was identified as a change in methodology in Reference 6. These long-term containment analyses continue to conservatively neglect any heat loss from the containment through the containment walls to the reactor building or environs (Assumption 8 of the same GE letter, Reference 5).

The metal-water reaction energy versus time relationship is calculated using the method described in USNRC RG 1.7 (Reference 7) as a normalized value (fraction of reactor thermal power). All of the energy from the metal-water reaction is assumed transferred to the reactor coolant in the first 120 seconds into the LOCA. The metal-water reaction energy represents a very small fraction of the total shutdown energy transferred to the coolant.

A summary of the codes, version and NRC approval is provided below:

Computer Code	Version	NRC Approved	Comments
SHEX	06	Yes	Note 2
M3CPT	05	Yes	NEDO-10320, Apr. 1971 (NUREG-0661)
LAMB	08	Note 1	NEDE-20566-P-A September 1986

Note 1: The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566-P-A), but no approving NRC safety analysis report exists for the use of LAMB in the evaluation of Reactor Internal Pressure Differences (RIPDs) or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566-P-A.

Note 2: The application of the methodology in the SHEX code to the containment response is approved by the NRC in the letter to G. L. Sozzi (GE) from A. Thadani (NRC), "Use of the SHEX Computer Program and ANSI/ANS 5.1-1979 Decay Heat Source Term for Containment Long-Term Pressure and Temperature Analysis," July 13, 1993 (Reference 5).

14.12.2 Short-Term Containment Pressure Response

The short-term containment response analysis was performed for the limiting DBA LOCA that assumes a double ended guillotine break of a recirculation suction line to demonstrate that operation at RTP (3952 MWt) does not result in exceeding the containment design limits. The short-term analysis covers the blowdown period during which the maximum drywell pressure and suppression chamber (torus) pressure occur. The analysis was performed at 102% of RTP (4031 MWt). The time-dependent results of the limiting short-term analysis are presented in Figures 14.12-1 through 14.12-6 and are summarized in Table 14.12-6. The maximum calculated containment pressure remains within the design value.

The short-term analysis was performed for three different initial containment conditions. The Design case (D) considers the most limiting initial containment conditions of 70°F in the drywell and 2.6 psig in the drywell and 1.5 psig in the wetwell (torus). The Bounding case (B) considers initial containment conditions of 130°F in the drywell and 2.6 psig in the drywell and 1.5 psig in the wetwell, bounding normal operation. A Reference case (R) is also evaluated that assumes initial

conditions of 150°F in the drywell and 2.6 psig in the drywell and 1.5 psig in the wetwell. It is conservative to use 130ºF as the initial drywell gas temperature instead of 150ºF because it leads to a higher peak drywell temperature in the analysis use for equipment qualification evaluation. A lower initial drywell gas temperature in the analysis results in more initial non-condensable gas in the drywell which leads to a higher pressure and accordingly higher peak temperature for the small steam line break analysis. The Design case (D) and Bounding case (B) were also performed at 3458 MWt (105% Power Uprate) conditions to provide comparison for evaluating the impact of operation at RTP (3952 MWt) conditions. The key parameters used to model and analyze the plant response at RTP are provided in Table 14.12-1.

The use of the Design (D) cases initial drywell temperature is to provide the most conservative hypothesized initial conditions in order to demonstrate that a DBA LOCA will not challenge the BFN containment design pressure of 56 psig. The Design Case initial temperature of 70°F is well below the lowest drywell initial temperature that can be achieved with BFN operating at power and is therefore very conservative for demonstrating the maximum BFN containment pressure response.

The initial drywell temperature for the Bounding (B) cases was developed with a conservative historical statistical basis, which also achieve a conservative prediction of the containment pressure response due to a DBA LOCA. The containment pressure response determined from the DBA LOCA using conservative initial conditions is then used to determine a conservative value of 'Pa' for 10 CFR Part 50 Appendix J leakage rate testing. The Bounding Case initial temperature of 130°F represents a lower statistical bound of the 5-year historical normal drywell operating temperature during power operation of BFN Units 1, 2, and 3.

Analysis of the BFN containment response of the limiting power and flow statepoints within the MELLLA+ operating domain confirmed that the results shown in Figures 14.12-1 through 14.12-6 and Table 14.12-6 remain bounding.

14.12.3 Drywell and Wetwell Temperature

The bounding drywell temperature occurs during a break of a small steam line (SSLB). A spectrum of steam line break sizes have been evaluated to ensure a bounding drywell environmental qualification temperature profile is established (Figure 14.12-7.a). The analysis was performed in accordance with NUREG-0588 (Reference 9), and the most limiting drywell temperature from this analysis is shown in Table 14.12-6. Although the drywell environment may see temperatures approaching 337°F, the most limiting temperature for the drywell shell has been analyzed to be within the design temperature of 281°F. The peak drywell wall temperature from the complete spectrum of analyzed break sizes is 280.8 F. The

maximum predicted drywell shell temperatures occur at the beginning of the event prior to the initiation of drywell sprays. This is because the maximum drywell atmosphere temperature and drywell pressure conditions occur during this early period with maximum heat transfer to the drywell shell. Drywell pressure and temperature, together with drywell shell temperature, decrease when drywell spray is initiated. The drywell pressure response to SSLB is shown in Figure 14.12.7.b.

The maximum DW airspace and DW shell temperature occurs early in the event with the assumption that a High Drywell Pressure (HDWP)/Low RPV Pressure (LRPVP) LOCA signal occurs in the accident unit 10 minutes after the accident initiation. This LOCA signal at 10 minutes will result in a further 10 minute delay in the initiation of drywell spray in the accident unit (20 minute total delay for drywell spray initiation).

The torus gas space peak temperature response was calculated assuming a heat and mass transfer model between the pool and torus gas space that is calculated mechanistically. Table 14.12-6 shows the calculated peak torus gas space temperature for the DBLOCA of 174°F. The torus gas temperatures are bounded by the torus design temperature of 281°F.

MELLLA+ does not result in a change in the break energy or break flow because 1) break sizes do not change and 2) MELLLA+ does not result in a change in reactor power or reactor pressure; therefore, enthalpy of the steam does not change. Consequently, the drywell and wetwell temperature response to SSLB is unaffected by operation in the MELLLA+ domain.

14.12.4 Long-Term Bulk Suppression Pool Temperature Response-Design Basis Accidents

The long-term bulk pool temperature response at RTP is evaluated for the limiting design basis LOCA (DBA-LOCA) originally analyzed (i.e., one core spray loop and one RHR loop with two RHR pumps, two heat exchangers and two RHR service water pumps with containment spray). This DBA-LOCA is an instantaneous guillotine break of the recirculation loop suction line (RSLB). A guillotine break of a Recirculation Discharge Line (RDLB), small break LOCAs and special events (Fire event, Station Blackout, and ATWS) were also analyzed at RTP conditions.

The GE Safety communication SC 06-01 (Reference 10) identified the potential that a single failure that eliminated only the RHR heat exchanger could prove more limiting than the typically analyzed scenario of the single failure of an entire AC electrical power source. The Browns Ferry RHR system is configured with two loops of RHR, with each loop having its own separate injection point to the reactor pressure vessel, and with each loop having its own separate return to the

suppression pool. Each loop is comprised of two RHR pumps with each pump having its own separate heat exchanger on its discharge. The current licensing basis analysis (Reference 8) for the short-term (first 10 minutes after the accident) evaluation of the RSLB assumed a Single Active Failure (SAF) where only two of the four RHR pumps were available. In order to address the issue identified in SC 06-01, the RSLB RTP analysis assumes that all four RHR pumps are running in the short-term phase of the RSLB DBA-LOCA. This assumption will maximize the ECCS pump heat addition to the suppression pool and thereby maximize the suppression pool temperature. The RDLB analysis for a RTP of 3458 MWt conservatively assumed that all four RHR pumps are running in the short-term phase of the RSLB DBA-LOCA. The 3952 MWt RTP analysis also conservatively assumes all four RHR pumps are running in the short-term phase of the RSLB DBA-LOCA. Therefore, the issue identified in SC 06-01 is addressed in the 3952 MWt RTP analysis.

The sensible and decay heat do not change in the MELLLA+ operating domain. Therefore, the long term suppression pool temperature response does not change as a result of operation in the MELLLA+ operating domain.

14.12.4.1 Suppression Pool Temperature Response – RSLB DBA-LOCA

The analysis of the RSLB DBA-LOCA was performed at 102% RTP (4031 MWt). The calculated suppression pool temperature response is presented in Figure 14.12-8, the drywell and wetwell temperature responses are presented in Figure 14.12-9, and the peak value for LOCA bulk pool temperature is shown in Table 14.12-6. The RTP analysis was performed using a decay heat table based on ANS/ANSI 5.1-1979 with 2-sigma adders with additional actinides and activation products per GE SIL 636 (Reference 11). No modifications were made to this standard.

The containment system response to the accident is divided into two analysis phases. The first phase, hereafter referred to as the short-term phase covers the period up to 10 minutes after the accident initiation. During the short-term phase, no operator action is credited in the analysis. The second phase, hereafter referred to as the long-term phase covers the period after 10 minutes following the accident initiation. During the long-term phase, operator actions such as those to reduce electrical loading on the emergency diesel generators and to re-align portions of the ECCS from core cooling mode to containment cooling mode are credited. The RSLB DBA-LOCA analysis assumes that offsite power is lost concurrently with the accident initiation and that offsite power is not available during the accident analysis period. Separate RSLB analysis cases are run for EPU with initial conditions to either maximize or minimize the containment drywell and wetwell pressure response while maximizing the suppression pool temperature response in order to determine the

sensitivity of the peak suppression pool temperature response to perturbed initial conditions. No containment leakage is assumed except for the RSLB cases with initial conditions to minimize the containment drywell and wetwell pressure response while maximizing the suppression pool temperature response, for which containment leakage (2% per day) and the leakage from MSIVs (150 scfh for all steam lines) are considered. In addition, the containment responses to various modes of containment cooling are evaluated. These three RHR cooling modes are: (1) Coolant Injection Cooling (CIC), where RHR flow is cooled by the RHR heat exchanger before being discharged into the reactor vessel; (2) Containment Spray Cooling (CSC), where RHR flow is cooled by the RHR heat exchanger and then discharged to the containment via the DW spray and wetwell spray headers; and (3) Suppression Pool Cooling (SPC), where RHR flow is cooled by the RHR heat exchanger and then discharged back to the suppression pool.

A complete Loss of Offsite Power (LOOP) is assumed to occur concurrent with the accident initiation. If a worst-case SAF such as failure of one emergency electrical power source (emergency diesel generator or loss of a 4KV shutdown board) is assumed concurrent with the accident, then less than the full complement of low pressure ECCS pumps (four RHR pumps and four CS pumps) would be available during the short-term phase of the accident. However, if no SAF is assumed, then the full complement of ECCS pumps would be available. The initial condition of no SAF during the short-term phase is limiting for the determination of ECCS pump NPSH during the accident because of the Browns Ferry ECCS pump suction configuration where each ECCS pump does not have a dedicated ECCS suction strainer and piping suction directly from the suppression pool (torus). For each Browns Ferry unit, there are four ECCS suction strainers installed in the torus. The torus water volume then communicates to the ECCS pump suctions via a torus ring header located below the torus. This configuration result in higher ECCS piping head loss when there are multiple ECCS pumps running. In addition, a larger number of running ECCS pumps will lead to higher pump heat addition to the suppression pool. Conformance with GEH SC 06-01 is made by assuming all low pressure ECCS pumps start during the short-term phase of the accident.

All ECCS pumps are assumed to be available for the first 600 seconds after accident initiation. No RHRSW flow is assumed to the RHR heat exchangers and there is no heat removal from the RHR heat exchangers during the short-term phase. RPV liquid is discharged from the break into the drywell causing rapid vessel depressurization and a rapid increase in the drywell pressure and temperature. For the first 600 seconds following the accident, four RHR pumps in LPCI mode (with two RHR pumps injecting liquid into the intact recirculation loop at a flow rate of 9,000 gpm per RHR pump and the other two RHR pumps into the broken recirculation loop

at a flow rate of 9,000 gpm per RHR pump) and four CS pumps, each with flow rate of 3,550 gpm, are used to cool the core. For the RSLB DBA-LOCA, the RHR flow into the broken recirculation loop will be directed to the RPV and RHR flow will not go into runout flow because the RHR injection point is between the RPV and the closed reactor recirculation discharge valve (the reactor recirculation discharge valve in each reactor recirculation loop receives an automatic closure signal during a LOCA). HPCI is assumed available and will start on either high DW pressure or low RPV level. However, HPCI will isolate on low steam pressure. The ECCS injection of suppression pool water, along with the assumed addition of feedwater produces a recovery of the reactor water level. This allows water heated by decay heat and vessel sensible energy to be discharged into the drywell, and subsequently into the suppression pool.

If the accident were to occur on either Unit 1 or 2 and a worst-case SAF such as failure of one emergency electrical power source (emergency diesel generator or loss of a 4KV shutdown board) is assumed concurrent with the accident, then less than the full complement of low pressure ECCS pumps (four RHR pumps and four CS pumps) would be available during the long-term phase of the accident. Assuming that one RHR pump is required for shutdown of the non-accident unit, only two RHR pumps and two RHR heat exchangers are assumed available for long-term containment cooling in the accident unit.

After 600 seconds, operator actions are credited. One loop of CS with two CS pumps continues to be available for RPV water makeup. One loop of CS with two pumps is secured because two CS pumps can supply adequate long-term core cooling after accident initiation. The CS pump flow rate for the remaining CS loop is assumed in the analysis as 3,125 gpm per loop. The throttling of CS flow is not a new operator action for EPU. One loop of RHR with two pumps is secured, and another loop of RHR with two pumps is switched to a RHR mode of containment cooling with its associated RHRSW flow activated for two heat exchangers. A conservatively low RHRSW flow value of 4,000 gpm to each in-service RHR heat exchanger is assumed in the analysis. The analysis assumes operator action to throttle RHR flow to 6,500 gpm per RHR pump. Three RHR cooling modes are investigated: (1) Coolant Injection Cooling (CIC) where RHR in LPCI mode with flow from the suppression pool is cooled by the RHR heat exchanger before being discharged into the reactor vessel; (2) Containment Spray Cooling (CSC) where RHR flow from the suppression pool is discharged as drywell and wetwell sprays; and (3) Suppression Pool Cooling (SPC) where the RHR flow from the suppression pool is cooled by the RHR heat exchanger before being discharged back into the suppression pool. The heat exchanger K-value and RHR pump flow rate are presented in Table 14.12-1. Initial conditions (initial DW pressure, initial wetwell pressure and initial DW temperature) are also perturbed in separate analysis cases to both maximize and minimize the peak containment pressure and thereby investigate the effect on peak suppression pool temperature. The resulting calculated peak bulk SP temperature for RSLB

DBA-LOCA at 10 minutes after the accident initiation is 152.8°F and the peak bulk SP temperature for RSLB DBA-LOCA is 179.0°F.

The possible effect of containment cooling interruption on the accident unit due to concurrent shutdown and cooldown of the non-accident units was also investigated. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA signal due to high drywell pressure (if high DW pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room and is a normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on both the accident and the non-accident unit. Therefore, there are no additional containment cooling interruptions on the accident unit due to interaction from the non-accident units.

14.12.4.2 Suppression Pool Temperature Response – Recirculation Discharge Line Break

The containment response during the first 10 minutes following the accident initiation for a RDLB was calculated using Browns Ferry specific inputs to maximize suppression pool temperature and minimize containment pressure, similar to the RSLB DBA-LOCA analysis. The key parameter differences between the RDLB and the RSLB during the short-term phase of the accident are: 1) the break area (4.2 ft2 for the RSLB versus 1.94 ft2 for the RDLB), and 2) the RHR flow rate and RHR injection path into the broken recirculation loop.

For the RDLB LOCA, all ECCS pumps are assumed to be available for the first 600 seconds. No RHRSW flow is assumed to the RHR heat exchangers and there is no heat removal from the RHR heat exchangers during the short-term phase. RPV liquid is discharged from the break into the drywell causing rapid vessel depressurization and a rapid increase in the drywell pressure and temperature. For the first 600 seconds following the accident, four RHR pumps in LPCI mode (with two RHR pumps injecting liquid into the intact recirculation loop at a flow rate of 9,000 gpm per RHR pump and the other two RHR pumps into the broken recirculation loop at a flow rate of 11,000 gpm per RHR pump) and four CS pumps, each with flow rate of 3,550 gpm, are used to cool the core. For the RDLB LOCA, the RHR flow into the broken recirculation loop discharges directly to the drywell and the RHR flow into the broken loop is assumed at runout conditions. HPCI is assumed available and will start on either high DW pressure or low RPV level. However, HPCI will isolate on low steam pressure. The ECCS injection of suppression pool water, along with the

assumed addition of feedwater produces a recovery of the reactor water level. This allows water heated by decay heat and vessel sensible energy to be discharged into the drywell, and subsequently into the suppression pool. The resulting calculated peak bulk SP temperature for the RDLB at 10 minutes after the accident initiation is 152.0°F.

14.12.4.3 Suppression Pool Temperature Response – Small Steam Break LOCA

For the Browns Ferry small break LOCA, a spectrum of small steam line breaks was evaluated. Initial reactor conditions are consistent with operation at 102% of 3952 MWt RTP, and the same decay heat, relaxation and metal-water reaction energies are assumed as is used for the large DBA-LOCA analysis. Consistent with the large DBA-LOCA assumptions, a complete LOOP is assumed. A worst-case single failure is also assumed for this analysis to minimize the available quantity of containment cooling. This single failure is either the failure to start an EDG or the loss of a 4KV shutdown board. The single failure assumption will result in no more than three CS pumps and three RHR pumps automatically starting on either low RPV level or High DW Pressure (HDWP) concurrent with Low RPV Pressure (LRPVP). For cases where HPCI is assumed available, HPCI will automatically start on either HDWP or on low RPV level.

Cases with HPCI (high pressure ECCS) available and with no HPCI available are evaluated to determine the effect of the availability of high pressure ECCS on the limiting peak pool temperature and the limiting drywell temperature. The condensate storage tank is assumed unavailable during the accident and HPCI pump suction is assumed available only from the suppression pool. For Browns Ferry, HPCI is qualified only for water temperatures up to 140°F. If HPCI is conservatively assumed available, HPCI will provide reactor inventory makeup until the reactor pressure decreases below the HPCI isolation pressure, after which low-pressure ECCS provides reactor inventory makeup. If HPCI is not available, ADS would be used to rapidly reduce reactor pressure to allow low-pressure ECCS to provide vessel makeup. Such use of ADS results in a faster heatup of the suppression pool. With reactor pressure at the time of peak pool temperature the same, the total (integrated) sensible heat addition to the suppression pool remains the same, but the total (integrated) decay heat to the pool at the time of peak suppression pool temperature is less for the fast pool heatup. In addition, the heat removed from the pool is greater for the faster pool heatup. Thus, a faster pool heatup will result in a lower peak suppression pool temperature. For this reason, the assumption of crediting the HPCI as available until it isolates on low steam pressure is conservative for the determination of a peak suppression pool temperature response.

Automatic starting of ECCS pumps will occur in accordance with their start logic and timing for electrical loading. Automatic start of CS and RHR will result in reactor vessel inventory makeup provided by three CS pumps and three RHR pumps in LPCI mode. Operators initiate depressurization of the RPV at 100°F/ hour when the suppression pool temperature reaches 120°F. At no sooner than 10 minutes after the start of the accident, operators will stop the third RHR pump and third CS pump. When containment conditions permit (drywell and wetwell pressures and drywell temperatures), operators will either re-align or start two RHR pumps in containment spray mode (two RHR pumps at 6,500 gpm each with two RHR heat exchangers with a K-factor of 265 BTU/sec-ºF per heat exchanger) and one CS loop (two CS pumps with maximum flow of 3,125 gpm/pump). For breaks greater than 0.01 ft^2, drywell and wetwell spray initiation is assumed delayed by up to 1,200 seconds (20 minutes) to address concerns related to ECCS interruption caused by a subsequent LOCA signal activated on HDWP concurrent with LRPVP. For the smallest break (0.01 ft^2), the late LOCA signal will occur much later in the event and the operator would inhibit the late LOCA signal and the additional drywell spray delay will not occur. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room which is a normal action per Browns Ferry procedures to prevent the generation of a LOCA signal that could result in interruption of containment cooling when core cooling has already

been confirmed. RPV depressurization is terminated when RPV pressure reaches 50 psig. Because Browns Ferry is a hot shutdown plant, entry into alternate shutdown cooling for entering cold shutdown is not required. Operators will maintain the plant at this pressure until Normal Shutdown Cooling (NSDC) can be restored. The resulting calculated peak bulk SP temperature for a steam line break is 182.7°F, which occurs for the 0.01 ft^2 break. Figure 14.12-10 shows the suppression pool temperature response for the limiting break size of 0.01 ft^2.

The peak suppression pool temperature of 182.7°F is for the case where HPCI is assumed available and the initial DW temperature is 70°F. The sensitivity of this peak suppression pool temperature due to initial DW temperature was investigated by setting the initial DW temperature to 150°F. The resulting calculated peak bulk SP temperature for a 0.01 ft^2 steam line break with initial DW temperature of 150°F is 182.7°F, which demonstrates the insensitivity of initial DW temperature on the peak suppression pool temperature. The peak SP temperature for the limiting 0.01 ft^2 steam line break where HPCI was assumed unavailable was 181.5°F, which demonstrates that the assumption of HPCI availability during the event is conservative.

The possible effect of containment cooling interruption on the accident unit due to concurrent shutdown and cooldown of the non-accident units was also investigated. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA signal due to high drywell pressure (if high DW pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room which is a normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on both the accident and the non-accident units. Therefore, there are no additional containment cooling interruptions on the accident unit due to interaction from the non-accident units.

The peak suppression pool temperature for small liquid line breaks is bounded by the containment analysis results for small steam line breaks. Because the break flow is an isenthalpic process, the steam line breaks result in superheated steam discharged to the drywell atmosphere, while the liquid line breaks result in two-phase break flow into the drywell atmosphere. Therefore, the drywell temperature response for the small liquid line breaks is less severe than for the small steam line breaks. The emergency operating instruction entry point for drywell spray is either not reached for the liquid line breaks or the duration of drywell spray operation is much less for the liquid line breaks than for the steam line breaks. Because the non-operation or limited operation of drywell sprays for the liquid line breaks allows a greater holdup of the sensible heat in the drywell, less heat is discharged to the

suppression pool for the liquid line breaks and the peak suppression pool temperature is consequently less for a given liquid line break size than for the same size steam line break.

14.12.4.4 Suppression Pool Temperature Response – Non-Accident Units

Evaluation of the containment response for the non-accident units was also evaluated based on the conservative assumption that the RSLB DBA-LOCA occurs concurrently with a LOOP, resulting in reactor isolation and scram on the non-accident unit. The term “non-accident unit” refers to the Browns Ferry unit that is both not experiencing a LOCA and has the minimum containment cooling equipment available. This evaluation is applicable to either of the following conditions: 1) a LOOP for all three Browns Ferry units (with no LOCA), or 2) a LOCA on any one unit concurrent with a simultaneous LOOP for the remaining two units. The bounding condition has been evaluated for two scenarios that either assume the CST is available or assume the CST is not available.

For Units 1 and 2, there are four 4kV shutdown boards (4kV shutdown board A, B, C and D) shared between the two units. Each 4kV shutdown board is supplied during a LOOP by a safety-related emergency diesel generator (EDG). Power distribution to 480V shutdown boards and 480V Reactor Motor Operated Valve (RMOV) boards is redundant in that each 480V board can be supplied power from two of the Unit 1 and 2 shared 4kV shutdown boards. For Unit 3, there are four dedicated 4kV shutdown boards (4kV shutdown board 3EA, 3EB, 3EC and 3ED). Each 4kV shutdown board for Unit 3 is supplied during a LOOP by a safety-related EDG. Power distribution to Unit 3 480V shutdown boards and 480V RMOV boards is redundant in that each 480V board can be supplied power from two of the 4kV shutdown boards designated for Unit 3. In addition, the 4kV electrical distribution system at Browns Ferry allows a Unit 3 EDG to either power a de-energized Unit 1 and 2 4kV shutdown board or to operate in parallel with a Unit 1 and 2 EDG for powering a Unit 1 and 2 4kV shutdown board.

Conservatively assuming that the DBA-LOCA occurs concurrently with a LOOP, reactor isolation and scram will occur on the non-accident units. Concurrent with the LOOP, the worst case single failure for containment cooling is the loss of a 4kV shutdown board (A, B, C or D) shared between Units 1 and 2. This single failure is more severe than loss of an EDG alone because it prevents repowering the lost (de-energized) 4kV shutdown board from one of the Unit 3 EDGs. For this assumed electrical power failure, only three RHR pumps would be available for either core or containment cooling between Units 1 and 2. The LOCA analysis assumes that two of these RHR pumps would be used for long-term containment cooling in the accident unit.

Paralleling of a Unit 3 EDG with the EDG supplying power to the non-accident unit (so that two EDGs are supplying power to one Unit 1 and 2 4kV shutdown board) is not assumed. Therefore, EDG power limitations are conservatively assumed that allow the starting and alignment of only one RHR pump and one RHR heat exchanger for containment cooling on the non-accident unit.

The loss of the 4kV shutdown board may also result in loss of the normally aligned power to the 480V shutdown board and the 480V RMOV board that supplies power to the RHR Shutdown Cooling (SDC) isolation valves for the non-accident unit. However, the Browns Ferry electrical system configuration is such that there are redundant means of re-powering the 480V shutdown boards and the 480V RMOV boards that supply power to both additional DW coolers and the RHR SDC isolation valves. Therefore, there is no loss of the ability to place RHR SDC into service due to electrical power limitations. Drywell cooling is initially lost for the non-accident unit because the LOOP signal in conjunction with a LOCA signal on the accident unit causes the loads to be stripped from the 4kV shutdown boards and then re-sequenced on as the EDGs repower the 4kV shutdown boards. Within 90 seconds after the LOOP, a minimum of four drywell coolers are automatically re-started and are available for DW cooling in the non-accident unit. Operators are able to manually restart DW coolers and restore power to RHR SDC isolation valves later in the event using the redundant power sources mentioned above.

The capability of the non-accident unit to achieve cold shutdown was analyzed at 102% of 3952 MWt RTP and ANS/ANSI 5.1-1979 with 2-sigma adders decay heat. The decay heat model includes additional actinides and activation products per GE SIL 636. The analysis includes the assumption of reactor shutdown initiated by a loss of offsite power (for all three Browns Ferry units) with concurrent loss of a 4kV shutdown board shared between Units 1 and 2. Two scenarios are evaluated. Scenario 1 assumes that the Condensate Storage Tank (CST) volume is available and HPCI provides high pressure inventory makeup to the RPV with HPCI pump suction from the CST. Scenario 2 assumes that the CST volume is not available and HPCI provides high pressure inventory makeup to the RPV with HPCI pump suction from the suppression pool. Initial conditions and key input parameters for the non-accident unit containment response evaluation are shown in Table 14.12-2.

Scenario 1 - CST Available

The event is initiated by LOOP. The LOOP causes a reactor scram, a containment isolation signal due to loss of power to the Nuclear Steam Supply System (NSSS) isolation relays, tripping of the feedwater pumps and loss of power to the drywell coolers. The MSIVs are assumed to be fully closed at 3.5 seconds after event initiation. In the analysis, the FW temperature is initially at or above 337°F (saturation temperature is at 100 psig). Following the closure of the MSIVs, the

feedwater is assumed to flash to steam and then is injected into the reactor vessel. The feedwater mass entering the vessel after closure of the MSIVs is conservatively assumed to come into thermal equilibrium with the downstream feedwater piping as the feedwater travels toward the vessel. Feedwater injection into the vessel is assumed to resume when the RPV pressure is reduced to below 220 psig which ensures that all hot feedwater at a temperature equal to and greater than 337ºF is injected into the vessel before the suppression pool temperature peaks. This assumption is conservative because the timing results in the FW enthalpy addition occurring late in the event when the suppression pool temperature is high and will therefore result in a more conservative (higher) suppression pool temperature response.

The MSRVs will automatically cycle to control RPV pressure. At 90 seconds into the event, four drywell coolers will have automatically restarted. The HPCI pump will automatically start on low RPV level with HPCI pump suction from the CST. At ten minutes after reactor shutdown, the operators align one loop of RHR (one RHR pump, one RHR heat exchanger and RHRSW cooling flow of 4500 gpm to the RHR heat exchanger) in suppression pool cooling mode with a flow rate of 9700 gpm. At approximately 20 minutes after the start of the event, operators are assumed to restart an additional four drywell coolers to provide additional cooling to the non-accident unit drywell and restore power to NSSS isolation relays. When the non-accident unit suppression pool temperature reaches 110°F, but no sooner than ten minutes after reactor shutdown, the operators commence manual reactor depressurization and reactor cooldown at a rate of 100°F/hr. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA signal due to high drywell pressure (if high drywell pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room and is a normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on the non-accident unit. HPCI is assumed to isolate on low RPV pressure when RPV pressure decreases to 150 psig. A single core spray pump is started to provide RPV inventory makeup after HPCI is no longer available. Further depressurization of the RPV to 100 psig is accomplished by opening MSRVs.

When RPV pressure reaches 100 psig, the analysis assumes that the operators will maintain the RPV at this pressure. Operators stop the RHR pump in suppression pool cooling and begin transitioning RHR to shutdown cooling mode. The transition to place shutdown cooling in operation is assumed to take 20 minutes. During this 20 minute transition period from RHR in suppression pool cooling to shutdown cooling there is no cooling of the suppression pool. Cooldown of the RPV to cold

shutdown conditions on the non-accident unit is accomplished with SDC. Cold shutdown is achieved when bulk reactor liquid water temperature is less than or equal to 212°F. The peak bulk suppression pool cooling temperature is 185.1°F. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.12-11.

Scenario 2 - CST Not Available

The event is initiated by LOOP. The LOOP causes a reactor scram, a containment isolation signal due to loss of power to the NSSS isolation relays, tripping of the feedwater pumps and loss of power to the drywell coolers. The MSIVs are assumed to be fully closed at 3.5 seconds after event initiation. In the analysis, the feedwater temperature is initially at or above 337°F (saturation temperature is at 100 psig). Following the closure of the MSIVs, feedwater is assumed to flash to steam and then is injected into the vessel. The feedwater mass entering the RPV after closure of the MSIVs is conservatively assumed to come into thermal equilibrium with the downstream feedwater piping as the feedwater travels toward the reactor vessel. Feedwater injection into the vessel is assumed to resume when the RPV pressure is reduced to below 220 psig which ensures that all hot FW at a temperature equal to and greater than 337ºF is injected into the vessel before the suppression pool temperature peaks. This assumption is conservative because the timing results in the FW enthalpy addition occurring late in the event when the suppression pool temperature is high and will therefore result in a more conservative (higher) suppression pool temperature response.

The MSRVs will automatically cycle to control RPV pressure. At 90 seconds into the event, four drywell coolers will have automatically restarted. The HPCI pump will automatically start on low RPV level with HPCI pump suction from the suppression pool. The CST volume is assumed to not be available, consistent with the assumptions used for the containment system response for a LOCA. HPCI provides reactor inventory makeup until suppression pool temperature reaches 140°F. If the suppression pool temperature reaches 140°F, HPCI is secured because HPCI availability cannot be assured with a SP temperature greater than 140°F. At ten minutes after reactor shutdown, the operators align one loop of RHR (one RHR pump, one RHR heat exchanger and RHRSW cooling flow of 4500 gpm to the RHR heat exchanger) in suppression pool cooling mode with a flow rate of 9700 gpm. At approximately 20 minutes after the start of the event, operators are assumed to restart an additional four drywell coolers to provide additional cooling to the non-accident unit drywell and restore power to NSSS isolation relays. When the non-accident unit suppression pool temperature reaches 110°F, but no sooner than ten minutes after reactor shutdown, the operators commence manual reactor depressurization and reactor cooldown at a rate of 100°F/hr. Prior to depressurizing the non-accident unit below the RPV pressure that would result in initiating a LOCA

signal due to high drywell pressure (if high drywell pressure exists) in conjunction with low RPV pressure, the operators recognize that a LOCA signal could occur and therefore inhibit the generation of a LOCA signal from the non-accident unit. This LOCA signal inhibit is performed by the operation of permanently installed hand switches in the Browns Ferry main control room and is a normal action per Browns Ferry procedures to prevent the generation of a false LOCA signal that could result in interruption of containment cooling on the non-accident unit. When the SP temperature reaches 140°F, the analysis assumes that HPCI is secured. A single core spray pump is started to provide RPV inventory makeup after HPCI is no longer available. Further depressurization of the RPV to 100 psig is accomplished by opening MSRVs.

When the RPV pressure reaches 100 psig, the analysis assumes that the operators will maintain the RPV at this pressure. Operators stop the RHR pump in suppression pool cooling and begin transitioning RHR to SDC mode. The transition to place SDC in operation is assumed to take 20 minutes (Browns Ferry operators confirmed that this assumed operator action time can be achieved). During this 20 minute transition period from RHR in suppression pool cooling to SDC, there is no forced cooling of the suppression pool from RHR. Cooldown of the RPV to cold shutdown conditions on the non-accident unit is accomplished with SDC. Cold shutdown is achieved when bulk reactor liquid water temperature is less than or equal to 212°F. The peak bulk suppression pool cooling temperature is 180.0°F.

14.12.5 Long-Term Bulk Suppression Pool Temperature Response-Special Events

14.12.5.1 Station Blackout

The containment response to a station blackout (SBO) was evaluated at 100% RTP (3952 MWt) in the MELLLA operating domain. This evaluation remains applicable and is unchanged for operation in the MELLLA+ operating domain. The evaluation used the NRC approved method of NUMARC 87-00 (Reference 12) and NRC Regulatory Guide 1.155 (Reference 13). BFN is evaluated as an “Alternate AC Approach” plant per Reference 12. Alternate AC approach would entail a short period of time in an AC Independent state (up to one hour) while operators initiate power from the backup source. The BFN SBO assessment assumes only one unit in station blackout with the other two units available to supply Alternate AC to blacked-out unit. Use of Alternate AC power is limited to providing the required cooling systems to certain areas (control room, control bay, and electrical board rooms). The SBO coping duration for BFN is four hours. The containment response was analyzed using the SHEX analysis code as was utilized for the design basis accident responses. In accordance with Reference 12 and 13 methodologies, the containment response was evaluated with consideration of both zero reactor coolant system leakage into the primary containment and 61 gpm

reactor coolant system leakage into the primary containment. Conservative Technical Specification limits were used as inputs to the containment response analyses. The specific values are shown in Table 14.12-3.

The SBO scenario is based on a LOOP which causes turbine trip and reactor scram on all three BFN Units. One BFN unit suffers a SBO and HPCI and RCIC are the only available sources of makeup to maintain reactor water level on the SBO unit. HPCI and RCIC start on low reactor level and take suction from the condensate storage tank (CST). Automatic and manual actuation of safety relief valves (MSRVs) provides reactor pressure control. Operators begin manual control of MSRVs to control pressure and cooldown rate at approximately 20 minutes into the SBO event. One hour after the SBO event initiation, alternate AC is available from non - SBO units to provide HVAC loads (control room, control bay and electrical board rooms) for the remainder of the SBO coping duration. The SBO coping duration ends at four hours and containment cooling capability is restored. The peak bulk suppression pool temperature for this analysis at 100% RTP conditions is 203.7°F and the peak drywell pressure is 43.4 psia. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.12-12.

In addition to the containment response, CST inventory, battery capacity, compressed gas capacity, the effects of loss of ventilation, and containment isolation capability were evaluated to verify the response to a SBO.

Condensate Inventory for Decay Heat Removal

Analyses have shown that the BFN condensate inventory is adequate to meet the SBO coping requirement for RTP conditions. The current CST inventory reserve for RCIC and HPCI use ensures that adequate water volume is available to remove decay heat, depressurize the reactor and maintain reactor vessel level above the top of active fuel (TAF) during the coping period.

Class 1E Battery Capacity

Evaluation of the BFN Class 1E Battery Capacity has shown that BFN has adequate battery capacity to support decay heat removal during a SBO for the required coping duration. The battery capacity remains adequate to support required coping equipment operation at RTP.

Compressed Gas Capacity

BFN meets the requirement for compressed gas capacity. An evaluation has shown that the BFN air operated MSRVs required for decay heat removal have sufficient compressed gas capacity for the required automatic and manual operation during the

SBO event at RTP conditions. Sufficient capacity remains to perform emergency RPV depressurization in case it is required. Adequate compressed gas capacity exists to support the MSRV actuations because the maximum number of MSRV valve operations is less than the capacity of the pneumatic supply.

Effects of Loss of Ventilation

Areas containing equipment necessary to cope with an SBO event were evaluated for the effect of loss of ventilation due to an SBO. The evaluation shows that equipment operability is maintained because the SBO environment is milder than the existing design and qualification bases.

Containment Isolation

The containment isolation capability was shown to be within the existing design and qualification basis.

14.12.5.2 Fire Event

The limiting NFPA 805 fire events were analyzed at RTP conditions in the MELLLA operating domain. This analysis remains applicable for operation in the MELLLA+ operating domain. The fuel heat-up analysis was performed using the NRC approved, AREVA LOCA Methodology (RELAX/HUXY). The containment analysis was performed using the GEH SHEX model. These analyses determined the affect of EPU on fuel cladding integrity and containment integrity as a result of the fire event. The two bounding cases described below are identified as “Case 1” and “Case 4.” Key inputs to the fire event analyses are shown in Table 14.12-4.

Case 1 : The bounding safe shutdown case for PCT has Multiple Spurious Operation (MSO) of 11 of the 13 MSRVs, which depressurize the reactor, and one RHR pump aligned in the LPCI/ASDC mode at 20 minutes, The analysis shows that the calculated 1330°F PCT is acceptable from a deterministic perspective (<1500°F).

Case 4: The bounding safe shutdown case for peak suppression pool temperature has reactor depressurization beginning at 25 minutes using three MSRVs. As the reactor is depressurized, condensate pumps replenish reactor inventory until hotwell inventory is depleted. After condensate is secured, one RHR pump is aligned into LPCI/ASDC mode. One RHRSW pump is initiated at 2 hours. Peak SP temperature reaches 207.7°F and this meets the containment integrity acceptance criteria of <281°F. The peak suppression pool temperature is less than qualification temperature of the torus attached piping. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.12-13.

14.12.5.3 Anticipated Transients Without Scram (ATWS)

The BFN ATWS evaluation reviewed the results of the ATWS analyses considering the limiting cases for RPV overpressure and for suppression pool temperature / containment pressure. Previous evaluations considered four ATWS events, Main Steam Isolation Valve Closure (MSIVC), Pressure Regulator Failure – Open (PRFO), Loss of Off-Site Power (LOOP), and Inadvertent Opening of a Relief Valve (IORV). Consistent with the event selection disposition contained in Section L.3.3 of Reference 1 (ELTR-1), these four events were analyzed at RTP for the containment system response (suppression pool temperature and containment pressure) from ATWS.

The limiting containment response for ATWS occurs with initial conditions of 3952 MWt / 85% core flow (MELLLA+ boundary) versus 3952 NWt / 99% core flow (MELLLA boundary). This is because operation in the high power/low flow statepoint in the MELLLA+ domain results in reduced effectiveness of the ATWS recirculation pump trip (ATWS-RPT) for lowering reactor power and there is a resultant increase in the integrated energy deposition to the suppression pool (torus) from MSRV discharge.

The 3952 MWt with MELLLA+ ATWS analyses for the containment response were performed using the NRC approved code ODYN, to determine the heat addition to the suppression pool from MSRV flow, and STEMP, to determine the suppression pool heatup due to energy input from the MSRVs. The key inputs and limiting results for the containment response to ATWS events are presented in Tables 14.12-5 and 14.12-6. The limiting ATWS event with respect to containment response and ECCS pump net positive suction head for Browns Ferry is LOOP where the peak suppression pool temperature is 174.5°F. The resulting time-dependent bulk suppression pool temperature response is presented in Figure 14.12-14. The peak suppression pool temperature and containment pressure results are well below the containment design temperature and pressure. Therefore, the Browns Ferry 3952 MWt with MELLLA+ ATWS analysis for the containment response complies with the acceptance criteria of 10CFR50.62.

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