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See Proposed Change To Tech Specs

SUBJECT: Application for amend to license DPR-23,increasing max allowable SW temp used to determine operability of USH from 95 F to 97 F.

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10 CFR 50.90

CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville SC 29550

MAY 27 1999

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United States Nuclear Regulatory Commission
Attn: Document Control Desk
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H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REQUEST FOR TECHNICAL
SPECIFICATIONS CHANGE ULTIMATE HEAT SINK (UHS)

Dear Sir or Madam:

Carolina Power & Light (CP&L) Company requests a change to the Technical Specifications (TS) for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 in accordance with 10 CFR 50.90. The requested change increases the maximum allowable Service Water (SW) temperature used to determine OPERABILITY of the Ultimate Heat Sink (UHS) from 95°F to 97°F. As a result of new analyses performed to support the increase of the maximum allowable SW temperature, TS changes are also requested to decrease the required actuation setpoint and ALLOWABLE VALUE for the Containment Pressure High High signal, decrease the required closure time for the Main Feedwater Isolation Valves, increase the required Isolation Valve Seal Water (IVSW) tank pressure and IVSW nitrogen bottle pressure, and increase the peak calculated containment internal pressure P_a. The TS Bases have been changed to reflect the TS changes, the results of the new analyses, and a faster closing time for the Main Steam Isolation Valves, which was credited in the analysis for a Main Steam Line Break inside containment.

In addition, CP&L's commitment to Inspection and Enforcement Bulletin (IEB) 80-06, "Engineered Safety Features (ESF) Reset Controls," must be modified as a result of the new analyses supporting the SW temperature increase. The Containment Spray (CS) system actuation circuitry will be modified so that it can be blocked while the CS pump suction is being switched from the Refueling Water Storage Tank following a large break Loss-of-Coolant Accident (LOCA). The new analyses show that containment pressure will be higher than the actuation setpoint at the time switchover is completed, and the CS pumps may need to be manually restarted following the switchover. The actuation circuitry currently does not allow the CS pumps to be

A001

restarted while the original actuation signal is present. The modification to the CS actuation circuitry will be consistent with the guidance of IEB 80-06.

Attachment I provides an affidavit as required by 10 CFR 50.30(b).

Attachment II provides a description of the current condition, a description of the proposed change, a safety assessment, a basis for a conclusion that the proposed change does not involve a significant hazards consideration and an environmental impact consideration which demonstrates that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)((9) and (10)).

Attachment III provides a markup of the revised TS and Bases.

Attachment IV provides retyped pages for the TS and Bases.

In accordance with 10 CFR 50.91(b), CP&L is providing the State of South Carolina with a copy of this letter with attachments.

CP&L requests that the proposed change be reviewed and approved by June 30, 2000, for implementation prior to restart after Refueling Outage (RO) 20. The proposed change does not involve a significant hazards consideration.

If you have questions concerning this matter, please contact Mr. Harold Chernoff.

Sincerely,

R. L. Warden
Manager - Regulatory Affairs

DNB/dnb
Attachments
I. Affidavit
II. Request For Technical Specifications Change, Ultimate Heat Sink (UHS)
III. Markup Of Current Technical Specifications And Bases
IV. Retyped Technical Specifications And Bases

c: M. K. Batavia, Chief, Bureau of Radiological Health (SC)
L. A. Reyes, NRC, Region II
R. Subbaratnam, NRC, NRR
NRC Resident Inspector, HBRSEP
Attorney General (SC) (w/out Attachments)

Affidavit

State of South Carolina
County of Darlington

D. E. Young, having been first duly sworn, did depose and say that the information contained in letter RNP-RA/99-0025 is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

Dale E Young

Sworn to and subscribed before me

this 27th day of May 1999

(Seal) Albert L Carron
Notary Public for South Carolina

My commission expires: March 22nd 2005

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE
ULTIMATE HEAT SINK (UHS)

Description of Current Condition

The Ultimate Heat Sink (UHS) provides a heat sink for the operating and decay heat produced by various plant components during normal operation, transients, and accidents. The Service Water (SW) system and the Component Cooling Water (CCW) system are used to transfer heat from plant components to the UHS. The SW system draws water directly from the UHS to provide cooling water to several plant components. Also, the SW system cools the CCW system, which in turn, cools other plant components. The CCW system serves as a second barrier to prevent leakage of potentially radioactive fluid directly to the SW system and environment from plant components containing reactor coolant.

The UHS for H. B. Robinson Steam Electric Plant, Unit No. 2 is Lake Robinson, as noted in Updated Final Safety Analysis Report (UFSAR), Section 9.2.4, "Ultimate Heat Sink." Lake Robinson was developed for use initially for condenser cooling of HBRSEP, Unit No. 1, a fossil plant. When HBRSEP, Unit No. 2, a nuclear plant, was licensed on July 31, 1970, the unit was designed to use Lake Robinson both for condenser cooling and UHS. HBRSEP, Unit No. 2 was licensed in accordance with the proposed draft General Design Criteria and was licensed prior to the promulgation of 10 CFR 50, Appendix A. Therefore, the UHS was not designed to satisfy the requirements of the final General Design Criteria. Additionally, the UHS was not designed to satisfy Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Position C.1, which stipulates a 30 day cooling supply. The UHS for HBRSEP, Unit No. 2 is capable of providing cooling water for at least 22 days following a design basis accident, as stated in the Bases to LCO 3.7.8, "Ultimate Heat Sink."

The two principal safety functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident. The basic performance requirements for the UHS are that a 22 day supply of water be available, and that the design basis temperatures of safety related equipment not be exceeded. These performance requirements are verified through periodic surveillances that ensure that lake water level is ≥ 218 feet mean sea level and SW inlet temperature is ≤ 95°F while the plant is operating in MODES 1, 2, 3, and 4. If either of these surveillances is not satisfied while the plant is operating in MODES 1, 2, 3, and 4, the plant is required to be in MODE 3 within 6 hours and in MODE 5 within 36 hours. No time is allowed for SW temperature to exceed 95°F without initiating preparations to shut down the plant. The current requirement for SW inlet temperature to be ≤ 95°F was incorporated into the Technical Specifications upon implementation of the Improved Technical Specifications in November 1997.

As a result of unusually hot, dry weather in 1998, it appeared that the UHS temperature could exceed 95°F, which would require shutting down the plant. Prior to 1998, the containment average air temperature and the thermal discharge limits of National Pollutant Discharge Elimination System (NPDES) permit were more limiting to plant operation during hot weather than the SW temperature. As a result of the monthly averaging requirement for peak daily thermal discharge limit being removed from the NPDES permit in October 1997, and measures to improve containment air cooling being implemented during the refueling outage in the spring of 1998, the SW temperature, which was formerly masked by other limitations, became more limiting during the hot, dry weather conditions in 1998.

In anticipation of the UHS temperature exceeding 95°F, Carolina Power & Light (CP&L) Company requested a change to Technical Specification LCO 3.7.8 by a letter dated June 26, 1998, that would allow plant operation above 95°F for up to 8 hours. The purpose of the change was to reduce the risk associated with plant shutdown transients. The Technical Specification change was supported by an engineering evaluation, which concluded that the components that rely on the SW system for cooling are able to operate at a SW temperature of up to 99°F. Also, a Probabilistic Safety Assessment (PSA) evaluation determined that allowing 8 hours for the UHS temperature to return to ≤ 95°F would not impact the PSA model success criteria, and if the UHS temperature exceeded the 95°F for less than 8 hours, not having to shut down the plant decreases the core damage probability. The SW system mitigates the containment response for a Main Steamline Break (MSLB) inside containment and a large break Loss of Coolant Accident (LOCA) inside containment. These events were not reanalyzed at the higher SW temperature, but the probability is very low that a MSLB or large break LOCA would occur during the limited time when the UHS was above 95°F.

Prior to the Technical Specification change being approved, unusually hot and dry weather conditions prompted CP&L to request a Notice of Enforcement Discretion (NOED) by a letter dated June 27, 1998 until the Technical Specification change could be approved. The request proposed a similar change to Technical Specification LCO 3.7.8 with an upper temperature limit of 99°F, and as a long term resolution for this condition, committed to perform an engineering analysis to justify an increase in the allowed SW temperature. The request for a NOED was accepted by the NRC on July 1, 1998.

Based on a request from the NRC Staff, CP&L subsequently submitted a supplement to the requested change to Technical Specification LCO 3.7.8 by a letter dated July 22, 1998, that limited the effective period of the change until September 30, 1998. The provisions of License Amendment No. 179, which were effective through September 30, 1998, were issued by letter dated July 29, 1998.

In March 1999, CP&L requested a Technical Specification change that would allow UHS temperature to be above 95°F for 8 hours and less than 99°F before a plant shutdown is required. Further evaluations and calculations performed since the summer of 1998 showed that the change did not increase the core damage frequency, had a negligible effect on the large early release frequency, and reduced the potential for plant shutdown transients. The proposed change did not

request a long-term increase in UHS temperature because supporting engineering analyses were still in progress.

In April 1999 after discussion with the Staff, CP&L requested a one-time Technical Specification change that would allow UHS temperature to be above 95°F and less than 99°F for 8 hours before a plant shutdown is required. The one-time change was requested because approval of the Technical Specification change submitted in March 1999 was not feasible by requested June 1, 1999.

The engineering analyses to support plant operation at a higher SW temperature for an unlimited period of time have now been completed. The analyses, in concert with the evaluations and calculations performed since the summer of 1998, support continuous plant operation at a SW temperature of 97°F. The proposed increase in SW temperature supplements the Technical Specification changes submitted in March 1999 and April 1999 in that the 8 hours would commence when the UHS temperature exceeds 97°F instead of 95°F.

Description of the Proposed Changes

The following is a list of proposed Technical Specification changes that are required for the increase in SW temperature from 95°F to 97°F. In addition to the Technical Specification that directly addresses the SW temperature, changes are proposed to other Technical Specifications to reflect the assumptions used in the supporting analyses and results from the supporting analyses. In Technical Specification LCO 3.7.8, "Ultimate Heat Sink (UHS)," the proposed change increases the maximum allowed SW temperature in surveillance requirement Surveillance Requirement (SR) 3.7.8.2 from 95°F to 97°F.

In Technical Specification LCO 3.3.2, "Engineered Safety Features Actuation System Instrumentation," the proposed change decreases the ALLOWABLE VALUE and NOMINAL TRIP SETPOINT for FUNCTIONS 2.c, 3.b(3), and 4.c, "Containment Pressure High High" from 20.45 psig and 20 psig, respectively, to 10.45 psig and 10 psig, respectively. These new values correspond with new analytical values used in the containment analysis supporting the SW temperature change.

In Technical Specification LCO 3.6.8, "Isolation Valve Seal Water (IVSW) System," the proposed change increases the minimum allowed IVSW tank pressure and IVSW dedicated nitrogen bottle pressure in surveillance requirements SR 3.6.8.1 and SR 3.6.8.5 from 44 psig to 44.6 psig. This new value is greater than the Technical Specification required value of 1.1 times the peak containment internal pressure (P_a) of 40.5 psig calculated for a large break LOCA inside containment by the new analysis supporting the SW temperature change.

In Technical Specification LCO 3.7.3, "Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulation Valves (MFRVs), and Bypass Valves," the proposed change decreases the maximum allowed MFIV closure time in surveillance requirement SR 3.7.3.2 from 80 seconds to

50 seconds. The new value is used in the MSLB inside containment analysis supporting the SW temperature change.

In Technical Specification Section 5.5.16, "Containment Leakage Rate Testing Program," the proposed change increases the peak containment internal containment pressure for the design basis LOCA, P_a, from 40 psig to 40.5 psig. This new value is calculated by the large break LOCA inside containment analysis supporting the SW temperature change.

In addition to the Technical Specifications, the analyses of record for the containment response to a MSLB inside containment and a large break LOCA inside containment, which are described in the Updated Final Safety Analysis Report (UFSAR), Section 6.2, "Containment Systems," are changed as a result of the proposed SW temperature change.

The Technical Specification Bases have been changed to reflect the above listed Technical Specification changes and the assumptions and results of the new analyses of record.

Safety Assessment

Evaluation of Components Served by SW

SW interfaces with the components listed in Table 1, "Components Interfacing with Service Water System." SW may be used to cool the component itself, e.g., cool a bearing, or may be used by the component to cool another medium, e.g., CCW heat exchangers. The components that use SW for a safety-related function are identified.

The components listed in Tables 1 were evaluated using a UHS temperature of at least 97°F to confirm that the SW cooled the component or other medium as required during normal operation, transients, and accidents. For components that use SW for a safety-related function, the evaluations ensure that results of the safety analyses are bounding and are provided below.

Emergency Diesel Generators (EDG)

SW is used by the *Emergency Diesel Generators (EDG)* heat exchangers to cool scavenge air, lube oil, and jacket water of the EDGs. The SW system, which cools these three heat exchangers in series, enables the EDGs to perform their intended function if the shell side outlet temperature of the scavenge air heat exchanger is less than or equal to the design temperature of 120°F, lube oil is less than or equal to 215°F, and jacket water is less than 195°F. Assuming the 100% rated electrical capacity of 2500 kW, SW temperature of 100°F, design SW flow rate of 505 gpm, 5 percent of each heat exchanger's tubes are plugged, and a design fouling factor, the scavenge air, lube oil, and jacket water heat exchangers will be within their required design temperature limits.

If the rated overload capacity of 2750 kW (110% of 2500 kW) is assumed, the lube oil and jacket water temperatures are below the stated temperatures, but the shell side outlet temperature of the scavenge air heat exchanger exceed the design temperature by about 3°F. By reducing SW temperature to 97°F and allowing only 2 percent tube plugging, the shell side outlet temperature of the scavenge air heat exchanger is acceptable. These heat exchangers will be limited to 2% tube plugging. Therefore, the EDGs are capable of performing their intended function with a SW temperature of 97°F.

Reactor Containment Air Recirculation Fan Coolers (RCFC)

SW is used by the *Reactor Containment Air Recirculation Fan Coolers (RCFCs)* to cool containment air temperature during normal operation, and remove heat from containment air following a MSLB or large break LOCA inside containment to mitigate the temperature and pressure excursion to maintain containment structural integrity and the environmental qualification of equipment required to mitigate the effects of the MSLB or large break LOCA. The SW system enables the RCFCs to perform their intended function during normal operation if containment air temperature is maintained ≤120°F, and during a MSLB or large break LOCA inside containment if containment pressure is maintained less than or equal to its 42 psig design limit, and containment pressure and temperature are maintained within equipment environmental qualification limits.

Operational data has shown that when the SW temperature is close to the 95°F upper limit during normal operation. The containment average air temperature can be as high as 23°F above the SW temperature. As a result, the containment average air temperature is not expected to exceed 120°F when SW temperature is 97°F. Containment average air temperature is separately controlled by Technical Specification LCO 3.6.5, "Containment Air Temperature," and appropriate REQUIRED ACTIONS are implemented if the CONDITIONS in Technical Specification LCO 3.6.5 are not satisfied, e.g., containment average air temperature exceeds 120°F. The results of the containment analyses for a MSLB inside containment and large break LOCA inside containment, which are provided later in the Safety Assessment, show that the SW system enables the RCFC to maintain containment pressure below its 42 psig design limit and containment pressure and temperature within environmental qualification limits for equipment required to mitigate a MSLB or large break LOCA.

In addition to reanalyzing the RCFC heat removal for the MSLB inside and large break LOCA inside containment, the RCFC were re-evaluated with respect to waterhammer and two-phase flow, as discussed in NRC Generic Letter 96-06, "Assurance Of Equipment Operability And Containment Integrity During Design-Basis Accident Conditions," at a SW temperature of 100°F. The waterhammer magnitudes were below waterhammer pulses measured during plant testing, and the RCFC can remove the design basis heat load with two-phase flow conditions for fouled and clean conditions. Therefore, the RCFC are capable of performing their intended function with a SW temperature of at least 97°F.

Containment Fan Motor Coolers

SW is used by the *Containment Fan Motor Coolers* to remove the motor heat load. The SW system enables the motor coolers to perform their intended function if the motor windings do not exceed the "hot spot" temperature limit. Westinghouse determined that the motor windings will not exceed the hot spot temperature limit of 110°C if the motor air inlet temperature is limited to 122.31°F. A heat balance equation shows that the motor air inlet temperature is less than 122.31°F with a SW flow rate of 37 gpm, 15 percent of the tubes plugged, and a SW temperature of 100°F. The SW design flow for these coolers is 50 gpm and tube plugging will be limited to 15%. Therefore, these coolers are capable of performing their intended function with a SW temperature of at least 97°F.

Safety Injection (SI) Pumps

SW is used by the *Safety Injection (SI) pumps* to cool the thrust bearing lube oil, via a cavity in the bearing housing, of the SI pumps. The SW system enables the SI pumps to perform their intended function if the bearing oil temperature is maintained within manufacturer recommendations. The manufacturer of the pumps confirmed that a SW temperature of 100°F will not cause detrimental effect on the pump thrust bearing. Therefore, the SI pumps are capable of performing their intended function with a SW temperature of at least 97°F.

Steam Driven Auxiliary Feedwater (SDAFW) Pump

SW is used by the *Steam Driven Auxiliary Feedwater (SDAFW) Pump*, as a backup to the Condensate Storage Tank, to cool the bearing lube oil of the SDAFW pump. The SW system enables the SDAFW to perform its intended function if the bearing lube oil temperature is less than 155°F. Assuming a design heat load of 23,400 BTU/hr, design SW flow rate of 11.3 gpm, design fouling factor, and a SW temperature of 100°F, the lube oil temperature is below the alarm setpoint. Also, The SDAFW pump is capable of pumping fluid at 100°F. Therefore, the SDAFW pump is capable of performing their intended function with a SW temperature of at least 97°F.

Motor Driven Auxiliary Feedwater (MDAFW) Pumps

SW is used by the *Motor Driven Auxiliary Feedwater (MDAFW) Pumps* to cool the bearing lube oil and the shaft packing of the MDAFW pumps. The SW system enables the MDAFW to perform their intended function if the bearing oil temperature and packing are maintained within manufacturer recommendations. The manufacturer of the pumps has confirmed that a SW temperature of 100°F will not adversely affect MDAFW pump operation. Therefore, the

MDAFW pumps are capable of performing their intended function with a SW temperature of at least 97°F.

Reactor Auxiliary Building Ventilation System

SW is used by the *Reactor Auxiliary Building Ventilation System* to cool the air in the SI/Containment Spray (CS) pump room, Residual Heat Removal (RHR) pump room, and MDAFW pump room. The SW system enables the system to perform its intended function if the air temperature in each room is maintained less than the maximum air temperature allowed for operation of the equipment located in the rooms. The maximum air temperature of the SI/CS pump room, RHR pump room, and MDAFW pump room has been calculated during accident conditions. Assuming an outside air temperature of 95°F, surrounding rooms at 104°F (unless conditions warrant higher temperatures, e.g., EDG room assumed to be 125°F), SW at 100°F, and one cooler operating in each room, the maximum room temperatures are approximately 125°F, 115°F, and 120°F for the SI/CS, RHR, and MDAFW pump rooms, respectively. These temperatures are below the temperatures at which the equipment in these rooms have been evaluated to be operable. Therefore, the equipment in these rooms are capable of performing their intended function with a SW temperature of at least 97°F.

Control Room Water Cooled Condensing Units (WCCUs)

SW is used by the *Control Room Water Cooled Condensing Units (WCCUs)* to cool the condensers of refrigerant units and maintain Control Room habitability. The SW system enables the WCCUs to perform their intended function if the amount of heat generated by the Control Room can be removed. Each of the WCCUs is considerably oversized for the Control Room heat load, and the heat transfer rate in the condenser is not appreciably affected by a small increase in SW temperature because the high pressure refrigerant on shell side of the condenser is at a considerably higher temperature than the SW. Therefore, the WCCUs are capable of performing their intended function with a SW temperature of at least 97°F.

Steam Generators (SGs)

SW is used by the *Steam Generators (SGs)* via a cross-connect to the AFW system to cool the Reactor Coolant System (RCS) for a plant shutdown during a Station Blackout (SBO). The SW system performs its design function if an adequate volume of water is available for plant decay heat removal during the SBO coping period. With an increased SW temperature, the amount of SW required to obtain the same amount of decay heat removal during the coping period increases, but the SW system supply capacity is considerably larger than the required amount. Additionally, the capability of the AFW system to cool the RCS for a plant shutdown following a fire, a MSLB, and a Loss of Offsite Power has been evaluated and found to be acceptable.

Therefore, the SG are capable of performing their intended function with a SW temperature of 97°F.

Component Cooling Water (CCW)

SW is used by the *Component Cooling Water (CCW) system heat exchangers* to reject heat from the CCW system. During normal power operation, the SW system enables the CCW heat exchangers to perform their intended function if the CCW heat exchanger outlet is maintained ≤105°F. During plant shutdown, the SW system performs its design function if the CCW heat exchanger can cool the RCS, using the RHR system/CCW heat exchanger, from 350°F to 200°F within 30 hours with the CCW heat exchanger outlet ≤125°F . With a SW temperature of 99°F, heat loads associated with normal power operation, and design heat exchanger fouling, the outlet temperature of the CCW heat exchanger has been calculated as ≤105°F with up to 20% tube plugging. With a SW temperature of 100°F, heat loads 3 hours after reactor shutdown, design heat exchanger fouling, and 5% tube plugging, RCS cooldown from 350°F to 200°F can be accomplished within 30 hours. Also, the CCW heat exchangers themselves are designed to 200°F, which is higher than the proposed operating temperature. The CCW heat exchangers will be limited to 5% tube plugging. Therefore, the CCW heat exchangers are capable of performing their intended function with a SW temperature of at least 97°F.

In conclusion, calculations and evaluations have shown that the plant components cooled by SW will perform their safety related and non-safety related function(s) with a SW temperature of 97°F.

Ultimate Heat Sink (UHS)

The effect of raising the maximum allowed SW temperature to as high as 100°F has been calculated with respect to the impact on the UHS capacity. The calculation determined that 22.1 days could elapse following an accident before a vortex would form at the suction of the SW pumps. The assumed lake temperature for the purpose of the calculation is 110°F.

Minor changes to the Bases to Technical Specification 3.7.8, "Ultimate Heat Sink," have been made to reflect the calculation assumptions. The need for the changes was identified while evaluating the SW increase, but are not as a result of the increase in SW temperature.

Containment Analysis for Large Break LOCA

The containment analysis performed for the LOCA, in support of a 97°F SW temperature limit, represents a change in licensing basis analysis methodology. The current containment analysis

method uses a mass and energy release model calculated by the FLASH computer code[1] and a containment response model calculated by the COCO computer code. The FLASH code uses three regions, each at different pressure, to simulate the RCS. Flow through a leak is calculated, in subcooled conditions, using Fauske's model for metastable flow for short pipes[2], or Moody's model for homogeneous equilibrium flow for long pipes[3]. Once the leaking region reaches saturation, Moody's correlation is used for both cases. Modifications were made to the original FLASH program to account for the single pass rod type reactor core, the location of the reactor coolant pump, the accumulator, and injection pumps characteristics. The mass and energy released were calculated from a spectrum of break sizes of the RCS cold leg, up to and including a double ended break. The reflood model in the FLASH code includes flow from the accumulators through the core with a conservative accounting of bypass of some accumulator flow through the RCS loops.

The new containment analysis for the LOCA uses the Westinghouse mass and energy release model[4] which accommodates the use of computer codes which were developed to comply with the ECCS acceptance criteria presented in 10 CFR 50.46 and 10 CFR 50, Appendix K. The system noding scheme used for mass and energy release is a 68 node model of the RCS. The mass and energy release model calculates results for a double ended guillotine break of the reactor coolant pump suction. The reflood phase of the transient is modeled with the WREFLOOD code[5]. In order to be consistent with 10 CFR 50, Appendix K, the WREFLOOD model assumes accumulator water bypasses the core and goes out through the break until the termination of bypass. The new mass and energy calculations model the current plant configuration for SI pumps of a single SI pump on each of the two emergency power buses.

The COCO model in the current containment response analysis has similarities to the COCO model used in the new containment response analysis[6]. For example, Tagami[7] is used to develop the modeling of the heat transfer coefficient to the containment surface. However, the current model uses an iterative method for modeling the containment fan cooler heat removal rate while the new containment analysis provides an explicit mathematical solution.

The containment analysis for a large break LOCA was performed in two separate steps. First, the mass and energy release rates from postulated breaks were calculated in accordance with

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[1] "Flash; a Program for Digital Simulation of the Loss of Coolant Accident," S.F. Margolis, and J. A. Redfield, Bettis Atomic Power Laboratory, Report WAPD-TM-534.
[2] "The Discharge of Saturated Water Through Tubes," by H. K. Fauske, AICHE, Reprint 30, Seventh National Heat Transfer Conference, AICHE and ASMR, Cleveland, Ohio, August 9 to 12, 1964.
[3] "Maximum Flow Rate of Single Component, Two-Phase Mixture," by F. H. Moody, Paper No. 64-HT-35, and ASME publication.
[4] Shepard, R. M., Massie, H. W., Mark, R. H., and Docherty, P. J., "Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A, June 1975.
[5] Kelly, R. D., et. Al., "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary), WCAP-8171 Non-Proprietary), June 1974.
[6] F. M. Bordelon, E. T. Murphy, "Containment Pressure Analysis Code (COCO) WCAP-8327, July 1974.
[7] Tagami, Takasi, "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period ending June, 1965, No. 1.

WCAP-10325-P-A[8], which is comprised of mass and energy release versions of the SATAN VI[9], WREFLOOD[10], FROTH, and EPITOME codes. Then, the resultant release rates were used to calculate the containment response, i.e., pressure and temperature profiles, in accordance with WCAP-8327[11]. The pressure and temperature profiles were used to confirm that the 1) containment pressure is maintained less than the 42 psig containment design limit; and 2) environmental conditions for equipment required to operate during the large break LOCA are bounding. The text that follows describes selected cases analyzed, assumptions, methodologies, and analyses separately for the mass and energy release calculations and the containment response.

Mass and energy release cases were analyzed for a double-ended hot leg (DEHL) guillotine break, and a double-ended pump suction (DEPS) break.

The DEHL guillotine break has been shown in previous Westinghouse studies to result in the highest blowdown mass and energy release rates. Although the core flooding rate would be the highest for this break location, the amount of energy released from the Steam Generator (SG) secondary is minimal because the majority of the fluid which exits the core bypasses the SGs venting directly to containment. As a result, the reflood mass and energy releases are reduced significantly as compared to either the DEPS break or cold leg break locations where the core exit mixture must pass through the SGs before venting through the break. For the DEHL guillotine break, generic studies have confirmed that there is no reflood peak (i.e., from the end of the blowdown period the containment pressure continually decreases). Therefore only the mass and energy releases for the DEHL guillotine break blowdown phase are calculated.

The cold leg break location has been found in previous studies to be much less limiting in terms of the overall containment energy releases. The cold leg blowdown is faster than that of the DEPS break, and more mass is released into the containment. However, the core heat transfer is greatly reduced (due to the break location the flow will bypass the normal path through the core and go through the path of least resistance to the broken loop) and this results in a considerably lower energy release into containment. Studies have determined that the blowdown transient for the cold leg is less limiting than that for the DEPS break and DEHL guillotine breaks. During reflood, the flooding rate is greatly reduced because the core vent paths include the resistance of the reactor coolant pump, in addition to Emergency Core Cooling System (ECCS) injection spill, thus the energy release rate into the containment is reduced. Therefore, the cold leg break was not included in the analysis.

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[8] "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," WCAP-10325-P-A, May 1983 (Proprietary), WCAP-10326-A (Non-Proprietary).
[9] "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant," WCAP-8303 (Proprietary), WCAP-8306 (Non-Proprietary), June 1974.
[10] "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)," WCAP-8170 (Proprietary), WCAP-8171 (Non-Proprietary), June 1974.
[11] "Containment Pressure Analysis Code (COCO)," WCAP-8327, July 1974 (Proprietary), WCAP-8326, July, 1974 (Non-Proprietary).

The DEPS break combines the effects of the relatively high core flooding rate, as in the DEHL guillotine break, with the addition of the stored energy in the SGs. As a result, the DEPS break yields the highest energy flow rates during the post-blowdown period by including the available energy of the RCS and secondary side in calculating the releases to containment.

Loss of offsite power is assumed for the mass and energy release calculation, which requires the EDGs to start to power the ECCS. This causes a delay in the start of ECCS which results in a higher containment pressure and temperature. Maximum containment backpressure equal to the design pressure is assumed, which reduces ECCS flow thereby reducing the condensation of steam and extending the time to the reflood phase. In addition, the single failure of an EDG and single failure of a CS pump are analyzed separately for a DEPS break. Single failure of an EDG corresponds with the minimum ECCS flow case and results in operation of one CS pump, two RCFCs, one SI pump, and one RHR pump, which reduces steam condensation and heat removal from containment. Single failure of a CS pump corresponds with the maximum ECCS flow case, i.e., two SI pumps and two RHR pumps, while reducing steam condensation and heat removal from containment by CS. Only the minimum ECCS flow case was analyzed for a DEHL guillotine break because minimum SI yields slightly higher mass and energy release rates and the CS pump(s) and RCFCs are not a factor as a result of their associated start time delays.

Selected system parameters used as inputs for the large break LOCA mass and energy release calculation are listed in Table 2, "Initial Conditions for Large Break LOCA Mass and Energy Release Calculation." Initial reactor power level is maximized to maximize the mass and energy release with respect to reactor coolant system (RCS) temperature, available decay heat energy, and initial core stored energy. Initial RCS temperatures reflect the highest average coolant temperature to maximize the initial fluid energy. RCS pressure is maximized to maximize initial RCS blowdown rate and to maximize the fluid density for more mass release. SG tube plugging is 0% to maximize the reactor coolant volume, maximize the heat transfer from the secondary to the primary after the blowdown phase, and reduce the RCS loop resistance, which reduces the pressure drop upstream of the break for the pump suction breaks and increases break flow. Secondary to primary heat transfer is maximized by assuming conservative coefficients of heat transfer (i.e., SG primary/secondary heat transfer, and RCS metal heat transfer). Initial SG mass is maximized to maximize the heat transfer from the secondary to the primary after the blowdown phase. Stored energy in the fuel is determined based on rod geometry, rod power, and limiting time in life (e.g., burnup). RCS fluid volume is maximized to maximize the amount of fluid available for blowdown. Core decay heat was calculated in accordance with ANSI/ANS-5.1-1979[12], with the consideration of the industry experience provided by NRC Information Notice 96-39[13].

The sources of mass and energy considered in the large break LOCA mass and energy release analysis are RCS water, SI accumulator water and nitrogen gas, ECCS pumped injection water, decay heat, core stored energy, RCS metal (including SG tubes), SG metal (including transition

[12] "American National Standard for Decay Heat Power in Light Water Reactors," August 1979.
[13] "Estimates of Decay Heat Using ANS 5.1 Decay Heat Standard May Vary Significantly," July 5, 1996.

cone, shell, wrapper, and other internals located below the top of the SG tubes), SG secondary energy (including fluid mass and steam mass), and secondary transfer of energy (feedwater into and steam out of the SG secondary).

The large break LOCA mass and energy analysis is divided into four phases: blowdown, refill, reflood, and post-reflood. Break mass and energy releases have been calculated for the blowdown, reflood, and post-reflood phases of the large break LOCA for the DEPS break case. For the DEHL guillotine break case, the releases were calculated only for the blowdown phase. The blowdown phase is the period of time from accident initiation (when the reactor is at steady state operation) to the time that the RCS and containment reach an equilibrium pressure. A SI signal is assumed to actuate at the low pressurizer pressure setpoint. Two SI accumulators are assumed to inject into the RCS cold legs when the RCS depressurizes to the accumulator pressure. The blowdown period ends when the RCS active core area is essentially empty. Blowdown typically lasts <30 seconds. Delivery of ECCS flow for the minimum ECCS case is assumed to occur after a delay time that includes generation of the SI signal, valve opening, and pump acceleration. For the maximum ECCS case, ECCS flow is assumed with a 16.4 second delay after SI signal generation. A mass and energy release version of the SATAN-VI code is used for computing the blowdown transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. A point kinetics model is used with weighted feedback effects. The major feedback effects include moderator density, moderator temperature, and Doppler broadening. A critical flow calculation for subcooled (modified Zaloudek), two-phase (Moody), or superheated break flow is incorporated into the analysis. Heat from the zirconium-water reaction is not included because Peak Cladding Temperature (PCT) analyses using the models of Appendix K to 10 CFR 50 show that less than 1.0% of the total core zirconium is reacted during the hypothetical large break LOCA. Thus, the energy release from the zirconium-water reaction will be small and will not significantly affect the mass and energy releases to containment.

The refill period is the period of time when the lower plenum is being filled by accumulator and pumped ECCS water. At the end of blowdown, a large amount of water remains in the cold legs, downcomer, and lower plenum. To conservatively consider the refill period for the purpose of containment mass and energy releases, it is assumed that this water is instantaneously transferred to the lower plenum along with sufficient accumulator water to completely fill the lower plenum. This allows an uninterrupted release of mass and energy to containment. Thus, the refill period is conservatively neglected in the mass and energy release calculation.

The core reflooding phase begins when the primary coolant system has depressurized (following blowdown) due to the loss of water through the break. The water from the lower plenum, supplied by the ECCS refills the reactor vessel and provides cooling to the core. This phase ends when the core is completely quenched. The model conservatively assumes quenching of the core at the 10-foot elevation on the active fuel for containment functional design calculations. During this phase, decay heat generation will produce boiling in the core resulting in a two-phase mixture of steam and water in the core. This two-phase mixture rises above the core and

subsequently enters the SGs. The most-important feature is the steam/water mixing model (described below), which is used during this phase. The WREFLOOD code is used for computing the reflood transient. The WREFLOOD code consists of two basic hydraulic models - one for the contents of the reactor vessel, and one for the coolant loops. The two models are coupled through the interchange of the boundary conditions applied at the vessel outlet nozzles and at the top of the downcomer. Additional transient phenomena such as pumped ECCS and accumulators, reactor coolant pump performance, and SG release, are included as auxiliary equations which interact with the basic models as required. The WREFLOOD code permits the capability to calculate variations during the core reflooding transient of basic parameters such as core flooding rate, core and downcomer water levels, fluid thermodynamic conditions (pressure, enthalpy, density) throughout the primary system, and mass flow rates through the primary system. The code permits hydraulic modeling of the two flow paths available for discharging steam and entrained water from the core to the break; i.e., the path through the broken loop and the path through the unbroken loop. The model assumes a complete mixing condition (i.e., thermal equilibrium) for the steam and ECCS injection during the reflood phase for each loop receiving ECCS water. The complete mixing process, however, is made up of two distinct physical processes. The first is a two-phase interaction with condensation of steam by cold ECCS water. The second is a single-phase mixing of condensate and ECCS water. Since the steam release is the most-important influence to the containment pressure transient, the steam condensation part of the mixing process is the only part that need be considered. (Spillage directly heats only the sump.)

Post-reflood describes the period following the reflood transient. For the pump suction break, a two-phase mixture exits the core, passes through the hot legs, is superheated in the SGs, and exits the break as superheated steam. After the broken loop SG cools, the break flow becomes two phase. The FROTH code is used for computing the post-reflood transient. The FROTH code calculates the heat release rates resulting from a two-phase mixture level present in the SG tubes. The mass and energy releases that occur during this phase are typically superheated due to the depressurization and equilibration of the broken loop and intact loop SGs. During this phase of the transient, the RCS has equilibrated with the containment pressure, but the SGs contain a secondary inventory at an enthalpy that is much higher than the primary side. Therefore, there is a significant amount of reverse heat transfer that occurs. Steam is produced in the core due to core decay heat. During the FROTH calculation, ECCS injection is addressed for both the injection phase and the recirculation phase. SG equilibration and depressurization is the process by which secondary side energy is removed from the SGs in stages. The FROTH computer code calculates the heat removal from the secondary mass until the secondary temperature is at the saturation temperature (T_{sat}) at the containment design pressure. After the FROTH calculations, SG secondary energy is removed based on first and second stage rates. The first stage rate is applied during the time interval from the broken loop equilibrium at containment design pressure to the estimated intermediate pressure. The second stage is the time interval from the estimated intermediate pressure equilibrium out to a SG pressure of 14.7 psia at 3600 seconds. These rates are applied simultaneously in the transient until the desired depressurization is achieved for each SG, which may occur over differing periods of time and rates for each SG. The EPITOME code continues the FROTH calculation for SG cooldown. The first stage rate is applied until the SG

reaches T_{sat} at the user specified intermediate equilibration pressure, when the secondary pressure is assumed to reach the actual containment pressure. Then the second stage rate is used until the final depressurization, when the secondary reaches the reference temperature of T_{sat} at 14.7 psia, or 212°F. The heat removal of the broken loop and intact loop SGs are calculated separately. During the FROTH calculations, SG heat removal rates are calculated using the secondary side temperature, primary side temperature and a secondary side heat transfer coefficient determined using a modified McAdam's correlation. SG energy is removed during the FROTH transient until the secondary side temperature reaches saturation temperature at the containment design pressure. The constant heat removal rate used during the first heat removal stage is based on the final heat removal rate calculated by FROTH. The SG energy available to be released during the first stage interval is determined by calculating the difference in secondary energy available at the containment design pressure and that at the (lower) user specified intermediate equilibration pressure, assuming saturated conditions. This energy is then divided by the first stage energy removal rate, resulting in an intermediate equilibration time. At this time, the rate of energy release drops substantially to the second stage rate. The second stage rate is determined as the fraction of the difference in secondary energy available between the intermediate equilibration and final depressurization at 212°F, and the time difference from the time of the intermediate equilibration to the user specified time of the final depressurization at 212°F. The secondary energy remaining after the intermediate equilibration is conservatively assumed to be released by imposing a mandatory cooldown and subsequent depressurization down to atmospheric pressure at 3600 seconds, i.e., 14.7 psia and 212°F. The methodology assumes that the energies in the system are taken out to these conditions in the first hour of the event. In actuality, the release of these energies to these conditions would take much longer, on the order of hours. There is the possibility that the remaining energies, for example, down to containment conditions of 120°F could be released, however this is not included in the releases discussed herein. This additional energy would tend to slightly increase the water temperature of the spilled fluid coming from the pump side of the break, but would not increase the amount of steam being released from the SG side of the break. It is expected that the effects on the long term cooldown would be insignificant. The mass and energy release rates are calculated by FROTH and EPITOME until the time of containment depressurization. After containment depressurization (14.7 psia), the mass and energy release available to containment is generated directly from core boiloff/decay heat.

The mass and energy release rates calculated by FROTH and EPITOME, a summary of which are provided in Table 3, "Large Break LOCA Mass and Energy Releases - Minimum ECCS," are input into the COCO model to calculate the containment response, i.e., pressure and temperature profiles during the large break LOCA. COCO is a mathematical model of a generalized containment; the proper selection of various options in the code allows the creation of a specific model for particular containment design. The values used in the specific model for different aspects of the containment are derived from plant-specific input data. The plant-specific system parameters used as inputs to COCO are listed in Table 4, "Large Break LOCA Containment Response Analysis (COCO) Parameters." The COCO model uses the following assumptions:

1. Homogeneous mixing is assumed. The steam-air mixture and the water phases each have uniform properties. More specifically, thermal equilibrium between the air and the steam is assumed. However, this does not imply thermal equilibrium between the steam-air mixture and the water phase.

2. Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties.

3. For the blowdown portion of the large break LOCA analysis, the discharge flow separates into steam and water phases at the breakpoint. The saturated water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment. For the post-blowdown portion of the large break LOCA analysis, steam and water releases are input separately.

4. The saturation temperature at the partial pressure of the steam is used for heat transfer to the heat sinks and the fan coolers.

Transient phenomena within the RCS affect containment conditions by means of convective mass and energy transport through the pipe break. For analytical rigor and convenience, the containment air-steam-water mixture is separated into a water (pool) phase and a steam-air phase. Sufficient relationships to describe the transient are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate boundary conditions. As thermodynamic equations of state and conditions may vary during the transient, the equations have been derived for possible cases of superheated or saturated steam and subcooled or saturated water. Switching between states is handled automatically by the code.

The significant heat removal source during the early portion of the transient is the containment structural heat sinks. Provision is made in the containment pressure response analysis for heat transfer through, and heat storage in, both interior and exterior walls. Walls are divided into a large number of nodes. For each node, a conservation of energy equation expressed in finite-difference form accounts for heat conduction into and out of the node and temperature rise of the node. Table 5, "Containment Heat Sinks," is the summary of the containment structural heat sinks used in the analysis. The thermal properties of each heat sink material are shown in Table 6, "Thermophysical Properties of Containment Heat Sinks."

The heat transfer coefficient to the containment structure for the early part of the event is calculated based primarily on the work of Tagami. From this work, it was determined that the value of the heat transfer coefficient can be assumed to increase parabolically to a peak value. In COCO, the value then decreases exponentially to a stagnant heat transfer coefficient which is a function of steam-to-air-weight ratio.

For a large break, the engineered safety features are quickly brought into operation. Because of the brief period of time required to depressurize the RCS, the CS does not have a major influence

on the blowdown peak pressure; however, CS does reduce the containment pressure after the blowdown and maintain a low long-term pressure and a low long-term temperature.

During the injection phase of post-accident operation, the ECCS pumps water from the Refueling Water Storage Tank (RWST) into the reactor vessel. Since this water enters the vessel at RWST temperature, which is less than the temperature of the water in the vessel, it is modeled as absorbing heat from the core until the saturation temperature is reached. SI and CS can be operated for a limited time, depending on the RWST capacity.

After the RWST is exhausted, RHR is operated to provide long term cooling of the core. In this operation, water is drawn from the containment sump, cooled in a RHR heat exchanger, then pumped back into the reactor vessel to remove core residual heat and energy stored in the vessel metal. The heat is removed from the RHR heat exchanger by CCW. The RHR heat exchangers and CCW heat exchangers are coupled in a closed loop system, where the heat sink for the CCW heat exchangers is the SW system.

CS is an active removal mechanism which is used for pressure reduction. During the injection phase of operation, the CS pumps draw water from the RWST and spray it into the containment through nozzles mounted high above the operating deck. As the spray droplets fall, they absorb heat from the containment atmosphere. Since the water comes from the RWST, the entire heat capacity of the spray from the RWST temperature to the temperature of the containment atmosphere is available for energy absorption. CS must be stopped to switch the CS pump suction from the RWST to the containment sump 63, 66.7, or 77 minutes from the beginning of the event based on the assumed single failure (see Table 4). The analysis assumes that CS is restarted at 100 minutes to continue containment heat removal and pressure reduction and is stopped again at 11 hours for the shift from cold leg recirculation to hot leg recirculation. For the DEHL guillotine break case, CS is not modeled because the blowdown phase terminates prior to CS initiation.

When a spray droplet enters the hot, saturated, steam-air containment environment, the vapor pressure of the water at its surface is much less than the partial pressure of the steam in the atmosphere. Hence, there will be diffusion of steam to the drop surface and condensation on the droplet. This mass flow will carry energy to the droplet. Simultaneously, the temperature difference between the atmosphere and the droplet will cause the droplet temperature and vapor pressure to rise. The vapor pressure of the droplet will eventually become equal to the partial pressure of the steam, and the condensation will cease. The temperature of the droplet will essentially equal the temperature of the steam-air mixture.

Analysis shows that the temperature of the (mass) mean droplet produced by the spray nozzles rises to a value within 99 percent of the bulk containment temperature in less than 2 seconds. Detailed calculations of the heatup of spray droplets in post-accident containment atmospheres by Parsly[14], show that droplets encountered in the containment spray reach equilibrium in a

[14] "Design Consideration of Reactor Containment Spray System Part VI, The Heating of Spray Drops in Air-Steam

fraction of their residence time in a typical pressurized water reactor containment. These results confirm the assumption that the containment spray will be 100 percent effective in removing heat from the atmosphere.

The RCFCs are another means of heat removal. Each RCFC has a fan which draws in the containment atmosphere from the upper volume of the containment. Since the RCFCs do not use water from the RWST, the mode of operation remains the same both before and after the ECCS is changed to the recirculation mode. The steam/air mixture is routed through the enclosed RCFC unit, past SW cooling coils. The fan then discharges the air through ducting containing a check damper. The discharged air is directed at the lower containment volume. The RCFCs heat removal capability assumed in the analysis for different containment air temperature is listed in Table 4.

The containment pressure and temperature results from each of the large break LOCA cases are shown in Figures 1 through 4 and Table 7, "LOCA Containment Response (COCO) Results." All cases resulted in a peak containment pressure that was less than the containment design pressure of 42 psig. The new large break LOCA containment analysis response result of 40.5 psig represents a change in the margin of safety for the containment as defined in the NRC Safety Evaluation (SE) of Amendment 79 to the Technical Specifications dated April 4, 1984. The margin of safety was defined by the SE as 2 psig which was the difference between the containment design pressure of 42 psig and the previous analysis result of 40 psig. The Technical Specifications allowable upper pressure limit for normal operation was chosen in Amendment No. 79 to be ½ of the margin of safety, defined as the difference between 42 psig and 40 psig, or +1.0 psig. The new analysis result represents a decrease in that margin of 0.5 psig. However, since the new analysis value assumes an initial containment pressure of +1.0 psig, which is the current maximum allowed Technical Specification value, and the previous analysis assumed an initial containment pressure of 0 psig, no change to the allowed containment pressure during normal operation is proposed.

The environmental qualification of equipment required to operate during a large break LOCA was evaluated using the composite (LOCA / MSLB) pressure and temperature profiles in Figures 5 and 6. The new accident temperature profile is generally bounded by the profile used for qualification testing of components, but since the accident profile exceeds the qualification profile in some cases, an accident degradation equivalency via the Arrhenius calculation was used to demonstrate that the components are qualified to the new accident temperature profile. Note that the component temperature profile shown on Figure 6 was a separate analysis run with initial containment pressure at 13.7 psia to maximize the component temperature for environmental qualification. The new accident pressure profile is also generally bounded by the pressure qualification profile. Equipment qualification data packages were evaluated for cases that were not enveloped by the qualification profile to demonstrate that the equipment are qualified to the new accident pressure profile.

Atmospheres," ORNL-TM-2412 Part VI, January 1970.

The available and required net positive suction head for ECCS pumps have been calculated using bounding values and are not affected by the new pressure and temperature profiles.

Normally, reanalysis of the large break LOCA mass and energy releases inside containment would be accompanied by a subcompartment analysis to ensure that the walls of subcompartments can maintain their structural integrity during the short pressure pulse that accompanies a high energy pipe rupture within the subcompartment. The licensing basis analysis for HBRSEP, Unit No. 2 is based on double ended ruptures of the main RCS piping. The effects of large breaks on subcompartments have not been considered because HBRSEP, Unit No. 2 has implemented Leak Before Break (LBB) in accordance with Generic Letter 84-04[15].

The effects of the revised containment response on the large break LOCA PCT has been evaluated. The evaluation included the effects of the new equipment qualification temperature envelope on setpoint uncertainties credited in the PCT analysis the effects of the new containment net free volume and new heat sinks calculations developed for the containment analyses. This evaluation found that the cumulative effects of these changes on large break LOCA PCT is -2°F.

Containment Analysis for MSLB

The containment analysis performed for the MSLB, in support of a 97°F SW temperature limit, represents a change in licensing basis analysis methodology. The original licensing basis of the plant did not include an explicit analysis of containment response to an MSLB. By letter dated January 5, 1983, the NRC approved a main steam line break analysis with continued feedwater addition that was submitted to the NRC in response to NRC Bulletin 80-04, "Analysis of a Pressurized Water Reactor Main Steam Line Break with Continued Feedwater Addition." The analysis was updated in support of dilution of the Boric Acid Injection tank allowed in Amendment No. 97 to the operating license, dated March 7, 1986. The analysis has since been updated and included in UFSAR Section 6.2.1.4, "Containment Analysis for Postulated Secondary System Pipe Ruptures."

The UFSAR analysis for mass and energy release during an MSLB uses a model[16] developed with the RELAP 5 code. The RELAP 5 code is primarily used to simulate the core power, core boundary conditions, other primary system conditions and secondary conditions to determine the impact on the reactor fuel during an MSLB. In order to model the cooldown in the affected core region, RELAP 5 is used to model the break flow from the main steam line. The break flows calculated by RELAP 5 were analyzed using the CONTEMPT-LT/028 computer code[17]. The CONTEMPT-LT/028 model uses a single control volume to simulate a dry containment. The

[15] " Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops," February 1, 1984.
[16] "Steamline Break Methodology for PWR's," XN-NF-84-93(P), Exxon Nuclear Company, Inc., November 1984
[17] D. W. Hargroves and L. J. Metcalfe, CONTEMPT-LT/028 - A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," NUREG/CR-0255, March 1979.

volume has a vapor and a liquid region, each of which are assumed homogeneous and in equilibrium. However, the regions are not necessarily in equilibrium with each other. The mass and energy added to the containment included three separate parts. First, the contents of unisolable portions of the steamlines and header were conservatively discharged to containment as a "step" input at the beginning of the accident. Second the dry steam break flows from the RELAP 5 calculation of the steam generator blowdown were added to the containment. Third, beyond 255 seconds into the event, the mass and energy release into containment was based upon extrapolation.

The new containment response analysis for the main steam line break uses the LOFTRAN Code[18] for calculating the mass and energy releases to containment. The LOFTRAN code is a multiloop version. The secondary side is effectively a one node, two region model of saturated steam and water. Heat transfer is assumed to occur to saturated water. If tube uncovery is predicted, the total heat transfer coefficient is accordingly reduced. The codes contain a detailed steam generator model which is used to predict tube uncovery. This model calculates the liquid volume in the steam generator shell and accounts for the detailed steam generator geometry. LOFTRAN uses a point kinetics model to describe the core nuclear power transient initiated by the cooldown from the main steam line break.

The mass and energy releases were used in COCO to calculate the containment response. The COCO model has been described previously. The COCO model is similar to the CONTEMPT model by having a single control volume with a vapor and liquid region. COCO models heat removal from condensing steam, from the coolers, and from sprays.

The containment analysis for a MSLB was performed in two separate steps similar to the containment analysis for a large break LOCA. First, the mass and energy releases from postulated breaks were calculated using the LOFTRAN computer code. Then, the resultant releases were used to calculate the containment response, i.e., temperature and pressure profiles, in accordance with COCO[19]. The pressure and temperature profiles were used to confirm that the 1) containment pressure is maintained less than the 42 psig containment design limit and 2) environmental conditions for equipment testing for equipment required to operate during the MSLB are bounding. The following describes the cases analyzed, assumptions, and methodologies for the mass and energy releases. The description of the COCO model provided for the large break LOCA containment analysis is also applicable to the MSLB containment analysis and has not been repeated.

Two power levels have been evaluated for the MSLB: 0% and 102%. One break area has been analyzed – full double-ended rupture (DER) downstream of the flow restrictor in one steamline.

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[18] "Mass and Energy Releases Following a Steam Line Rupture, Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," WCAP-8822-S1-P-A, September 1986, and "Mass and Energy Releases Following a Steam Line Rupture, Supplement 2 - Impact of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture For Dry and Subatmospheric Containments," WCAP-8822-S2-P-A, September 1986.

[19] "Containment Pressure Analysis Code (COCO)," WCAP-8327, July 1974.

Note that a DER is defined as a rupture in which the steam pipe is completely severed and the ends of the break displace from each other. The break area in the forward-flow direction is 1.4 ft^2 which is the effective blowdown area equivalent to the flow restrictor at the SG outlet nozzle. The reverse flow break area is 1.479 ft^2, which is the cross-sectional area of the flow-restricting venturi.

MSLBs can be postulated to occur with the plant in operating condition ranging from hot shutdown to full power. Since SG mass decreases with increasing power level, breaks occurring at-lower power level will generally result in a greater total mass release to the containment. However, because of increased stored energy in the primary side of the plant, increased heat transfer in the SGs, and additional energy generation in the fuel, the energy release to the containment from breaks postulated to occur during "at-power" operation may be greater than for breaks postulated to occur with the plant in a hot-shutdown condition. Additionally, steam pressure and the dynamic conditions in the SGs change with increasing power and have a significant influence on the rate of blowdown. Because of the opposing effects (mass versus energy release) of changing power level on MSLB releases, the two extremes, 102% and 0%, have been analyzed.

The plant initial conditions are assumed to be at conservatively biased values corresponding to the initial power. Table 8, "Plant Parameters and Initial Condition Assumptions - MSLB," identifies selected plant parameters corresponding to limiting MSLB cases analyzed.

The following five cases were analyzed to determine the most limiting combination of single failure and initial reactor power level with respect to containment pressure and temperature:

- Hot zero power (HZP) with single failure of one E-Bus.
- HZP with single failure of one Steam Line Check Valve (SLCV).
- 102% power with single failure of one E-Bus.
- 102% power with single failure of one SLCV.
- 102% power with single failure of one feedwater regulating valve.

The loss of one E-bus results in the loss of one SI pump and the loss of one train of containment cooling (one CS pump and two RCFC). At HZP, the MFRVs are closed, therefore, the single failure of a MFRV was only considered at 102% power. The failure of a MSIV is bounded by the loss of SLCV. Therefore, the loss of one MSIV was not analyzed. With a single failure of the SLCV in the affected steamline, MSIV closure in the intact steamlines is required to terminate the blowdown. A delay time of 4 seconds was assumed (2-second signal processing plus 2-seconds for MSIV closure) with full steam flow assumed through the valve during the valve stroke. An additional case at HZP was analyzed to determine the effect of AFW runout. The single failure for this case is assumed to be the SDAFW pump flow control valve.

LOFTRAN considers two important sources of latent heat energy: 1) the reactor vessel and primary system piping thick metal and 2) the fluid inventory in the secondary side of the SG which have been isolated from the broken steam line. The energy transferred from the reactor

vessel and primary piping is modeled as a lumped system where the important variables are metal mass, heat capacity, and the heat transfer coefficient. Addition of fluid to the primary by the Safety Injection system is also modeled. The reverse heat transfer from the secondary side fluid to the primary coolant, which is cooled by the steam release from the faulted steamline and associated SG, is defined by the McAdams correlation.

The ability of the SG feeding the broken steam line to transfer heat from the primary to the secondary affects the mass and energy release. The film coefficient on the outside of the SG tubes during a blowdown will generally be due to some form of stable boiling forced by the continual depressurization and the addition of AFW. This situation exists for long periods after the break occurs and keeps the secondary side heat transfer coefficient large. If the RCPs are assumed to continue operating, the primary side coefficient is large due to forced convection. Therefore, offsite power is assumed available in order for the RCPs to remain running to transfer heat from core decay, thick metal, and from the SGs on the intact steam lines. The heat transfer coefficient is reduced when sufficient mass has been lost from the SG to lower the water level below the tube bundle. The model used to predict tube uncovery calculates a liquid volume which includes a correlation that predicts voiding in the tube region. The SG is modeled as two nodes on the secondary side, saturated steam and water, and multiple nodes on the primary side. Variable noding is used to calculate heat transfer coefficient for the uncovered tube area. Superheating of the steam was not considered because it has little impact on the peak temperature and pressure of large dry containments.

Feedwater addition is considered in the model to maximize the water inventory available for release. Main feedwater flow was conservatively modeled by assuming an increase in feedwater flow in response to increases in steam flow following initiation of the MSLB. This maximizes the total mass addition prior to feedwater isolation. During actual plant operation, MFRVs are not in service at power levels up to approximately 10-15% of full power; rather the Main Feedwater Bypass Valves are used to provide flow as necessary to the SGs at zero power. For the analysis, the bypass valves are assumed to be closed at HZP and 100% power.

For a MSLB at power, the analysis assumes that the MFRVs open in 0.2 seconds and allow 116% of nominal feedwater flow to pass through. This high flowrate is assumed until a SI signal is generated, at which time, the MFRVs receive an isolation signal and begin to close. From the time of isolation signal until closure of the MFRVs 30 seconds later, feedwater flow is assumed to linearly decrease. For the case where a MFRV is assumed as the single failure, feedwater isolation is delayed an additional 20 seconds (total of 50 seconds) until the MFIV closes. After isolation, feedwater leakage through the MFIV and MFRV is assumed to be 75 gpm if the MFIV in that line is closed, and 125 gpm if the MFIV in that line is failed open.

Following feedwater isolation, as the SG pressure decreases, some of the fluid trapped in the feedwater lines between the SG nozzle and the MFIV may flash to steam if the feedwater temperature exceeds the saturation pressure. The unisolable feedwater line volume is an additional source of high energy fluid that was assumed to be discharged out of the break. The unisolable volume in the feedwater lines is 355.4 ft^3 per line (SG nozzle to the MFIV). If a

MFRV is assumed as the single failure, 1818 ft^3 (SG nozzle to the Main Feedwater Pump discharge) is assumed for the feedwater line in which the MFRV fails.

Generally, within the first minute following a MSLB, the AFW system will be initiated on one of several protection system signals. AFW flow is modeled by assuming constant flow following the initiation signal. For the analysis, a 1.5 second delay is assumed for SI signal processing. Both MDAFW pumps are assumed to start with no delay upon receipt of the SI signal. The SDAFW pump is assumed to start with no delay when the MSIVs close on a Steam Line Isolation. Addition of AFW to the SGs will increase the secondary mass available for release to containment as well as increase the heat transferred to the secondary fluid. The AFW flow control valves are set to supply a fixed flow to each SG, regardless of the backpressure in the SG.

Following the MSLB, AFW flow is assumed to flow to the faulted SG to maximize the mass release. No AFW flow is assumed to be directed to either of the 2 unaffected SGs. The maximum AFW flow to the affected SG is 1209 gpm which reflects total flow from each of the 3 AFW pumps. This value includes flow at the flow controller setpoint plus uncertainty. A single failure of a full-open flow control valve on the SDAFW pump results in a total flowrate of 1325 gpm to the affected SG. The single failure of the SDAFW pump flow control valve has been analyzed as a separate case for the MSLB mass and energy releases.

Conservative initial SG masses in the affected steamline SG were used in the analyzed cases. The use of high initial SG masses maximizes the SG inventory available for release to containment. Lower initial SG masses in the unaffected SGs were used in the analyzed cases. The use of reduced initial SG masses minimizes the availability of the heat sink afforded by the SGs in the intact steamlines.

Once steamline isolation is complete, the intact SGs become sources of energy which can be transferred to the SG with the broken line. This energy transfer occurs via the primary coolant. As the primary plant cools, the temperature of the coolant flowing in the SG tubes drops below the temperature of the secondary fluid in the intact SGs resulting in energy being returned to the primary coolant. This energy is then available to be transferred to the SG with the broken steamline.

Piping discharge resistances were not included in the calculation of the releases resulting from the steamline ruptures (Moody Curve for an $f(l/d) = 0$ was used).

Core decay heat generation assumed is based on the ANSI/ANS-5.1-1979. LOFTRAN uses a point kinetic model to describe the core nuclear power transient initiated due to the cooldown following a MSLB.

The contribution to the mass and energy releases from the secondary plant steam piping was included in the mass and energy release calculations. The flowrate was determined using the Moody correlation, the cross-sectional flow area of the MSIV and the initial steam pressure. The

unisolable steamline mass is included in the mass exiting the break from the time of steamline isolation until the unisolable mass is completely released to containment.

As the primary side of the plant cools, the temperature of the reactor coolant drops below the temperature of the RCS piping, the reactor vessel and the RCPs. As this occurs, the heat stored in the metal is available to be transferred to the SG with the broken line.

The rod control system was assumed to be in manual operation for the MSLB analyses.

The protection systems and equipment available to mitigate the effects of a MSLB accident inside containment include Reactor Trip, SI, Steam Line Isolation, MSIVs, MFRVs, RCFCs, and CS. For the cases with an assumed SLCV failure, the first protection system signal actuated was Containment High Pressure which initiated SI; the SI signal produced a Reactor Trip signal. Feedwater isolation occurred as a result of the SI signal. Finally, steamline isolation occurred via a Containment Pressure High High signal.

For the cases in which the SLCV is assumed to function, the first protection signal actuated was the Steam Line High Differential Pressure Between Steam Header and Steam Lines, which initiated SI; the SI signal produced a Reactor Trip signal. Feedwater isolation occurred as a result of the SI signal.

Minimum SI flowrates corresponding to the failure of one SI train were assumed in this analysis. A minimum SI flow is conservative since the reduced boron addition maximizes a return to power resulting from the RCS cooldown. The higher power generation increases heat transfer to the secondary side, maximizing steam flow out of the break. The delay time to achieve full SI flow was assumed to be 9.9 seconds for this analysis.

Conservative core reactivity coefficients corresponding to end-of-cycle conditions, including moderator temperature density coefficients, were used to maximize the reactivity feedback effects resulting from the MSLB. Use of maximum reactivity feedback results in higher power generation if the reactor returns critical, thus maximizing heat transfer to the secondary side of the SGs.

The mass and energy release rates calculated by LOFTRAN, a summary of which are provided in Table 9, "MSLB Mass and Energy Releases - HZP with SLCV Failure," are input into the COCO model to calculate the containment response, i.e., pressure and temperature profiles during the MSLB. The description of the COCO model for the large break LOCA containment analysis is applicable to the MSLB containment analysis.

The containment temperature and pressure results for two of the MSLB cases are shown in Figures 7 through 10 and in Table 10, "MSLB Containment Response (COCO) Results." Table 10 summarizes the various MSLB cases analyzed. The limiting containment temperature resulting from a MSLB is for the 102% case with the single failure of one MFRV. The limiting temperature of approximately 274°F was reached at approximately 35 seconds into the event.

The limiting containment pressure occurred for the case at HZP with the failure of SLCV in the affected steamline. The limiting pressure of 41.85 psig was reached at approximately 612 seconds into the event. All cases resulted in a peak containment pressure that was less than the containment design pressure of 42 psig.

An evaluation of the potential effects of the new environmental qualification temperature envelope on the Low Pressurizer Pressure - Reactor Trip and the Low Pressurizer Pressure - Safety Injection Trip setpoint uncertainties and the effect of the new Containment Pressure High High setpoint of 10 psig have been performed for the MSLB analyses of UFSAR Section 15.1.5.2, "[MSLB] Analysis Basis." This evaluation found that in one case, HZP with RCPs off at 1000 pcm, the new setpoint uncertainties result in a 0.35 second delay for borated water to enter the RCS, but that there is a substantial margin to the minimum Departure from Nucleate Boiling Ratio (DNBR).

The environmental qualification of equipment required to operate during a MSLB was evaluated using the pressure and temperature profiles in Figures 5 and 6. The new accident temperature profile is generally bounded by the profile used for qualification testing of components, but since the accident profile exceeds the qualification profile in some cases, an accident degradation equivalency via the Arrhenius calculation was used to demonstrate that the components are qualified to the new accident temperature profile. Note that the component temperature profile shown on Figure 6 was a separate analysis run with initial containment pressure at 13.7 psia to maximize the component temperature for environmental qualification. The new accident pressure profile is also generally bounded by the pressure qualification profile. Equipment qualification data packages were evaluated for cases that were not enveloped by the qualification profile to demonstrate that the equipment are qualified to the new accident pressure profile. Therefore, the equipment required to operate during a MSLB is qualified for the environmental conditions that will be experienced inside containment.

Associated Changes

Containment Pressure High High Actuation Setpoint

The new containment analyses for a MSLB inside containment and a large break LOCA inside containment credit actuation of the Containment High High Pressure signal at an analytical limit of 12 psig (corresponding to a NOMINAL TRIP SETPOINT of 10 psig) which is a lower pressure than the 20 psig NOMINAL TRIP SETPOINT listed in Technical Specifications Table 3.3.2-1. Therefore, new values are proposed for the ALLOWABLE VALUE and NOMINAL TRIP SETPOINT for Functions 2.c, 3.b(3), and 4.c in Table 3.3.2-1 to reflect the value used in the new containment analyses.

Containment pressure is monitored to sense an accident condition and initiate necessary Engineered Safety Features. Specifically, a Containment High High Pressure signal initiates

Containment Spray, Containment Isolation (Phase B), and Steamline Isolation. Containment is isolated to prevent the release of radioactivity to the outside environment in the event of a large break LOCA. (Note that the Phase A signal isolates "non-essential" process lines penetrating containment and initiates seal injection. The Phase B signal isolates "essential" process lines penetrating containment.) Containment Spray is initiated to mitigate the pressure and temperature resulting from a large break LOCA or MSLB to ensure containment integrity and minimize leakage. Also, Containment Spray removes elemental iodine, should it be released in the event of a large break LOCA, to reduce the offsite thyroid dose. The steam lines are isolated to prevent blowdown of more than one SG in the event of a MSLB. For these functions, decreasing the actuation setpoint of the Containment High High Pressure signal has a beneficial effect, i.e., less time for potential leakage to the outside environment, earlier mitigation of the containment pressure and temperature increases, earlier removal of iodine, and less blowdown of unaffected SGs. Therefore, decreasing the actuation setpoint for the Containment High High Pressure signal beneficially affects accident results.

Decreasing the actuation setpoint for the Containment High High Pressure signal will not increase the probability of an actuation when not required. The difference between the new 10 psig setpoint and the containment operating pressure of -0.8 psig and +1.0 psig required by Technical Specification LCO 3.6.4, "Containment Pressure," is sufficiently large to prevent an actuation when not required. Also, an actuation requires redundant signals from at least two pressure instruments.

MFIV Closure Times

The new MSLB containment analysis credited a closure time for the MFIVs of 50 seconds, which is shorter than the 80 seconds listed in Technical Specification Surveillance Requirement (SR) 3.7.3.2. Therefore, a new closure time is proposed for SR 3.7.3.2 to reflect the time used in the new MSLB containment analysis.

The MFIVs are motor-operated valves. The gearing and motor on the MFIVs are in the process of being modified to improve their performance. The improvements were identified and being implemented as a result of the H. B. Robinson, Unit No. 2 motor operated valve program implemented in accordance with NRC Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The modifications change the MFIVs design closure time from 68 seconds to 39.2 seconds. The 50 second closure time proposed for SR 3.7.3.2 provides margin above the design closure time. One MFIV has already been modified. The modifications for two remaining MFIVs are scheduled for the Fall 1999 outage. The modifications to the MFIVs will be completed and the new closure time will be verified prior to implementation of the proposed change.

MSIV Closure Time

The new MSLB containment analysis credited a closure time for the MSIVs of 2 seconds, which is shorter than the 5 seconds required by Technical Specification SR 3.7.2.1. The new closure time has been added to the Bases for Technical Specification LCO 3.7.2, "Main Steam Isolation Valves." A change to SR 3.7.2.1 is not proposed. SR 3.7.2.1 is intended to provide assurance that the MSIVs perform within the new MSLB analysis assumptions. A 5 second closure time for SR 3.7.2.1 will be used to verify operability of the MSIV and to provide assurance that the MSIVs will close within 2 seconds for accident conditions.

The MSIVs are 26 inch swing disc type isolation valves. The MSIVs are opened by an air operator, and closed with the air operator, steam flow, and a spring (for 9 inches of the actuator's 19-7/16 inch stroke). When the MSIVs receives a closing signal, the air operator and spring drive the swing disc down into the steam flow. The disc is aligned such that forward steam flow (SG to turbine) helps swing the disc to the fully closed position. The disc enters the steam flow after about 3 inches of actuator stroke. An engineering evaluation, which is based on the forces acting on the disc during closure and plant data at 100% steam flow, concludes that the MSIVs will close within two seconds. For MSLB events, the steamline flow exceeds 100% in the intact steamlines with the MSIVs that need to close to isolate blowdown from the intact SGs. The current surveillance is performed at MODE 3 conditions. As stated in the Bases for SR 3.7.2.1, the surveillance is normally performed upon returning the unit to operation following a refueling. The MSIVs should not be tested at power, since even a partial stroke increases the risk of MSIV closure and the resultant plant transient. A previous test at 8% steam flow did not provide adequate motive force to close the MSIVs in 2 seconds. As tests at higher powers to confirm the 2 second closure time may cause undesirable plant transients, the MSIVs will continue to be surveilled in accordance with the existing SR 3.7.2.1. Testing closure of the MSIVs within 5 seconds combined with trending of the test results provides assurance that the MSIVs have not degraded and will close within the 2 seconds assumed in the MSLB containment analysis.

Containment Leak Rate Testing (CLRT)

The CLRT program described in Technical Specification Section 5.5.16 is based on the peak calculated containment internal pressure for a large break LOCA, P_a. The 40 psig value specified in Technical Specification Section 5.5.16 does not bound the 40.5 psig peak pressure calculated by the new large break LOCA containment analysis. Therefore, Technical Specification Section 5.5.16 must be changed to reflect the new analysis value. The CLRT program implemented at HBRSEP, Unit No. 2, which includes integrated leak rate testing and local leak rate testing, historically has been performed at the containment design pressure of 42 psig or higher. Therefore, the operability of containment and its penetrations will not be affected by the proposed change.

Isolation Valve Seal Water (IVSW) System

The IVSW system assures the effectiveness of certain containment isolation valves during condition which requires containment isolation by providing a water seal at the valves. The IVSW system is designed to provide seal water at a pressure of at least 1.1 times the peak containment internal pressure, P_a, during a large break LOCA. The 44 psig values specified in Technical Specification SR 3.6.8.1 for IVSW tank pressure and SR 3.6.8.5 for IVSW dedicated nitrogen bottle pressure are based on a peak LOCA pressure of 40 psig, but are less than 1.1 times the 40.5 psig peak pressure calculated by the new large break LOCA containment analysis. Therefore, the proposed change increases the required surveillance values to 44.6 psig, which is more than 1.1 times the new analysis value. The current operating pressure of the IVSW tank and dedicated nitrogen bottle is 51 to 53 psig, which exceeds the proposed surveillance value of 44.6 psig. Therefore, the design and operation of the IVSW tank and dedicated nitrogen bottle are not affected by the proposed change and the change in the peak containment pressure calculated by the new large break LOCA containment analysis. The surveillance procedures for the IVSW tank and dedicated nitrogen bottle will be changed to specify the new required surveillance value.

Containment Spray (CS) Actuation Signal and Block

CS initiates on a Containment High High Pressure signal, which has a NOMINAL TRIP SETPOINT of 10 psig. Figure 2 shows that containment pressure reaches 10 psig within the first few seconds after the initiation of a large break LOCA. CS flow out of the spray header initiates about 23 or 38 seconds (Table 4) after containment pressure reaches 10 psig. The suction of the CS pump is shifted from the RWST when RWST decreases to 9%, which occurs between about 63 minutes to 77 minutes (See Table 4) after the start of a large break LOCA. To accomplish the shift, the CS pumps must be stopped. The CS pump circuit breakers have an anti-pump feature that will prevent the CS pump motors form being restarted after being stopped while the original actuation signal is present. Figure 2 shows that containment pressure is above 10 psig when the switchover is required, hence, CS actuation circuitry must be blocked during the switchover to allow the CS pumps to be restarted manually upon completion of the switchover for further containment heat removal and pressure reduction. For the large break LOCA inside containment, the analysis assumed that CS is reinitiated at 100 minutes. The analysis shows that the containment pressure will be above 10 psig when recirculation is switched from the cold-leg to the hot-leg. In accordance with current procedures and system capabilities, CS will not be reinitiated after recirculation is switched from the cold-leg to the hot-leg.

In response to Inspection and Enforcement Bulletin (IEB) 80-06, "Engineered Safety Features (ESF) Reset Controls," CP&L committed to ensure that an automatic actuation SI and manual signals to CS would not be inhibited when another signal is present and blocked, reset, or overridden. This commitment was implemented through a modification which prevents resetting CS when an ESF actuation signal is present. As the new analysis shows that the CS pumps may be stopped and restarted prior to the containment pressure being below the Containment Pressure

High High setpoint, the signal must be blocked during the switchover to allow stopping of the pumps during the switchover to prevent damage to the pumps. The modification will follow the guidance of IEB 80-06 because the safety-related equipment actuated by the CS actuation circuitry will remain in its emergency mode when the actuation signal is blocked. Operators will be able to change the state of actuated equipment while the signal is blocked. Blocking of the CS actuation circuitry and subsequent operation of the actuated equipment to effect the switchover, including restoration of equipment after the switchover, will be procedurally controlled. In addition, the device used to block the signal will be periodically tested to ensure that the signal is restored when unblocked.

No Significant Hazards Consideration

Carolina Power & Light (CP&L) Company has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. The conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change increases the maximum allowable Service Water (SW) temperature, which is used to determine OPERABILITY of the Ultimate Heat Sink (UHS), from 95°F to 97°F. As a result of the new analyses to support the increase in SW temperature, the proposed change also decreases the required actuation setpoint for the Containment Pressure High High signal from 20 psig to 10 psig, decreases the closure time credited for the Main Feedwater Isolation Valves (MFIVs) in the analysis from 80 seconds to 50 seconds, increases the required operating pressure for the Isolation Valve Seal Water (IVSW) and IVSW nitrogen bottle pressure from 44 psig to 44.6 psig, decreases the closure time for Main Steam Isolation Valves (MSIVs) credited in the analysis from 5 seconds to 2 seconds, and increases the peak calculated containment internal pressure for a large break Loss of Coolant Accident (LOCA), P_a, from 40 psig to 40.5 psig. In addition, the Containment Spray (CS) actuation circuitry will be modified to allow the CS pumps to be restarted after they have been stopped while the original actuation signal is present.

   SW temperature is not itself an initiator of accidents evaluated in the Safety Analysis report (SAR). The components provided SW flow that are required to perform a safety-related function are designed to operate at temperatures above the temperatures to which SW will be increased. Therefore, these components are not more likely to fail and initiate an accident. The components have been shown to perform their intended safety related function with the higher SW temperatures. Containment analyses have been performed that show that containment integrity and equipment environmental qualification are maintained.

   The modification to the Containment High High Pressure actuation setpoint will not increase the probability of an unwanted actuation. Changing the actuation setpoint will not change the reliability of this function. The Containment Pressure High High Pressure function will 1) initiate Containment Spray sooner, which will mitigate the pressure and temperature transient sooner, and 2) isolate leakage of radioactivity from containment through "essential" process lines sooner in an accident. Also, the lower actuation

setpoint, in conjunction with other analysis assumptions, has been evaluated to result in a slight decrease (-2°F) in the large break LOCA Peak Cladding Temperature.

Crediting faster MFIV closure in the Main Steam Line Break (MSLB) containment analysis will not change the probability of MFIV failure or the probability that the MFIV will initiate an accident because a physical modification is not associated with the proposed change. (The physical modification is being implemented in accordance with 10 CFR 50.59). Since there is no physical modification, the amount of feedwater addition to containment during a MSLB if the Main Feedwater Regulating Valve (MFRV) fails open will not change, although the amount calculated by the analysis will be reduced.

Crediting faster MSIV closure in the MSLB containment analysis will not change the probability of MSIV failure or the probability that the MSIV will initiate an accident because a physical modification is not involved. Since there is no physical modification, the amount of blowdown from the unaffected SGs and the amount of radioactivity released to the environment by a MSLB will not be adversely affected, although the amount calculated by the analysis will be reduced. Crediting a faster closure time does not require crediting a faster MSIV opening time because of the valve design, and opening a MSIV is not postulated for an analyzed accident.

Changing the minimum operating pressure of the IVSW components does not involve a physical modification, hence, will not affect the probability that components will fail or initiate an accident. The IVSW system will perform its containment isolation function by providing a water seal at the higher pressure calculated by the new large break LOCA containment analysis.

The Containment Leakage Rate Testing (CLRT) program historically has performed integrated leak rate testing and local leak rate testing at pressures higher than the peak containment pressure calculated by the new large break LOCA containment analysis. The components which are tested by the CLRT program are designed for operation at a pressure higher than the pressure to which they are tested. The current CLRT program ensures that the containment leakage is less than that used to calculate the doses for a large break LOCA accident.

The modification to the CS actuation circuitry will not affect the reliability of the circuit. The modification will be tested periodically to ensure reliability and to confirm the capability of restoring CS after being blocked. Blocking the actuation circuitry will be procedurally controlled and will allow the CS pumps to be restarted, after being stopped, when an actuation signal is present. The analysis results show that containment pressure and temperature are within design limits when CS is stopped for the switchover.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the SAR.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The components provided SW flow have been shown to perform their safety related function with the higher service temperature, hence, will not exhibit any new type of failure mechanism or mode as a result of the increased temperatures.

Decreasing the Containment High High Pressure actuation setpoint only changes the time at which the signal is generated, not how it is generated, or how the actuated equipment responds to the signal, hence, will not introduce any new types of failures.

Crediting faster MFIV and MSIV stroke times in the MSLB containment analysis does not involve a physical modification, hence, can not introduce any new failure modes.

The IVSW components and the components tested by the CLRT program are designed for pressures that are higher than the pressures at which they are proposed to operate and be tested. As the functions of these components are not changing, and the components are capable of withstanding the higher pressure, a higher operating or testing pressure will not create any new failure mechanisms or accidents.

The modification to the CS actuation circuitry will be tested periodically to ensure proper operation and reliability of the circuit. Even if one of the blocking circuits should fail during operation, a single failure of a CS pump has been considered in the containment analysis, hence, is not a new type of failure or accident.

Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does this change involve a significant reduction in a margin of safety?

Containment structural integrity, containment leakage, fuel cladding, equipment environmental qualification, EDG electrical capacity, and UHS capability were considered to determine if the proposed change involves a significant reduction in a margin of safety.

Containment pressure is limited to the design pressure of 42 psig to maintain structural integrity. A structural integrity test at 115% of the design pressure (48.3 psig) has confirmed the containment's structural capability. The new containment analyses for large break LOCA and MSLB using a SW temperature of 100°F show that the containment pressure does not exceed 42 psig. The margin of safety for containment is not reduced by the proposed change because the design pressure is not exceeded.

The containment leakage rate, L_a, is limited to 0.1% of the containment air weight per day. L_a is based on the peak calculated containment internal pressure, P_a, for the design basis LOCA. The offsite doses resulting from an accident are based on L_a. If containment leakage does not exceed L_a, the margin of safety is not reduced. The leakage rates for Type A, B, and C containment penetrations are measured periodically throughout plant life to ensure that containment leakage is $\leq L_a$. The leakage rate acceptance criteria are $\leq 0.75\ L_a$ for Type A tests, and $\leq 0.60\ L_a$ for Type B and Type C tests. As a result of using a SW temperature of 97°F in the new large break LOCA containment analysis, P_a has changed from 40 psig to 40.5 which changes the pressure at which the Type A, B, and C containment penetration leakage is measured. Historically, containment leakage rate testing has been performed at the containment design pressure of 42 psig or higher. The margin of safety related to containment leakage is not reduced by the proposed change because containment leakage is $\leq L_a$.

Fuel cladding integrity is evaluated by determining the effect on the Peak Cladding Temperature (PCT) and the Departure to Nucleate Boiling Ratio (DNBR) for postulated accident. The PCT for a large break LOCA changes by -2°F as a result of the proposed change including associated changes. The DNBR for a non-limiting case of the MSLB changes, but the margin to the DNBR limit is very large. Therefore, fuel cladding integrity is not adversely affected.

Safety-related equipment is potentially required to function in an adverse environment during and following an accident. Using a SW temperature of 97°F, the new large break LOCA and MSLB containment analyses yield temperature and pressure profiles show that the temperature and pressure profiles for equipment required to operate during and following an accident are qualified. The margin of safety related to equipment environmental qualification is not reduced by the proposed change because equipment required to operate during and following an accident are environmentally qualified.

The Emergency Diesel Generators (EDGs) provide emergency electrical power to run safety-related equipment following an accident that is accompanied by a loss of offsite power. The EDGs are rated at 110% capacity for 2 hours out of each 24 hours and tested between 106% to 110% for at least 1.75 hours. Since the EDG can provide 110% for 1.75 hours, the margin of safety is not reduced. Using a SW temperature of 97°F, a calculation shows that adequate cooling is provided for the EDG to produce 110% electrical output.

The UHS is required to provide cooling water for at least 22 days following a design basis accident. The UHS is able to provide cooling water for 22.1 days at a temperature of 100°F. Therefore, the cooling capability of the UHS would not be adversely affected.

Based on the above, it may be concluded that the proposed change does not involve a significant reduction in the margin of safety.

Environmental Impact Consideration

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; or (3) result in an increase in individual or cumulative occupational radiation exposure. CP&L has reviewed this request against these criteria and determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9).

Proposed Change

The proposed change increases the maximum allowable Service Water (SW) temperature, which is used to determine OPERABILITY of the Ultimate Heat Sink (UHS), from 95°F to 97°F. As a result of the new analyses to support the increase in SW temperature, the proposed change also decreases the required actuation setpoint for the Containment Pressure High High signal from 20 psig to 10 psig, decreases the closure time credited for the Main Feedwater Isolation Valves (MFIVs) in the analysis from 80 seconds to 50 seconds, increases the required operating pressure for the Isolation Valve Seal Water (IVSW) and IVSW nitrogen bottle pressure from 44 psig to 44.6 psig, decreases the closure time for Main Steam Isolation Valves credited in the analysis from 5 seconds to 2 seconds, and increases the peak calculated containment internal pressure for a large break Loss of Coolant Accident (LOCA), P_a, from 40 psig to 40.5 psig. In addition, the Containment Spray (CS) actuation circuitry will be modified to allow the CS pumps to be restarted after they have been stopped while the original actuation signal is present.

Basis

The proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons.

1. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not involve a significant hazards consideration.

2. The proposed change does not allow for an increase in plant power level and does not increase the production nor alter the flow path or method of disposal of radioactive waste or byproducts. There will be a slight increase in the temperature of the plant cooling water effluent when the Service Water temperature exceeds 95°F, but the effect is very small because the temperature of the plant cooling water effluent is dominated by the significantly larger discharge from Circulating Water system. The plant effluent discharge temperature limits are not being changed and the discharge temperature will not exceed the limits specified in National Pollutant Discharge Elimination Permit SC 0002925. Therefore, the proposed change does not result in a significant change in

the types, or significant increase in the amounts, of any effluent that may be released offsite.

3. The proposed change involves plant modifications to change the Containment High High actuation setpoint and to add the Containment Spray actuation signal block. The Containment Spray actuation signal block will require periodic maintenance and surveillances. The work required to implement and test these modifications and to perform periodic maintenance and surveillances will occur in non-radiation areas. Therefore, the proposed change does not increase individual or cumulative occupational radiation exposure.

Table 1 - Components Interfacing with Service Water System

	Safety Related
Reactor Containment Air Coolers And Fan Motor Coolers	X
Emergency Diesel Generators (Jacket Water, Lube Oil, Scavenge Air)	X
Safety Injection Pumps (Thrust Bearings)	X
Auxiliary Feedwater Pumps (Bearing Oil Coolers)	X
Reactor Auxiliary Building Ventilation (SI, CS, RHR, And AFW Pump Room Air Coolers)	X
Control Room Water Cooled Condensing Units	X
Component Cooling Water Heat Exchangers	X
Steam Generators (via AFW system for plant cooldown during Station Blackout)	
Station Air Compressor and After Cooler	
Condensate Pump Motors	
Heater Drain Pumps And Motors	
Emergency Diesel Generator Air Dryers	
Auxiliary Building Air Handling Units (Evaporative Air Coolers)	
Turbine Lube Oil Coolers	
Electrohydraulic Control Unit Coolers	
Seal Oil Unit Coolers	
Instrument Air Compressors and After Coolers	
Primary Air Compressor, Intercooler, and After Cooler	
Seal Water Booster Pumps	
Main Feedwater Pumps And Motors	
Circulating Water Pumps	
Traveling Screen Wash	
Isolation Valve Seal Water tank	
Condenser Vacuum Pump Heat Exchangers	
Isolated Phase Bus Duct Heat Exchanger	
Condensate Polisher Air Compressor Aftercooler	
Condensate Polisher Waste Water Sump Pumps	
Condensate Polisher Sample Cooler	
Main Steam Sample Roughing Cooler	
Secondary System Sampler Cooler / Chiller	
Auxiliary Boiler Sample Coolers	
Hydrogen Coolers	
Exciter Air Coolers	
Radiation Monitors (R-16, R-18)	

Table 2 - Initial Conditions for Large Break LOCA Mass and Energy Release Calculation

Parameter			Value
Core Rated Thermal Power (MWt)			2346
Reactor Coolant System Fluid Volume Ft^3			9586
Reactor Coolant System Total Flowrate (lbm/hr)			973,000,000
Reactor Coolant System Pressure (psia)			2280
Vessel Outlet Temperature (°F)			610.3
Core Inlet Temperature (°F)			548.5
Vessel Average Temperature (°F)			579.4
Initial Steam Generator Steam Pressure (psia) (at 0% power)			1021
Steam Generator Tube Plugging (%)			0
Initial Steam Generator Secondary Side Mass (lbm)			
102% power			94,503
0% power			137,294
Assumed Maximum Containment Backpressure (psia)			56.7
Accumulator			
Water Volume (ft^3) per accumulator			841
N_2 Cover Gas Pressure (minimum)(psig)			600
Temperature (°F)			130
Safety Injection Flowrate (lbm/sec)	Injection	Recirculation	Piggy Back
Minimum ECCS	4119	3877	425
Maximum ECCS	5838	3877	1052
Safety Injection Delay, total (sec) (from beginning of event)			
Minimum ECCS (SI / RHR)			27.7 / 41.7
Maximum ECCS (SI / RHR)			9.9 / 16.4

Table 3 - Large Break LOCA Mass and Energy Releases - Minimum ECCS

	Reactor Vessel Side of Break		Steam Generator Side of Break	
Time (Seconds)	Flow (lbm/sec)	Energy (BTU/sec)	Flow (lbm/sec)	Energy (BTU/sec)
0.00000	.0	.0	.0	.0
0.00103	82365.3	44573.2	39618.3	21399.7
0.00206	40639.7	21952.1	40313.1	21773.9
0.101	40102.2	21732.2	19713.5	10635.9
0.202	40673.3	22197.4	22178.2	11978.1
0.301	41398.2	22799.7	23342.2	12613.7
0.402	42176.3	23481.5	23249.1	12567.5
0.602	42826.5	24394.4	22110.8	11962.9
0.701	42111.3	24227.3	21855.6	11831.1
0.902	39535.1	23132.5	21680.9	11745.5
1.40	35231.4	21460.7	21018.9	11393.6
1.70	33014.7	20601.2	20719.7	11230.1
1.90	31209.1	19889.1	20150.9	10919.0
2.20	27668.3	18276.1	19177.1	10387.9
2.30	25944.3	17333.0	18782.8	10173.0
2.50	20498.7	13919.7	17990.0	9741.9
2.70	18694.1	12804.5	17279.4	9356.8
3.10	14645.1	10142.2	16275.6	8817.1
3.40	12575.4	8813.5	15660.0	8488.9
3.60	11768.7	8310.2	15288.7	8291.1
3.90	10944.0	7812.6	14577.6	7909.6
4.40	9876.7	7210.7	13917.6	7561.1
4.80	9212.2	6860.6	14624.2	7956.6
5.40	8708.8	6604.3	14625.1	7969.0
5.80	8344.7	6416.4	14413.0	7858.8
6.80	7933.9	6059.9	13773.3	7505.8
7.20	7968.6	5947.3	13471.8	7335.3
7.60	8515.3	6345.5	13261.4	7216.6
8.00	7153.8	5957.4	12797.8	6958.1
8.80	6686.5	5571.5	11953.3	6499.7
9.60	6391.8	5158.9	11130.7	6049.9
10.6	5928.9	4735.8	10193.7	5540.5

Table 3 - Large Break LOCA Mass and Energy Releases - Minimum ECCS

	Reactor Vessel Side of Break		Steam Generator Side of Break	
Time (Seconds)	Flow (lbm/sec)	Energy (BTU/sec)	Flow (lbm/sec)	Energy (BTU/sec)
12.4	4762.6	3790.2	8439.2	4594.3
13.6	4138.0	3299.6	7605.2	3833.9
13.8	4031.0	3247.3	7302.3	3624.3
14.0	3921.7	3200.1	7430.1	3638.1
14.4	3704.0	3127.1	6690.8	3193.9
14.6	3591.1	3098.9	6971.9	3291.3
14.8	3478.8	3079.6	6376.1	2984.0
15.2	3211.6	3044.7	6593.4	3015.4
15.4	3063.9	3038.3	5849.1	2651.6
15.8	2546.0	2853.0	5382.8	2363.7
16.2	2126.3	2581.4	4430.1	1882.9
17.0	1496.5	1857.4	3590.1	1414.1
17.2	1374.3	1709.4	3503.9	1347.7
18.2	853.6	1070.1	4318.6	1453.4
18.6	678.7	852.9	3845.1	1258.2
19.4	397.6	501.1	2652.0	828.6
20.8	79.8	101.3	988.5	299.8
21.2	.0	.0	.0	.0
22.1	.0	.0	.0	.0
22.2	70.0	82.3	.0	.0
22.3	21.8	25.6	.0	.0
22.6	21.2	25.0	.0	.0
23.1	51.8	61.0	.0	.0
24.2	88.0	103.6	.0	.0
25.3	115.2	135.6	.0	.0
26.3	135.2	159.2	.0	.0
27.3	159.2	187.5	3584.4	523.5
28.0	160.1	188.6	3667.4	538.8
28.3	159.7	188.2	3646.6	536.8
30.3	157.6	185.6	3492.6	520.6
31.3	156.6	184.4	3417.0	512.4
32.3	155.6	183.3	3343.6	504.5
33.3	154.7	182.2	3272.6	496.7
34.3	153.8	181.2	3204.0	489.2

Table 3 - Large Break LOCA Mass and Energy Releases - Minimum ECCS

	Reactor Vessel Side of Break		Steam Generator Side of Break	
Time (Seconds)	Flow (lbm/sec)	Energy (BTU/sec)	Flow (lbm/sec)	Energy (BTU/sec)
35.0	153.2	180.5	3157.5	484.1
35.3	153.0	180.2	3137.9	481.9
36.3	152.1	179.2	3074.2	474.9
37.3	151.4	178.3	3012.8	468.0
38.3	150.6	177.4	2953.5	461.4
39.3	149.9	176.6	2896.3	455.0
40.3	149.2	175.8	2841.0	448.8
41.3	148.5	174.9	2787.5	442.8
43.3	147.2	173.4	2685.8	431.6
45.3	146.0	171.9	2590.3	421.0
46.3	141.0	166.0	1615.2	311.6
47.3	143.7	169.2	269.0	146.4
49.3	142.2	167.5	267.4	144.5
53.4	139.4	164.2	264.3	140.8
61.3	134.7	158.7	259.0	134.6
69.3	130.8	154.0	254.6	129.4
77.3	127.4	150.1	250.7	124.8
86.4	124.4	146.5	247.2	120.7
104.3	120.3	141.6	242.3	114.9
112.3	119.0	140.1	241.1	113.1
120.3	117.9	138.9	241.1	111.8
124.3	117.5	138.4	241.7	111.4
132.3	116.7	137.4	244.3	110.9
140.3	116.0	136.5	248.6	110.9
148.3	115.1	135.5	254.5	111.0
164.3	112.9	132.9	267.9	111.1
166.3	112.5	132.5	269.6	111.1
174.3	111.0	130.7	275.9	110.6
176.3	110.6	130.2	277.4	110.5
184.3	108.8	128.1	282.9	109.7
200.3	105.0	123.6	292.3	107.4
232.3	97.1	114.3	308.6	102.2
232.5	97.1	114.3	308.7	102.2
232.6	101.5	127.7	358.9	98.8

Table 3 - Large Break LOCA Mass and Energy Releases - Minimum ECCS

	Reactor Vessel Side of Break		Steam Generator Side of Break	
Time (Seconds)	Flow (lbm/sec)	Energy (BTU/sec)	Flow (lbm/sec)	Energy (BTU/sec)
252.6	100.4	126.2	360.1	98.3
257.6	101.0	127.0	359.4	97.9
272.6	100.1	125.9	360.3	97.6
292.6	100.7	126.6	359.7	99.1
312.6	99.5	125.1	360.9	98.5
317.6	100.1	125.8	360.3	98.2
332.6	99.2	124.7	361.3	97.7
337.6	99.7	125.4	360.7	97.4
357.6	98.5	123.8	362.0	96.8
362.6	99.0	124.5	361.4	96.5
377.6	98.0	123.3	362.4	96.1
382.6	98.6	123.9	361.9	95.7
397.6	97.6	122.7	362.9	95.3
402.6	98.2	123.4	362.3	94.9
422.6	97.2	122.2	363.2	94.3
427.6	97.8	123.0	362.6	93.9
442.6	97.1	122.0	363.4	93.4
447.6	97.6	122.8	362.8	95.3
462.6	96.9	121.8	363.6	94.7
467.6	97.4	122.5	363.0	94.3
487.6	97.2	122.2	363.3	93.4
502.6	96.4	121.1	364.1	92.9
522.6	96.8	121.7	363.6	93.9
547.6	96.1	120.8	364.3	92.7
567.6	96.4	121.2	364.1	91.6
592.6	95.5	120.0	365.0	92.4
642.6	95.7	120.3	364.7	91.3
687.6	94.9	119.2	365.6	90.4
752.6	94.8	119.2	365.6	89.1
847.6	93.9	118.0	366.5	88.4
852.6	53.1	66.8	407.3	98.3
1021.8	53.1	66.8	407.3	98.3
1021.9	57.6	71.7	402.8	95.7
1143.9	57.6	71.7	402.9	95.7

Table 3 - Large Break LOCA Mass and Energy Releases - Minimum ECCS

	Reactor Vessel Side of Break		Steam Generator Side of Break	
Time (Seconds)	Flow (lbm/sec)	Energy (BTU/sec)	Flow (lbm/sec)	Energy (BTU/sec)
1144.0	56.1	64.5	404.3	27.8
2442.0	47.0	54.1	413.5	29.5
2442.1	47.0	54.1	44.6	12.3
3042.0	44.5	51.3	47.0	12.8
3042.1	44.5	51.3	571.3	124.8
3600.0	42.3	48.7	573.6	125.2
3600.1	36.4	41.9	579.5	118.8
4620.0	33.5	38.5	582.4	119.4
4620.1	31.5	36.3	26.1	3.8
6000.1	28.9	33.2	28.8	4.2
10000.0	24.9	28.7	32.7	4.8
39600.0	17.3	19.9	40.4	5.9
100000.0	13.3	15.3	44.4	6.4
500000.1	7.6	8.8	50.0	6.8
1000000.0	5.7	6.5	52.0	7.0

Table 4 - Large Break LOCA Containment Response Analysis (COCO) Parameters

Service water temperature (°F)			100
RWST water temperature (°F)			100
Initial containment temperature (°F)			130
Initial containment pressure (psia)			15.7
Initial relative humidity (%)			30
Net free volume (ft^3)			2013007
Containment High setpoint (psig)			5.5
Reactor Containment Air Recirculation Fan Cooler Delay time (seconds after setpoint actuation) With Offsite Power (HVH-1&3 / HVH-2&4) Without Offsite Power (HVH-1&3 / HVH-2&4)			30.4 / 35.4 41 / 46
Reactor Containment Air Recirculation Fan Coolers Heat Removal Rate (BTU/sec)	16538.24 @ 300°F	7456.49 @ 200°F	1820.44@ 130°F
Containment High High setpoint (psig)			12
Containment Spray Delay time (seconds after setpoint actuation) With Offsite Power Without Offsite Power			 23.5 38.2
Containment Spray Flowrate (gpm @ 45 psig containment backpressure)			928
Containment Spray Termination Time for Switchover to Cold-Leg Recirculation (minutes) Minimum ECCS Maximum ECCS (single failure 1 RCFC / single failure 1 CS pump)			 77 63 / 66.7
Containment Spray Restart Time following Switchover to Cold-Leg Recirculation (minutes)			100
RHR Heat Exchangers			
UA (BTU/hr-°F)			29400000
Tube Side flow (gpm - Train A / Train B Recirculation Phase)			3833 / 3877
Shell side flow (gpm Component Cooling Water)			8320
Component Cooling Water Heat Exchangers			
UA (BTU/hr-°F)			2200000
Shell side flow (lbm/hr per HX)			4460000
Tube side flow (gpm Service Water)			5000
Additional heat loads (BTU/hr)			7100000

Table 5 - Containment Heat Sinks

No.	Material	Heat Transfer Area ft^2	Thickness ft
1	Containment Cylinder	46926	
	Stainless Steel		0.00158
	Insulation & Epoxy		0.1045
	Carbon Steel		0.03285
	Concrete		3.5
2	Uninsulated Portion of the Containment Cylinder	3462	
	Epoxy		0.0005
	Carbon Steel		0.090026
	Concrete		3.5
3	Containment Dome	6456	
	Stainless Steel		0.00158
	Insulation & Epoxy		0.1045
	Carbon Steel		0.0417
	Concrete		2.5
4	Containment Dome	20094	
	Epoxy		0.0005
	Carbon Steel		0.0417
	Concrete		2.5
5	Interior Unlined Concrete	59846	
	Epoxy		0.001297
	Concrete		1.97
6	Interior Unlined Concrete	3659	

Table 5 - Containment Heat Sinks

No.	Material	Heat Transfer Area ft^2	Thickness ft
	(W/internal steel) Flooded		0.00292
	Epoxy		1.74
	Concrete		0.0221
	Carbon Steel		8.46
	Concrete		
7	Interior Unlined Concrete (W/internal Steel) Dry	7318	
	Epoxy		0.00292
	Concrete		1.74
	Carbon Steel		0.0221
	Concrete		8.46
8	Interior Lined Concrete	8847	
	Stainless Steel		0.00198
	Concrete		3.388
9	Structural and Misc Exposed Steel - Epoxy coated carbon steel	101757	
	Epoxy		0.000583
	Carbon Steel		0.035065
10	Structural and Misc Exposed Steel - Bare Stainless Steel	2708.	
	Stainless Steel		0.01425
11	Galvanized Steel	54865	
	Zinc		0.0000833
	Carbon Steel		0.01102
12	Insulted Copper Cable (Used for EQ Calc only)	0.059	

Table 5 - Containment Heat Sinks

No.	Material	Heat Transfer Area ft^2	Thickness ft
	Hyplon		0.00125
	Ethylene Propylene Rubber		0.0025
	Copper		0.005667
13	Carbon Steel Plate (Used for EQ)	0.0872	
	Carbon Steel		0.005208

Table 6 - Thermophysical Properties of Containment Heat Sinks

Material	Thermal Conductivity (Btu/hr-ft - °F)	Volumetric Heat Capacity (Btu/ft^3 - °F)
Stainless Steel	9.4	60.1
Carbon Steel	29.53	56.9
Zinc	65.3	40.7
Concrete	1.05	22.5
Insulation & Epoxy	0.01088	0.58
Epoxy	0.23	18.3
Hyplon	0.125	32.537
Ethylene Propylene Rubber	0.1445	20.5
Copper	219.0	50.778
Carbon Steel (EQ component)	27.0	48.02

Table 7 - LOCA Containment Response (COCO) Results

Case	Peak Press. (psig)	Peak Steam Temp. (°F)
Double-Ended Pump Suction Break		
Minimum ECCS	40.50@ 850.807 sec	261. 76 @ 850.76 sec
Maximum ECCS	38.17 @ 18.66 sec	258.43 @ 18.66 sec
Double-Ended Hot Leg Break (Minimum ECCS)	39.83 @ 18.66 sec	259.8 @ 18.66 sec

Table 8 - Plant Parameters and Initial Condition Assumptions - MSLB

	102%	0%
Nominal Core Power, Mwt	2300	2300
Reactor Coolant Pump Heat, Mwt	15	15
Initial Power Fraction of Nominal	1.02	0.01
Total Nominal Heat Input, Mwt	2361	23
Reactor Coolant Flow (total), gpm	97300000	97300000
Reactor Coolant System Pressure, psia	2250	2250
RCS Average Temperature, F	579.4	547.0
SG Water Mass (lbm/SG)		
Affected SG	94503	137294
Unaffected SG	88641	135000
Steam Pressure, psia	850	1021
Steam / Main Feedwater Flow(total), lbm/hr	10110000	0
Feedwater Temperature, F	441.5	100
Feedwater Enthalpy (BTU/lbm)	421	70.24

Table 9 - MSLB Mass and Energy Releases -(HZP with SLCV Failure)

Time (sec)	Mass Flow (Lbm/sec)	Energy Release (BTUs/sec)
.00000	.000000	.00000000
2.5000	5186.08	6204614.5
4.5000	4695.00	5628028.5
6.5000	4310.72	5174249.0
8.5000	4010.13	4817854.5
10.500	3765.34	4526759.0
12.500	3528.53	4245599.5
14.500	2393.22	2880490.0
16.500	1317.94	1587421.2
18.500	1238.66	1491926.6
20.500	1170.46	1409685.6
22.500	1113.51	1340962.0
24.500	1062.90	1279840.3
26.500	1016.39	1223643.3
28.500	973.889	1172246.3
30.500	935.501	1125814.0
32.500	900.986	1084050.2
34.500	869.965	1046503.9
36.500	842.384	1013113.1
38.500	817.684	983203.87
40.500	793.507	953921.68
42.500	770.588	926160.56
44.500	750.374	901669.87
46.500	732.619	880155.75
48.500	716.972	861192.81
50.500	703.053	844322.06
52.500	690.505	829111.81
54.500	679.272	815495.25
56.500	669.357	803477.18
58.500	660.582	792839.06
60.500	652.910	783536.68
62.500	645.972	775126.43
64.500	639.769	767605.93
66.500	634.201	760855.75
68.500	629.221	754818.81

Table 9 - MSLB Mass and Energy Releases -(HZP with SLCV Failure)

Time (sec)	Mass Flow (Lbm/sec)	Energy Release (BTUs/sec)
70.500	624.710	749349.37
72.500	620.615	744385.37
74.500	616.953	739945.37
76.500	613.683	735981.06
78.500	610.762	732440.81
80.500	608.137	729258.62
82.500	605.771	726389.87
84.500	603.635	723800.87
86.500	601.711	721467.62
88.500	599.990	719382.06
90.500	598.463	717530.87
92.500	597.107	715886.50
94.500	595.894	714416.62
96.500	594.803	713093.62
98.500	593.815	711895.68
102.50	592.088	709802.75
106.50	590.609	708009.93
110.50	589.299	706421.06
114.50	588.099	704966.31
118.50	586.965	703592.25
122.50	585.865	702258.81
126.50	584.775	700937.18
130.50	583.675	699603.93
134.50	582.555	698245.75
138.50	581.402	696848.56
142.50	580.212	695405.43
146.50	578.979	693911.25
150.50	577.701	692360.93
154.50	576.373	690751.56
158.50	574.995	689081.12
162.50	573.566	687348.50
166.50	572.084	685552.87
170.50	570.551	683694.06
174.50	568.965	681771.87
178.50	567.328	679786.75
182.50	565.639	677739.75
186.50	563.900	675631.68

Table 9 - MSLB Mass and Energy Releases -(HZP with SLCV Failure)

Time (sec)	Mass Flow (Lbm/sec)	Energy Release (BTUs/sec)
190.50	562.111	673463.87
194.50	560.273	671236.12
198.50	558.386	668948.93
206.50	554.468	664199.12
214.50	550.419	659291.43
222.50	546.368	654382.68
230.50	542.252	649394.06
238.50	533.138	638340.87
246.50	461.314	551319.00
254.50	347.471	413523.62
262.50	239.602	283502.81
270.50	191.383	225607.84
278.50	180.162	212172.15
286.50	178.274	209913.04
294.50	177.858	209415.50
302.50	177.623	209133.81
310.50	177.455	208933.54
318.50	177.339	208793.76
326.50	177.254	208692.78
334.50	177.191	208617.17
342.50	177.141	208557.67
350.50	177.100	208508.28
358.50	177.064	208465.28
366.50	177.031	208426.26
374.50	176.998	208386.85
382.50	176.971	208354.39
390.50	176.943	208320.48
398.50	176.914	208285.89
406.50	176.885	208251.04
414.50	176.855	208215.62
422.50	176.825	208179.39
430.50	176.794	208142.14
438.50	176.762	208103.79
446.50	176.729	208064.21
454.50	176.695	208023.34
462.50	176.659	207981.06
470.50	176.623	207937.35

Table 9 - MSLB Mass and Energy Releases -(HZP with SLCV Failure)

Time (sec)	Mass Flow (Lbm/sec)	Energy Release (BTUs/sec)
478.50	176.585	207892.17
486.50	176.546	207845.48
494.50	176.506	207797.17
502.50	176.463	207746.68
510.50	176.420	207694.31
518.50	176.374	207640.10
526.50	176.328	207584.26
534.50	176.279	207526.60
542.50	176.229	207466.71
550.50	176.178	207405.21
558.50	176.125	207341.87
566.50	176.070	207276.37
574.50	176.013	207208.34
582.50	175.954	207137.56
590.50	175.893	207063.79
598.50	175.828	206986.76
606.50	175.761	206906.20
614.50	89.3699	103971.67
622.50	10.6214	12218.310
630.50	10.5566	12143.799
638.50	10.5105	12090.789
646.50	10.4778	12053.135
654.50	10.4558	12027.892
662.50	10.4415	12011.381
670.50	10.5015	12080.447
678.50	10.4465	12017.209
686.50	10.4172	11983.418
694.50	10.5377	12122.038
702.50	10.4755	12050.514
710.50	10.4434	12013.556
718.50	10.4994	12077.952
726.50	10.4400	12009.664
734.50	10.6050	12199.464
742.50	10.5094	12089.459
750.50	10.4588	12031.295
758.50	10.5448	12130.244
766.50	10.4625	12035.573

Table 9 - MSLB Mass and Energy Releases -(HZP with SLCV Failure)

Time (sec)	Mass Flow (Lbm/sec)	Energy Release (BTUs/sec)
774.50	10.4198	11986.448
782.50	10.5408	12125.683
790.50	10.4745	12049.378
798.50	10.5496	12135.742
806.50	10.4848	12061.171
814.50	10.4303	11998.521
822.50	10.5754	12165.408
830.50	10.4929	12070.583
838.50	10.4502	12021.387
846.50	10.5125	12093.099
854.50	10.4443	12014.651
862.50	10.6122	12207.820
870.50	10.5134	12094.068
878.50	10.4606	12033.427
886.50	10.5371	12121.433
894.50	10.4572	12029.445
902.50	10.4157	11981.737
910.50	10.5331	12116.797
918.50	10.4711	12045.500
926.50	10.5650	12153.511
934.50	10.4722	12046.730
942.50	10.4231	11990.194
950.50	10.5561	12143.192
958.50	10.4836	12059.809
966.50	10.4464	12017.021
974.50	10.4931	12070.716
982.50	10.4336	12002.354
990.50	10.5880	12179.984
998.50	10.5012	12080.133
1000.0	.000000	.00000000

Table 10 - MSLB Containment Response (COCO) Results

Case	Peak Press. (psig)	Peak Steam Temp. (°F)
HZP with single failure of one E-Bus	41.61 psig @ 614.00sec	267.40°F @ 33.472 sec
HZP with single failure of one SLCV	41.85 psig @ 612.21sec	264.21°F @ 612.19 sec
HZP with Auxiliary Feedwater Runout	38.40 psig @ 613.23sec	267.21°F @ 33.579 sec
102% power with single failure of one E-Bus	40.63 psig @ 614.10sec	273.50°F @ 34.50 sec
102% power with single failure of one SLCV	41.19 psig @ 611.78 sec	263.203°F @611.77 sec
102% power with single failure of one MFRV	38.92 psig @ 611.73sec	273.59°F @ 34.50 sec

Figure 1 - Double Ended Pump Suction Break - Minimum ECCS - Temperature

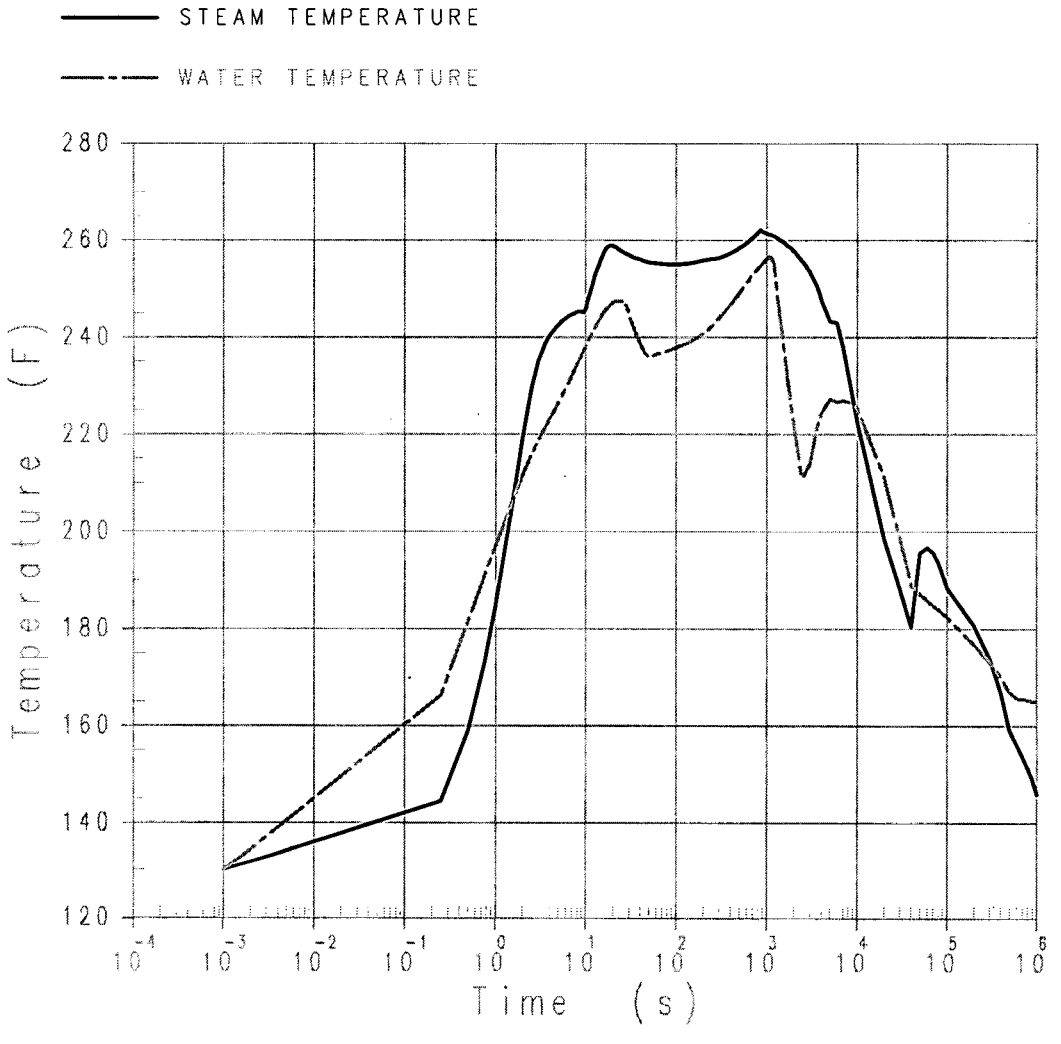

Figure 2 - Double Ended Pump Suction Break - Minimum ECCS - Pressure

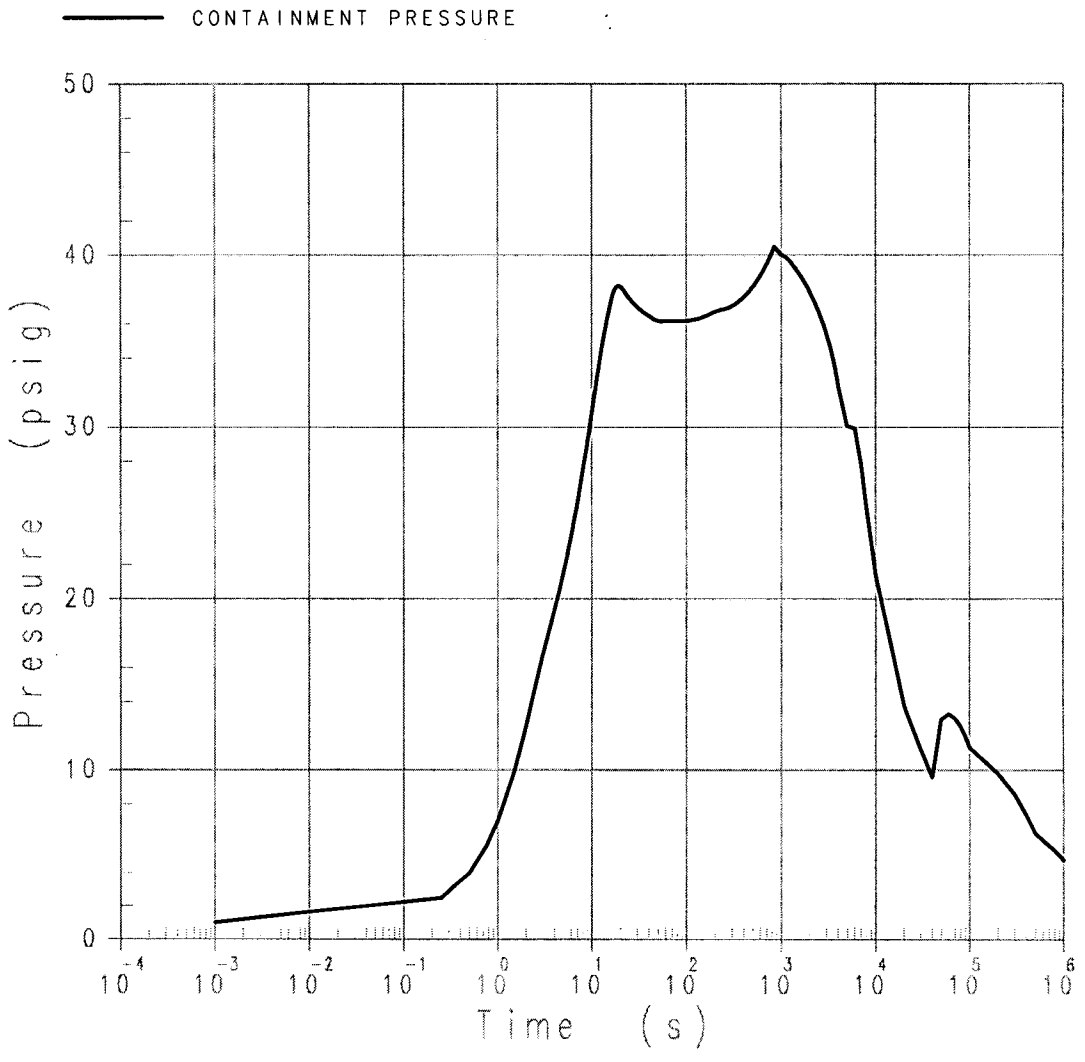

Figure 3 - Double Ended Hot Leg - Minimum ECCS - Temperature

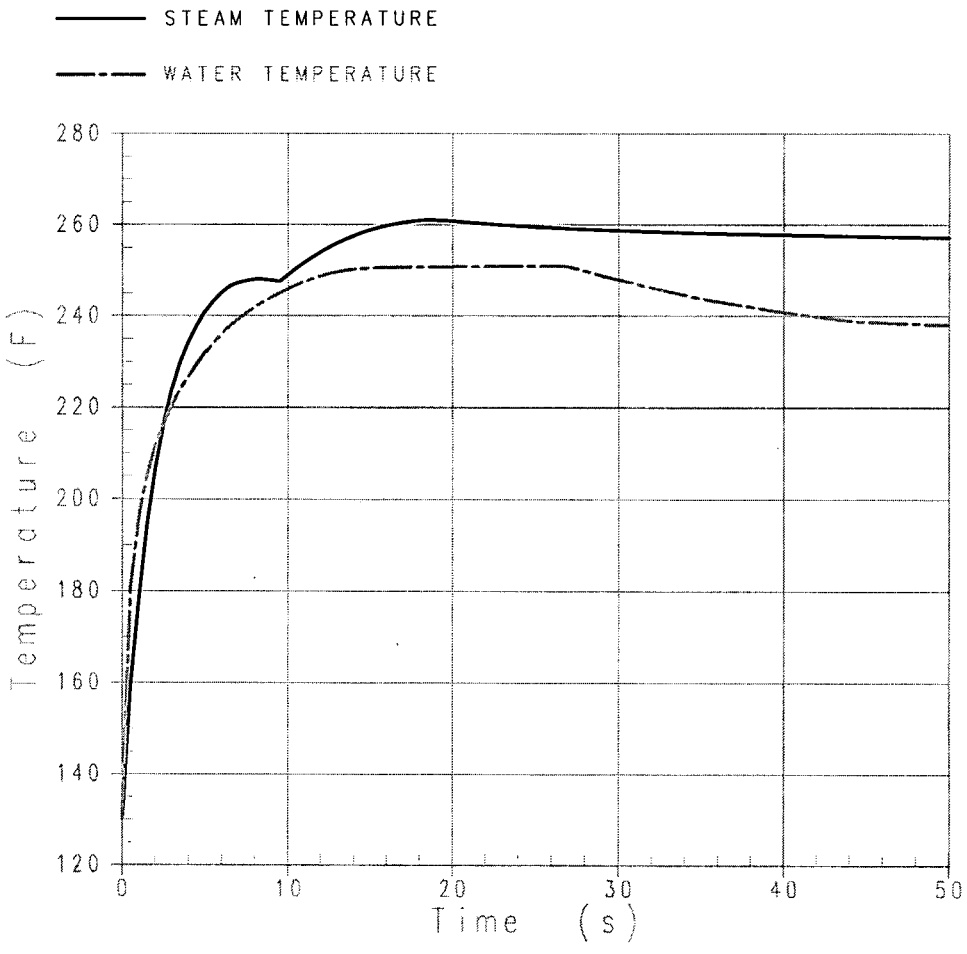

Figure 4 - Double Ended Hot Leg - Minimum ECCS - Pressure

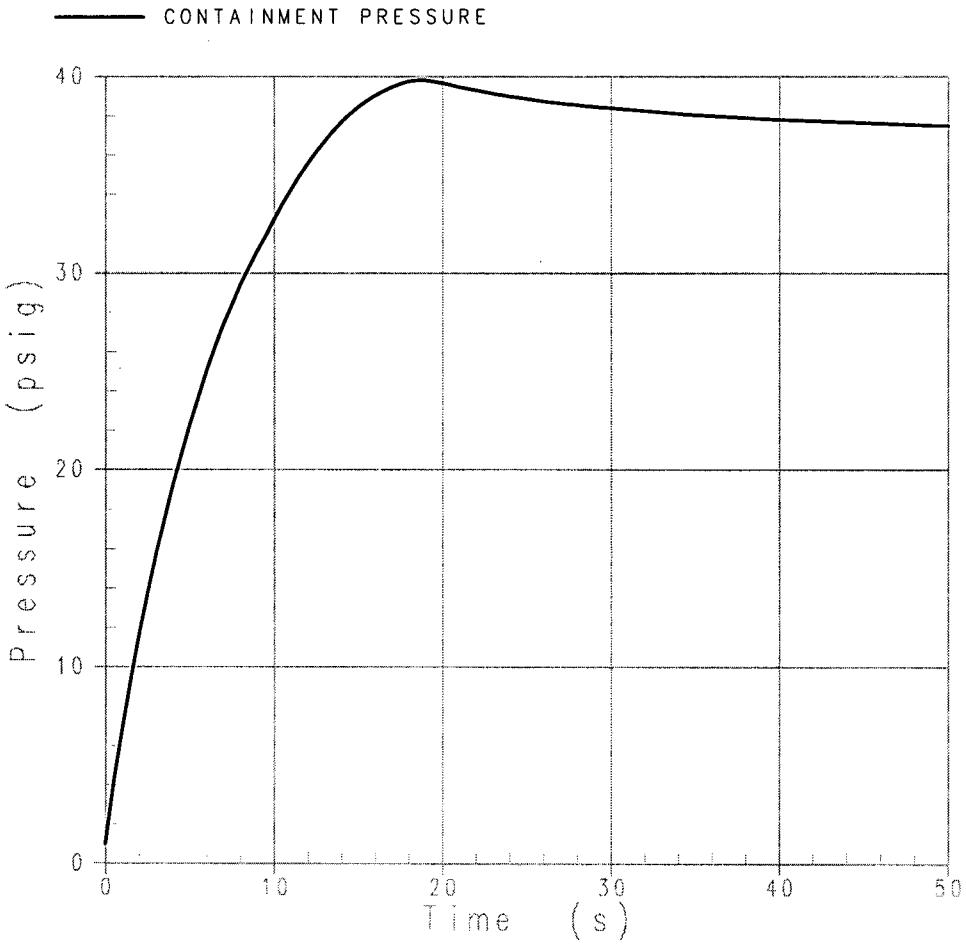

Figure 5 - Containment Pressure Environmental Qualification Envelope

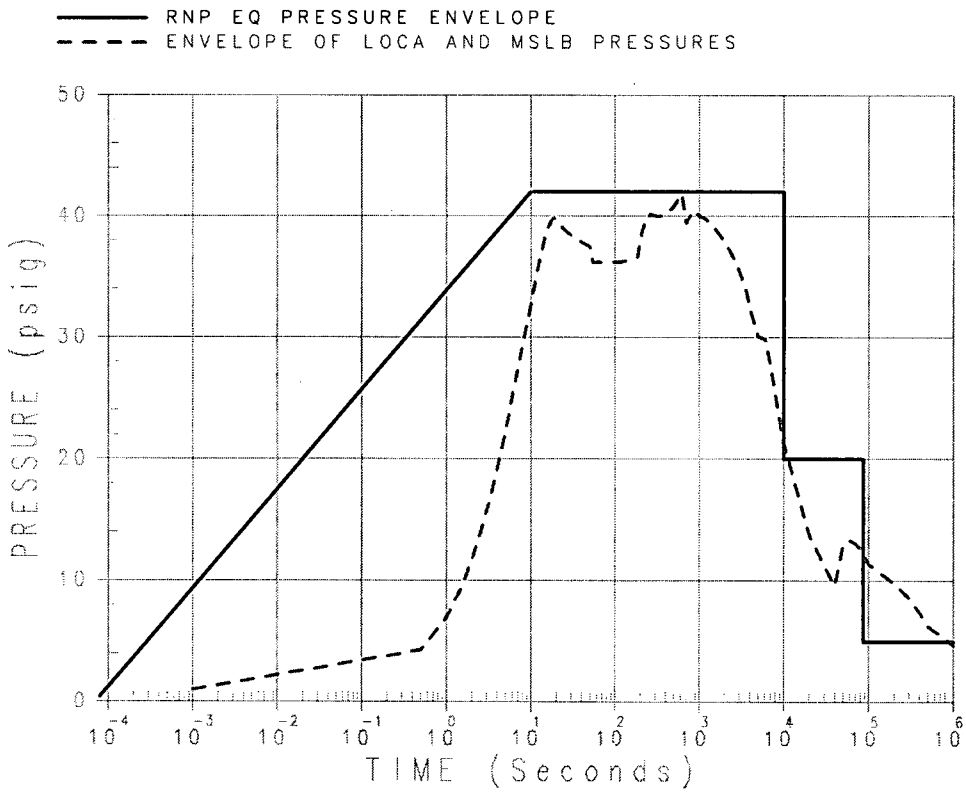

Figure 6 - Containment Temperature Environmental Qualification Envelope

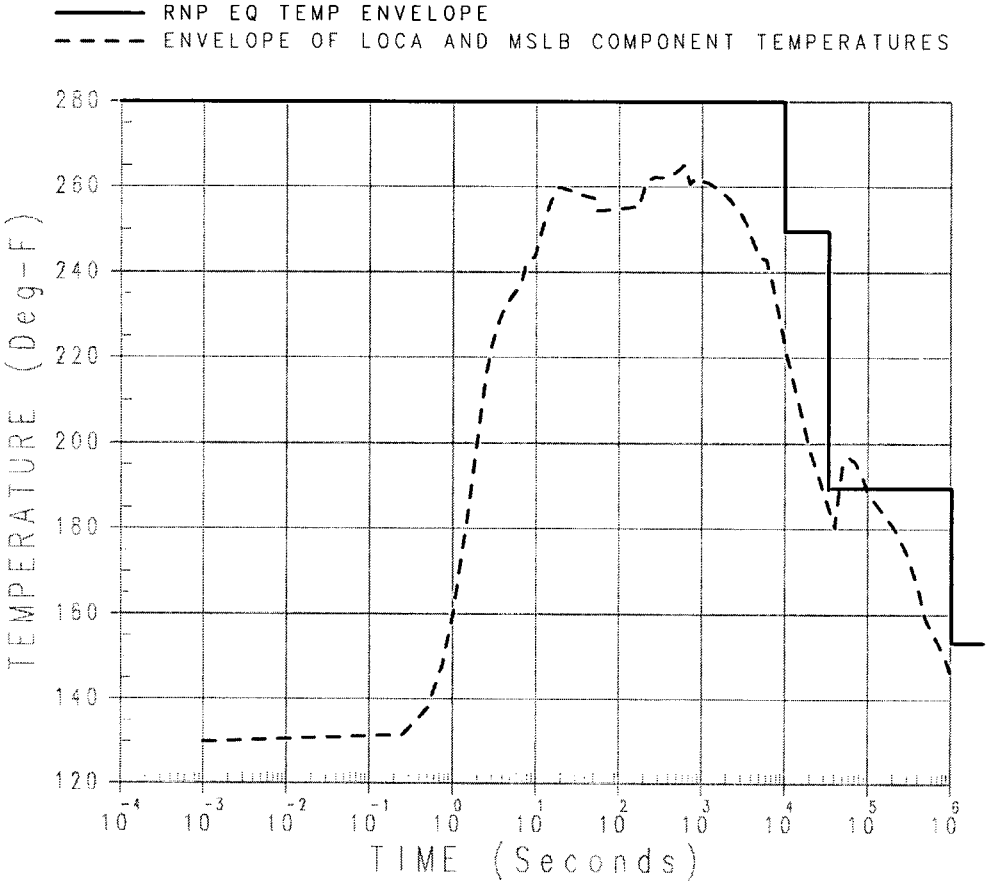

Figure 7 - MSLB 102% Power with Check Valve Failure - Pressure

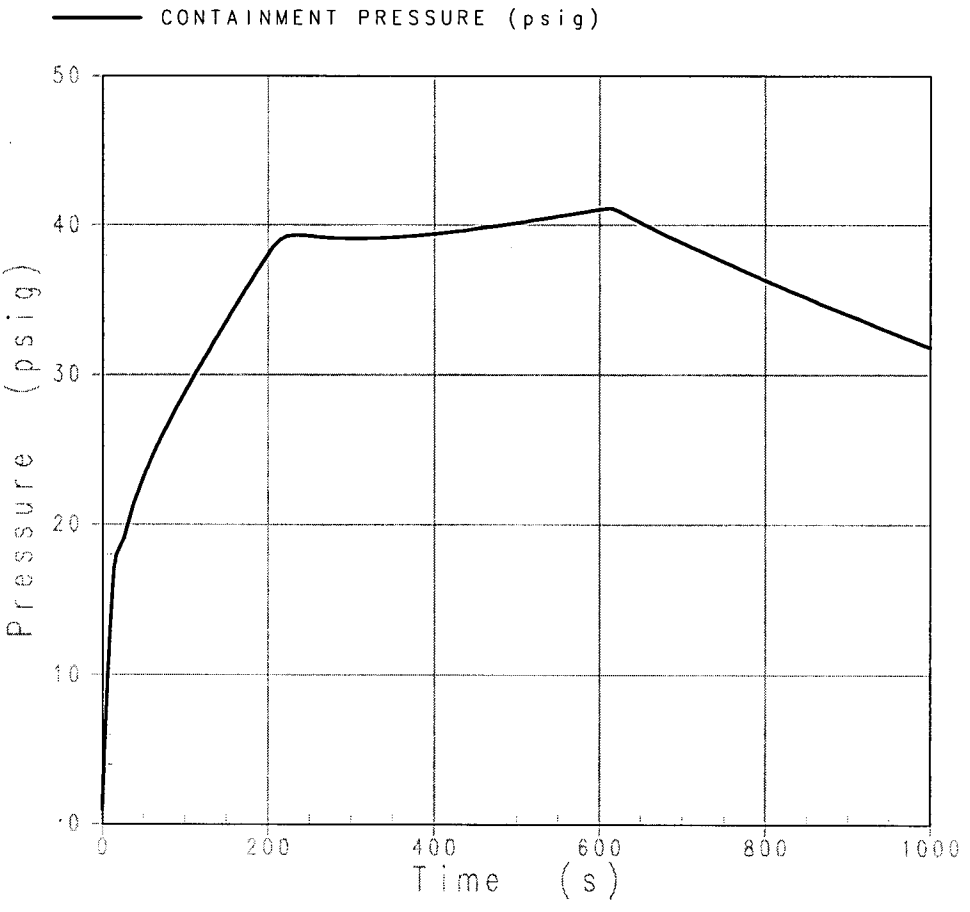

Figure 8 - MSLB 102% Power with Check Valve Failure - Temperature

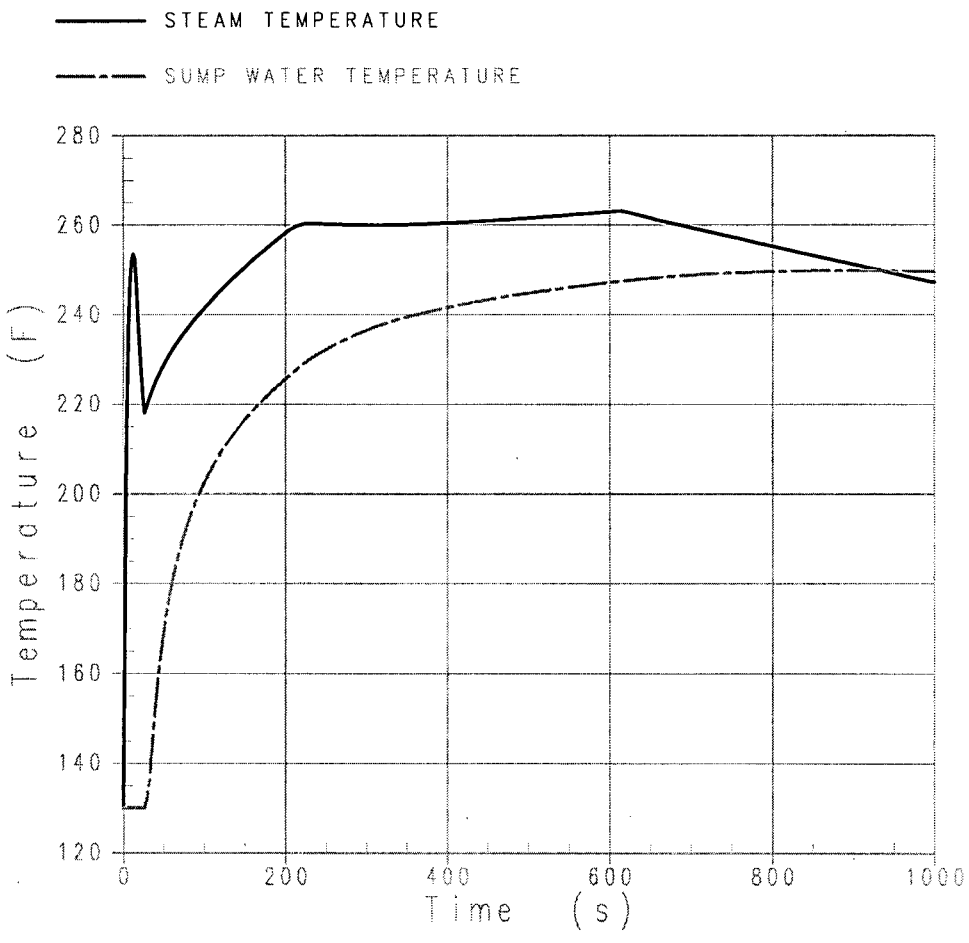

Figure 9 - MSLB HZP with Check Valve Failure - Pressure

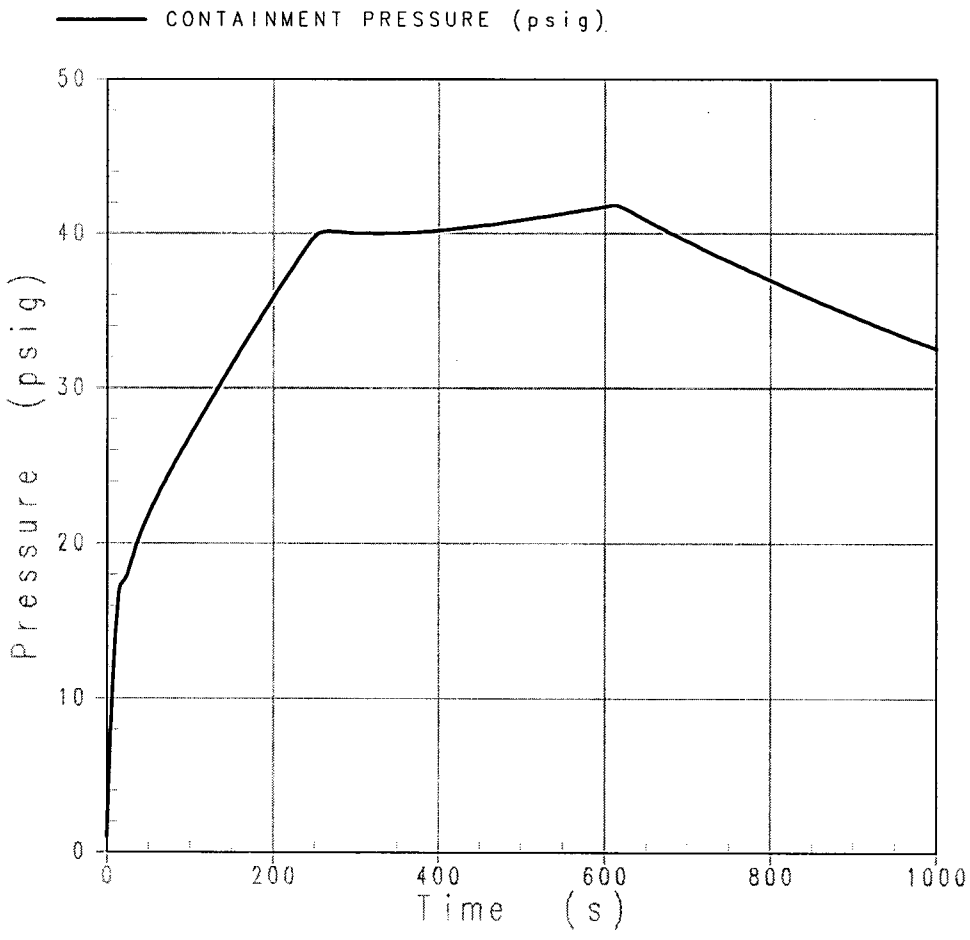

Figure 10 - MSLB HZP with Check Valve Failure - Temperature