Attachment 1
Proposed Revisions to Chapter 15 of the Oconee UFSAR
and the Technical Specification Bases

The current small break LOCA (SBLOCA) analyses are based on the CRAFT2 Evaluation Model. A new Evaluation Model, based on the RELAP5/MOD2-B&W computer code, was approved by the staff in a safety evaluation dated February 18, 1997. This attachment provides a markup of Section 15.14 of the Oconee UFSAR to incorporate revised SBLOCA analyses based on the new Evaluation Model. The revised SBLOCA analyses credit operator actions to raise steam generator levels to the loss of subcooled margin setpoint and depressurize the steam generators.

This attachment also provides a markup of the Bases to Technical Specification 3.3 to incorporate the changes associated with the new SBLOCA analyses. Attachment 1A provides change pages for the Bases of Technical Specification 3.3.

15.14.2.5 Long-Term Cooling

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

Conformance with the acceptance criteria must be demonstrated in a LOCA analysis which is conducted within the guidelines of 10CFR50 Appendix K, "ECCS Evaluation Models." Appendix K outlines the assumptions and analytical methods which have been accepted by the Nuclear Regulatory Commission (NRC) for evaluating the consequences of LOCA. The ECCS evaluation model applicable to Oconee is detailed in the following section.

15.14.3 ECCS EVALUATION MODEL

15.14.3.1 Methodology and Computer Code Description

The evaluation model which has been approved by the NRC for simulating the response of Oconee type plants to a large break LOCA is detailed in the B&W topical report, "B&W's ECCS Evaluation Model" (Reference 1). The evaluation model consists of an integrated application of computer codes that calculate the system response from blowdown to peak clad temperature (PCT). Each code consists of models and assumptions which have been shown to be in accordance with Appendix K.

The CRAFT2 code (Reference 2), which is a modified version of the FLASH-2 code (Reference 3), solves the evolution of system hydrodynamics and core power generation during blowdown. The REFLOD3 code (Reference 4) is used to determine the length of the refill period and the flooding rates during reflood. The CONTEMPT code (Reference 5) calculates the Reactor Building pressure response. The THETA1-B code (Reference 6) is used with the output from CRAFT2, REFLOD3, and CONTEMPT to determine the fuel thermal and mechanical response and the PCT. ~~For the small break LOCA (SBLOCA) which results in core uncovery, the core mixture level is calculated by the FOAM2 code (Reference 7). Clad metal-water reaction and hydrogen generation is calculated by the QUENCH code (Reference 8).~~ The code interfaces for the large break LOCA are shown in Figure 15-44. ~~and for SBLOCA in Figure 15-45.~~

(Insert 1)

15.14.3.2 Simulation Model

The large break LOCA CRAFT2 simulation model is shown in Figure 15-47 and Figure 15-48. A noding scheme of 53 volumes and 86 junctions has been justified by sensitivity studies. All nodes except the pressurizer and the secondary side of the steam generators are treated as homogeneous. For break locations other than the pump discharge, the nodalization is appropriately modified. The core is divided into three radial regions - one for the hot fuel assembly, one for the eight assemblies surrounding the hot assembly, and one for the remainder of the core. Each radial region is divided into six axial levels in order to represent various axial flux shapes. The calculated flow, inlet enthalpy, power, and pressure transient from the CRAFT2 hot assembly is used as input to the THETA1-B code for the hot pin thermal response.

(Insert 2)

~~The SBLOCA CRAFT2 nodalization scheme is shown in Figure 15-49. This nodalization includes a revision to the nodalization given in Reference 1 which separated the original single volume representing the core, core bypass, upper plenum, and upper head, into two volumes representing the core - core bypass, and the upper plenum - upper head (Reference 9). Since the SBLOCA is a moderate transient, a less detailed model than that used for a large break can be utilized. All nodes use a bubble rise model and those flowpaths associated with the reactor vessel include a dual representation which allows for the prediction of counter-current flow. A change to the evaluation model was made in order to yield better~~

Insert 1

The small break LOCA (SBLOCA) evaluation model, which has been approved by the NRC, is detailed in the topical report "BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants" (Reference 38). The SBLOCAs are analyzed with the RELAP5/MOD2-B&W computer code (Reference 39). The SBLOCA evaluation model has been shown to conform to the requirements of Appendix K.

Insert 2

The RELAP5 SBLOCA nodalization is detailed in Reference 38. A detailed nodalization of the primary loop and reactor vessel is included. The secondary side nodalization is sufficient for modeling the effects of emergency feedwater delivery and steaming.

agreement between the CRAFT2 and FOAM2 predicted steam escape rates. A bubble velocity multiplier of 2.38 is applied in the core and a multiplier of 2.0 in the remaining vessel volumes was incorporated (Reference 9).

As a result of NUREG-0737, Section II.K.3.30, the SBLOCA evaluation model has been modified (Reference 32). The modifications include a non-equilibrium pressurizer model, revised emergency feedwater model, two-phase RC pump model, and a more mechanistic steam generator model. The more mechanistic steam generator model includes a more detailed nodalization than is given in Figure 15-49.

15.14.3.3 Thermal Hydraulic Assumptions

Thermal hydraulic conditions and parameters are assumed in accordance with Appendix K.

15.14.3.3.1 Sources of Heat

The reactor is initially operating at 102 percent of 2,772 MWt, the maximum rated power for an Oconee class plant. Core peaking factors are obtained from the analysis based on the criteria of 10CFR50.46. Core stored energy and fuel temperatures are calculated using the TACO2 or TACO3 code (References 10 and 35). Fission product decay heat is given by 1.2 times the ANS standard and decay of actinides is also assumed greater than the ANS decay curve. Direct moderator heating accounts for 2.7 percent of the fission energy released during the blowdown. Metal-water reaction is calculated using the Baker-Just equation without steam limiting. Heat transfer from non-fuel sources is accounted for, as is primary to secondary heat transfer.

15.14.3.3.2 Fuel Mechanical and Thermal Response

The detailed fuel response throughout the duration of the transient is predicted by the CRAFT2 and THETA1-B codes. Thermal expansion, elastic and plastic deformation, and the events leading to possible clad rupture are considered. Approved models for heat capacity and conductivity in the fuel, and gap conductance and heat transfer are used.

As a result of ongoing research programs, the NRC developed new models for fuel clad swelling and rupture which indicated that vendor evaluation models might be nonconservative (Reference 11). These new models for cladding, swelling and rupture are described in NUREG-0630. In response to this NRC concern, a bounding assessment of the impact of NUREG-0630 on the LOCA linear heat rate limits has been performed (Reference 12). The results of this assessment are discussed in Section 15.14.4.2, "Limiting Linear Heat Rate Analysis (LOCA Limits)."

15.14.3.3.3 Blowdown Model

Break flow is calculated using the orifice equation for qualities up to 0.0 at which time a switch to the Moody correlation occurs. Discharge coefficients of 0.6, 0.8, and 1.0 are applied to each break. ECCS bypass is predicted to occur as long as the flow velocity is calculated to be sufficient to carry the ECCS fluid away from the core. The end of blowdown is considered either when zero leak flow occurs or when ECCS water starts entering the core. Friction and form loss factors account for system pressure drops and compare well with measured plant data. Single-phase and two-phase pump models are derived from homologous relationships. Core flow including cross flow is based on a correlation of experimental data. The critical heat flux (CHF) correlations used are the B&W-2, BWC, Barnett, and modified Barnett. Pre-CHF heat transfer uses the Dittus-Boelter correlation for forced convection, the Thom correlation for nucleate boiling, and the Schrock and Grossman correlation for forced convection vaporization (THETA1-B only). Post-CHF heat transfer does not permit use of nucleate boiling heat transfer until that correlation gives a lower heat flux than transition boiling.

A SBLOCA does not progress as rapidly as a large break LOCA. Thus, for a SBLOCA, the timing of ECCS injection is not as significant as with a large break LOCA. For this reason, the worst single failure for a SBLOCA remains the loss of one bus of emergency power. With the selection of an adverse break location, one half of the available HPI train would inject into the broken loop. With these assumptions the ECCS is reduced to the two CFTs, one LPI train, and one half of one HPI train. For conservatism, no credit is taken for the HPIS for large breaks. For a core flood line break, the available equipment are one core flood tank and HPI train.

For the SBLOCA which does not depressurize to below the core flood tank setpoint (600 psig), only one half of one HPI train was available if the break is assumed to be in the cold leg pump discharge. This was identified as an unacceptable scenario (Reference 16). In order to deliver the required HPIS flow of 350 gal/min at 600 psig (Reference 17), the HPIS was modified to allow cross connecting of the pump discharges in order to balance the flow from two HPI pumps into the four injection locations (Reference
6 18, 19). This manual realignment of the HPIS is assumed to be completed within ten minutes of HPIS
6 actuation.

(Insert 3)

15.14.4 BREAK SPECTRUM ANALYSIS

The LOCA analysis has been performed using the ~~B&W Evaluation Model~~ NRC-approved evaluation models in accordance with 10CFR50 Appendix K for a complete spectrum of break sizes and locations.

~~This~~ The LBLOCA analysis is given on a generic basis for an Oconee type plant in BAW-10103A Rev. 3, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS" (Reference 20).

The reference analysis was revised and expanded to account for an improved system loop pressure distribution (Reference 21). ~~to change the location of the worst small break from the cold leg pump suction to the pump discharge (Reference 9), and to examine the impact of a delayed reactor coolant pump trip on SBLOCA (Reference 14). The effects of these reanalyses on the reference analysis will be presented in the following sections.~~

15.14.4.1 Large Break LOCA

A spectrum of large breaks from 0.5 ft^2 up to and including the cross sectional area of the largest pipe in the system was analyzed for both double-ended and longitudinal split breaks in all locations. The methodology used to identify the worst break was as follows. A double-ended break with discharge coefficient C_D = 1.0 was analyzed at the hot leg, cold leg pump suction, and pump discharge. The cold leg pump discharge was determined to be the worst break location. The break size was then varied from 0.5 ft^2 to the geometric maximum for both double-ended and split breaks. The results of these analyses are shown in Table 15-6 and Figure 15-50. A symmetric power shape with an axial peaking factor of 1.7 and a peak linear heat rate of 18 kw/ft is assumed.

The worst break was identified as the double-ended cold leg break at the pump discharge with C_D = 1.0. This break of 8.55 ft^2 area yielded a predicted PCT of 2079°F and a maximum local metal-water reaction of 4.29 percent. The same break size at the pump suction showed a predicted PCT of 1920°F and a metal-water reaction of 3.04 percent. The largest hot leg break of 14.14 ft^2 resulted in a PCT of 1953°F and a metal-water reaction of 3.28 percent. The range of break sizes smaller than the full area double-ended break at the pump discharge all showed less severe consequences.

A series of large breaks are analyzed from an initial condition where three or two reactor coolant pumps are in operation. Five possible break locations associated with these modes of operation were identified.

Three pump operation:

Insert 3

The performance of the ECCS is also evaluated assuming that one of the three HPI pumps is initially unavailable. The limiting single failure leaves only one HPI pump available to inject following SBLOCAs. With only one HPI pump operating, the realignment to cross connect the pump discharges described above cannot be performed due to pump runout concerns at low primary pressure. Significantly less HPI flow capacity results, and the power level must be reduced to 75% full power for the SBLOCA analyses to meet the acceptance criteria.

6 code to specify the initial fuel temperatures. Thus, the LOCA linear heat rate limits in Figure 15-51,
6 Figure 15-111, and Figure 15-112. are not affected by this error.

2 LOCA limit analysis values for MK-B9 fuel are given in Table 15-27, along with a time sequence of
6 events for ECCS equipment. The LOCA limit analysis valves for MK-B10T fuel are given in
6 Table 15-31. These results indicate a maximum PCT of 2108°F and a local metal water reaction rate of
6 3.87 percent. LOCA limit analyses have been performed for one batch of MK-B10 fuel which has a
6 higher initial pin pressure than the other batches. LOCA limit analyses have also been performed for
6 demonstration lead test assemblies for the MK-B11 fuel design. These limits and the results of the
6 analyses are not presented in detail due to their limited applicability.

15.14.4.3 Small Break LOCA

(Insert 4)

1 The SBLOCA is considered to be those break sizes less than 0.5 ft² and greater than the capacity of the
1 normal makeup system. This corresponds to a minimum break size of approximately 0.0008 ft². In
0 addition to the cold leg break locations, the HPI line break and core flood line with a maximum break
0 size of 0.44 ft² were considered. The reference analysis considered three cases; the 0.44 ft² core flood line break, the 0.5 ft² pump discharge break, and the 0.04 ft² pump suction break since it had been previously determined to be the limiting small break (Reference 22). The results of these analyses determined that the core remained covered and assured that the criteria were met.

Subsequent evaluations determined that the worst case small break should be at the pump discharge rather than the pump suction. A break at the pump discharge could result in one half of the HPI going out the break, and an insufficient flow rate would be delivered to the vessel thereby uncovering the core for an extended period of time. As described in Section 15.14.3.3.6, "ECCS Performance and Single Failure Assumption," the HPIS was modified in order to deliver a higher flowrate. Additional analyses of
6 SBLOCA were performed taking credit for operator action to align HPIS flow to both loops within ten
2 minutes (Reference 9 on page 15-66 page=.). The HPI flow rates assumed in core flood line, RCP
2 discharge, and HPI line small break LOCA analyses are given in Table 15-28, Table 15-29, and
2 Table 15-30, respectively. HPI flow rates are obtained from Reference 37. Break sizes of 0.04, 0.055, 0.07, 0.085, 0.10, and 0.15 ft² were performed at the pump discharge to supplement the reference analyses. Two modifications to the evaluation model were also included in the analyses. The results of the analyses showed that minor core uncovery occurred for the 0.055, 0.07, and 0.085 ft² breaks. The worst case PCT resulting from the 0.07 ft² break was 1092°F.

The SBLOCA has been analyzed assuming that the reactor coolant pumps trip and coast down on reactor trip. With no forced flow the liquid in the system would collapse to the lower elevations and an inner vessel mixture level could be tracked to determine the occurrence of core uncovery. For cases in which the pumps remained running, the circulation of the two-phase mixture provided adequate core cooling. Subsequent evaluations of the effect of a delayed pump trip revealed that a pump trip at high system void fractions would result in a collapsed mixture level well below the top of the core (Reference 14). This situation could only occur for a limited range of break sizes and a certain time window when pump trip would be unacceptable. In order to avoid an inadvertent pump trip in the time window, operating procedures were revised to instruct the operator to trip the reactor coolant pumps on loss of primary system subcooled margin (Reference 15). Analyses have shown that this action will prevent the criteria of 10CFR 50.46 from being exceeded for any SBLOCA.

The SBLOCA evaluation model has been revised due to NUREG-0737, Section II.K.3.30. NUREG-0737, Section II.K.3.31 requires that analyses are performed with the revised evaluation model to show compliance with 10CFR 50.46. Compliance with 10CFR 50.46 is demonstrated by a qualitative assessment of the SBLOCA spectrum and then a quantitative evaluation of the critical break sizes. The analyses documented in Reference 29 demonstrate that the previous SBLOCA evaluation model predicts

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The SBLOCA is considered to be those break sizes greater than the normal makeup capacity and less than 0.5 ft^2. Break locations in both the cold leg pump suction and discharge piping are considered, along with a spectrum of break sizes (0.04, 0.07, 0.1, 0.125, 0.15, 0.175, 0.2, 0.3, and 0.5 ft^2). This approach ensures that the limiting case is identified. In addition, two special cases are analyzed. These are the 0.44 ft^2 core flood line break, and the 0.025 ft^2 HPI injection line break. These two cases are unique due to the different fractions of the ECCS flow that can spill out the break and not contribute to core cooling. Breaks at the connection of the HPI injection line to the cold leg are limited in size to the injection line itself. A larger break at this location, which would be a nozzle break, is not required per the NRC-approved evaluation model. Transition break sizes of 0.625 and 0.75 ft^2 are also analyzed to show a continuous plant response as the break size approaches the large breaks which are analyzed with a different evaluation model.

The SBLOCA analyses have demonstrated that the ECCS supplies sufficient emergency coolant injection to meet the 10 CFR 50.46 acceptance criteria for all SBLOCAs. The HPI flowrates assumed in the core flood line, pump discharge, and HPI line break analyses are shown in Tables 15-28, 15-29, and 15-30, respectively. Due to the possibility for spilling of HPI water for cold leg pump discharge breaks and HPI line breaks, credit is taken in the analyses for realigning the HPI System by opening valves HP-409 and/or HP-410 within 10 minutes after ES actuation.

The SBLOCA analyses assume that the operator manually controls the Emergency Feedwater System to raise the steam generator levels to the loss of subcooled margin setpoint. Operator action to begin raising levels to the loss of subcooled margin setpoint, which enhances primary-to-secondary heat transfer, is credited starting at 20 minutes for one steam generator, and 30 minutes for the second steam generator. For SBLOCAs below a break size of 0.06 ft^2, credit is also taken for the operator to manually steam the steam generators at 60 minutes. This action is very effective in cooling and depressurizing the primary, decreasing break flow, and increasing ECCS flow. The normal method of steaming the steam generators is remotely using the Turbine Bypass System. The analyses credit steaming the steam generators locally with the atmospheric dump valves.

The limiting SBLOCA was determined to be a 0.15 ft^2 break at the cold leg pump discharge, with a peak clad temperature of 1288°F. Subsequent to the analysis of the break spectrum, an error in the decay heat associated with the actinides was identified. A reanalysis of the limiting case to address the decay heat error resulted in an increase in the peak clad temperature to 1325°F. An additional reanalysis was performed to incorporate a reduction in HPI flow. The peak clad temperature increased to 1380°F for this reanalysis.

SBLOCA analyses are also performed assuming that one of the three HPI pumps is initially unavailable, and that a single failure leaves only one pump available for credit in the analysis. In this situation a significant fraction of the HPI flow is spilled out the break. The realignment of the HPI System described above cannot be performed with only one HPI pump operating. For the limiting break sizes and locations, the available HPI flow is only capable of cooling the core for initial power levels of up to 75% full power. These analyses also assume that the operator raises the steam generator levels to the loss of subcooled margin setpoint as described above. Steaming of the steam generators at 25 minutes using the atmospheric dump valves is also credited. The peak clad temperature for the one HPI pump SBLOCA analysis is 1607°F for the 0.075 ft^2 cold leg pump discharge break.

~~more conservative results than the revised SBLOCA evaluation model. Therefore, analyses performed with the previous version of the SBLOCA evaluation model remain bounding.~~

15.14.5 EVALUATION OF NON-FUEL CORE COMPONENT STRUCTURAL RESPONSE

The temperature transient in the core can produce significantly higher than normal temperatures in components other than fuel rods. Therefore a possibility of eutectic formation between dissimilar core materials exists. Considering the general area of eutectic formation in the entire core and reactor vessel internals, the following dissimilar metals are present, with major elements being in the approximate proportions shown:

Type-304 Stainless Steel

19% chromium

10% nickel

remainder iron

Control Rod Poison Material

80% silver

15% indium

5% cadmium

Zircaloy-4

98% zirconium

1-3/4% tin

Inconel

53% nickel

19% chromium

3% molybdenum

5% Nb-Ta

1% titanium

0.5 % aluminum

remainder iron

All these alloys have relatively high melting points (greater than 2,700°F) except those for silver, cadmium, and indium. The melting point of the silver-indium-cadmium alloy is about 1,470°F.

The binary phase diagram indicates that zirconium in the proportion 75 to 80 percent has a eutectic point with either iron, nickel, or chromium at temperatures of approximately 1,710, 1,760, and 2,380°F, respectively. If these dissimilar metals are in contact and if those eutectic points are reached, then the

15.14.6 CONFORMANCE WITH ACCEPTANCE CRITERIA

The ~~B&W~~ NRC-approved ECCS Evaluation ~~Model~~ Models used for the LOCA analysis for Oconee class plants ~~has~~ have been shown to be within the guidelines of 10CFR50 Appendix K. ~~This model has~~ These models have been used to perform detailed sensitivity studies to assure that any adverse phenomena are identified and adequately addressed. These analyses have demonstrated that the consequences of hypothetical LOCA's up to and including a double-ended break of the largest pipe in the RCS are within the limits prescribed in 10CFR50.46, as follows:

15.14.6.1 Peak Cladding Temperature

The maximum peak cladding temperature was calculated to be 2108°F, which is less than the 2200°F limit.

15.14.6.2 Maximum Cladding Oxidation

The maximum local metal-water reaction was calculated to be 3.87 percent, which is less than the 17 percent limit.

15.14.6.3 Maximum Hydrogen Generation

The worst case core average hydrogen generation was calculated to be less than the 1 percent limit.

15.14.6.4 Coolable Geometry

Changes in core geometry due to thermal and irradiation effects and mechanical loading have been calculated and show that no gross core blockage or disfiguration will occur. The core will maintain a coolable geometry.

15.14.6.5 Long-Term Cooling

Subsequent to the blowdown, refill, and reflood phases of a LOCA, long-term cooling to remove core decay heat for an extended period of time must be established. The ECCS is designed to perform this function. Operator action is assumed to be available fifteen minutes following a LOCA. Several operational modes are available to provide the necessary cooling and also to assure that adequate coolant circulation exists to prevent any concentration of boric acid in a region of the RCS (Refer to Section 6.3.3.2.1, "Boron Precipitation Evaluation"). Redundancy in the design of the ECCS and multiple available flowpaths for removing core heat provide for sufficient long-term cooling.

15.14.7 ENVIRONMENTAL EVALUATION

The evaluation of the environmental consequences for the LOCA includes the assumption that one percent of the fuel rods in the core have been defective prior to the initiation of the accident. This results in the coolant fission product inventory given in Table 15-14 (Reference 28) for the worst time in life (up to 400 EFPD) for each isotope. The fission product release to the Reactor Building includes the coolant activity plus the gap activity from all fuel rods. The total core gap activities are given in Table 15-4.

Of the iodine released, 50 percent is assumed to plate out and the other half is assumed to remain in the Reactor Building atmosphere where it is available for leakage. No credit is taken for removal of airborne iodine by the Reactor Building Spray System (RBSS). To facilitate environmental dose calculations, all isotopes of iodine have been equated into dose equivalent curies of iodine-131. The dose equivalency factor is determined by considering the concentration and specific dose of each iodine isotope present over

the period of interest. The iodine dose to the thyroid per curie is obtained from the values given in TID-14844. The iodine activity released to the reactor building is 1.43×10^6 dose equivalent curies of iodine-131.

While the Reactor Building leakage rate will decrease rapidly as the pressure decays, the leakage is assumed to remain constant at the rate of 0.25 percent of Reactor Building volume per day for the first 24 hrs. Thereafter, since the Reactor Building has returned to nearly atmospheric pressure, the rate is assumed to be reduced to 0.125 percent of the Reactor Building volume per day and to remain at this value for the duration of the accident.

It is assumed that 50 percent of the Reactor Building leakage will go into the penetration rooms which will be maintained at a negative pressure. The atmosphere in these rooms is discharged through charcoal filters to the unit vent. The charcoal filters are assumed to be 90 percent efficient for iodine removal. The remaining 50 percent of the Reactor Building leakage is assumed to escape directly to the atmosphere. By this method a maximum of 55 percent of the iodine released from the Reactor Building is ultimately released to the atmosphere. Atmospheric dilution of the leakage discharged from the unit vent is calculated using the elevated release dispersion factor of 3.35×10^{-5} sec/m^3. Dilution of the other leakage from the Reactor Building is calculated using the ground release dispersion factor of 1.16×10^{-4} sec/m^3. A breathing rate of 3.47×10^{-4} m^3/sec is assumed for the 2 hr. exposure. For the 30-day exposure, a breathing rate of 2.32×10^{-4} m^3/sec is assumed.

The total integrated thyroid doses resulting from this LOCA fission product release are 5.0 rem for the 2 hr. exposure at the 1 mi exclusion distance, and 5.5 rem for the 30-day exposure at the 6 mi low population distance. The corresponding whole body doses are 0.01 rem and 0.01 rem.

15.14.8 CONCLUSIONS

A complete spectrum of LOCA's have been conservatively analyzed with the ~~B&W~~ NRC-approved evaluation models which conform~~s~~ to 10CFR50 Appendix K. The results of these analyses meet the acceptance criteria of
6 10CFR50.46. The off-site environmental consequences are within the dose limits of 10CFR100. Therefore, the consequences of all design basis LOCA's have been shown to be acceptable.

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39. J. R. Biller, et al, RELAP5/MOD2-B&W, BAW-10164P, Rev. 3, July 1996

5 **Table 15-26. Deleted Per 1995 Update**

6 6 **Table 15-27. Results of LOCA Limits Analysis. MK-B9 BOC Fuel Time Sequence of Events vs. Core Elevation, ft.**

		2	4	6	8	10
6	Allowable peak linear heat rate, kW/ft	16.2	17.5	17.0	17.0	17.0
6	ECCS Actuation Setpt. reached in hot leg, sec	0.4	0.5	0.5	0.5	0.5
6	CFTs Begin Injecting, sec	17.8	15.8	15.8	15.9	15.9
6	End of Blowdown, sec	25.2	24.4	24.0	24.4	24.4
6	LPI Begins Injecting, sec	48.4	48.5	48.5	48.5	48.5
6	Peak cladding temp of unruptured node/time, °F/sec	1862/38.5	2034/90.8	1980/116.7	1917/122.3	1846/147.9
6	Peak cladding temp of ruptured node/time, °F/sec	1894/38.8	1681/38.0	1596/33.0	1502/33.0	1476/36.9
2	Initial pin pressure, psia	1045	1045	1045	1045	1045
6	Rupture time, sec	24.6	25.8	26.7	31.9	36.9
6	Local metal-water reaction, %	2.22	2.91	2.81	2.55	2.47

2 **Table 15-28. HPI Flow Assumed in Core Flood Line Small Break LOCA Analyses**

	RCS Pressure (psig)	HPI Flow Rate (gpm)
2	0	465.5
2	600	440
2	1300	365
2	1500	340
2	1600	325
2	3000	325

insert new
Table 15-28

2 **Table 15-29. HPI Flow Assumed in RCP Discharge Small Break LOCA Analyses**

2 Flow rates prior to credit for operator realignment of HPI at 10 minutes

RCS Pressure (psig)	Total Pump Flow (gpm)	Broken Leg Flow (gpm)	Intact Leg Flow (gpm)
0	515	257.5	257.5
600	450	225	225
1200	380	190	190
1500	342	171	171
1800	300	150	150
2400	260	130	130

Flow rates after credit for operator realignment of HPI at 10 minutes

RCS Pressure (psig)	Total Pump Flow (gpm)	Broken Leg Flow (gpm)	Intact Leg Flow (gpm)
0	548	164.4	383.6
600	500	150	350
1200	437.1	131.1	306
1500	404.3	121.3	283
1800	364.3	109.3	255
2400	260	78	182

insert new Table 29

Table 15-28 HPI Flow assumed in Core Flood Line Small Break LOCA Analyses

Flow rates prior to credit for operator realignment of HPI at 10 minutes

RCS Pressure (psig)	HPI Flow (gpm)
0	428
600	428
1200	333
1500	294
1600	280
1800	250
2400	127

Flow rates after to credit for operator realignment of HPI at 10 minutes

RCS Pressure (psig)	HPI Flow (gpm)
0	817
600	817
1200	653
1500	573
1600	544
1800	482
2400	230

Table 15-29 HPI Flow assumed in RCP Discharge Small Break LOCA Analyses

Flow rates prior to credit for operator realignment of HPI at 10 minutes

RCS Pressure (psig)	Broken Leg Flow (gpm)	Intact Leg Flow (gpm)
0	243	185
600	243	185
1200	189	144
1500	167	127
1600	159	121
1800	142	108
2400	72	55

Flow rates after to credit for operator realignment of HPI at 10 minutes

RCS Pressure (psig)	Broken Leg Flow (gpm)	Intact Leg Flow (gpm)
0	243	574
600	243	574
1200	189	464
1500	167	406
1600	159	385
1800	142	340
2400	72	158

Table 15-30. HPI Flow Assumed in HPI Line Small Break LOCA Analyses

Flow rates prior to credit for operator isolation of broken HPI line at 20 minutes*

RCS Pressure (psig)	HPI Flow Rate (gpm)
0	350.5
600	293.7
1200	213.7
1500	169
1600	150
1800	112.5

Flow rates after credit for operator isolation of broken HPI line at 20 minutes*

RCS Pressure (psig)	HPI Flow Rate (gpm)
0	368
600	333.5
1200	289.5
1500	262.5
1600	253.8
1800	235.0

*Although the Oconee HPI System is not configured so that the operator can accomplish this action, an evaluation has been made which shows that the 10 minute realignment referred to in Table 15-29 provides sufficient flow at Oconee to bound the Table 15-30 line isolation assumption.

insert new Table 30

Table 15-30 HPI Flow assumed in HPI Line Small Break LOCA Analyses

Flow rates prior to credit for operator realignment of HPI at 10 minutes

RCS Pressure (psig)	HPI Injected Flow (gpm)	HPI Spilled Flow (gpm)
0	181	259
600	124	320
1200	48	384
1500	0	408
1600	0	408
1800	0	408
2400	0	408

Flow rates after to credit for operator realignment of HPI at 10 minutes

RCS Pressure (psig)	HPI Injected Flow (gpm)	Intact Leg Flow (gpm)
0	570	259
600	513	320
1200	366	383
1500	279	407
1600	264	407
1800	232	407
2400	103	407

delete

(19)

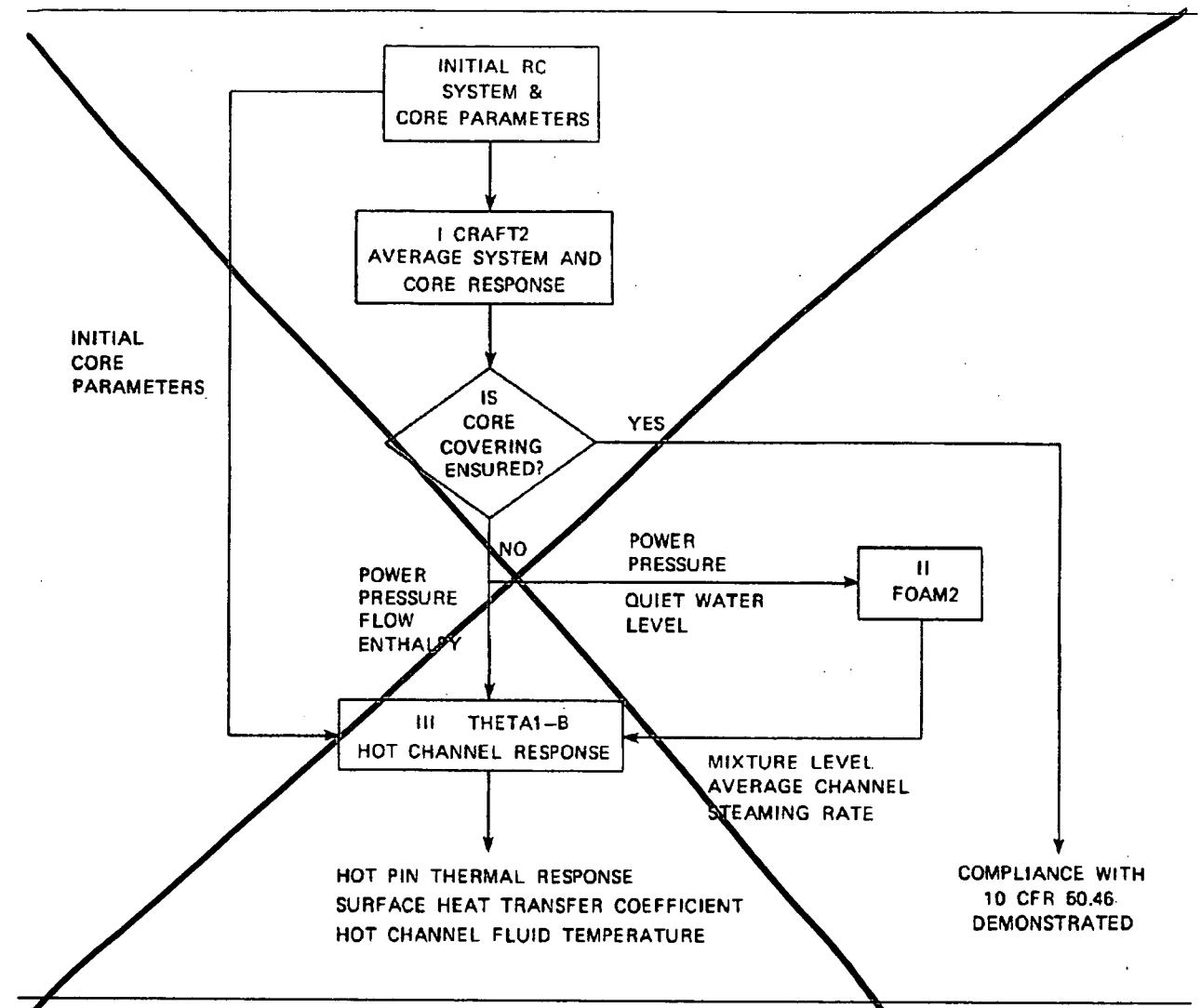

Figure 15-45.
LOCA - Small Break Analysis Code Interfaces

Figure 15-46.
Deleted per 1990 Update

delete

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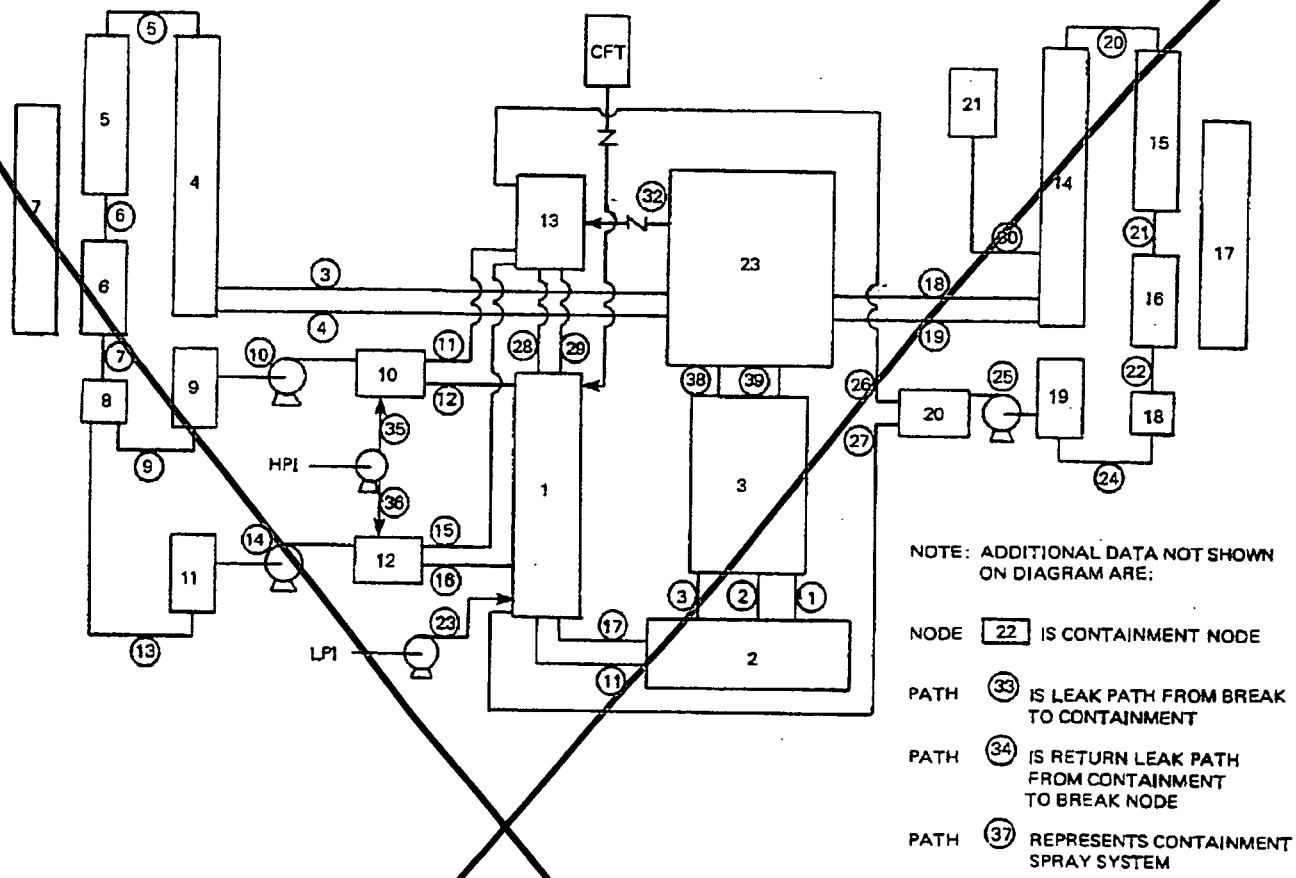

NODE NO.	IDENTIFICATION
1	DOWNCOMER
2	LOWER PLENUM
3	CORE
4,14	HOT LEG PIPING
5,15	SG & UPPER HEAD
6,16	STEAM GENERATOR TUBES
7,17	SECONDARY SG
8,18	SG LOWER HEAD
9,13,19	COLD LEG PIPING
10,12,20	COLD LEG PIPING
13	UPPER DOWNCOMER
21	PRESSURIZER
22	CONTAINMENT
23	UPPER PLENUM

PATH NO.	IDENTIFICATION
1,2	CORE
3,4,18,19	HOT LEG PIPING
5,26	HOT LEG, UPPER
6,21	SG TUBES
7,22	SG LOWER HEAD
8	CORE BYPASS
9,13,24	COLD LEG PIPING
10,14,25	PUMPS
11,12,15,16,26,27	COLD LEG PIPING
17,31	DOWNCOMER
23	LPI
28,29	UPPER DOWNCOMER
30	PRESSURIZER
32	VENT VALVE
33,34	LEAK & RETURN PATH
35,36	HPI
37	CONTAINMENT SPRAYS

Figure 15-49.
LOCA - CRAFT2 Small Break System Nodalization

Bases

Specification 3.3 assures that, for whatever condition the reactor coolant system is in, adequate engineered safety feature equipment is operable.

For operation up to 60% FP, two high pressure injection pumps are specified. Also, two low pressure injection pumps and both core flood tanks are required. In the event that the need for emergency core cooling should occur, functioning of one high pressure injection pump, one low pressure injection pump, and both core flood tanks will protect the core, and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2,200°F and the metal-water reaction to that representing less than 1 percent of the clad.(1) Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.

The requirement to have three HPI pumps and two HPI flowpaths operable during power operation above 60% FP is based on considerations of potential small breaks at the reactor coolant pump discharge piping for which two HPI trains (two pumps and two flow paths) are required to assure adequate core cooling.(2) The analysis of these breaks indicates that for operation at or below 60% FP only a single train of the HPI system is needed to provide the necessary core cooling.

Insert A (see attached)

The requirement for a flowpath from LPI discharge to HPI pump suction is provided to assure availability of long term core cooling following a small break LOCA in which the BWST is depleted and RCS pressure remains above the shutoff head of the LPI pumps.

The borated water storage tanks are used for two purposes:

(a) As a supply of borated water for accident conditions.

(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

Three-hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature above 50°F to lessen the potential for thermal shock of the reactor vessel during high pressure injection system operation. The boron concentration is set at the amount of boron required to maintain the core 1 percent $\Delta k/k$ subcritical at 70°F without any control rods in the core. The minimum boron concentration is specified in the Core Operating Limits Report.

It has been shown that the containment temperature response following a LOCA or main steam line break accident will be within the equipment qualification analysis conditions with one train of Reactor Building spray and two Reactor Building coolers operable.(4) Therefore, a maintenance period of seven days is acceptable for one Reactor Building cooling fan and its associated cooling unit provided two Reactor Building spray systems are operable or one Reactor Building spray system provided all three Reactor Building cooling units are operable.

Valve LPSW-108 is the LPSW isolation valve on the discharge side of each Unit's RBCUs. This valve is required to be locked open in order to assure the LPSW flowpath for the RBCUs is available.

Three low pressure service water pumps serve Oconee Units 1 and 2 and two low pressure service water pumps serve Oconee Unit 3. There is a manual cross-connection on the supply headers for Unit 1, 2, and 3. One low pressure service water pump per unit is required for normal operation.

The Units 1 and 2 LPSW system requires two pumps to meet the single failure criterion provided that one of the Units has been defueled and the following LPSW system loads on the defueled Unit are isolated: RBCUs, Component Cooling, main turbine oil tank, RC pumps, and LPI coolers. In this configuration, if two of the three LPSW pumps are inoperable, 72 hours are permitted by TS 3.3.7.b to restore two of the three LPSW pumps to operable status. At all other times when the RCS of Unit 1 or 2 is ≥ 350 psig or ≥ 250°F, all three LPSW pumps are required to meet the single failure criterion. When all three LPSW pumps are required to be operable and one of the three pumps is inoperable, 72 hours are permitted by TS 3.3.7.b to restore the pump to operable status.

The operability of redundant equipment(s) is determined based on the results of inservice inspection and testing as required by Technical Specification 4.5 and ASME Section XI.

REFERENCES

(1) ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Babcock & Wilcox, Lynchburg, Virginia, June 1975.

(2) Duke Power Company to NRC letter, July 14, 1978, "Proposed Modifications of High Pressure Injection System".

(3) UFSAR, Section 9.3.3.2

(4) UFSAR, Section 15.14.5

(5) BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, BAW-10192P, Framatome Technologies, Inc., Lynchburg, Virginia, February 1994.

6. UFSAR Section 15.14.4

Insert A:

The small break LOCA (SBLOCA) analyses have been performed with an approved Evaluation Model.(5) Certain operator actions support the HPI System operability requirements.(6) Because the possibility exists for spilling of HPI water for cold leg pump discharge breaks and HPI line breaks, the full power SBLOCA analyses credit operator action to realign the HPI System by opening valves HP-409 and/or HP-410 within 10 minutes after ES actuation. The SBLOCA analyses assume that the operator manually controls the Emergency Feedwater System to raise the steam generator levels to the loss of subcooled margin setpoint. Operator action to begin raising levels to the loss of subcooled margin setpoint, which enhances primary-to-secondary heat transfer, is credited starting at 20 minutes for one steam generator, and 30 minutes for the second steam generator. For SBLOCAs from full power below a break size of 0.06 ft^2, credit is also taken for the operator to manually steam the steam generators at 60 minutes. This action is very effective in cooling and depressurizing the primary system, decreasing break flow, and increasing ECCS flow. The normal method of steaming the steam generators is remotely using the Turbine Bypass System. The analyses credit steaming the steam generators locally using the atmospheric dump valves. Therefore, the atmospheric dump valves must be operable to support operability of the HPI System.

The SBLOCA analyses are also performed assuming that one of the three HPI pumps is unavailable and that a single failure leaves only one pump available to mitigate the event. In this situation, a significant fraction of the HPI flow is spilled out the break. The realignment of the HPI System using HP-409 and/or HP-410 cannot be performed with only one pump operating. Analyses demonstrate that a single train of HPI is capable of mitigating the spectrum of SBLOCAs for initial power levels up to 75% FP. Thus, the 60% FP operability requirements in Technical Specification 3.3.1 are bounded by the analyses. These analyses also assume that the operator raises steam generator levels to the loss of subcooled margin setpoint as described above. In addition, steaming of the steam generators using the atmospheric dump valves is credited for the spectrum of SBLOCAs at 25 minutes following ES actuation for SBLOCAs with a break area less than 0.06 ft^2.(6) Since the atmospheric dump valves support operability of the HPI System during certain SBLOCAs, operability requirements for these valves are controlled by Selected Licensee Commitment 16.9.9.

Attachment 1A
Technical Specification 3.3 Bases Change Pages

Remove Page	Insert Page
3.3-6	3.3-6
3.3-7	3.3-7
---	3.3-8

Bases

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(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

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(3) UFSAR Section 9.3.3.2

(4) UFSAR, Section 15.14.5

(5) BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants, BAW-10192P, Framatome Technologies, Inc., Lynchburg, Virginia, February 1994.

(6) UFSAR Section 15.14.4

Attachment 2

Technical Justification

The RCP discharge small break LOCA, described in the revised Bases for Technical Specification 3.3.1, is the limiting accident to be mitigated by the HPI system. Above 60% FP, Specification 3.3.1 provides additional requirements to ensure that two HPI trains will always be available to mitigate this accident. This is accomplished by requiring equipment that would allow a train which did not automatically actuate to be manually aligned from the control room.

As reported in LER 269/90-15, the HPI system operability requirements below 60% power had been based on the small break LOCA on the discharge side of the RCPs. The generic reactor coolant pump discharge small break LOCA analyses assume an even flow split between the injection line connected to the broken cold leg and the injection line connected to the intact cold leg. This is because the back pressure on each line is assumed to equal RCS pressure. In addition, HPI flow from the injection line connected to the broken leg is injected into the reactor coolant pump discharge volume. A computer model then determines how much of this injection flow is lost out the break.

The discovery reported in LER 269/90-15 was that another break location could cause the stated flow split assumption to be non-conservative. If the break location is postulated to be on the HPI injection line itself (0.025 ft^2 break size), downstream of the last check valve but upstream of the RCS cold leg, the appropriate back pressure assumption would be containment pressure for the broken injection line and RCS pressure for the intact injection lines. In addition, none of the HPI flow through the broken injection line would reach the RCS. The resulting flow split from this asymmetric pressure boundary condition would cause less injection flow to reach the reactor.

If a break is postulated at the intersection of the HPI injection line and the RCS cold leg (HPI nozzle), the break size could range from 0.025 ft^2 to 0.5 ft^2. The HPI system is not required to mitigate large break LOCAs, defined as being greater than 0.5 ft^2 in area. A break at the HPI nozzle could result in a break size greater than 0.025 ft^2 with the more severe HPI spilling assumption that the

injection line connected to the broken cold leg is exposed to containment pressure. Since 10 CFR 50.46 addresses breaks in the RCS piping, the break at an HPI injection nozzle is not included in the break spectrum. Based upon discussions with Framatome Technologies Incorporated (FTI), the break of an HPI injection nozzle is not required to be analyzed, according to the licensing history associated with Babcock & Wilcox pressurized water reactors.

Although the HPI line break scenario (0.025 ft^2 break size) had been analyzed based on full power initial conditions and HPI system operability requirements, this scenario had not been considered in developing the less restrictive HPI requirements for operation at or below 60% power. Until an explicit analysis of this scenario determined the maximum allowable power level, the immediate corrective action was to apply the more restrictive requirements for operation above 60% power (existing Specification 3.3.1.c) to all operating modes for which the HPI system is required. Margin in the analyses for operation above 60% power ensured that this corrective action was conservative. This problem is being corrected by the proposed changes to Specification 3.3.1 submitted to the staff on March 31, 1997, which impose the additional restrictions at 75% power. The analyses supporting the proposed 75% FP Technical Specification submitted on March 31, 1997, are bounding for the operability requirements contained in current specification 3.3.1 for less than or equal to 60% power.

In summary, the available HPI flow is calculated for all small break LOCAs, except the HPI line break, by assuming all injection lines are exposed to RCS pressure. This results in an approximately even flow split between the injection lines. For the HPI line break (0.025 ft^2 break size), it is assumed that the broken HPI line is exposed to containment pressure, resulting in a large percentage of the HPI flow being lost to containment.

Analyses have been performed by FTI to confirm that the plant response to all small break LOCAs from an initial condition of 75% FP, with only one HPI pump/train available to mitigate the event (following an assumed single failure of the second HPI train), meets the acceptance criteria of 10 CFR 50.46. These analyses use the recently approved FTI LOCA Evaluation Model as described in FTI topical report BAW-10192P. This topical report was approved by the staff in a safety evaluation dated February 18, 1997. Therefore, these analyses justify the acceptability of Technical Specification 3.3.1 with respect to the concerns identified in LER 269/90-15.

The approach taken in the selection of the break spectrum to be analyzed began with a review of the break spectrum results beginning from 100% power and assuming two HPI pumps and trains were available. The full power break spectrum assumes operator action to cross-connect HPI trains at ten minutes in order to balance the HPI flow. The full power break spectrum also assumes operator action to begin manually increasing steam generator levels from the natural circulation level setpoint to the loss of subcooled margin level setpoint within 20 minutes following reactor trip for one steam generator, and within 30 minutes for the second steam generator. For full power SBLOCAs below a break size of 0.06 ft^2, credit is also taken for the operator to manually steam the steam generators at 60 minutes. These actions are required by the Emergency Operating Procedure following actuation of the HPI System and a loss of subcooled margin. Training exercises demonstrate that the operator action times assumed in the SBLOCA analyses will be met.

A complete break spectrum was then analyzed, including the core flood line break and HPI injection line break, to determine the limiting break size and location. The limiting break size was a 0.15 ft^2 on the reactor coolant pump discharge with a peak cladding temperature of less than 1400°F.

Another reactor coolant pump discharge cold leg break spectrum, along with the core flood line break and HPI line break, was then analyzed from 75% FP with only the flow from one HPI pump. No cross-connecting of the HPI System is credited with only one HPI pump in operation. The break is assumed to occur in the loop with the operable HPI train. Steam generator levels are manually raised by the operator similar to the full power analyses. In addition, credit is taken at 25 minutes after HPI actuation for the operator to steam the steam generators. This action is very effective in cooling and depressurizing the primary system, decreasing break flow, and increasing ECCS flow. The normal method of steaming the steam generators is remotely using the Turbine Bypass System. The analyses credit steaming the steam generators locally using the atmospheric dump valves (ADVs). Again, these actions are required by the Emergency Operating Procedure and the times assumed for operator action have been verified through training exercises.

It was determined from the results of the analyses from 75% FP that the limiting break is a 0.075 ft^2 break on the reactor coolant pump discharge piping, with a peak cladding

temperature of less than 1700°F. For SBLOCAs of 0.06 ft^2 or less, the depressurization of the steam generators by opening the ADVs is required to successfully mitigate the event. For larger break sizes, opening the ADVs is not required. The methodology to model enhanced steam generator heat transfer by crediting raising steam generator levels to the loss of subcooled margin setpoint and depressurizing the steam generators has been demonstrated in FTI topical report BAW-10192P.

Thus, the analyses submitted to the staff on March 31, 1997 also justify the adequacy of Technical Specification 3.3.1 as it is currently written and eliminate the need for the administrative restrictions associated with LER 269/90-15. The additional operator actions credited in the revised small break LOCA analyses include raising steam generator levels to the loss of subcooled margin setpoint and depressurizing the steam generators. Both of these operator actions have been in the Oconee Emergency Operating Procedure for over 10 years and do not introduce any new actions by the Operations staff. Therefore, Duke concludes that there is no increased burden on the operators associated with including these operator actions in the licensing basis safety analyses.

The operator action of depressurizing the steam generators would normally be performed remotely by the operators using the Turbine Bypass System. However, the analysis assumes the Turbine Bypass System is unavailable and credits the use of the atmospheric dump valves. Operability requirements for these valves are currently contained in Selected Licensee Commitment (SLC) 16.9.9. SLC 16.9.9 includes surveillance testing of these valves each refueling outage. The ADVs are manual valves that are located on the turbine deck just outside the control rooms. Since these valves are easily accessible, there is a high level of confidence that the valves can be opened within the time limits assumed in the small break LOCA analyses. The March 31, 1997 Duke submittal provides additional information regarding these valves. The proposed Bases of Technical Specification 3.3.1 state that the operability requirements for the ADVs are controlled by SLC 16.9.9. This SLC, which currently addresses the support function of the ADVs for the auxiliary service water pump, will be revised upon approval of this license amendment to incorporate the additional operability requirements for the ADVs as proposed in the March 31, 1997 submittal to the staff. Duke believes these administrative controls for the atmospheric dump valves will be adequate to address their support function for the HPI System during certain small break LOCAs. As a long-term enhancement for

these administrative controls, Duke has proposed a new Technical Specification for the ADVs in the March 31, 1997 submittal.

In summary, this proposed license amendment updates the UFSAR and Technical Specification Bases to incorporate a complete break spectrum of SBLOCAs based on the new Evaluation Model approved by the staff in BAW-10192P. Operator actions that have been included in the Emergency Operating Procedure for years are credited in these new analyses. The results of the SBLOCA analyses meet the acceptance criteria of 10 CFR 50.46.

ATTACHMENT 3
NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Pursuant to 10 CFR 50.91, Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated:

No. None of the proposed changes has any impact upon the probability of any accident which has been evaluated in the UFSAR.

None of these changes have any impact upon the ability of the HPI System to mitigate the consequences of a small break LOCA, which is addressed below. The small break LOCA is the limiting design basis accident with respect to HPI System operability requirements.

The proposed changes to the Bases of Specification 3.3.1 and Chapter 15 of the Oconee UFSAR include operator actions that have not previously been reviewed and approved by the staff for licensing basis small break LOCA analyses. However, these operator actions have been included in the Emergency Operating Procedure for over 10 years and crediting these actions in the safety analyses does not result in any change to the operator's response to a small break LOCA. These actions are simply changes to the assumptions contained in the licensing basis small break LOCA analyses. The operability requirements for the HPI System contained in Specification 3.3.1 are supported by a spectrum of small break LOCA analyses based on the approved Evaluation Model described in FTI topical report BAW-10192P. These small break LOCA analyses demonstrate that the acceptance criteria of 10CFR 50.46 are satisfied.

The operability requirements in Technical Specification 3.3.1.c assure that the HPI System can withstand the worst single failure and still result in two HPI pumps injecting through two trains. The full power small break LOCA analyses supporting this proposed license amendment have been performed in accordance with the

approved Evaluation Model described in FTI topical report BAW-10192P.

When at or below 75% FP, one HPI train provides sufficient flow to mitigate a small break LOCA. The 60% power level currently in Specification 3.3.1 is justified by analyses using the Evaluation Model described in FTI topical report BAW-10192P, considering the worst case break location and size described in LER 269/90-15 and Attachment 2 to this submittal. The proposed changes to the Bases of Technical Specification 3.3.1 describe the operator actions credited to justify the adequacy of the current specification and eliminate the need for the administrative restrictions imposed by LER 269/90-15. These requirements ensure that, following the worst single failure, one train of HPI would remain available to mitigate a small break LOCA.

In summary, the technical analyses described in this license amendment justify the adequacy of this specification and assure that operability of the HPI System is maintained in a manner consistent with the requirements of the design basis accidents. Therefore, it is concluded that this amendment request will not significantly increase the probability or consequences of an accident previously evaluated.

(2) <u>Create the possibility of a new or different kind of accident from any kind of accident previously evaluated:</u>

No. The proposed changes to the Bases of Technical Specification 3.3.1 and Chapter 15 of the Oconee UFSAR do not result in any new operator actions or changes in plant operation. The proposed changes involve crediting operator actions in the licensing basis small break LOCA analyses that have been included in the Emergency Operating Procedure for years. No new initiating events or potentially unanalyzed conditions have been created. Therefore, this proposed amendment will not create the possibility of any new or different kind of accident.

(3) Involve a significant reduction in a margin of safety.

No. The HPI System requirements associated with the proposed UFSAR and Technical Specification Bases changes are supported by analyses which demonstrate that the acceptance criteria of 10 CFR 50.46 are not violated for any small break LOCA. These analyses were performed in accordance with the Evaluation Model described in FTI topical report BAW-10192P. Therefore, it is concluded that the proposed amendment request will not result in a significant decrease in the margin of safety.

Duke has concluded, based on the above, that there are no significant hazards considerations involved in this amendment request.

Attachment 4

ENVIRONMENTAL IMPACT ANALYSIS

Pursuant to 10 CFR 51.22 (b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22 (c) 9 of the regulations. The proposed amendment does not involve:

1) A significant hazards consideration.

This conclusion is supported by the No Significant Hazards Consideration evaluation which is contained in Attachment 3.

2) A significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed amendment will not change the types or amounts of any effluents that may be released offsite.

3) A significant increase in the individual or cumulative occupational radiation exposure.

The proposed will not increase the individual or cumulative occupational radiation exposure.

In summary, the proposed amendment request meets the criteria set forth in 10 CFR 51.22 (c) 9 of the regulations for categorical exclusion from an environmental impact statement.