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SUBJECT: Provides response to request for addl info re Tech Specs Change 96-10 for high pressure injection sys requirements. Submittal include commitment to include atomospheric dump valves in QA-5 program,developed by Duke.

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June 17, 1998

U. S. Nuclear Regulatory Commission
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Washington, DC 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
High Pressure Injection System Requirements
Response to Request for Additional Information
Technical Specification Change # 96-10

On March 31, 1997, Duke Energy Corporation (Duke) submitted proposed Technical Specification changes for the High Pressure Injection (HPI) System. This submittal was made to address deficiencies in the current HPI Technical Specification. Supplemental information to support this proposed license amendment was submitted to the staff in a letter dated February 9, 1998.

Based on conversations with the staff in April of 1998, Attachment 1 provides additional information to support the proposed license amendment. Attachment 2 is a revision to the bases of the proposed Technical Specifications to reference the approved LOCA Evaluation Model.

This submittal includes a commitment to include the atmospheric dump valves in the QA-5 program that is being developed by Duke. This is the only commitment contained in this submittal.

Please address any questions to J. E. Burchfield, Jr. At (864)-885-3292.

Very Truly Yours,

W. R. McCollum, Jr.
Site Vice President
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A001/1

Attachment

xc:

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Attachment 1
NRC Request for Additional Information
Technical Specification Change 96-10

Question 1: The February 9, 1998 submittal states that neither the atmospheric dump valve nor its associated piping are QA or seismic; however, that is consistent with the original design basis and the QA classification is not based directly on functional definitions. You also reference an August 3, 1995 staff SER as the basis for not requiring these components to meet safety qualifications. In reviewing the referenced information, the staff does not come to the same conclusion. Although the original design basis does not include these components as QA-1, they were not originally credited in the 10 CFR 50.46 LOCA analysis. The August 3, 1995 safety evaluation concludes that components that were credited in the, then current, SBLOCA analysis needed to be at least QA-5, if not QA-1. The basis for this conclusion was, in part, that non QA-1 components were already credited in the licensing basis SBLOCA analysis. It does not appear that the SE approved the use of any additional or all non QA-1 components for future SBLOCA analysis. As a result, if licensing credit is now requested for new components in the SBLOCA analysis, those components should be QA-1. Crediting the equipment that is currently credited (QA-1 or QA-5) continues to be acceptable, however, any new (not previously credited) equipment should be QA-1.

Duke Response: Duke's February 9, 1998, submittal indicated that a portion of the Atmospheric Dump Valve (ADV) flow path is designed as QA-1 and seismic. The bypass and block valves and their associated piping are QA-1 and seismic, as they form the pressure boundary of the main steam line. However, the throttle and isolation valves and their associated piping, which are downstream of the block valve, were not designed as QA-1 or seismic. This categorization is consistent with the original design and licensing basis of the plant.

General Design Criterion 1 of 10 CFR 50, Appendix A, states that:

> "Structures, systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards *commensurate with the importance of the safety functions to be performed."* (emphasis added)

It has been previously recognized by the staff that, depending upon the importance of the function, it is acceptable to credit non-safety equipment in the mitigation of some design basis accidents. For example, many licensees have safety-related feedwater or main steam isolation systems that, if a single failure occurs, would rely on backup non-safety equipment to complete the isolation function.

Duke's proposed Technical Specification changes rely on the ADV flow path for those cases where a single failure disables part of the HPI System. If a single failure is not assumed in the safety-related HPI System, operator actions to depressurize the steam generators using the ADV flow path are not needed to assist in mitigation of a small break LOCA. The most limiting case for the ADV flow path is as follows:

- One HPI pump is out of service and the plant is operating under the proposed Technical Specifications at a reduced power of 75% FP.

- A small break LOCA occurs on the discharge side of the reactor coolant pumps.

- A single failure disables one of the two remaining HPI pumps.

For the above scenario, one ADV flow path would be needed to depressurize one of the two steam generators. The proposed Technical Specifications for the ADV flow paths are conservative in that they require both ADV flow paths to be operable. The analyses described in the February 9, 1998, submittal conservatively assume two failures: 1) one train of HPI, and 2) one ADV flow path. Although credited in the full power analyses, the ADV flow path is not a critical assumption in meeting the acceptance criteria of 10 CFR 50.46.

As stated above, a portion of the ADV flow path is designed as QA-1 and seismic. Duke believes that including the remainder of the flow path in its QA-5 program provides quality standards commensurate with the importance of the safety function. The QA-5 program applies testing and maintenance quality controls consistent with those applied to QA-1 components. Duke believes the QA-5 program applies acceptable quality controls in lieu of reclassifying the ADV flow path to QA-1. This determination is based on the low likelihood of a SBLOCA that would require operation of these

valves, the fact that a seismic event is not postulated to occur coincident with a small break LOCA, the ADVs are manual valves, and the proposed surveillance requirements for the ADV flow path that will assure continued operability of these valves. It should be noted that the QA-5 program is currently under development. It may be necessary to adopt an interim method of testing and maintenance that appropriately applies the QA-5 concepts until the formal program is implemented.

Question 2: For each of the 12 SER restrictions in the NRC SER for BAW-10192-P-A, "BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," describe how the restriction is met.

Duke/Framatome Technologies Incorporated (FTI) Response:
Each of the 12 SER restrictions in the SER for BAW-10192-PA have been reviewed for applicability to SBLOCA analyses. The responses related to SBLOCA are as follows:

<u>NRC Restriction #1:</u> The LOCA methodology should include any NRC restrictions placed on the individual codes used in the evaluation model (EM).

<u>Duke/FTI Response:</u> The SBLOCA analyses are performed with the RELAP5/MOD2-B&W code, which was submitted as topical report BAW-10164. The NRC restrictions were stated in an SER dated April 18, 1990, and in an SER dated March 14, 1995. Some of the NRC restrictions placed on that code are not applicable to the B&W NSSS design or are not SBLOCA-related. The restrictions that are applicable to the Oconee HPI Tech Spec revision request are addressed as follows:

NRC SER Dated April 18, 1990: RELAP5/MOD2-B&W (BAW-10164 Revision 0 & 1)

<u>Restriction (1):</u> The Chen-Sundaram-Ozkaynak film-boiling correlation in the core heat transfer model and B&W auxiliary feedwater model for the once through steam generators were not reviewed and, therefore, should not be used in licensing calculations without prior review and approval by the NRC.

<u>Duke/FTI Response:</u> The CSO correlation is not used in any EM analyses and the B&W auxiliary feedwater model was reviewed and approved in Revision 3. FTI has verified that the Condie-Bengston IV correlation was used for the film boiling correlation.

Restriction (2): Prerupture cladding swell is not modeled because BWFC indicated that the swell is generally less than 20 percent with insignificant flow diversion effects. The acceptability of neglecting the effects of prerupture swelling is part of the LOCA EM review based on BWFCs analysis of the flow diversion effects. The SER on report BAW-10168P will address the resolution of this matter.

Duke/FTI Response: The maximum prerupture strain is 20 percent of the rupture stain. Since the maximum NUREG-0630 rupture strain is less than 90 percent, the resultant maximum prerupture strain must be less than 20 percent.

Restriction (3): The built-in-kinetics for decay heat calculations in the RELAP5/MOD2-B&W code are based on the 1973 and 1979 standards of the American Nuclear Society (ANS). Appendix K requires the use of a value that is 1.2 times the 1971 ANS standard for decay heat calculations. BWFC should ensure that the decay heat used in licensing LOCA analysis complies with Appendix K.

Duke/FTI Response: The 1973 option with a 1.2 multiplier is consistent with 1.2 times ANS 1971 decay heat and this is the option used for all Oconee SBLOCA analyses.

Restriction (4): The LOCA assessments of the Extended Henry-Fauske and Moody critical flow models were based on the use of the static properties as input to the critical flow tables. The LOCA licensing calculations should be performed accordingly.

Duke/FTI Response: FTI has verified that static properties have been used as input to the critical flow tables for Oconee SBLOCA analyses.

Restriction (5): The interphase drag model of the RELAP5/MOD2-B&W code tends to overpredict interphase drag. This overprediction may cause nonconservative predictions of loop seal clearing phenomena in that liquid is cleared even when the steam flow is not sufficiently high to drag the liquid out of the loop seal. Therefore, this model may not accurately calculate the core uncovery and the peak cladding temperature (PCT). A resolution requiring a sensitivity study to choose a proper loop seal nodalization that results in the highest PCT calculation will be addressed in the LOCA EM review.

Duke/FTI Response: Loop seal clearing is a phenomenon related to non-B&W designed plants. The reactor vessel vent valves in the B&W-designed plants provide a direct core

steam venting path to cold leg break locations, such that loop seal clearing is not necessary for the B&W plant analyses.

Restriction (6): Even though noncondensible gases are not modeled in the SBLOCA system analysis, BWFC demonstrated the negligible effect that all sources of noncondensible gases will have on the overall response of the system for the range of SBLOCAs. However, BWFC noted that a 50 psi increase above the steam generator control pressure of 1150 psia could result from a worst case release of noncondensible gases. The staff believes that this pressure increase generally would not substantially reduce the injection capabilities of the charging and safety injections (SI) systems. However, because the performance characteristics of the SI pumps vary widely in the plants, verification should be made on a plant specific basis to ensure that a 50 psi pressure increase will not greatly reduce SI flow such that the PCT would increase by more than 50F. Otherwise, additional information should be provided to justify neglect of noncondensible gases, or the effect of the pressure increase caused by noncondensible gases should be included in the analysis.

Duke/FTI Response: This limitiation is related to non-B&W designed plants and the reflux cooling mode of heat transfer. The reactor vessel vent valves in the B&W-designed plants provide a direct venting path for any core-generated noncondensibles to reach and be discharged out of the cold leg break. Therefore, this verification is not needed for the B&W plant analyses.

Restriction (7): For a complete safety analysis, an approved core thermal hydraulic code and CHF correlation should be used with the RELAP5/MOD2-B&W code. The noding details and inputs should be justified on a plant specific basis. The choice of constitutive models including the empirical models and correlations should be justified to ensure their use is within the ranges of applicability.

Duke/FTI Response: There are three approved CHF correlations that have been used for the core heat transfer in RELAP5/MOD2 EM analyses. These are BWC for any LOCA non-mixing vane grid assemblies (Mark-B9 and Mark-B10), BWUMV for SBLOCA mixing vane grid assemblies (Mark-B11), and BWCMV for LBLOCA mixing vane grid assemblies (Mark-B11). FTI has verified the use of the BWUMV CHF correlation for the SBLOCA analyses based on the Mark-B11 fuel assembly type.

NRC SER Dated April 18, 1990: RELAP5/MOD2-B&W (BAW-10164 Revisions 2 & 3)

First Limitation: Use of Wallis and UPTF parameters at the tube bundle and steam generator plenum inlet are acceptable. The parameters used in the CCFL model for any other application must be validated, and the validation reviewed and approved by the staff for that application (see section 3.1.3 of this evaluation).

Duke/FTI Response: This is a limitation for recirculating steam generator plants. It is not applicable to B&W-designed plants.

Second Limitation: The BWUMV correlation is limited to pressures above 1300 psia.

Duke/FTI Response: The code heat transfer logic is programmed to use BWUMV above 1500 psia. It will linearly interpolate between Barnett and BWUMV between 1500 and 1300 psia. Below 1300 psia, Barnett and modified Barnett are used. Therefore, BWUMV cannot be used below 1300 psia.

Third Limitation: For large break LOCA ECCS evaluation model calculations, form losses due to ruptured cladding should not be excluded using the user option described in Section 3.2.4 of this evaluation.

Duke/FTI Response: The form losses due to cladding rupture are not excluded from the Oconee analyses.

Fourth Limitation: The value of the user-specified parameters listed in Table 1 of this evaluation (i.e. those used for the benchmark calculations) are the only acceptable values for LOCA licensing calculations.

Duke/FTI Response: FTI has reviewed the Oconee SBLOCA analyses for use of the user-specified parameters listed in Table 1. The parameters used were consistent with the table values except for the omission of the B&W slug-drag model inside the SG tubes. This option was used in the benchmark cases reported in the EM appendices. Although use or non-use of this option has a negligible effect on the SBLOCA results, it is in direct contradiction of the restriction. FTI reanalyzed the two most limiting Oconee SBLOCA cases to confirm that the PCTs were similar, and in both cases the PCT was slightly less than the PCT predicted without the B&W slug-drag model option in the tubes.

NRC Restriction #2: The guidelines, code options, and prescribed input specified in Tables 9-1 and 9-2 in both Volume I and II of BAW-10192P should be used in LBLOCA and SBLOCA evaluation model applications, respectively.

Duke/FTI Response: The guidelines, code options and prescribed inputs specified in Tables 9-1 and 9-2 in Volume II (SBLOCA) of BAW-10192P provide a summary of the inputs and models used in SBLOCA analyses. FTI has verified the proper application of these inputs and models, with the clarification of two areas for which the EM interpretation is unclear. These two areas are the (1) rupture temperature ramp rate and (2) two-phase pump degradation multipliers.

The plastic-weighted time averaged ramp rate option was not specifically used for the SBLOCA analyses because a composite set of pin parameters is used to maximize the PCT prediction. See the response to NRC Restriction 5 for clarification of the time-in-life (TIL) input method.

The two-phase pump degradation model was listed as the default option. There is no default option in RELAP5/MOD2-B&W so this designation is unclear. For SBLOCA analyses, the two-phase pump degradation multiplier has little impact because the reactor coolant pumps (RCPs) coast down on loss of offsite power or on manual trip by the operator on loss of subcooling margin. Without the RCPs, the SBLOCA evolves into a low flow condition that is unaffected by the two-phase head difference multiplier table used. Nonetheless, a basis for the head multiplier curve is required. FTI used the M1 degradation model for the Oconee SBLOCA analyses. This model was shown to be the most limiting for the higher flow LBLOCA analyses, therefore it was used for the SBLOCA analyses.

NRC Restriction #3: The limiting linear heat rate for LOCA limits is determined by the power level and the product of the axial and radial peaking factors. An appropriate axial peaking for use in determining LOCA limits is one that is representative of the fuel and core design and that may occur over the core lifetime. The radial peaking factor is then set to obtain the limiting linear heat rate. For this demonstration, calculations were performed with the axial peak of 1.7. The general approach is acceptable for demonstrating the LOCA limits methodology. However, as future fuel or core designs evolve, the basic approaches that were used to establish these conclusions may change. FTI must revalidate that acceptability of the evaluation model peaking methods if: (1) significant changes are found

in the core elevation at which the minimum core LOCA margin is predicted or (2) the core maneuvering analyses radial and axial peaks that approach the LOCA LHR limits differ appreciably from those used to demonstrate Appendix K compliance.

Duke/FTI Response: This limitation is primarily for LBLOCA analyses. SBLOCA analyses are analyzed at the highest core linear heat rate (LHR) limit with a core exit-skewed power peak to produce conservative PCT results.

NRC Restriction #4: The mechanistic ECCS bypass model is acceptable for cold leg transition ($0.75ft^2$ to $2.0ft^2$) and hot leg break calculations. The nonmechanistic ECCS bypass model must be used in the large cold leg break ($\geq 2.0\ ft^2$) methodology since the demonstration calculations and sensitivities were run with this model.

Duke/FTI Response: The SBLOCA evaluations cover break sizes up to $0.75\ ft^2$, therefore no bypass model is used.

NRC Restriction #5: Time-in-life LOCA limits must be determined with, or shown to be bounded by, a specific application of the NRC-approved evaluation model.

Duke/FTI Response: The small break evaluation model includes a provision for predicting cladding swelling and rupture based on NUREG-0630 data. Flow diversion is modeled through the use of hot and average channels, cross flow, and the axial resistance due to flow blockage at the ruptured location. Once rupture has been calculated, the heat transfer, heat conduction, and metal-water reaction models are updated for the resultant strain and the availability of interior clad surface for oxidation.

Time-in-life (TIL) calculations for SBLOCA applications are not required unless the fuel pin heatup is sufficient to cause cladding rupture. FTI evaluates the likelihood of rupture by analyzing the SBLOCA with a composite set of pin conditions that provide a conservative PCT prediction. End-of-life pin pressures are used to maximize the cladding hoop stresses, thereby improving the likelihood of rupture for those cases that do experience heatup. To maximize the cladding temperatures, the beginning-of-life (BOL) fuel stored energy and BOL oxide thicknesses are used. FTI has also used a constant normalized heating ramp rate limit of zero for some cases to maximize the likelihood of cladding rupture. Any case that predicts cladding rupture with these

conditions is further parameterized by adjusting the time of rupture (via pin pressure or normalized heating ramp rate changes) to push rupture to the time of peak cladding temperature. This composite method ensures that the calculated PCT will bound any PCT predicted by a consistent TIL analysis with appropriate TIL pin parameters. A pure TIL calculation (with fuel stored energy, pin pressure, and cladding oxide thickness consistent with the TIL that produces the worst rupture time) would be performed if the composite case is judged to be overly conservative. The consistent case would also use the plastic-weighted normalized heating ramp rate to predict the fuel pin swell and rupture performance.

NRC Restriction #6: LOCA limits for three-pump operation must be established for each class of plants by application of the methodology described in this report. An acceptable approach is to demonstrate that three-pump operation is bounded by four pump LHR limits.

Duke/FTI Response: For SBLOCA analyses, three pump operation is bounded by four pump full power operation because the three pump operation is at a reduced total core power (75 percent for Oconee). The reduced power level lowers the predicted PCT because of the decreased core liquid boiloff rate.

NRC Restriction #7: The limiting ECCS configuration, including minimum versus maximum ECCS, must be determined for each plant or class of plant using this methodology.

Duke/FTI Response: This determination is applicable for LBLOCA analyses and relates specifically to ECCS flow effects on containment pressure. SBLOCA break sizes that would not unchoke before the core is refilled are unaffected by the containment pressure used as a boundary condition. If the SBLOCA analyses were performed with a maximum ECCS flow, there would be less core uncovering with little to no core heatup.

NRC Restriction #8: For the small break model, the hot channel radial peaking factor to be used should correspond to that of the hottest rod in the core, and not to the radial peaking factor of the 12 hottest bundles.

Duke/FTI Response: There are twelve assemblies modeled in the hot bundle and each pin is peaked to the hot pin radial value.

NRC Restriction #9: The constant discharge coefficient model (discharge coefficient = 1.0) referred to as the "High or Low Break Voiding Normalized Value," should be used for all small break analyses. The model which changes the discharge coefficient as a function of void fraction, i.e., the "Intermediate Break Voiding Normalized Value," should not be used unless the transient is analyzed with both discharge models and the intermediate void method produces the more conservative results.

Duke/FTI Response: FTI has verified that a constant discharge coefficient of 1.0 was used for the SBLOCA analyses.

NRC Restriction #10: For a specific application of the FTI small break LOCA methodology, the break size which yields the local maximum PCT must be identified. In light of different possible behaviors of the local maximum, FTI should justify its choice of break sizes in each application to assure that either there is no local maximum or the size yielding the maximum local PCT has been found. Break sizes down to 0.01 ft^2 should be considered.

Duke/FTI Response: For 75 percent power analyses, the 0.07 ft^2 break size gave a local maximum PCT of 1862 °F, where both the 0.075 ft^2 and the 0.065 ft^2 break sizes yielded lower PCTs. The 100 percent power analyses yielded a local maximum PCT of 1369 °F for the 0.15 ft^2 break, with the 0.125 ft^2 and 0.175 ft^2 breaks giving lower PCTs.

Duke's February 9, 1998, submittal provided the analysis results for the spectrum of break sizes for the full power and reduced power cases. The fineness of these break spectrums is sufficient for identifying the limiting break sizes. A break size of 0.01 ft^2 for the full power and 75% full power cases has also been evaluated and is non-limiting.

NRC Restriction #11: B&W-designed plants have internal reactor vessel vent valves (RVVVs) that provide a path for core steam venting directly to the cold legs. The BWNT LOCA evaluation model credits the RVVV steam flow with the loop steam venting for LBLOCA analyses. The possibility exists for a cold leg pump suction to clear during blowdown and then reform during reflood before evaluation model analyses predict average core quench. Since the REFLOOD3B code cannot predict this reformation on the loop seal, FTI is required to run the RELAP5/MOD2-B&W system model until the whole core quench, to confirm that the loop seal does not

reform. This demonstration should be performed at least once for each plant type (raised loop and lowered loop) and be judged applicable for all LBLOCA break sizes.

Duke/FTI Response: This is a LBLOCA specific limitation. The SBLOCA analyses will generally retain a cold leg pump suction loop seal during the PCT prediction.

Question 3: Describe why the analysis for the core flood line break and the HPI line break, performed at 65% power, bounds operation at 75% power, which is permitted by the TS. Additionally, verify that all the analyses performed at 100% and 75% power include 2% margin for instrument uncertainty.

Duke/FTI Response: The core flood line break has been analyzed by FTI at 100 percent power and 65 percent power and the core remains continuously covered and cooled with no core uncovering. The PCT is equal to the initial cladding temperature at break opening time. During the first 10 minutes following Engineered Safeguards actuation, the ECCS systems (one HPI pump feeding two cold legs and one core flood tank) combine to provide core boiloff makeup and establish a core mixture level at ten minutes post-LOCA that is more than 4 ft above the top of the core for the 100 percent power case. The minimum mixture level for the 65 percent power case was several feet higher. For both analyses the one HPI pump can provide 428 gpm of flow to the core at ten minutes. This flow is capable of removing 428gpm*0.1375lbm/s/gpm*(1150 - 88)Btu/lbm = 62,500 Btu/s of decay heat energy via boiloff at 14.7 psia. The decay heat fraction at 10 minutes post-LOCA is 0.0282 times the initial power level. The decay heat power for a 77 percent power case is 0.77 * 2568MWt * 0.0282 * 948Btu/s/MWt = 52,900 Btu/s. The HPI flow exceeds the decay heat boiloff, so there will be at least 4 feet of mixture above the top of the core at 10 minutes post LOCA. Therefore, the CFT line break at 75 percent power need not be analyzed because the other two bounding CFT line break cases and a hand calculation show that the core will not uncover.

The HPI line break has been analyzed by FTI at 100 percent power and 65 percent power and the core remains continuously covered and cooled with no core uncovering. The PCT is equal to the initial cladding temperature at break opening time. The 65 percent power case had a mixture level that was 5 ft above the top of the core at the time of the steam generator blowdown (~1600 seconds). The difference in the integrated decay heat for the first 1600 seconds with a 10 percent power difference is 0.1 * 2568MWt * 948 Btu/s/MWt * 46.859

full power seconds = 11.4E6 Btu. The amount of saturated liquid boiloff required to match this integrated power difference is determined by taking the power divided by h_{fg} or 11.4E6/613 Btu/lbm = 18,600 lbm at 1200 psia. This liquid mass is equivalent to approximately 18,600 lbm *0.02232 ft3/lbm = 415 ft^3 of liquid inventory. The mixture level remaining above the top of the core will supply this needed inventory with little to no core uncovering. Nonetheless, because of the small margin available, FTI analyzed the HPI line break at 77 percent power to confirm these extrapolations. The RELAP5/MOD2 analysis confirmed that the core did not uncover and the PCT of 715 °F was realized near the break opening time.

The 100 percent power and 75 percent power SBLOCA analyses increased the core power by 2 percent to account for the thermal power measurement uncertainty.

Question 4: Please reference the new SBLOCA methodology in the TS basis.

Duke Response: Attachment 2 provides a revision to the proposed TS bases that references the SBLOCA methodology. In addition, Attachment 2 clarifies the TS bases with respect to passive failures to reflect the information provided in response to Question 5.

Question 5: This license amendment request redefines the function of the HPI system. The submittal and the response to the staff questions states that the HPI system is not required for long term cooling because at Oconee "long term cooling is defined as decay heat removal via the LPI system." This does not assure long term decay heat removal following a small break LOCA in which the RCS pressure remains above the shutoff head of the LPI pumps. As a result, please retract the statements declaring the HPI system is not required for long term cooling and verify all requirements for long term cooling continue to be met.

Response: Duke recognizes that long-term cooling is typically considered to begin when decay heat is removed via sump recirculation. Duke's analyses also clearly conclude that operation of the HPI System during the recirculation mode is required for certain small break LOCAs. This mode of operation is referred to as the HPI piggyback mode. Thus, the HPI System does assure adequate decay heat removal following a small break LOCA in which the RCS pressure remains above the shutoff head of the LPI pumps. It was never intended in previous submittals to imply that this mode of operation was not necessary. However, the objective

of Duke's previous submittals has been to clarify that single passive failures of the HPI System were not contemplated in the system design. The fact that the Technical Specifications require operation with the suction header cross-connected supports this position.

Duke has reviewed the available regulatory guidance regarding single failures. Duke's UFSAR states that the ECCS is designed to accommodate a single active failure in the short term or a single active or passive failure in the long-term. As with the regulations, long-term cooling is not explicitly defined in Oconee's licensing basis.

10 CFR 50, Appendix A, states, in part:

> "The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are still under development."

The principal design criteria for Oconee, given in Section 3 of the UFSAR, were developed in consideration of the seventy General Design Criteria for Nuclear Power Plant Construcution Permits proposed by the AEC in a proposed rule-making published for 10 CFR Part 50 in the Federal Register of July 11, 1967. At this time, as is evident in the current statements in Appendix A of 10 CFR 50, the requirements for consideration of passive failures were not fully developed. Based on recent discussions with the staff, SECY 77-439 was identified as providing additional guidance regarding application of the single failure criterion. A review of SECY 94-084, Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs, also references SECY 77-439 as providing the staff's guidance on passive failures.

SECY 77-439 states:

> "In the study of passive failures it is current practice to assume fluid leakage owing to gross failure of a pump or valve seal during the long-term cooling mode following a LOCA (24 hours or greater after the event) but not pipe breaks. No other passive failures are required to be assumed because it is judged that compounding of probabilities associated with other types of passive failures, following the pipe break associated with a LOCA, results in probabilities sufficiently small that they can be reasonably discounted without

> substantially affecting overall systems reliability."

The engineering judgment employed by the staff in 1977 is supported by the findings of the recently completed HPI System Reliability Study. This study concluded that operation with the suction and discharge headers cross-connected has a negligible impact on reliability due to the low likelihood of passive failures as compared to active failures.

SECY 77-439 further states:

> "In some licensing review areas, the staff does impose a passive failure in addition to the initiating event, while in others it does not. As previously mentioned, an example of the application of a passive failure requirement is the approach to long-term recovery subsequent to a loss-of-coolant accident. Applicants are required to consider degradation of a pump or valve seal and resulting leakages in addition to the initiating failure (LOCA). The rationale for applying this type of failure is a recognition of the relatively extended periods of required operation of systems that are expected to be on a standby status throughout the plant life. The likelihood of accelerated wear of such components as pump and valve seals would be increased after the adverse conditions following a LOCA. Extended operation during the long term (up to months) requires that these types of failures be considered in designing the plant."

Thus, it is clear that the intent of considering passive failures is to address long-term operation (up to months) of systems required to remove decay heat. Operation of the HPI System in the piggyback mode is required for some small break LOCAs. However, this is not an extended or indefinite mode of operation. Operation of the HPI System in the piggyback or recirculation mode may last from a few hours to a day or so. As the decay heat load drops, the RCS will depressurize to the point where the LPI pumps are used for extended (up to months) decay heat removal. Thus, for this reason, passive failures in the HPI System were not considered in the design and licensing of the plant.

Oconee has also reviewed BAW-10103A, Rev. 3, ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, regarding long term cooling requirements. Chapter 10 of BAW-10103A, Rev. 3, addresses the establishment of long-term cooling in a

generic manner for the lowered-loop B&W class plants. This topical report states:

> "The exact duration of long-term cooling will vary depending on several factors, including the size of the break and the radiation release. A realistic assessment of the duration of the long-term cooling period for the worst case is approximately one month. As a maximum, assuming the worst-case calculational results and corresponding radiation releases to the building, long-term cooling may be required for periods on the order of one year."

BAW-10103A, Rev. 3, is consistent with SECY 77-439 in that it characterizes the duration of long-term cooling as several months. This topical report does not describe the HPI piggyback mode as a long-term means of core cooling. The topical report specifically refers to long-term cooling via operation of the LPI System taking suction from the reactor building sump. The only mention of cooling by the HPI System is as an alternative for a short duration while maintenance is being performed on normal plant equipment.

It should be noted that the HPI System does contain isolation valves that would allow the operators to address certain passive failures. In addition, Duke's reliability analyses have concluded that postulated passive failures have a negligible impact on system reliability. In summary, based on the limited period of time the HPI System may operate in the piggyback mode, Duke concludes that passive failures do not need to be postulated for this system in the deterministic design basis analyses for the plant.

Question 6: Operation with the HPI suction cross-connected has been approved by the NRC; however, it is not clear if operation with the HPI discharge cross-connected has been approved. Please describe if allowing the HPI discharge to be cross connected has been approved. Given that running one pump to all four injection nozzles poses pump runout problems, why is cross connecting the discharge acceptable (in the three pump configuration)? If two of the three pumps fail randomly will the third fail as a result of pump runout? The submittal states that the cross-connected configuration increases margin, however, if the isolated configuration meets all the acceptance criteria why is the additional margin necessary? It would appear that maintaining train independence whenever possible is advantageous.

Duke Response:

Operation with the discharge header cross-connected has been approved by the staff in the context that the current Technical Specifications and the accident analyses require cross-connecting trains following a small break LOCA to achieve HPI flow through both trains. The Technical Specifications also require the suction header to be cross-connected. Interim guidance regarding opening HP-116 was implemented in 1978 prior to implementation of the HP-409/HP-410 modification to require cross connecting trains to achieve HPI flow through both trains. However, as described in the March 31, 1997, Duke submittal, the current Technical Specifications create confusion when assessing operation with HP-116 open. This is because the Technical Specifications require two independent trains, yet at the same time require operation with the suction header cross-connected.

Duke's proposed Technical Specification changes explicitly address requirements for the discharge header. The discharge header may be isolated or cross-connected at full power with three HPI pumps operable. If only two HPI pumps are operable, the discharge header must be isolated between the two operable pumps. This guidance assures that, assuming a single failure, one pump will not inject down both headers. Duke believes it is beneficial for our operators to have these requirements explicitly addressed in the Technical Specifications. From a safety standpoint, operation with the discharge header cross-connected at full power maximizes the initial HPI flow. Assuming the worst case single failure will result in a minimum of two HPI pumps initially injecting down one train. Thus, from a deterministic perspective, this approach maximizes injection and improves safety margins.

Duke has evaluated the impact of the staff's beyond design basis scenario where two pumps fail during an accident from full power and only one HPI pump operates. If the discharge header is cross-connected, this HPI pump would be susceptible to runout at RCS pressures less than about 600 psig. The core flood tanks inject at 600 psig. Thus, if the scenario is a larger small break LOCA that results in relatively rapid depressurization of the RCS, it is likely that the core flood tanks will provide adequate injection to assure core cooling. If the depressurization is slower, time is available for the operators to throttle HPI flow to prevent runout. Thus, this beyond design basis concern is not considered significant from a risk perspective. In addition, the following assessment addresses system

reliability with respect to the suction and discharge headers cross-connected.

Suction Cross Connect

The HPI pump train suction is isolated by closing valve HP-99 or HP-100. In this configuration, failures due to the LDST interface would not fail HPI Pump C. However, the system would not be single failure proof for the limiting break locations. The Oconee Nuclear Station HPI Reliability Study indicated that with the suction trains isolated, the failure probability of HPI during the injection mode following a HPI line break or RCP discharge break LOCA increased from 3.6E-3 to 1.1E-2. In this configuration, the following single failures could result in inadequate flow reaching the core:

1. Failure of Borated Water Storage Tank (BWST) suction valve HP-24 to open
2. A failure occurs in the power supply to HP-24 prior to its opening.

For the recirculation mode of core cooling, the cross-connect has less of an impact. This is because other failure modes (namely, failure of Letdown Storage Tank (LDST) check valve to close, failure of operators to initiate high pressure recirculation, and common cause failure to open of LPI to HPI supply valves) dominate the system failure probability. Even with the HP-97 failure mode removed from the results, the change in recirculation mode injection reliability is not significantly affected by availability of the cross-connect. However, due to its significance during the injection mode of core cooling, the cross-connect capability is seen to enhance the overall reliability of the HPI system in mitigating accidents, as stated in the HPI System Reliability Study.

Discharge Cross Connect

On the discharge side, valves HP-409 and HP-410 provide the cross-connect feature. Following an HPI line break or reactor coolant pump discharge break, one of these valves may be opened, if needed, to decrease HPI flow to the break and increase flow to the reactor vessel via the intact injection path. Not having this cross-connect available significantly increases the failure probability of HPI during the injection mode following a HPI line break or RCP discharge break LOCA, from 3.6E-3 to 2.0E-2. For example, the following accident sequences would no longer be

recoverable and would result in inadequate flow reaching the core:

1. HPI Train A emergency injection valve HP-26 fails to open
2. HPI Pump C fails to start.

As is true on the suction side, for the recirculation mode of core cooling, the cross-connect capability does not measurably impact system reliability. However, due to its significance during the injection mode of core cooling, the cross-connect capability is again seen to enhance the overall reliability of the HPI system in mitigating accidents.

The above analysis addresses the reliability aspects of cross-connecting the HPI pump discharge via HP-409 and HP-410. The HPI System Reliability Study also addressed operation with the pump discharge header cross-connected. This would typically involve operation with valve HP-116 open. The study concluded that this mode of operation did not adversely impact the reliability of the HPI System.

In conclusion, overall system reliability is enhanced by maintaining the HPI train suction cross-connect valves normally open, and retaining the ability to cross-connect the discharge headers.

NRC Question 7: The Oconee HPI reliability study that was submitted to the NRC in December of 1997 concluded that the HPI system can reliably perform its function with the three pump configuration. The proposed Oconee TS will allow indefinite operation with only two operable HPI pumps at reduced power. Can the system reliably perform its function in the two pump configuration allowed by TS?

Duke Response: In responding to this question, it is helpful to revisit the HPI system success criteria. The HPI System Reliability Study indicated that successful mitigation of the design basis HPI line break during the injection mode of core cooling requires two HPI pumps injecting through all four cold leg nozzles. Recent analysis by Framatone currently under review by the NRC indicates that one HPI pump is sufficient to cool the core at a reduced power level (up to ~ 77% of full power).

The configuration of the HPI system is different depending on which HPI pump is assumed to be unavailable. If HPI Pump A or B is unavailable, the system is left in its normal configuration, with cross-connect valve HP-116 closed to

separate the trains. HPI Pump B or A is aligned to the 'A' cold legs and the HPI Pump C is aligned to the 'B' cold legs. If HPI Pump C is unavailable, the HPI Pump A and B discharge paths are similarly isolated from each other by closing HP-115. In addition, HP-116 would be opened and HP-27 closed, allowing HPI Pump B to supply the 'B' cold legs on an ES signal. In the latter configuration, both pumps are required to operate, with HPI Pump A supplying normal injection and HPI Pump B supplying the RCP seals. Both of these system configurations are analyzed below.

HPI Pump A or B Unavailable

With HPI Pump A or B out of service, the impact of two pump operation may be estimated by deleting the HPI Pump B logic from the design basis injection and recirculation mode logic of the HPI Study model, modifying the existing logic to require only one HPI pump injecting through two cold leg nozzles for system success, and resolving for the top gates of interest. (For the injection mode solve, the top gate is named HCH15INJ. For the recirculation mode solve, the top gate is HPR001.) Table 1 lists the top ten cut sets for the injection mode solve. In this configuration, the failure probability is found to decrease slightly from the base case value of 3.6E-3 to 2.6E-3. The decrease occurs primarily because the number of running pumps required (one) is less than required for the base case (two). Thus, the reliability for providing the high pressure injection function in this configuration remains high (>99%).

A resolve of the recirculation top gate indicates that the base case failure probability of 1.1E-2 is not increased even though only two pumps are assumed to be available. This is because pump train failures are not dominant failure modes.

HPI Pump C Unavailable

With HPI Pump C out of service, the impact of two pump operation may be estimated by deleting the HPI Pump C logic and the HPI Pump B start failure mode from the design basis injection and recirculation mode logic of the HPI Study model, modeling HP-115 and HP-27 as normally closed and HP-116 as normally open, modifying the existing logic to require only one HPI pump injecting through two cold leg nozzles for system success, and resolving for the top gates of interest. The top ten cut sets for the injection mode solve are the same as those shown in table 1 below. In this configuration, the failure probability is found to decrease from the base case value of 3.6E-3 to 2.6E-3, as is the case

if HPI Pump B is assumed to be unavailable. Again, the decrease occurs primarily because the number of running pumps required (one) is less than required for the base case (two), so that the reliability for providing the high pressure injection function remains high (>99%).

For the recirculation mode of core cooling, the results indicate that the base case failure probability of 1.1E-2 is not increased even though only two pumps are assumed to be available. This is because pump train failures are not dominant failure modes.

Impact on Core Damage Frequency

For the core damage frequency determination, the model is resolved assuming that only two HPI pumps are available. With HPI Pump A, B, or C unavailable, the core damage frequency shows an increase to 4.6E-5 (excluding seismic), compared to the base case value of 4.3E-5. This result is slightly higher than the base case result primarily because there is one HPI pump available to supply the RCP seals instead of two, leading to an increase in the seal LOCA CDF.

A final point to consider is that two pump operation is likely to be an infrequent mode of operation. The length of time in this mode of operation is constrained by the outage scheduling constraints.

In conclusion, operation with only two HPI pumps available continues to assure high reliability of the HPI system. In terms of core damage frequency, the impact of having one less HPI pump available is seen to be not significant. There is no negative impact on the LOCA sequences. There is a small increase associated with the RCP seal LOCA frequency. However, the assessment assumes that the third HPI pump is unavailable for an entire year. In reality, it is expected that the duration of two pump operation would be significantly less than a year. For example, if the HPI system were configured in two pump operation for a month, the increase in CDF is estimated to be very small (less than 1E-6).

Table 1

Top 10 Cut Sets For HPI System Failure During Injection Mode
(1 Pump Required, HPI Pumps A and C Available)

Cut Set	Event Name	Event Description	Event Probability	Cut Set Probability
1	H3BBREAK HHP0486CVT	HPI Line Break Occurs on 3B Side Check Valve 3HP-486 Transfers Closed	5.00E-1 8.94E-4	4.47E-4
2	H3BBREAK HHP0487CVT	HPI Line Break Occurs on 3B Side Check Valve 3HP-487 Transfers Closed	5.00E-1 8.94E-4	4.47E-4
3	HNMOPRSDHE HNMOPRSLHE HPI0095VVT	Operators Overcharge the LDST Operators Fail to Recognize Pressure Instrument Failure Instrument Root valve HPIIV0095 Transfers Shut	1 1 2.46E-4	2.46E-4
4	HLDXMTRCOM HNMOPLVDHE	Common Cause Failure of LDST Level Transmitters Operators Fail to Recognize Level Instrument Failure	2.09E-4 1	2.09E-4
5	HLDPXTRCOM HNMOPRSDHE HNMOPRSLHE	Common Cause Failure of LDST Pressure Transmitters Operators Overcharge the LDST Operators Fail to Recognize Pressure Instrument Failure	1.34E-4 1 1	1.34E-4
6	HHPSVLVCOM	Common Cause Failure Of MOVs 3HP-24 And 3HP-25 To Open On Demand	1.24E-4	1.24E-4
7	H3ABREAK HHP0152VVT	HPI Line Break Occurs on 3A Side Manual Valve 3HP-152 Transfers Closed	5.00E-1 2.43E-4	1.22E-4
8	H3ABREAK HHP0153VVT	HPI Line Break Occurs on 3A Side Manual Valve 3HP-153 Transfers Closed	5.00E-1 2.43E-4	1.22E-4
9	H3BBREAK HHP0126VVT	HPI Line Break Occurs on 3B Side Manual Valve 3HP-126 Transfers Closed	5.00E-1 2.43E-4	1.22E-4
10	H3BBREAK HHP0127VVT	HPI Line Break Occurs on 3B Side Manual Valve 3HP-127 Transfers Closed	5.00E-1 2.43E-4	1.22E-4