September 29, 2003

Mr. Ronald A. Jones
Vice President, Oconee Site
Duke Energy Corporation
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB8083, MB8084, AND MB8085)

Dear Mr. Jones:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 335, 335, and 336 to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated March 20, 2003, as supplemented by letters dated July 22 and August 5, 2003.

The amendments revise the TS and the licensing basis in the Updated Final Safety Analysis Report (UFSAR) to support installation of a passive low-pressure injection (LPI) cross connect inside containment. The changes to the TS add requirements for the passive LPI cross connect and eliminate requirements associated with the capability to cross connect, by manual action, the trains outside containment. The changes to the UFSAR revise the licensing basis for a portion of the core flood and LPI/Decay Heat Removal (DHR) piping to allow the exclusion of dynamic effects associated with postulated pipe rupture of that piping by application of leak-before-break technology for Unit 1. The changes to the UFSAR also revise the licensing basis for selected portions of the LPI/DHR piping to adopt design requirements of Standard Review Plan Section 3.6.2, Branch Technical Position MEB 3-1.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/
Leonard N. Olshan, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosures:
1. Amendment No. 335 to DPR-38
2. Amendment No. 335 to DPR-47
3. Amendment No. 336 to DPR-55
4. Safety Evaluation

cc w/encls: See next page

September 29, 2003

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Vice President, Oconee Site
Duke Energy Corporation
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cc w/encls: See next page
DOCUMENT NAME:ML032721590 *No major changes to SE

OFFICE	PDII-1/PM	PDII-1/LA	OGC	EMEB/SC	SPLB/SC	PDII-1/SC
NAME	LOlshan	CHawes	AKermandez	KManoly*	DSolorio*	JNakoski
DATE	9/9/03	9/9/03	9/24/03	7/23/2003	8/20/2003	9/29/03

OFFICE	EMCB/SC	SRXB/SC	IROB/SC
NAME	TChan*	JUhle*	KKavanagh for TBoyce
DATE	3/23/03	9/26/03	9/26/03

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DUKE ENERGY CORPORATION

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 335
Renewed License No. DPR-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Renewed Facility Operating License No. DPR-38 filed by the Duke Energy Corporation (the licensee) dated March 20, 2003, as supplemented by letters dated July 22 and August 5, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended (1) to authorize revision to the Updated Final Safety Report as set forth in the application for amendment dated March 20, 2003, as supplemented by letters dated July 22 and August 5, 2003; and (2) by page changes to the Technical Specifications as indicated in the attachment to this license amendment; and Paragraph 3.B of Renewed Facility Operating License No. DPR-38 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 335, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 29, 2003

DUKE ENERGY CORPORATION

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 335
Renewed License No. DPR-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. DPR-47 filed by the Duke Energy Corporation (the licensee) dated March 20, 2003, as supplemented by letters dated July 22 and August 5, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended (1) to authorize revision to the Updated Final Safety Analysis Report as set forth in the application for amendment dated March 20, 2003, as supplemented by letters dated July 22 and August 5, 2003; and (2) by page changes to the Technical Specifications as indicated in the attachment to this license amendment; and Paragraph 3.B of Renewed Facility Operating License No. DPR-47 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 335, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 29, 2003

DUKE ENERGY CORPORATION

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 336
Renewed License No. DPR-55

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Renewed Facility Operating License No. DPR-55 filed by the Duke Energy Corporation (the licensee) dated March 20, 2003, as supplemented by letters dated July 22 and August 5, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended (1) to authorize revision to the Updated Final Safety Analysis Report as set forth in the application for amendment dated March 20, 2003, as supplemented by letters dated July 22 and August 5, 2003; and (2) by page changes to the Technical Specifications as indicated in the attachment to this license amendment; and Paragraph 3.B of Renewed Facility Operating License No. DPR-55 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 336, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 29, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 335

RENEWED FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 335

RENEWED FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 336

RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Appendix A Technical Specifications and associated Bases with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3.4.14-1	3.4.14-1
3.5.3-1	3.5.3-1
3.5.3-2	3.5.3-2
3.5.3-3	3.5.3-3
B 3.3.8-1 through -19	B 3.3.8-1 through -19
B 3.4.14-1 through -5	B 3.4.14-1 through -6
B 3.5.3-1 through -9	B 3.5.3-1 through -10

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 335 TO RENEWED FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 335 TO RENEWED FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 336 TO RENEWED FACILITY OPERATING LICENSE DPR-55

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated March 20, 2003, as supplemented by letters dated July 22 and August 5, 2003, Duke Energy Corporation (the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3, Technical Specifications (TS) and changes to the licensing basis in the Updated Final Safety Analysis Report (UFSAR) to support installation of a passive low-pressure injection (LPI) cross connect inside containment. The requested changes to the TS would add requirements for the passive LPI cross connect and eliminate requirements associated with the capability to cross connect, by manual action, the trains outside containment. The requested changes to the UFSAR would revise the licensing basis for a portion of the core flood (CF) and LPI/Decay Heat Removal (DHR) piping to allow the exclusion of dynamic effects associated with a postulated pipe rupture of that piping by application of leak-before-break (LBB) technology for Unit 1. The requested changes to the UFSAR would also revise the licensing basis for selected portions of the LPI/DHR piping to adopt the design requirements of Standard Review Plan (SRP) 3.6.2 Branch Technical Position (BTP) MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment." The supplements dated July 22 and August 5, 2003, provided clarifying information that did not change the scope of the March 20, 2003, application nor the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

The emergeny core cooling system (ECCS) includes two redundant LPI trains. The LPI system provides emergency core cooling injection from the borated water storage tank to the primary system during a loss-of-coolant accident (LOCA). It also circulates water between the primary system and the reactor building sump to provide long-term cooling. In MODES 1, 2, and 3 both trains of LPI must be operable. This ensures that 100 percent of the core-cooling requirements can be provided even in the event of a single active failure. Only one train is required for MODE 4 without a single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

With the current LPI piping configuration, operators must manually open LPI discharge header valves to mitigate certain single failures, including failures during a postulated CF line break (CFLB). The LPI passive cross-connect modification will ensure that each train is capable of delivering flow to the core without significant operator actions during an event considering an LPI component or train failure. This modification will cross-connect the LPI trains at a location inside containment downstream of check valves LP-47 and LP-48. It will also add two new check valves (LP-176 and LP-177) downstream of the cross-connect piping to prevent a blowdown of both CF tanks through the break of a CF line. Flow restricting devices will be installed to limit LPI flow under low back-pressure conditions to mitigate LPI pump net positive suction head availability concerns and also to divert sufficient flow to the intact header during a postulated CFLB. These devices are also sized to ensure sufficient flow in the event of a LBLOCA and sufficient cool-down capacity during a unit shutdown condition.

Requirements regarding exclusion of the dynamic effects of pipe rupture from the licensing basis of a nuclear power plant are addressed in Title 10 of the *Code of Federal Regulations* (10CFR) Part 50, Appendix A, General Design Criteria (GDC) 4:

> Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

The licensee’s submittal and supplements contain the LBB analysis mentioned in GDC 4 to support the exclusion of the dynamic effects of pipe rupture from the Unit 1 licensing basis for segments of the CF and LPI/DHR piping systems. The NRC staff used draft SRP 3.6.3, "Leak-Before-Break Evaluation Procedures," (August 28, 1987), and NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," (November 1984) to conduct the LBB review. LBB evaluations also rely in part on the capability of a facility’s reactor coolant system (RCS) leakage detection system. NRC Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary (RCPB) Leakage Detection Systems," (1973), provides staff guidance on the design and evaluation of RCS leakage detection systems.

The licensee referenced the requirements of GDC 4 of Appendix A to 10 CFR Part 50 for the treatment of postulated pipe ruptures in nuclear power units. GDC 4 requires that structures, systems and components important to safety be appropriately protected against dynamic effects associated with postulated pipe ruptures. SRP Section 3.6.2 BTP MEB 3-1 provides NRC staff guidance regarding the treatment of postulated pipe ruptures that complies with the requirements of GDC 4.

The licensee has proposed to use the guidance in BTP MEB 3-1 for the treatment of postulated pipe breaks for the LPI/DHR piping from the containment penetrations to valves LP-176 and LP-177 including the cross connect piping between the two trains. The licensee proposed to add a description of the criteria for the treatment of pipe breaks in Section 3.6.1.2.1 of the UFSAR. The proposed UFSAR description is contained in Attachment 3 of the licensee's March 20, 2003, submittal.

The licensee evaluated the leakage detection systems using the guidelines of RG 1.45 that a leakage of 1 gallon per minute (gpm) can be detected in 1 hour.

The licensee is using the guidance from the NRC-approved NUREG-1430, Revision 2, "Standard Technical Specifications B&W Plants," and the guidance from NUREG-800, "Standard Review Plan," as appropriate. The licensee must submit an analysis to demonstrate that the emergency core coolant systems (ECCS), including the modifications to the LPI system, continue to satisfy the requirements of 10 CFR 50.46 and the acceptance criteria of 10CFR Part 50, Appendix K, "ECCS Evaluation Models." The LPI system with the proposed modification must be in compliance with GDC 34, "Residual heat removal," and GDC 35, "Emergency core cooling."

3.0 TECHNICAL EVALUATION

3.1 Use of Leak-Before-Break Technology for Unit 1

This section of this Safety Evaluation (SE) applies to Unit 1 only and describes: (1) the scope (i.e., piping segments evaluated) of the licensee's LBB evaluations, (2) the analysis methodology used by the licensee in its LBB evaluation, and (3) the results of the licensee's analysis and its conclusions regarding the application of LBB to the subject piping segments.

3.1.1 Scope of the Licensee's LBB Evaluation

In the submittal, the licensee clearly defined the scope of the high energy piping within the Unit 1 CF and LPI/DHR piping system for which it sought to apply LBB. The piping includes a 14-inch CF line connecting the CF tank with the reactor pressure vessel (RPV) and two 10-inch LPI/DHR lines that connect two new check valves, 1LP-176 and 1LP-177, to two 90-degree tee fittings of the CF line. The Alloy 82/182 weld between the RPV CF nozzle and the safe end, which is at an operating temperature of 557 °F, is excluded from the LBB application due to the concern of primary water stress corrosion cracking (PWSCC). However, the Alloy 82/182 welds at the two CF tanks are included, since PWSCC is of no concern at these locations due to the low operating temperature of 125 °F. The added passive LPI cross connect piping is also excluded from the LBB application.

The 14-inch CF line was constructed from wrought austenitic A-358, Type 304 (schedule 30), A-376, Type 304 (schedule 140), and A-376, Type 316 (schedule 140) stainless steel. The corresponding wall thicknesses for the piping are 0.375-inch, 1.25-inch, and 1.25-inch. The 10-inch LPI/DHR line was constructed from wrought austenitic A-376, Type 304 (schedule 140) stainless steel, and the corresponding wall thicknesses for the piping is 1 inch. The piping welds were fabricated using a gas tungsten arc welding (GTAW) process and a shielded metal arc welding (SMAW) process. No cast austenitic stainless steel (CASS) was used to construct the analyzed piping segments. However, Inconel Alloy 82/182 material was used in the

fabrication of piping welds at both CF tanks, which is at a low operating temperature of 125 °F. These welds are included in the analyzed piping segments.

3.1.2 Licensee's LBB Evaluation Methodology

The licensee's LBB evaluation methodology is summarized in the report prepared by Framatome Advance Nuclear Power (Framatome) entitled, "Leak-Before-Break Analysis of the Core Flood and Low Pressure Injection/Decay Heat Removal Piping Systems of Oconee Unit 1," (Framatome report) and additional information was provided in the supplements. The following description briefly addresses general aspects of the licensee's methodology that are consistent with draft SRP 3.6.3 and NUREG-1061, Volume 3. Specific aspects of the licensee's methodology, which are not specified in draft SRP 3.6.3 and NUREG-1061, Volume 3, are discussed in additional detail.

Consistent with the guidance provided in draft SRP 3.6.3 and NUREG-1061, Volume 3, the licensee first established that no active degradation mechanisms (flow accelerated corrosion, stress corrosion cracking, fatigue) were expected in the subject piping segments. Further, the licensee established that no unanalyzable loading events (water hammer) would be expected to occur in the subject piping segments. The evaluation of these topics was provided in Section 2.2 of the Framatome report.

Next, the licensee established material property parameters, operating conditions, and piping moments and membrane stresses for use in its LBB analyses. The material property parameters used in the licensee's analysis were given in Section 3.3 of the Framatome report, where both the tensile and fracture toughness (J-R) properties of the base metals and GTAW and SMAW welds were addressed. Based on consideration of the highest stress locations coincident with the worst material properties, the licensee identified two locations for LBB analysis: the CF piping adjacent to the CF tank nozzle and the RPV CF nozzle safe end to the CF piping. Materials applicable to these locations are Type 304 stainless steel and GTAW welds for the CF piping adjacent to the CF tank nozzle and Type 316 stainless steel and SMAW welds for the RPV CF nozzle safe end to the CF piping. The tensile and J-R properties for materials at these critical locations were obtained from the experimental work documented in the Electric Power Research Institute (EPRI) Report NP-4768, "Toughness of Austenitic Stainless Steel Pipe Welds." These properties are summarized in the Framatome report in Tables 3-5 and 3-6, with corresponding J-R curves shown in Figures 3-2 to 3-6.

The LBB analysis consists of a leakage flaw size calculation using loading associated with normal operating conditions and a critical flaw size calculation using loading associated with faulted conditions. The pipe loadings associated with normal operating conditions are axial forces and moments due to pressure, dead weight, and thermal expansion; and the pipe loadings associated with faulted conditions are axial forces and moments of normal operating conditions in conjunction with safe shutdown earthquake and seismic anchor motion loads. In the licensee's critical flaw size calculation, the absolute sum method was used to add the individual axial forces and moments into the combined axial forces and moments.

Based on the material property, operating condition, and loading information noted above, the licensee implemented its LBB evaluation. This process required, first, determination of the leakage flaw size; (i.e., the length of a through-wall circumferential flaw at the two critical locations in the analyzed piping segments that would generate a leakage rate of 10 gpm,

10 times the leakage detection capability of 1 gpm at Oconee Unit 1). This determination was based on the normal operating moments and stresses and the crack morphology parameters (surface roughness and number of turns) associated with fatigue type of cracks. The licensee then determined the critical flaw sizes for the critical locations that would be predicted to lead to piping failure under the faulted loading conditions. These critical flaw size calculations were performed using an elastic-plastic fracture mechanics (EPFM) technique. The last step in the licensee's evaluation process was the calculation of ratios (margins) between the critical flaw size and the leakage flaw size for the two critical locations. This relationship between the critical flaw size and leakage flaw size results from the guidance in draft SRP 3.6.3 and NUREG-1061, Volume 3, that specifies that a margin of two should be maintained in an acceptable LBB evaluation. A similar process was repeated for assumed axial flaws.

Several additional considerations that were raised by the NRC staff regarding the impact of some recent generic material information on the licensee's EPFM analysis, and the assessment of a stress corrosion cracking (SCC) type of degradation on LBB due to the implications of the V. C. Summer PWSCC experience, were addressed by the licensee in the supplements. They are (a) the variability of strain-hardening parameters, (b) thermal aging of stainless steel weld materials, (c) exclusion of fatigue crack growth analysis, (d) the validity of the J-estimation scheme, and (e) the assessment of the implication of PWSCC by performing a sensitivity study using crack morphology parameters characteristic of transgranular SCC.

The licensee addressed Item (a) by performing a sensitivity study using a wide range of strain hardening parameters (the Ramberg-Osgood parameters) in the LBB analysis. The results indicate that the variability of strain-hardening parameters has only minor effect on the LBB margins. Nevertheless, the licensee revised its results using the Ramberg-Osgood parameters that produced the most conservative results. The licensee addressed Item (b) by using the lower-bound, unaged J-R curve for SMAW, submerged arc welds (SAW), and GTAW from NUREG/CR-6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds," in its revised analysis, and it also showed a minor impact. The licensee responded to Item (c) by conducting a review of the detailed stress analysis to determine the effects of the transients at the critical location close to the RPV. The stress analysis showed that this location experiences negligible pressure and thermal transient stresses due to the only significant transient, the check valve test transients (240 cooldown cycles). Therefore, an explicit fatigue crack growth analysis is not necessary. The licensee responded to Item (d) by performing a comparative study using the original General Electric (GE)/EPRI J-integral estimation scheme in the flaw stability analysis. The results indicate that using the original GE/EPRI J-integral estimation scheme moderately decreases the flaw-size margins. Finally, the licensee addressed Item (e) by performing a sensitivity study using a wide range of crack morphology parameters characteristic of transgranular SCC based on information in NUREG/CR-6443, "Deterministic and Probabilistic Evaluations for Uncertainty in Pipe Fracture Parameters in Leak-Before-Break and In-Service Flaw Evaluations." In the analyses, the licensee maintained a margin of ten for the leakage estimation and studied the reduction of margins on flaw sizes for the assumed cases where crack morphology parameters characteristic of transgranular SCC were used.

3.1.3 Results/Conclusions from the Licensee's LBB Analysis

For circumferential flaws, the results of the licensee's LBB analysis for all critical locations in the 14-inch CF line and the two 10-inch LPI/DHR lines for Unit 1 is given in Table 2 of the July 22,

2003, letter. These results indicate that using the unaged lower-bound J-R curve for welds from NUREG/CR-6428 reduced the margin for SMAW welds from 2.8 to 2.7 and using the original GE/EPRI J-integral estimation scheme in the flaw stability analysis further reduced the margin from 2.7 to 2.4. Given the way in which the licensee's analysis was conducted (as noted in Section 3.1.2 of this SE), an acceptable LBB analysis result would be achieved if, for each critical location, the margin on flaw sizes is greater than two. This is what the margins in Table 2 show for all materials at the two critical locations. The corresponding results of the licensee's LBB analysis for axial flaws are listed in Table 7-2 of the March 20, 2003, submittal. Revision of Table 7-2 addressing staff concerns is not necessary because the circumferential flaws are more limiting.

Finally, as requested by the NRC staff, the licensee presented the results of its leakage rate sensitivity study in the August 5, 2003, supplement. For the sensitivity analysis, the licensee focused on the limiting SMAW weld. As noted in Section 3.1.2 above, the licensee developed a number of cases using different combinations of flaw morphology parameters consistent with both intergranular and transgranular SCC and determined the margins on flaw sizes for them. In this study, the margins for the leakage rate calculations were kept at ten for all cases. Based on the range of flaw morphology parameters, the licensee concluded that by maintaining this factor of ten for the leakage rate as required by the draft SRP 3.6.3 and NUREG-1061, Volume 3, there is still margin left for the ratios on flaw sizes for all cases.

3.1.4 NRC Staff Evaluation

3.1.4.1 Scope of the Licensee's LBB Evaluation

The NRC staff reviewed the scope of the licensee's LBB evaluation and concluded that the licensee adequately defined the analyzable segments of the piping system (as given in Section 3.1.1 of this SE) for which it sought LBB approval. Regarding the concern of PWSCC, the NRC staff concluded that the licensee 's exclusion of the Alloy 82/182 weld between the RPV CF nozzle and the safe end, which is at an operating temperature of 557 °F, from the LBB application is a key step to make the proposed lines an acceptable candidate for LBB approval at this time. The NRC staff also agrees with the licensee that inclusion of Alloy 82/182 welds at the two CF tanks, which is at an operating temperature of 125 °F, in the LBB application poses no concern. This conclusion is based on the test data in EPRI report, "Crack Growth of Alloy 182 Weld Metal in PWR Environments (PWRMRP-21)."

Since no CASS piping, elbows, or safe ends were present in the subject piping segments for LBB application, the NRC staff agrees with the licensee's conclusion that the selected materials being analyzed at the critical locations would be limiting. In addition, the NRC staff reviewed the tensile and fracture toughness material property parameters provided in the licensee's analysis for aged SMAW welds. The NRC staff concludes that the material property parameters used by the licensee were consistent with those used by the NRC staff for independent analyses in other recent LBB applications.

3.1.4.2 Licensee's LBB Evaluation Methodology

The NRC staff has reviewed the licensee's LBB evaluation methodology summarized in the Framatome report and additional information provided in the supplements, and the NRC staff has determined that the licensee's LBB methodology is in accordance with draft SRP 3.6.3 and

NUREG-1061, Volume 3. The qualitative evaluation of potential degradation mechanisms (corrosion, water hammer, thermal stratification, erosion, and creep) is consistent with plant-specific and industry service data and is acceptable to the staff. The leakage flaw size and the critical size calculations are based on (a) EPFM, which reflects the fracture phenomenon of the ductile piping materials; (b) loadings with adequate summation method; and (c) a factor of ten for the leakage estimation and a factor of two for the flaw-size margin. Therefore, the NRC staff considers the leakage flaw size and critical size calculations appropriate.

The NRC staff has also reviewed the licensee's response to the additional considerations regarding some recent generic material information related to the EPFM analysis and the potential SCC type of degradation. The tensile and J-R properties for materials at the critical locations are from the experimental work documented in EPRI Report NP-4768. Due to lack of actual plant-specific test data for these materials, the NRC staff considers that using the EPRI properties in the licensee's LBB analysis is appropriate only if the resulting LBB margins are large enough to account for uncertainties in these properties. To respond to this concern, the licensee performed a sensitivity study using a wide range of Ramberg-Osgood parameters in its LBB analysis. The results in the July 22, 2003, supplement indicate that the variability of Ramberg-Osgood parameters has only minor effect on the LBB margins. Based on this, the NRC staff determined that using generic tensile property is appropriate. For the J-R curve, the NRC staff has been using the lower-bound, unaged curve from NUREG/CR-6428 for SMAW, SAW, and GTAW welds as a proper reference toughness property in other recent LBB applications. The licensee's revised LBB analysis using this J-R curve showed that the margin for the limiting SMAW weld only decreased from 2.8 to 2.7, indicating that the J-R curve used in licensee's original LBB analysis for the SMAW weld was close to the lower-bound, unaged curve that is acceptable to the NRC staff. Further, the qualitative argument for excluding the fatigue crack growth analysis based on identification of the probable transients and the review of the piping stress analysis is acceptable, and the revised margin on flaw sizes using the original GE/EPRI J-integral estimation scheme as suggested by the NRC staff still meets the draft SRP requirement of 2. Finally, the NRC staff's concern with the sensitivity study using a crack morphology parameters characteristic of transgranular SCC was fully addressed by the licensee based on information in NUREG/CR-6443, and the results indicated that not only a margin of ten for leakage estimation is maintained for all cases being studied, but there is still margin left for the ratios of allowable flaw size to leakage flaw size for them.

3.1.4.3 Licensee's Results/Conclusions

The NRC staff's review confirmed the licensee's conclusion that the subject piping segments can be shown to exhibit LBB behavior consistent with the guidance in draft SRP 3.6.3 and NUREG 1061, Volume 3. The NRC staff's conclusion is based on the licensee's revised margins on flaw sizes that were obtained addressing all NRC staff's concerns mentioned in Section 3.1.2. For the segments of the piping system covered under the submittal, the licensee was able to show that a margin of two between the critical flaw size and leakage flaw size existed, while a margin of ten existed between the projected leakage rate and the sensitivity of the licensee's RCS leakage detection system. Based upon this information, the NRC staff concludes that LBB behavior had been demonstrated for the subject piping segments.

The NRC staff also evaluated the information provided by the licensee in the August 5, 2003, supplement regarding the sensitivity of its LBB analysis to changing flaw morphology

parameters. The changes in leakage identified in the licensee's analysis when going from a fatigue flaw morphology to a SCC flaw morphology were consistent with NRC staff expectations. The NRC staff concluded that, although the licensee's analysis did not demonstrate that the standard margin of two on flaw sizes would be met if an SCC-type flaw was assumed, the licensee's analysis did confirm that the factor of ten is maintained for the leakage-rate estimate and a lesser margin is maintained for the ratio of the critical flaw size to the leakage flaw size.

Considering the types of material from which the subject piping segments were constructed and their operating environment, no operating experience exists that would indicate the presence of any active SCC mechanism in these piping segments. Based on this experience, the NRC staff concludes that there is a lower likelihood of SCC in these piping segments when compared to traditional fatigue or corrosion-fatigue cracking mechanisms, such that the NRC staff can accept that the lesser margins demonstrated by the licensee's analysis were sufficient to confirm that LBB may still be granted on the segments of the piping system for which it was requested.

The NRC staff concludes that LBB behavior has been demonstrated for the segments of the Unit 1 CF and LPI/DHR piping system defined in Section 3.1.1 of this SE. Therefore, consistent with 10 CFR Part 50, Appendix A, GDC 4, the licensee shall be permitted to exclude consideration of the dynamic effects associated with the postulated rupture of the analyzed segments of the subject piping system from the Unit 1 design and licensing basis.

3.2 Use of Standard Review Plan Section 3.6.2 Branch Technical Position MEB 3-1

Attachment 3 of the March 20, 2003, submittal contains the proposed revision to UFSAR Section 3.6.1.2.1 that adopts SRP Section 3.6.2 BTP MEB 3-1 for the treatment of pipe breaks for the LPI system inside containment. The LPI piping is upstream of check valves LP-176 and LP-177 and the CF piping is downstream of the valves. BTP MEB 3-1 provides current NRC staff guidance for the treatment of pipe breaks. The NRC staff requested that the licensee provide a comparison of the current pipe break requirements at Oconee with the guidance provided in BTP MEB 3-1. The licensee's response indicated that the plant licensing basis as described in UFSAR Section 15.14.3.3 included a CF line break. The licensee also indicated that sufficient separation existed between the trains of CF, prior to implementation of the cross connect modification, to prevent interaction between the two trains. The addition of the new cross-connect piping between the two trains will cause a structural interaction between the two trains that did not exist prior to the modification. Therefore, the licensee proposed to use the pipe break postulation criteria provided in MEB 3-1 because of the potential structural interaction between the trains. As discussed below, the licensee adopted the guidance provided in MEB 3-1 to limit the number of postulated breaks in the LPI piping inside containment.

Attachment 5 of the licensee's submittal provides the licensee's technical justification for adopting BTP MEB 3-1 provisions for postulating pipe breaks for the LPI system. The submittal indicates that piping upstream of valves LP-47 and LP-48 qualifies as moderate energy per MEB 3-1, B.2.e, footnote 5. MEB 3-1, B.2.e allows postulation of leakage cracks instead of pipe breaks in the piping of systems that qualify as high energy fluid systems for only a short operational period, but qualify as moderate energy fluid systems for the major operational period. The NRC staff requested that the licensee describe the operating conditions under

which these sections of piping qualify as high energy. In response, the licensee indicated that the piping upstream of LP-47 and LP-48 is only postulated to be used at high energy conditions during decay heat removal operation of the LPI system during cooldown and heatup of the reactor coolant system. The licensee further indicated that its review of historic data found that the LPI system experiences high energy conditions a total of approximately 80 hours during startup and shutdown for each refueling outage. On the basis of the licensee's description of the LPI system operation, the NRC staff finds that the piping upstream of LP-47 and LP-48 qualifies as moderate energy per the guidance in MEB 3-1, B.2.e, footnote 5.

Attachment 5 of the licensee's submittal indicates that the piping between valves LP-176 and LP-48, the piping between valves LP-177 and LP-47, and the crossover piping is classified as high energy. The NRC staff requested that the licensee provide a comparison of the highest calculated stresses in these piping segments with the criteria for postulating pipe breaks and pipe cracks specified in BTP MEB 3-1. In response, the licensee indicated that the maximum calculated stress is less than the stress thresholds specified in BTP MEB 3-1 for postulating either pipe cracks or pipe breaks. Therefore, the licensee did not postulate pipe breaks or pipe cracks in this portion of piping. On the basis of the stresses reported by the licensee, the NRC staff agrees that BTP MEB 3-1 does not require postulation of pipe breaks or pipe cracks in this portion of piping.

Attachment 5 of the licensee's submittal indicates that the stress analysis model of the piping system includes piping upstream and downstream of valves LP-47 and LP-48. The submittal also indicates that valves LP-47 and LP-48 form the boundary between the high and moderate energy portions of the piping. The submittal cites footnote 3 of MEB 3-1 as justification for not considering the valves' terminal ends for the purpose of postulating breaks. However, footnote 3 of MEB 3-1 contains the following statement: "In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve) a terminal end of such runs is the piping connection to this closed valve." The NRC staff requested that the licensee provide additional justification as to why valves LP-47 and LP-48 should not be considered as terminal ends for the purpose of postulating pipe breaks. In response, the licensee stated that the piping model that includes the valves satisfies the intent of the footnote in that the valves are modeled in the piping run and they are not independently supported in such a way as to represent a terminal end.

Attachment 5 of the submittal indicates that rupture restraints will be installed at Unit 1 to protect against postulated breaks at the CF reactor vessel nozzle. The NRC staff requested that the licensee describe the criteria used to design the rupture restraints, including the criteria used to develop the break loads. In response, the licensee indicated that the rupture restraints were designed to the faulted allowable limits for various structural steel members, anchor bolts and structural welds. The licensee also indicated that, for supplied items, American Society of Mechanical Engineers (ASME) Level D allowables were used for qualification. The NRC staff finds the use of faulted allowable or ASME Level D limits acceptable for design of pipe whip restraints. The licensee also indicated that the pipe break loads were developed using criteria that satisfy the guidance specified in SRP Section 3.6.2. Therefore, the NRC staff finds the criteria used by the licensee to develop pipe break loads acceptable.

The licensee proposed to amend UFSAR Section 3.6 to adopt the criteria specified in SRP 3.6.2 BTP MEB 3-1 for postulation of pipe breaks for the LPI system inside containment. BTP MEB 3-1 provides criteria for determinating pipe rupture locations that satisfies the

requirements of GDC 4. Therefore, the NRC staff finds acceptable the licensee's proposed UFSAR Section 3.6.1.2.1 that adopts BTP MEB 3-1 for the treatment of postulated pipe ruptures for the LPI system inside containment.

3.3 Leakage Detection Capability

In Section 5 of Attachment 8 to the March 20, 2003, letter, the licensee addressed the leakage detection system for the purpose of demonstrating the acceptability of using 1 gpm as the minimum detectable leak rate.

The LBB analysis is based on the minimum detectable through-wall leakage for the applicable piping systems. The licensee stated that the RCS pressure boundary leak detection system is consistent with the guidelines of RG 1.45 such that a leakage of 1 gpm can be detected in an hour. Therefore, the licensee's LBB analysis uses the 1 gpm limiting condition for operation as an upper limit for RCS leakage.

The licensee identified that leakage can be continuously monitored in the control room by surveillance of the following detection systems:

- the reactor building atmosphere particulate monitoring system,
- the reactor building normal sump level indicators,
- the reactor building iodine, gaseous radioactivity and area monitoring systems, and
- the reactor coolant constant inventory measurement system.

As a result of issues raised by the NRC staff during a May 1, 2003, meeting with the licensee, the licensee, in Attachment 1 to the July 22, 2003, letter, provided additional information related to the sensitivities of the airborne radioactivity detectors. The July 22, 2003, letter stated that the licensee performed a more thorough evaluation of airborne radioactivity monitor leak detection capability. This evaluation established that the particulate monitor is capable of detecting 1 gpm within 1-hour based on the most conservative (Unit 3) RCS radioactivity levels during normal operation. The licensee also stated that the gaseous radioactivity monitor was not capable of detecting 1 gpm within 1-hour. However, the licensee noted that the reactor building normal sump level monitor is capable of identifying a 1 gpm leak rate within 10 minutes, which is well below the 1-hour recommended by RG 1.45 and is well below the sensitivity (1 gpm) necessary to support the LBB analysis. The licensee also identified that continuous RCS leak monitoring is performed by observation of makeup flow and letdown storage tank level. Additionally, an inventory balance is performed every 24 hours to quantify RCS leakage.

Based on the NRC staff's review of the information provided by the licensee, the NRC staff concluded that the RCS leakage detection system at Oconee is consistent with the guidelines of RG 1.45 and that the air particulate radioactivity monitor and the normal sump level monitor have the necessary sensitivity and response time to support the licensee's LBB analysis. Therefore, the RCS leakage detection system is adequate to support the proposed license amendment.

3.4 LOCA Reanalysis

Framatome performed a reanalysis of the LOCA events and verified that the LPI system modification does not adversely impact the results of the limiting large-break loss-of-coolant accident (LBLOCA) and reduces the peak cladding temperature results for a CF line break by ensuring LPI flow to the intact injection line. Neither the steam generator tube rupture event nor any additional LOCA event are affected. The reanalysis demonstrates that the ECCS, including the modifications to the LPI system, continue to satisfy the requirements of 10 CFR 50.46 and the 10 CFR Part 50, Appendix K acceptance criteria. This LPI system modification will facilitate plant operations and enhance plant safety. Framatome performed a reanalysis of LOCA events to verify that the LPI systems modification will not adversely impact the LOCA results. The reanlaysis verified that the LPI system modification does not adversely impact the results of the limiting LBLOCA and reduces the peak cladding temperature results for a CFLB by ensuring LPI flow to the intact injection line. The limiting LBLOCA event was reanalyzed incorporating a number of changes. The new analysis includes replacement steam generators, the LPI cross-connect modification and evaluation model enhancements. The evaluation model used to perform the LOCA analyses is described in the topical report BAW-10164P-A Revision 4, "RELAP5/MOD2-B&W-- An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," dated November 2002. Sensitivity cases performed for the LPI cross-connect modification concluded that there is no impact on the LOCA event from this modification and the results are within 10 CFR 50.46 acceptance criteria.

The proposed LPI system modification would not affect any plant safety limit, set points or design parameters. After installation of the passive LPI cross connect, the valves LP-9 and LP-10 will no longer be required to be manually opened to mitigate a single failure. Either LPI train will be able to inject into the header without operator action. This LPI system modification will facilitate plant operation and enhance plant safety.

In summary, the NRC staff has determined that the proposed LPI system modification meets the current requirements of 10 CFR 50.46; 10 CFR Part 50, Appendix K; GDC 34; and 35.

3.5 Technical Specifications Changes

The licensee proposes to add TS requirements associated with the passive LPI cross-connect modification. Since Oconee TS are common to three units, notes will be used to indicate the applicable requirement for each unit based on the status of the modification. The TS Surveillance Requirements (SRs), Limiting Conditions of Operation (LCO) and Actions are re-numbered and re-lettered accordingly. Such re-numbering and re-lettering changes are administrative in nature and the NRC staff therefore approves them. When the proposed modification is made to all units, the TS pertaining to the crossover valves outside containment will be removed in a future license amendment request (LAR).

Note 3 of LCO 3.5.3 of TS 3.5.3, "Low Pressure Injection (LPI)," currently requires the LPI discharge header valves (LP-9 and LP-10) outside containment to be manually operable to open on each unit in order to assure that adequate flow can be established to both CF lines in the event of a CFLB coincident with a single failure on the unaffected LPI train opposite the CFLB break, during MODE 1, 2, or 3. This note is modified to indicate that this requirement pertains to the unit only if the passive LPI cross-connect modification has not been completed on the unit. Note 4 is added to require that the LPI discharge header crossover valves inside

containment be open on each unit after completion of the installation of the passive LPI cross-connect modification on the respective unit. The NRC staff approves these changes because they are consistent with the modification and will ensure that flow can be established to both CF lines in the event of a CFLB coincident with a single active failure, which is the basis of the LCO.

Action B is modified to address the condition when one or more required LPI discharge header crossover valves outside containment are manually inoperable to open in MODE 1, 2, or 3. Action B.1 is modified to indicate that the action pertains to the crossover valves outside containment. The 7-day Completion Time in Action A for restoring an inoperable LPI train was approved by amendment issued June 18, 2003. Consistent with this approval, the licensee commited to having four compensatory measures in place prior to implementing that revision to the TS. The NRC staff, in its safety evaluation that supported the amendment issued June 18, 2003, determined that the licensee's commitment to implement these four compensatory measures formed part of the basis for accepting the amendment. The licensee has commited to implement the same four compensatory measures prior to implementing this current amendment. The NRC staff has similarly determined that the licensee's commitment forms part of the basis for accepting the current amendment, because it is necessary to indicate that the Action applies to the crossover valves outside the containment (since the modification adds crossover valves inside containment), and the Completion Time is consistent with the Completion Time of restoring an inoperable LPI train; both conditions render the system incapable of mitigating the effects of a single failure.

Action C is added to address the condition when one or more required LPI discharge header crossover valves inside containment is not open in MODE 1, 2 or 3. Action C.1 requires these valves to be opened within 7 days. The NRC staff approves these changes, subject to the regulatory commitment described above, because an Action is required for the units with crossover valves inside containment. Additionally, the Completion Time is consistent with the Completion Time of restoring an inoperable LPI train; both conditions render the system incapable of mitigating the effects of a single failure.

Surveillance Requirement (SR) 3.5.3.7 requires each LPI discharge header crossover valve, LPI cooler outlet throttle valve, and LPI isolation valve be opened manually every 18 months. Such manual actions are currently relied upon in the event that one of these valves fails to function properly during a Design Basis Event. With implementation of the modification on each unit, these valves will no longer be required to be opened manually to mitigate such failures since either LPI train will be capable of injecting into the intact header without operator action. As a result, SR 3.5.3.7 is modified to indicate that the SR is not applicable after completion of the passive LPI cross connect modification. A new SR is not needed for the new valves inside containment since SR 3.5.3.1 applies. The NRC staff approves these changes, since SR 3.5.3.1 applies to the new valves inside containment and ensures the valves are open, and SR 3.5.3.7 is only necessary for units without the valves inside containment.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (68 FR 22745). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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