UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 10, 2008

Mr. Dave Baxter
Vice President, Oconee Site
Duke Power Company LLC
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, ISSUANCE OF AMENDMENTS REGARDING ALLOY 600 CONCERNS IN THE PRESSURIZER (TAC NOS. MD9389, MD9390, AND MD9391)

Dear Mr. Baxter:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 364, 366, and 365 to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3, respectively. The amendments consist of changes to the Updated Final Safety Analysis Report (UFSAR) in response to your application dated August 1, 2008, supplemented by letter dated September 25, 2008.

These amendments revise the UFSAR to describe a design change that mitigates Alloy 600 concerns in the pressurizer.

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

Sincerely,

Leonard N. Olshan, Sr. Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 364 to DPR-38
2. Amendment No. 366 to DPR-47
3. Amendment No. 365 to DPR-55
4. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 364
Renewed License No. DPR-38

1. The U.S. 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 Carolinas, LLC (the licensee), dated August 1, 2008, and supplemented September 25, 2008, 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 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. 364, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Further, Renewed Facility Operating License DPR-38 is hereby amended to authorize revision to the Updated Final Safety Analysis Report as required by 10 CFR 50.71(e) to describe a design change that mitigates Alloy 600 concerns in the pressurizer.

4. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John Stang

Melanie C. Wong, Chief For
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-38
and the Technical Specifications

Date of Issuance: November 10, 2008

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 366
Renewed License No. DPR-47

1. The U.S. 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 Carolinas, LLC (the licensee), dated August 1, 2008, and supplemented September 25, 2008, 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 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. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 366, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Further, Renewed Facility Operating License DPR-47 is hereby amended to authorize revision to the Updated Final Safety Analysis Report as required by 10 CFR 50.71(e) to describe a design change that mitigates Alloy 600 concerns in the pressurizer.

4. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John Stang For

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-47
and the Technical Specifications

Date of Issuance: November 10, 2008

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 365
Renewed License No. DPR-55

1. The U.S. 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 Carolinas, LLC (the licensee), dated August 1, 2008, and supplemented September 25, 2008, 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 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. 365, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Further, Renewed Facility Operating License DPR-55 is hereby amended to authorize revision to the Updated Final Safety Analysis Report as required by 10 CFR 50.71(e) to describe a design change that mitigates Alloy 600 concerns in the pressurizer.

4. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Melanie C. Wong, Chief For
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-55
and the Technical Specifications

Date of Issuance: November 10, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 364

RENEWED FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 366

RENEWED FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 365

RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

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

Remove Pages	Insert Pages
Licenses	Licenses
License No. DPR-38, page 3	License No. DPR-38, page 3
License No. DPR-47, page 3	License No. DPR-47, page 3
License No. DPR-55, page 3	License No. DPR-55, page 3

Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. <u>Maximum Power Level</u>

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 364 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

(b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or

Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 366 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

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(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

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Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 365 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

(b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 364 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 366 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-47

AND

AMENDMENT NO. 365 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

By application dated August 1, 2008, to the U.S. Nuclear Regulatory Commission (NRC) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML082190505), as supplemented by letter dated September 25, 2008 (ADAMS Accession No. ML082950501), Duke Energy Carolinas, LLC (the licensee) submitted License Amendment Request (LAR) No. 2008-08 requesting approval of changes to the Updated Final Safety Analysis Report (UFSAR) to describe a design change that mitigates Alloy 600 concerns in the Oconee Nuclear Station, Units 1, 2, and 3 (ONS, Units 1, 2, and 3) pressurizers. The concern was to evaluate small areas of carbon and low alloy steel that may be exposed to the reactor coolant after replacing the pressurizer vent and thermowell with primary water stress corrosion cracking (PWSCC) resistant materials.

The supplements dated September 25, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff original proposed no significant hazards consideration determination as published in the *Federal Register* on September 9, 2008 (73 FR 52415).

2.0 REGULATORY REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59, "Changes, tests, and experiments," addresses criteria with which a licensee may evaluate a facility modification to assess whether regulatory approval is required prior to implementing the modification. 10 CFR 50.59 states:

> A licensee may make changes in the facility as described in the final safety analysis report (as updated), make changes in the procedures as described

in the final safety analysis report (as updated), and conduct tests or experiments not described in the final safety analysis report (as updated) without obtaining a license amendment pursuant to [10 CFR] 50.90 only if:

(i) A change to the technical specifications incorporated in the license is not required, and

(ii) The change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section

Paragraph (c)(2) of 10 CFR 50.59 goes on to list eight specific criteria which would result in a licensee needing to request a license amendment pursuant to 10 CFR 50.90 prior to implementing the proposed change, including that the change would:

(viii) Result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses

The American Society of Mechanical Engineers (ASME) Code of record for the fourth 10year inservice inspection program at ONS, Units 1, 2, and 3 is the 1998 Edition, including the 2000 Addenda, of Section XI of the ASME Code. The ASME Code, Section XI, 1998 Edition with no addenda has been approved for Section XI, Repair/Replacement program activities. The original pressurizer construction code is ASME Code, Section III, 1965 Edition through Summer 1967 Addenda. The licensee is currently also using the 1983 and 1989 Edition of the ASME Code, Section III.

3.0 EVALUATION

The information provided by the licensee in support of the LAR No. 2008-08 to evaluate small areas of carbon and low alloy steel that may be exposed to the reactor coolant after replacing the pressurizer vent and thermowell with primary water stress corrosion cracking resistant (PWSCC) materials has been reviewed and the basis for disposition is documented below.

Licensee's Request, as stated

This License Amendment Request (LAR) requests review and approval of a revision to the Oconee Nuclear Station (ONS) Updated Final Safety Analysis Report (UFSAR) Section 5.2.3.2 to reflect pending and as-built changes due to Alloy 600 mitigation activities for components of the ONS Units 1, 2, and 3 Pressurizers. Section 5.2.3.2 of the ONS UFSAR describes the materials selected for the ONS Reactor Coolant System (RCS). It states that these materials were chosen for their corrosion resistant properties for the expected service conditions. As part of the ongoing mitigation of Alloy 600 components within the RCS, a mitigation technique was selected for the Pressurizer thermowell and vent nozzle which exposes small areas of carbon steel (CS) to the RCS environment. The current UFSAR terminology implies that no corrosion evaluation is required because the materials were selected for their corrosion resistant properties. The [ONS,] Unit 2 design change exposes CS and includes a corrosion evaluation. The inclusion of a corrosion evaluation was conservatively characterized as a methodology change during the 10 CFR 50.59 process.

It was during the preparation of the [ONS,] Unit 2 design change package that the need for prior NRC approval was first identified. This same work was previously completed on [ONS,] Units 1 and 3 without prior NRC approval. A cause investigation into this difference substantiated that there were no technical concerns with the package and its installation. The reviewer for the 10 CFR 50.59 evaluation for the [ONS,] Unit 2 design change package concluded that inclusion of a corrosion evaluation where none previously existed constitutes a methodology change requiring prior NRC approval. This reviewer was not involved in the [ONS,] Unit 1 or Unit 3 changes. This conclusion is considered to be conservative; however, per 10 CFR 50.59, NRC approval is being sought for the change.

In summary, the method chosen to alleviate Alloy 600 concerns for the Pressurizer thermowell and vent nozzles will allow a small portion of the Pressurizer CS to be exposed to stagnant RCS water. We request NRC approval of the design change and accompanying UFSAR revision.

Licensee's Basis for Request, as stated

Each of the original [ONS,] Unit 1, 2, and 3 Pressurizers contain nine small bore Alloy 600 (SB 166) components and their attachment butt welds, including the Pressurizer steam space vent nozzle, the six level and single sample tap safe ends, and the water space thermowell. Due to [PWSCC] concerns for these small bore Alloy 600 components and past Alloy 600 operational experience at other utilities, the replacement of these components was considered to be a priority for ONS. Therefore, starting in early 2005, ONS contracted with outside vendors to provide new component designs to replace these Alloy 600 components with PWSCC resistant materials.

Vent Nozzle Technical Evaluation

An outside vendor was contracted for the Pressurizer vent nozzle component replacement. The design was based on a half-nozzle replacement, where the existing Alloy 600 vent nozzle is severed mid-wall in the Pressurizer head. The nozzle hole is then over-bored to accept a new stainless steel vent nozzle, and an external weld pad is welded to the Pressurizer head to form the new pressure boundary weld. The drawing of this replacement design is included in Attachment 2[1]. The new stainless steel vent nozzle is welded to the external weld pad, maintaining a 1/16" minimum gap between the bottom of the new nozzle and the top of the old nozzle remnant, thereby allowing the Pressurizer steam environment to enter the gap between the new vent nozzle, the existing vent nozzle remnant, and the bored hole in the Pressurizer head. The vent nozzle remnant and the existing J-groove weld remain attached to the Pressurizer head inside surface. This change was implemented during refueling outage 3EOC22 [ONS, Unit 3 End-of-Cycle 22] in the spring of 2006 for ONS, Unit 3 and in the fall of 2006 during 1EOC23 for ONS, Unit 1. This change is planned for

1. Included in Attachment 2 are ONS, Units 1, 2, and 3, reference drawings of the pressurizer modifications and they can be found in the licensee's submittal dated August 1, 2008.

implementation in the fall of 2008 for ONS, Unit 2. The [ONS,] Unit 1 and 3 component replacements occurring without NRC prior approval has been entered into the corrective action program, and a cause evaluation has been performed.

The replacement Pressurizer vent nozzle leaves a small area of the Pressurizer head CS exposed to the Pressurizer steam space environment. The [ONS,] Unit 1 head material is SA212, Grade B and the [ONS,] Unit 2 and 3 material is SA516, Grade 70. The exposed area is of a cylindrical geometry, with a 1.65" diameter over the outermost 1.5" and a 1.45" diameter over the remaining inner 2.75" of Pressurizer head thickness, resulting in an exposed CS area on the order of 20 sq in. The new stainless steel vent nozzle is designed to have a 0.010" radial gap between the new nozzle and Pressurizer head bore diameter (Attachment 2). The original Alloy 600 vent nozzle was designed for a 0.010" radial gap between the existing nozzle and the Pressurizer head bore diameter.

Thermowell Technical Evaluation

A second, independent firm was contracted for the Pressurizer thermowell component replacement. The design removes the existing thermowell and a portion of the J-groove attachment weld with a combination of mechanical boring and Electro Discharge Machining (EDM) to reduce the potential for foreign material introduction into the Pressurizer. An external weld pad is welded to the outside surface of the Pressurizer shell to form the new pressure boundary weld and a new Alloy 690 thermowell is inserted and welded to the external weld pad, thereby allowing the Pressurizer water space environment to enter the gap between the new thermowell and the Pressurizer shell bore. This design change was implemented in the fall of 2006 for ONS, Unit 1 and the fall of 2007 for ONS, Unit 3. This change is planned for implementation in the fall of 2008 for ONS, Unit 2. The [ONS,] Unit 1 and 3 component replacements occurring without NRC prior approval has been entered into the corrective action program, and a cause evaluation has been performed.

The replacement Pressurizer thermowell leaves a small area of the Pressurizer shell CS exposed to the Pressurizer water space environment. The ONS, Unit 1 shell material is SA212, Grade B and the ONS, Unit 2 and 3 material is SA516, Grade 70. The exposed area is of a cylindrical geometry, with a 1.510" maximum diameter through the 6.188" Pressurizer shell thickness, resulting in an exposed area on the order of 30 sq in. The new Alloy 690 thermowell has a 1.493" to 1.498" outside diameter, so the radial gap between the new nozzle and Pressurizer head bore diameter is a maximum of 0.0085".

Proposed changes to the Section 5.2.3.2 of the ONS, [1, 2, and 3] UFSAR are based on exposing materials as part of Alloy 600 component mitigation and updating the effect this exposure has on the design life of the affected components.

The corrosion evaluations for each of the Pressurizer vent and thermowell designs meet the requirements of the ASME Code, Section III, Subsection NB, Subparagraph NB-3121, which states that corrosion effects must be considered in the design. Each of these corrosion evaluations was documented in the individual components' supporting design calculations. Corrosion rates were taken from appropriate, available Electric Power Research Institute (EPRI) and generally available industry standards (see References 6.3[2], 6.4[3], and 6.5[4]). Similar corrosion rates for similar materials were reviewed and accepted by the NRC in an [Safety Evaluation (SER)] dated 11/15/07 (Reference 6.2[5]). No proprietary vendor sources were used in the development of either of the corrosion evaluations.

Vent Nozzle Summary Information

The corrosion evaluation for the Pressurizer vent nozzle is documented in a calculation. Based on extensive experimental test data and power operating data, the evaluation determined that a 0.001 inch/year long term corrosion rate for stagnant locations such as the exposed CS due to the Pressurizer vent nozzle repair was appropriate for operating conditions and a 0.009 inch/year corrosion rate was appropriate for shutdown conditions. These values were used in an 18 month operating and 2 month shutdown cycle, resulting in a conservative general long term corrosion rate of 0.0018 inches per year and the conclusion that the long term corrosion rate and the overall release of iron into the RCS is expected to be negligible during the 60 year replacement vent nozzle component design life.

Thermowell Summary Information

The corrosion evaluation for the Pressurizer thermowell is documented in a calculation. Based on several corrosion rate references, the evaluation determined that a 0.0006 inch/year corrosion rate was appropriate for high temperatures during 85% of an operating year, 0.0022 inch/year at intermediate temperatures during 5% of the operating year, and a 0.008 inch/year corrosion rate was appropriate for shutdown conditions during the remaining 10% of the year. These values were used together, resulting in a conservative general corrosion rate of 0.00142 inches per year and the conclusion that this total general corrosion rate is not considered significant during the 60 year replacement thermowell component design life.

The conservative corrosion rates developed within both the corrosion calculations are considered to be 'negligible' or 'not significant', and therefore are acceptable for the specific Pressurizer vent and Pressurizer thermowell design applications. Similar corrosion rates for similar materials were reviewed and accepted by the NRC in an [SE]

2 Reference 6.3 "Boric Acid Corrosion Guidebook", Revision 1 EPRI TR-1000975 is not included in this SE.
3 Reference 6.4 "Evaluation of Yankee Vessel Cladding Penetrations", WCAP-2855 is not included in this SE.
4 Reference 6.5 "Low Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement Program", WCAP-15973, Revision 1 is not included in this SE.
5 Reference 6.2 is NRC SE for Crystal River, Unit 3 (ML073100991) is not included in this SE.

dated 11/15/07 (Reference 6.2). These expected extremely low rates of material loss and iron release rates will provide an acceptable level of safety.

No similar ONS design changes have been previously reviewed by the NRC. However, other utilities have implemented similar Alloy 600 component mitigation strategies, and these changes were reviewed by the NRC as part of their Relief Request submittals. One example is Crystal River, [Unit 3 (CR-3)] resubmittal of Relief Requests #07-001-RR, Rev 0 and #07-002-RR, Rev 0 (CR-3 Letter 3F0407-15, dated April 12, 2007). A corrosion evaluation summary was included for completeness on page 13 of Attachment B (Relief Request #07-001-RR, Rev 0) in paragraph 2.g[6]. The corrosion evaluation in the CR-3 document contains a similar level of detail as presented here, as well as providing an identical conclusion as to the significance of the corrosion that would be introduced by the implementation of these Alloy 600 mitigation techniques. Reference 6.2 is the NRC SE for the CR-3 Request for Relief.

Corrosion evaluations were performed as part of the Alloy 600 component replacement design for the Pressurizer vent nozzle and the thermowell connection. These evaluations met all ASME Code requirements. Supporting corrosion rates for the specific designs were taken from standard, non-proprietary industry sources (References 6.3, 6.4, and 6.5). The resulting corrosion rates are considered 'negligible' and 'not significant', respectively for the vent and thermowell replacement designs. The corrosion rates are acceptable for the replacement component design life and will provide an acceptable level of safety.

NRC Staff Evaluation

ASME Code, Section III, Subsection NB, Subparagraph NB-3121 states that:

> Material subject to thinning by corrosion, erosion, mechanical abrasion, or other environmental effects shall have provisions made for these effects during the design or specified life of the component by a suitable increase or increase in or addition to the thickness of the base metal over that determined by the design formulas.
>
> Materials added or included for these purposes need not be of the same thickness of all areas of the component if different rates of attack are expected for the various areas. It should be noted that the tests on which the design fatigue curves are based did not include tests in the presence of corrosive environments which might accelerate fatigue failure.

The potential of corrosion mechanisms affecting the pressurizer nozzles and thermowell penetrations were evaluated by the licensee. This evaluation was performed because the repair configuration leaves portions of the CS material inside the pressurizer penetrations exposed to the primary reactor coolant. The exposure of the CS material is caused by the existence of a small gap at the junction between the original (Alloy 600) and new (Alloy 690) nozzles. The licensee's analysis evaluated the long-term impact of the newly exposed CS material to the reactor coolant.

6 The Crystal River, Unit 3 submittal dated April 12, 2007 (ML071090182), is not included in this SE.

The results of the licensee's analysis for the pressurizer vent nozzle are documented in AREVA Document 9001050-001[7]. For the pressurizer thermowell component replacement, the results of the corrosion rate analysis are contained in Structural Integrity Associates (SIA) SIR-07-006, Revision 0, Appendix B[8].

The licensee has performed calculations based on the ASME Code, Section III requirements and determined that the corrosion rate for the pressurizer vent nozzle is approximately 0.001 inch/year at normal operating conditions and 0.009 inch per/year at shutdown conditions for stagnant locations such as the exposed CS due to the pressurizer vent nozzle repair. The licensee noted that these values were used to evaluate an 18-month operating cycle and 2-month shutdown cycle. This assumption resulted in a conservative general long term corrosion rate of 0.0018 inches/year for the exposed CS area from the pressurizer vent nozzle repair.

The pressurizer thermowell calculations were based on several corrosion rate references as noted above by the licensee. The licensee found that a 0.0006 inch/year corrosion rate was applicable for high temperatures during 85 percent of an operating year, that 0.0022 inch/year corrosion rate was applicable at intermediate temperatures during 5 percent of the operating year, and a 0.008 inch/year corrosion rate was applicable for shutdown conditions during the remaining 10 percent of the year. As a result, when these values were used together, a general corrosion rate of 0.00142 inches per year was obtained. Therefore, the total general corrosion rate was not considered significant during the 60 year replacement thermowell component design life.

The NRC asked if the licensee had any plans in preventing corrosion on the affected CS areas. In its letter dated September 25, 2008, the licensee noted, that since the corrosion rates were very low, these rates did not support options for preventing corrosion. However, the licensee noted that it controls the RCS chemistry which will help prevent the corrosion rate from increasing. In addition, the ASME, Section XI requires periodic inservice examinations of the various components of the ONS, Unit 1, 2, and 3 pressurizers. Therefore, if there is any evidence of degradation occurring in the subject locations it would be more than likely found by the licensee during these examinations.

There have been many studies done regarding the corrosion of CS exposed to the RCS environment as noted in the above references. The NRC staff has reviewed these studies and the CR, Unit 3 NRC SE dated November 15, 2007, regarding the effects of the RCS environment on newly exposed CS material. The NRC staff found that the corrosion rate results calculated for ONS, Units 1, 2, and 3 were similar to CR, Unit 3 in that the total corrosion rates were on the order of 0.001 to 0.002 inch/year. Based on these studies, the NRC staff has determined that the corrosion rate of the CS material is insignificant as compared to the 60-year design life of the subject components and that the corrosion rate of the exposed CS in the long term will not compromise the structural integrity of the ONS, Unit 1, 2, and 3 pressurizers. The NRC staff has determined that the licensee's evaluation regarding the corrosion rate of the exposed CS areas meets the ASME Code, Section III requirements and is acceptable.

The licensee's design change to the ONS, Unit 1, 2, and 3 pressurizer vent nozzle components and thermowells to mitigate and prevent PWSCC concerns is standard practice throughout industry. Therefore, the licensee's design change to the ONS, Units 1, 2, and 3 pressurizers meets the 10 CFR 50.90 process and the ASME Code, Section XI repair/replacement requirements. Therefore, the NRC staff has determined that the licensee's design change and UFSAR revision is acceptable.

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7 The AREVA Document 9001050-001 is not included in this SE and is considered proprietary information.

8 The SIA SIR-07-006, Revision 0, Appendix B report is not included in this SE and is considered proprietary information.

In conclusion, the NRC staff has reviewed the licensee's submittal and concludes that the licensee's evaluations regarding the corrosion rate of the exposed CS areas meets the ASME Code, Section III requirements. The NRC staff concludes that the design change to the ONS, Unit 1, 2, and 3 pressurizer vent nozzle components and thermowells to mitigate and prevent PWSCC concerns meets the 10 CFR 50.90 process and the ASME Code, Section XI repair/replacement requirements. Therefore, the NRC staff concludes that the licensee's design change to prevent PWSCC, CS corrosion evaluation, and UFSAR revision are acceptable.

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 a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. 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 (73 FR 52415, September 9, 2007). 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.

Principal Contributor: T. McLellan, NRR/DCI/CVIB

Date: November 10, 2008

November 10, 2008

Mr. Dave Baxter
Vice President, Oconee Site
Duke Power Company LLC
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, ISSUANCE OF AMENDMENTS REGARDING ALLOY 600 CONCERNS IN THE PRESSURIZER (TAC NOS. MD9389, MD9390, AND MD9391)

Dear Mr. Baxter:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 364, 366, and 365 to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3, respectively. The amendments consist of changes to the Updated Final Safety Analysis Report (UFSAR) in response to your application dated August 1, 2008, supplemented by letter dated September 25, 2008.

These amendments revise the UFSAR to describe a design change that mitigates Alloy 600 concerns in the pressurizer.

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

Sincerely,

/RA/

Leonard N. Olshan, Sr. Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 364 to DPR-38
2. Amendment No. 366 to DPR-47
3. Amendment No. 365 to DPR-55
4. Safety Evaluation

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GHill (6)
RidsNrrDirsItsb Resource
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RidsNrrDciCvib Resource

License Amendment No.: ML082820593 (*) Input from SE

OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	NRR/CVIB/BC	OGC	NRR/LPL2-1/BC
NAME	LOlshan	MO'Brien (GLappert for)	MMitchell (*)	AJones	MWong for
DATE	10/14/08	10/14/08	10/6/08	10/22/08	11/05/08

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