Duke Energy®

**DAVE BAXTER**
*Vice President*
*Oconee Nuclear Station*

**Duke Energy**
*ON01VP / 7800 Rochester Highway*
*Seneca, SC 29672*

**864-873-4460**
*864-873-4208 fax*
*dabaxter@dukeenergy.com*

June 30, 2009

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station, Units 1, 2, and 3
Docket Nos. 50-269, 50-270, 50-287
10 CFR 50.59 Annual Report

Attached are descriptions of Oconee facility changes, tests, and experiments which were completed subject to the provisions of 10 CFR 50.59 between January 1, 2008, and December 31, 2008. This report is submitted pursuant to the requirement of 10 CFR 50.59 (d)(2).

If there are any questions, please contact Corey Gray at (864)873-6325.

Sincerely,

Dave Baxter
Vice President
Oconee Nuclear Station

Attachment

IE47
NRR

xc: Mr. Luis Reyes
Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, GA 30303

Mr. Eric Riggs
Senior NRC Resident Inspector
Oconee Nuclear Site

Mr. John Stang
Senior Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

**Type:** Nuclear Station Modification (OD200443 / UFSAR Change 08-68)

**Title:** Control Room Chart Recorder Replacement

**Description:** Design change OD200443 replaces selected obsolete chart recorders. A total of twelve new multi-point graphic recorders will be installed in the Control Room, six on 2UB1, four on 2VB1 and two on 2VB2. New multi-system monitors are installed on 2UB2, 2AB2, and on the operator's desk. A new OAC workstation is installed in 2AB2. Direct and immediate trend of the information recorded by the obsolete recorders was never considered essential for operator information or action, nor is the recording device the primary indicator for the recorded parameter. Regulatory Guide 1.97 requirements for recording of parameters are met post-modification. The new recorders and monitors/workstation are not initiators of any analyzed accident, are not required for the mitigation of any analyzed accident, and will not adversely affect any SSCs required to mitigate an accident. Therefore, this design change does not adversely affect analyzed accidents, malfunctions, or their consequences.

This design change does not modify SSCs with the potential to affect fuel cladding, RCS boundary/pressure, or containment boundary/pressure. Therefore, design basis limits for fission product barriers are not affected. Replacement of recorders and addition of monitors/workstation does not involve a method of evaluation; the recorders only record various plant parameters and the new monitors/workstation enhance the ability of operators to interface with existing plant computer systems. Selected Licensee Commitment (SLC) Table 16.11.3-2, Technical Specification Bases 3.3.8, and UFSAR Sections 7.4.2.2.3 and 7.5.2.53 are revised due to this design change.

**Type:** Nuclear Modification (OD200547 / UFSAR Changes 08-17 and 08-18)

**Title:** Replace Unit 2 Control Rod Drive Control System (CRDCS) with a new Digital Control Rod Drive Control System (DCRDCS)

**Description:** A License Amendment Request (LAR) was submitted to the NRC by letter dated January 15, 2004 to obtain NRC review and approval of RTB Technical Specification (TS) changes required because of RTB replacement. Supplement 1 to the LAR was submitted on March 15, 2004. The NRC response dated November 2, 2004, found the proposed TS changes acceptable (License Amendment No. 343 for ONS Unit 2). This 50.59 does not further address those TS changes. No additional TS changes are required due to design change OD200547. This 50.59 addresses only those aspects of OD200547 which are outside of the bounds of the License Amendment No. 343, which affected TS 3.3.4. The reactor trip function is independent of and separate from the CRD Control System.

Design change OD200547 replaces the Unit 2 CRDCS with a new Digital Control Rod Drive Control System (DCRDCS) and consists of System Logic Equipment, Motor Control Equipment, RTB Switchgear and Associated Equipment, and DCRDCS Man/Machine Interfaces. A Failure Modes and Effects Analysis (FMEA) was performed for the new DCRDCS to determine if adverse effects (i.e., loss of reactor control, uncontrolled rod withdrawal, reactor trip, or prevention of reactor trip) could result from the credible failure of a single component. The results of the FMEA were "All operations critical to the safe and effective performance of the DCRDCS maintain sufficient redundancy such that no credible single failure can compromise the design".

This design change is designed to comply with the safety, reactivity rate limits, startup considerations, and operational design basis requirements for CRD. The DCRDCS is not required for accident mitigation, post accident response or offsite release mitigation. It does not perform any plant protective functions. UFSAR requirements for the CRDCS were reviewed and functions required to be performed by the CRDCS, as given in the UFSAR, are retained in the new design. The DCRDCS is nonsafety equipment and is not relied upon to mitigate accidents and DCRDCS failures will not prevent safety-related SSCs from fulfilling their design functions. Consequently, the DCRDCS is not considered to be an SSC important to safety. The activity does not affect the frequency of occurrence of evaluated accidents or the likelihood of occurrence of a malfunction of an SSC important to safety. The activity does not increase the consequences of an accident or malfunction.

DCRDCS is not an initiator of accidents and therefore does not create the possibility for an accident of a different type. DCRDCS is not an important to safety SSC, nor does it adversely affect an important to safety SSC, and therefore, it does not create a possibility for a malfunction of an important to safety SSC with a different result.

Based on the above, there were no safety concerns. No Technical Specifications or Bases need to be changed due to this Design Change. There were UFSAR and Selected Licensee Commitments changes required. Prior NRC review and approval is not required.

**Type:** Nuclear Station Modification (ON-23098 / UFSAR 07-30)

**Title:** Upper Surge Tank

**Description:** The implementation of NSM ON-23098 was broken into two phases. Phase 1 included the structural steel work to the Upper Surge Tank (UST) platform and the UST Dome Tank and a 10 CFR 50.59 screening was performed for that phase. The remainder of the NSM is to be implemented under Phase 2. This 10 CFR 50.59 summary only addresses Phase 2.

Phase 2 of NSM ON-23098 will make some modifications to eliminate single active failures associated with the Upper Surge Tank (UST). These modifications involve both "active type" and "passive type" isolation. "Active type" isolation requires a signal to be sent to a valve to close to assure flowpath isolation. "Passive type" isolation uses a check valve to prevent reverse flow.

This modification is to add four air-operated valves (AOV) that automatically close when the UST level drops below 7.5 feet. New valves 2C-903 and 904 are to isolate flow to the hotwell and to the Auxiliary Boiler Feedwater (FDW) pump. A bypass valve, 2C-912, is provided around 2C-903 and 904. Valves 2C-906 and 907 isolate flow to the Powdex Backwash Pump. Unit 1 NSM ON-13098 added pressure switches in the suction line to both the Auxiliary Boiler FDW pump and Powdex Backwash pump to trip them on low suction pressure for either units' closing of the pump's supply valves. NSM ON -23098 is to add a switch to the Auxiliary Boiler FDW pump pressure switch that was added by NSM ON-13098 to allow it to be bypassed when needed. The switch added by NSM ON-13098 trips the Auxiliary Boiler Feedwater pump on low suction pressure that would be indicative of valves 1C-903/904 or 2C-903/904 closing. This prevents the boiler from tripping on low Feedwater level and possibly from damage due to overheating as the Feedwater is boiled away.

The modification is also to add a new Condensate Recirculation path to the UST. This path will allow flow from the Condensate Booster Pump suction line to the UST Riser. A manual throttle valve, 2C-899, and flow indication (locally and on the Operator Aid Computer) are provided. This new Condensate Recirculation path should provide significant operational flexibility during unit start-ups.

Other changes by this modification include:

- upgrade of the hotwell level control system, including replacement of valve 2C-192, and level transmitters LT-17 and LT-19
- removal of electric motor operated valves 2C-152 and 153

- upgrade of the UST level transmitters to reduce instrument uncertainty
- upgrade of the Emergency Feedwater (EFW) Pump recirculation path to the UST Dome Tank to Class F, QA Condition 1; including seismic qualification of the Dome Tank.

Expansion joints are to be installed in Class G piping connections to the UST Dome tank. There are currently three somewhat independent paths of water from the UST to the condenser. The modification will combine the supplies to these paths with a common header.

The loss of air to the new AOVs could occur due to a loss of the non-safety related Instrument Air System or due to a loss of air locally at the valve (e.g., loss of the safety related power supply to the solenoid valve on one of the new valves. A loss of the Instrument Air System would result in the same effect in both the existing and new design. The effect would be that the flowpath(s) from the USTs to the hotwell would be isolated due to fail closed AOVs. If the air supply is lost to one of the new valves (2C-903 or 2C-904), then all three flowpaths would be isolated. But, there are manual actions that could be taken to bypass the failed closed AOV. If the hotwell level is not replenished over time, a trip of the unit could occur. Thus, there is a possible increase in the potential for a turbine/reactor trip if makeup to the hotwell is not able to be achieved. But there are other means for a reactor/turbine trip to occur. UFSAR Section 15.8 provides a number of means for a turbine trip. The potential cause for a turbine trip is described as including a generator trip, low condenser vacuum, loss of turbine lubrication oil, turbine thrust bearing failure, turbine overspeed, main feedwater pump trip, high steam generator level, or a reactor trip. The loss of hotwell level could, if low enough, cause the main feedwater pumps to lose suction pressure and ultimately trip. The trip of the main feedwater pumps are listed in the UFSAR section described above. This small potential for a localized loss of air to one of the new valves is considered to be a negligible increase to the overall turbine trip potential and thus is not a "more than a minimal increase" in the frequency of occurrence of an accident previously evaluated in the UFSAR.

The two AOVs (2C-903 and 2C-904) in the common header from the USTs to the hotwell will isolate the UST if a low UST level is detected or if air is lost to the valves. These two AOVs are to be used as the QA-1 Class F boundary so that the UST tank contents will be isolated even in the event of a single failure. The potential for "more than a minimal increase" in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR was investigated with respect to the UST's makeup going into a common header before going to the three separate pathways. Equipment important to safety affected by this modification includes the UST (assured source of EFW), condenser hotwell (one of the potential long term sources of EFW), and EFW System (provides feedwater in the event of a loss of main feedwater). There are currently three somewhat independent paths of water from the UST to the condenser.

The UFSAR describes the three separate pathways from the UST to the condenser hotwell, but the context of this wording is in relation to the pathways being automatically isolated on a low UST level. The "important to safety" aspect of the condenser hotwell is its function of supplying an EFW supply of water after the UST source has been exhausted. The flowpath from the condenser to the EFW pumps is not adversely affected with the new valves since that flowpath is not used when supplying the EFW pumps via the hotwell. The "important to safety function" of the existing AOVs and the new AOVs in the new design is considered to be their closure on low UST level. This function is enhanced in the new design. Supplying the hotwell from the UST is considered more of an operational issue versus an "equipment important to safety" issue. Thus, the use of a common header with two AOVs in series is not considered to cause a "more than minimal" increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The EFW System is used to mitigate accidents involving the loss of main feedwater. The modification will not change the design function of the EFW supply sources as evaluated in the UFSAR. Thus, in an accident involving loss of main feedwater, the EFW System will still be able to mitigate the event as currently described in the UFSAR. There is no adverse effect on containment integrity and no new release paths are created. The design is such that all valves, piping, components, and circuitry which are required to assure the UST is not prematurely depleted are QA-1 and seismically qualified.

The UST will be designed to provide a source of water to the EFW System even in the event of a single failure. The hotwell backup source is not designed to provide the additional EFW water supply in the event of a single failure. The flowpath from the UST to the hotwell is not required to be designed to withstand a single failure for the function of allowing water to flow. This path is designed such that a single failure does not allow UST flow to be depleted to the Hotwell. Thus, the EFW function is not adversely affected with respect to mitigating loss of feedwater scenarios previously evaluated in the UFSAR.

If air is lost to either new valve 2C-906 or 2C-907, UST supply to the polishing demineralizer backwash pumps could be lost. The polishing demineralizers are used for normal plant operation and they do not serve a safety function nor are they designed to the single failure criterion.

If air is lost to new valves 2C-903 or 2C-904, the Unit 2 supply to the Aux Boiler could be lost. The Aux Boiler is not required to be designed so that it is available following a single failure. A switch was added to the Aux Boiler FDW pump pressure switch by NSM ON-13098 to allow it to be bypassed when needed.

**Type:** Nuclear Station Modification OD102245

**Title:** Increase Unit 1 Reactor Coolant Pumps Seal Staging Flow and Revise Tech Spec Bases 3.10.1

**Description:** Design change OD102245 is to shorten the staging coils in each Reactor Coolant Pump (RCP) seal staging assembly. Shortening the staging coils will increase the seal return flow. The current flow rate to the seals (through the seal staging coils) is approximately 1.5 gallons per minute (gpm) at normal Reactor Coolant System (RCS) pressures. The design of the current seals with respect to their pressure breakdown is such that each of the 3 seals breaks down about 1/3 of the Reactor Coolant System pressure. The seals' staging coils are to be shortened such that they each achieve a flowrate of approximately 2.2 gpm. The pressure breakdown is to remain approximately the same as the current design.

The RCP seals have experienced failures and other problems due to excessive heat, especially around the #3 seal. The design change is desired so that the water temperatures within the seal cavities can be reduced. Increasing the flow rate reduces the temperature rise of the fluid as it passes through regions containing heat generating items such as RCP bearings or seals. This leads to reduced temperatures in the downstream seal cavities. Thus, the goal of the design change is to reduce the water temperature to the #3 seal of each RCP (and reduce the operating temperature of the RCP seals) by increasing the seal return flow.

The existing time to isolate the seal return line in an event in which the SSF is utilized is also being changed from 20 minutes to 15 minutes to assist in preventing overheating of the seal o-rings. The o-rings in the seal package are the limiting item regarding temperature within the seals.

The bases to Technical Specification (TS) 3.10.1 are to be revised to reduce the maximum total combined RCS leakage limit for Units 1, 2, and 3. This limit is used to assist in determining Reactor Coolant Makeup System operability. The bases are also to be revised to provide a different way to determine a RCP seal return flow rate if the flow rate is not available for Unit 1. The bases to TS 3.10.1 is also to be revised to change the Units 2 and 3 maximum total combined RCS leakage limit from 24.7

gpm to 15.0 gpm. Units 2 and 3 have an existing value that can be used in this combined RCS leakage limit determination if the seal return flowrate for a RCP is not available. This value is changing for Units 2 and 3. The SSF RC Makeup System Design Basis Specification is to be revised to incorporate these new limits and values.

The SSF RC Makeup System Design Basis Specification is also to be revised to delete an Operable But Degraded/Non-Conforming (OBD/N) condition that could occur if pressurizer ambient heat loss exceeds the capacity of the pressurizer heaters. The pressurizer ambient heat loss issue has been resolved but the leakage limit is currently still in effect.

The time to initiate SSF RC Makeup System seal injection is not changed. The time to isolate the seal return flow in an SSF event is changed and is discussed later in this response.

The design change is QA-1. The staging coils and the screens that cover them are non-code (i.e., not pressure retaining) parts. The changes to the coils do not cause the seismic qualification of the RCPs or the seal assemblies to be adversely affected.

The shortening of the RCP seal staging coil increases the portion of the RCP seal injection flow that provides cooling to the shaft seals. This increase to the shaft seal cooling causes a reduction in the seal injection flow that goes through the labyrinth seal and back into the RCS. The purpose of this labyrinth seal is to serve as a buffer to keep reactor coolant from entering the upper portion of the pump during normal operation. As long as there is any flow from the seal injection area across the labyrinth seal into the RCS, this function is accomplished. It also restricts flow of reactor coolant to the seal assembly in the event seal injection flow is lost. This function is not affected by this change. Thus, this reduction in labyrinth seal flow does not cause any adverse effects to the RCPs.

During an SSF event, seal injection and seal cooling are lost and hot Reactor Coolant will begin to flow up into the seal area. Seal return flow is isolated to minimize seal heat as well as conserve RCS inventory. The o-rings in the seal package are the limiting item within the seals. To maintain the o-rings within their allowed temperatures, the seal return line needs to be isolated within 15 minutes versus the existing 20 minutes. This change ensures that the o-rings will stay within the allowed temperature even if the seal injection is not started until 20 minutes. If the SSF is needed for mitigating an event, the power is to be transferred to the SSF within 10 minutes. The isolation of the seal return line within 15 minutes would still be after the SSF power transfer. The seal return isolation valve is powered from the SSF so isolation of the seal return line can be performed from the SSF after transfer of power.

The potential for thermal shocking the seals once SSF RCMU flow commences was reviewed and was determined to have no change in effect as compared to the current design. The design temperature in the seal return line was raised to a slightly higher value. The relief valve design was reviewed for effects of the higher seal return flow and was determined to be acceptable. The seal return filter is still designed for the new seal return flowrate. The increased seal return flowrates are still within the design of the seal return coolers. The calculation for the required differential pressure that the seal return isolation valve must close against was reviewed and determined to not be adversely affected.

**Type:** UFSAR Change 08-06

**Title:** Revisions to UFSAR 6.2.1.1.3.3, 6.2.1.4, 6.2.2.3 and TS Bases B3.6.5 for a change in reactor building spray header fill time

**Description:** The proposed activities are revisions to UFSAR Section 6.2.1.1.3.3 (Steam Line Break Containment Pressure and Temperature Response) and Section 6.2.1.4 (Mass and Energy Release Analyses for Postulated Secondary System Pipe Ruptures Inside Containment) for an analysis using a revised reactor building spray header fill time. The net effect of this change was an increase in the peak containment pressure and an increase in the amount of time in which the containment temperature is above the Equipment Qualification (EQ) envelope. The increase in peak containment pressure is bounded by the internal containment design pressure. The increase in the amount of time in which containment temperature is above the EQ envelope does not cause the equipment internal temperature to increase above the EQ envelope. The method used for the steam line break containment pressure and temperature analysis is described in topical report DPC-NE-3003-PA, "ONS Mass and Energy Release and Containment Response Methodology."

Additionally, applicable Technical Specifications have been reviewed and no changes are required for these changes. This 10CFR50.59 evaluation concluded that no prior NRC approval is necessary for these changes.

**Type:** Nuclear Station Modification (OD300050 / UFSAR Change 08-72)

**Title:** O2C24 Reload Core Design

**Description:** This activity installs the core designed for Oconee Nuclear Station Unit 2 Cycle 24, which is the first Oconee core design to employ Mk-B-HTP fuel. Mk-B-HTP fuel and the methods necessary to evaluate have been granted NRC approval as published in the DPC-NE-2015-PA methodology report. The O2C24 Reload Design Safety Analysis Review (REDSAR), performed in accordance with Engineering Directives Manual EDM-501, "Engineering Change Program for Nuclear Fuel", and the O2C24 Reload Safety Evaluation confirm the UFSAR accident analyses remain bounding with respect to predicted O2C24 safety analysis physics parameters (SAPP), and fuel thermal and mechanical performance limits. The SAPP method is described in the DPC-NE-3005-PA methodology report.

Except for the Mk-B-HTP fuel design change, the O2C24 core reload is similar to past cycle core designs, with a design generated using NRC approved methods. The O2C24 Core Operating Limits Report (COLR) was prepared in accordance with Technical Specification 5.6.5. Additionally, applicable Technical Specifications have been reviewed and no additional changes are required for the operation of O2C24. The 10CFR50.59 evaluation, which concluded that no prior NRC approval is necessary for O2C24 operation, combined with the DPC-NE-2015-PA methodology report will serve as justification to update the UFSAR per 10CFR50.71(e).

**Type:** UFSAR Change 08-28

**Title:** Containment Overpressure Revision to UFSAR Section 6.1.3

**Description:** This activity is a change to the UFSAR - described safety analysis and the NPSH analyses for the LPI and BS pumps. Inputs and assumptions and descriptive details from the long term containment response analysis and the NPSH analyses have changed. The evaluation methodology for NPSH analysis has changed in the following ways:

- Minimum available containment overpressure decreased from 2.2 psi to 0.44 psi.
- Two alternative references are used for determining piping and pipe fitting friction losses.
- Composite hydraulic modeling has been replaced by train-specific modeling.

- Instrument uncertainty considerations are no longer applicable. Containment overpressure is credited only in the sump recirculation mode of operation for the BS and LPI systems. This mode of operation is used exclusively for accident mitigation. Therefore, changes to containment overpressure do not have the potential to impact the normal (non-emergency) operation of either system. It follows that such changes also cannot increase accident probability or create new accidents.

NPSH analyses for the LPI and BS pumps demonstrate that the NPSH requirements for these pumps are satisfied with the revised containment overpressure credit. With adequate NPSH, there is no adverse impact on pump performance. Therefore, the probability and consequences of malfunction of these pumps is not affected, and any malfunctions would be expected to have the same results as previously analyzed in the UFSAR.

Changing the magnitude of overpressure credit is determined not to be a departure from the methodology used to evaluate NPSH. While there has been a change in an element of the methodology, the change meets the conditions for conservatism stipulated in industry guidance, NEI 96-07, Rev. 1. Changes meeting these conditions are not considered departures from approved methodology. Alternative approaches to evaluating friction losses in piping and fittings utilize commonly accepted industry references which have been approved by the NRC for similar applications. Such methodology changes are not considered departures from approved methodologies. Removal of instrument uncertainty and usage of a train-specific modeling approach in the NPSH analyses are trivial methodology changes which have been implemented for accuracy purposes only and can be easily judged to produce results which are essentially the same as for previous methods. Based upon the above reasoning, all questions in the evaluation have been answered "NO". Therefore, it is concluded that this activity can be performed under the provisions of 10CFR50.59 and prior NRC approval is not required.