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10 CFR 52 App. D
10 CFR 50.59

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Southern Nuclear Operating Company
Vogtle Electric Generating Plant Units 3 and 4
Report of 10 CFR 50.59 Changes, Tests and Experiments
and 10 CFR 52 Appendix D Departure Report

Ladies and Gentlemen:

The U.S. Nuclear Regulatory Commission (NRC) issued the Vogtle Electric Generating Plant (VEGP) Units 3 and 4 combined licenses (COLs) (License Nos. NPF-91 and NPF-92, respectively) to Southern Nuclear Operating Company (SNC) on February 10, 2012.

In accordance with 10 CFR 50.59(d)(2), VEGP Units 3 and 4 is required to submit a report to the NRC containing a brief description of any changes, tests or experiments made pursuant to 10 CFR 50.59(c), including a summary of the evaluation of each. This 10 CFR 50.59 report is for the 6-month period ending June 7, 2012. During that period there were no changes, tests or experiments made pursuant to paragraph (c) of 10 CFR 50.59.

Additionally, in accordance with 10 CFR 52, Appendix D, paragraphs X.B.1 and X.B.3.b, VEGP Units 3 and 4 is required to submit a semi-annual report to the NRC containing a brief description of any plant-specific departures from the DCD, including a summary of the evaluation of each. This 10 CFR 52 Appendix D departure report, provided as Enclosure 1 to this letter, is for the period of December 15, 2011 to June 7, 2012.

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If you have any questions regarding this letter, please contact Mr. Wes Sparkman at (205) 992-5061.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

C. R. Pierce

C. R. Pierce
Regulatory Affairs Director

CRP/GAB/dmw

Enclosure 1: Vogtle Electric Generating Plant (VEGP) Units 3 and 4, Semi-Annual Departure Report for the Period of December 15, 2011 to June 7, 2012

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Southern Nuclear Operating Company

ND-12-1274

Enclosure 1

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Semi-Annual Departure Report

for the Period of

December 15, 2011 to June 7, 2012

LDCR / Departure Number: 2012-007

Title: Simplification of DCD P&ID Figures

Activity Description:

Detailed figures were provided in the generic AP1000 DCD, Revision 19. This activity substitutes 46 DCD piping & instrumentation diagrams with simplified schematics such that all required information is maintained. It has been verified that the simplified figures together with associated DCD and FSAR text continue to provide sufficient understanding of design bases, safety analyses and facility operation. There is no change to the system design described in the plant-specific DCD figures or supporting analysis. The actual system piping and instrumentation diagrams are not altered by this activity. This is a change to the level of detail documented in the plant-specific DCD. The figure simplification effort removes extraneous detail from the plant-specific DCD figures.

Summary of Evaluation:

There is no design function related to replacing existing generic AP1000 DCD figures with simplified figures. This simplification effort does not impact the design function of any SSC The actual system piping and instrumentation diagrams are not altered by this activity. This activity only simplifies plant-specific DCD figures; no new design changes are proposed. The facility is not being changed by this activity. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-008

Title: Safety-Related Concrete Admixtures

Activity Description:

This activity updates admixtures used in the production of concrete structures and modules described in the plant-specific DCD. Admixtures are used to obtain certain concrete characteristics that would not be obtainable with a plain mix. The types of concrete admixtures are being revised to account for technology improvements that will allow for the production of conventional concrete and Self-Consolidating Concrete (SCC). Type B, C and F admixtures are added, Type D and vinsol admixtures are removed and Type A admixture use is clarified in the plant-specific DCD. These changes are consistent with ASTM C494, ACI 349, and ACI 237R.

Summary of Evaluation:

By allowing the use of admixture types B, C, and F and preventing the use of type D and vinsol, the concrete's design function is unchanged. The use of Self-Consolidating Concrete has no effect on structural analysis. There is no adverse impact on concrete parameters such as strength, density, and durability. Additionally, these admixtures do not impact security barriers or radiation protection and shielding safety analyses. These changes do not affect any procedure, method of evaluation, or test and experiment. The changes do not have an impact on ex-vessel severe accident consequences and do not impact core concrete interactions or containment pressurization due to core concrete interactions. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-013

Title: CA Module Liner Plate Material Change

Activity Description:

During design finalization, the design of the CA01, CA02, CA05, and CA20 structural wall modules was changed to use ASTM A572 steel for liner plates in lieu of ASTM A36 steel. The requirements of American Institute of Steel Construction (AISC) N690-1994 continue to apply to the design of the attachments to the higher strength plates.

In generic AP1000 DCD, Subsection 3.8.3.6, the reference to A36 steel for structural module plates permits the use of "steel with equal or better material properties." Since A572 steel has equal or better material properties as compared to A36 steel, this portion of the plant-specific DCD remains unchanged.

Summary of Evaluation:

This activity eliminates the specific requirement to use A36 steel for containment internal modules and modules in the auxiliary building. The use of the higher strength A572 plate and resultant change in shear stud spacing satisfy the requirements and acceptance criteria in AISC N-690-1994 and plant-specific DCD Subsection 3.8.3.1.3. The geometric configuration, thickness, and strength of these structures are not adversely affected. There is no change in the design, analysis, or operation of the RCS or other plant systems. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-014

Title: RNS Pump Seal Cooler Nozzle Change

Activity Description:

The RNS pump seal coolers are modified to add additional 1" vent and drain lines to the RNS process side of the pump seal coolers. These vent and drain lines include ASME-III isolation valves, because they are part of the RNS pressure boundary. The additional 1" vent and drain lines are needed because of the possibility for air to be trapped in the seal cooler. Trapped air would degrade the ability of the cooler to remove heat. The pump seal cooler is considered its own entity, separate from the pump.

Summary of Evaluation:

This change does not adversely impact the design function of the RNS pump seal coolers, which is to cool the seals to prevent seal degradation and pump outleakage. The change adds an additional vent and drain line to the coolers in order to remove any air which may accumulate inside the cooler. Accumulated gases could adversely affect the ability of the cooler to perform its design function of removing heat. Integrity of the RNS pressure boundary is maintained by the addition of AP1000 Code Letter C isolation valves on each of the added vent and drain lines. (Note that these vent and drain lines are being added outside of the reactor coolant pressure boundary (RCPB); therefore, RCPB integrity is unaffected by this change.) As defined in Table 3.2-1 of the plant-specific DCD, AP1000 Code Letter C is equivalent to ANS Equipment Safety Class 3, RG 1.29 Seismic Design Requirement I, and ASME Code Section III, Class 3. These valves will be designed to RG 1.26 NRC Quality Group C and 10 CFR 50, Appendix B requirements. They are in Environmental Zone 6 and function code PB (pressure boundary). Inspection, test and maintenance requirements are as defined in Table 3.2-1 of the plant-specific DCD. These valves are normally closed valves which are only opened for maintenance purposes to fill and drain the system. These maintenance procedures will include controls to verify that these valves are closed when maintenance is complete. The system pipe stress analyses consider the addition of these 1-inch lines and valves. The changes have no effect on any other analysis. These changes do not affect any procedure, method of evaluation, or test and experiment. The changes do not have an impact on ex-vessel severe accident consequences. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-015

Title: Discrepancy in DCD Section 9.3.5.2.2 "Sumps and Drain Tanks"

Activity Description:

Plant-specific DCD Subsection 9.3.5.2.2 is being corrected to present the plant design and be consistent with plant-specific DCD Subsection 9.3.5.1.2. The plant-specific DCD contains an inconsistency in that Subsection 9.3.5.2.2 incorrectly states that each sump is fitted with a vent connection to exhaust potential sump gases into the Radiologically Controlled Area Ventilation System (VAS) exhaust system. The VAS is a ventilation system in the Auxiliary and Annex Buildings. The liquid radwaste system (WLS), as described in plant-specific DCD Subsection 9.3.5.1.2 accurately described the venting as the radioactive sump vents are directed to the ventilation system exhaust ducts serving the areas where the sump is located and that the containment sump vents directly to the containment. This activity corrects the Radioactive Waste Drain System (WRS) sump venting described in the plant-specific DCD in that the containment sump is vented to containment rather than the VAS.

Summary of Evaluation:

The plant equipment and design intent and philosophy have not changed. The inconsistency in the plant-specific DCD has been eliminated regarding a generalized statement about the WLS and WRS sump vents' repository. The AP1000 was designed with the correct vent philosophy provided in plant-specific DCD Subsection 9.3.5.1.2, so there is no impact on SSCs. The plant design has not changed, but the plant-specific DCD is being clarified regarding a generalized statement about the non-safety related WLS and WRS sump venting.

There is no effect on structural analysis. The change does not impact security barriers or radiation, protection and shielding safety analyses, nor does the change affect any procedure, method of evaluation, or test and experiment. The physical design of the sump vents has not changed, so there is no impact on ex-vessel severe accident consequences, containment venting and containment integrity. The VAS supply and exhaust ducts that ventilate the middle annulus are not affected by this departure and continue to be designed to be isolated for holdup and deposition of containment radioactive releases during a severe accident as discussed in the AP1000 Probabilistic Risk Assessment. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-016

Title: Implementation of Integrated FSAR (IFSAR) Revision 0

Activity Description:

The NRC issued VEGP Units 3 and 4 a Combined License (COL) on February 10, 2012, based on FSAR Revision 5 from Part 2 of the Combined License Application (COLA). That FSAR incorporated by reference (IBR) the AP1000 Design Control Document (DCD), Revision 19 and the Early Site Permit Application (ESPA) Site Safety Analysis Report (SSAR), as amended, and other documents from NEI. That FSAR included departures from the AP1000 generic DCD and variances from the ESPA SSAR, as amended, which were reviewed and approved by the NRC.

Rather than maintaining 3 separate licensing documents as modified by departures and variances, this activity integrates COLA Part 2, FSAR Revision 5 with DCD Revision 19 and the amended ESPA SSAR licensing documents, including departures and variances, into one composite Integrated FSAR (IFSAR). The resultant integrated FSAR will be issued as IFSAR, Revision 0.

During the integration, numerous editorial changes were required to allow the integrated FSAR to flow properly from section to section, have proper cross-referencing, consistent tables and figures, elimination of historical information related to COL information items in the DCD, elimination of locator text related to IBR information, renumbering, and other non-technical changes. Each category of change included a justification for deviation from the source licensing document.

Summary of Evaluation:

The changes to the FSAR, SSAR, and DCD that are being made to accomplish the integration into a single composite and comprehensive document are non-technical in nature. Each category of change (e.g. renumbering tables or figures) was identified by a Justification For Deviation (JFD) and each JFD was justified to be a non-technical change to the FSAR, SSAR or DCD. The technical content and licensing basis are not changed by this activity. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-017

Title: Turbine Building Changes

Activity Description:

This departure makes changes in the turbine building to the El. 82'-9" basemat area, concrete base pads, general layout arrangement, and various plant-specific DCD text changes for consistency with the Condensate Polishing System (CPS) resin rinse effluent design function. The turbine building El. 82'-9" basemat area is expanded north of column line 18 and south of column line 13.1. The concrete base pads that support structural columns 14, 15, 16, and 17 are lowered from El. 100'-0" to El. 90'-0", and a ditch has been created in the middle of the base pad for column 17. Stairwell S09 is removed, a new material handling elevator is added, and stairwell S11 is relocated to the Northeast corner of El. 82'-9". Various DCD text changes are made to account for CPS resin rinse effluent being discharged to the turbine building sumps, and an additional pump is added to each sump to account for the additional volume and prevent overflowing.

Summary of Evaluation:

Implementing these changes has no adverse effect on structural analysis. The changes do not impact security barriers or radiation, protection and shielding safety analyses, nor does the change affect any procedure, method of evaluation, or test and experiment. There is no impact to ex-vessel severe accident consequences, containment venting, and containment integrity. The design functions of the turbine building and its structures, systems, and components as described in the plant-specific DCD or UFSAR continue to be met. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-019

Title: Professional Engineer (PE) Signatures for ACI 349 documents

Activity Description:

This departure clarifies the method of conformance to Regulatory Guide 1.142 in Appendix 1A of Chapter 1 and the endorsed ACI-349-01 Section 1.2.1 which states that "Copies of structural drawings, typical details, and specifications for all reinforced concrete construction shall be signed by a licensed engineer." This departure clarifies that this may be accomplished by a simple signature as long as the signature is confirmable by documentation to be that of a licensed professional engineer. This departure also includes minor editorial changes that do not affect the design information or design process but are included for consistency within the document.

Summary of Evaluation:

This departure does not change the conformance with ACI-349 but merely indicates the method by which the conformance is being accomplished. This change has no effect on structures, systems or components. Additionally, the change does not impact security barriers or radiation, protection and shielding safety analyses. The change does not affect any procedure, method of evaluation, or test and experiment. The change does not have an impact on ex-vessel severe accident consequences and does not impact containment pressurization. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-020

Title: Concrete Beneath Sumps KB10 & KB13 and Elevator Pits

Activity Description:

The plant-specific DCD, Subsection 3.8.5.4.2 states the concrete basemat is placed in a single pour. To prevent voiding beneath sumps and around elevator pits in the Nuclear Island, concrete will be placed underneath these areas prior to the single pour of the remaining basemat. The plant-specific DCD text will be clarified to state the concrete in these four areas is poured prior to the remaining single pour.

Summary of Evaluation:

By clarifying the areas around the sumps and elevator pits in the Auxiliary Building may be poured prior to the remaining single pour, the Nuclear Island basemat's design function is unchanged and not adversely affected. The activity will have negligible effect on the overall structural integrity, settlement, or construction sequence. Placement of concrete locally below elevation 66'-6" in the sump and pit areas prior to the overall basemat concrete placement does not affect the assumptions or analyses performed. Pre-placement of the sump and pit concrete has the benefit of allowing for conventional concrete placement and vibration techniques, while minimizing the potential for voids in these areas. The changes do not have an impact on ex-vessel severe accident consequences and do not impact core concrete interactions or containment pressurization due to core concrete interactions. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-021

Title: Codes and Standards to be used for Civil/Structural Fabrication

Activity Description:

Plant-specific DCD Subsections 3.8.3.2 and 3.8.4.2 are updated to remove codes and standards listed in the plant-specific DCD that are referenced within the top level (parent) structural design codes. These top level codes are ACI 349-01 and AISC N690-1994. The codes and standards removed are related to welding procedures and concrete specifications. Plant-specific DCD Subsections 3.8.3.2 and 3.8.4.2 are updated to remove the revisions or dates for standards and specifications related to the detailing, placement, and specification of concrete. These standards and specifications do not include design or analysis requirements. Reference to the NCIC weld acceptance criteria is removed from plant-specific DCD Subsections 3.8.3.2 and 3.8.4.2 since it is referenced by the top level codes. The top level structural design codes (ACI 349-01 and AISC N-690-1994) are identified as Tier 2* information in these plant-specific DCD sections and are not changed.

Construction and fabrication requirements for seismic Category II structures are removed from plant-specific DCD Subsection 3.7.2 since this subsection is about seismic analysis and not construction requirements. The seismic interaction between seismic Category I and seismic Category II structures are covered in Subsection 3.7.2.8 and are not changed.

Summary of Evaluation:

The codes and standards specified in the plant-specific DCD and the daughter standards cited in the parent codes can provide multiple versions of the standards and specifications for fabrication. The changes to the plant-specific DCD clarify the standards and specifications to use. As a result of advances in industry standard practices and material manufacture, more recent versions of the standards and specifications should be specified for the purposes of fabrication and construction.

Clarification of the requirements for fabrication and construction of steel structures does not change the design, analysis, or configuration of the AP1000 Seismic Category I and Seismic Category II structures. There is no adverse effect on the design function of these structures. The clarification of the requirements for fabrication and construction of steel structures has no impact on the procedures used to operate and control the AP1000 plant. The clarification of the requirements for fabrication and construction of steel structures has no impact on the design, analysis, and acceptance criteria for the AP1000 structures. The clarification of the requirements for fabrication and construction of steel structures does not require testing or an experiment. The clarification of the requirements for fabrication and construction of steel structures does not alter the response of systems, structures, and components in the AP1000 to an ex-vessel severe accident. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-025

Title: WLS Containment Sump Module and PSS Containment Atmosphere Radiation Monitor Seismic Requirements

Activity Description:

This activity is being made to enhance the functionality of containment sump level instrumentation post-Safe Shutdown Earthquake (SSE). Prior to this change, there was an inconsistency between the Reactor Coolant Pressure Boundary (RCPB) leak detection functionality and seismic classification and the plant-specific DCD design requirements. Specifically, the containment sump module (KQ11), three containment sump level instruments (WLS-LT-034, WLS-LT-035, WLS-LT-036), and Primary Sampling System (PSS) radiation particulate monitoring instruments require modifications to comply with the current licensing commitments regarding plant operation post-SSE.

Containment sump level monitoring, through the containment sump level instruments, is clarified to be the primary method of RCPB leakage detection in containment after an SSE. It provides conformance to position 6 of Regulatory Guide 1.45, although using different technology than envisioned in that guidance (sump level rather than airborne radioactivity). The containment sump level instruments indication in the main control room display remains non-seismic; however, SC-I local readout of the instruments is provided outside of containment and is qualified to be operable post-SSE.

The Containment Atmosphere Radioactivity Monitor ^{18}F particulate monitor remains seismic Category I, but the remaining tubing is not seismically qualified. This leakage detection system can be reasonably expected to remain functional following seismic events of lesser severity than the SSE; however, no special qualification program is used to assure operability under such conditions and no credit is taken for its functionality. It is clarified that the Containment Atmosphere Radioactivity Monitor is not the instrument used to provide RCPB leakage detection following seismic events that do not require plant shutdown in conformance to the intent of position 6 of Regulatory Guide 1.45; conformance to this position is provided by the containment sump level via the seismic Category I Containment Sump Level Monitoring system.

Summary of Evaluation:

By enhancing the functionality of the containment sump level instrumentation post-SSE, the Reactor Coolant Pressure Boundary leakage detection function is unchanged. There is no effect on structural analysis. Additionally, enhancing the functionality of the containment sump level instrumentation post-SSE does not impact security barriers or radiation, protection and shielding safety analyses. These changes do not affect any procedure, method of evaluation or test and experiment. RCPB leakage detection instrumentation is not credited in the ex-vessel severe accident assessment. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-026

Title: Containment Vessel: Structural Course and Plate Layout

Activity Description:

Design details and descriptions, as stated in the plant-specific DCD, contained implied and stated fabrication and construction details including weld seams, course elevations, plate geometry, and attachments for the containment air baffle for the Containment Vessel. Plant-specific DCD Tier 2 Subsections 3.8.2.1.1, 3.8.2.6, and 3.8.4.1.3, and plant-specific DCD Figures 3.8.2-1 Sheet 1 and 3.8.4-1 Sheet 1 are revised to remove details regarding the fabrication and erection of the Containment Vessel. These details and figures are not intended to show required design and fabrication details. These changes are necessary to ensure that the plant-specific DCD description is consistent with actual design and fabrication methods.

Design details and descriptions, included in the plant-specific DCD, provide fabrication and construction details (e.g., size and number of panels and detail design of supports and attachment) for the Containment air baffle and are unnecessary detail in the plant-specific DCD. Plant-specific DCD Tier 2 Subsection 3.8.4.1.3 and plant-specific DCD Figure 3.8.4-1 Sheet 1 are revised to remove details regarding the fabrication and construction of the containment air baffle. These details and figures are not intended to show required design and fabrication details. These details are inconsistent with the design finalization of the baffle and the fabrication details of the baffle, baffle panels, and supports.

Summary of Evaluation:

The removal of the design and fabrication details does not adversely affect the containment vessel and containment air baffle design functions. It does not affect the method of performing or controlling design functions, nor does it have an effect on an evaluation for demonstrating that intended design functions will be accomplished. It removes detailed plant-specific DCD information that is inconsistent with design and fabrication details.

The removal of fabrication and construction details does not impact the design function of any SCC. The pressure retention and structural integrity function of the containment vessel is not adversely affected. The containment air baffle design function of providing for an air flow path for the passive containment cooling system is not adversely affected. The containment vessel design function to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents is not adversely affected. The facility is not being adversely changed by this activity. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-035

Title: DCD Clarifications Needed to Support Construction

Activity Description:

The change activity clarifies requirements and commitments in the licensing basis for concrete and structural steel used in the nuclear Island structures. Departures to plant-specific DCD Subsection 3.8.4.6 and 19.58.2.2 and Tables 3.8.4-3, 14.3-5, and 19.59-18 are included as part of the activity. The information clarified includes test age of concrete, conformance with ACI standards, aggregate testing, use of air entraining admixtures, incorporation of waterstops, and ASTM specification tabulated.

Summary of Evaluation:

The change activity clarifies requirements and commitments in the licensing basis for concrete and structural steel used in the nuclear Island structures. The nuclear island structures, consisting of the containment vessel, shield building, and auxiliary building are founded on the 6-foot-thick, cast-in-place, reinforced concrete basemat foundation. The primary functions of the nuclear island structures are to provide support, protection, and separation for the seismic Category I mechanical and electrical equipment located in the nuclear island. The nuclear island structures provide protection for the safety-related equipment against the consequences of either a postulated internal or external event. The nuclear island structures are designed to withstand the effects of natural phenomena such as hurricanes, floods, tornados, tsunamis, and earthquakes without loss of capability to perform safety functions. The nuclear island structures are designed to withstand the effects of postulated internal events such as fires and flooding without loss of capability to perform safety functions.

The clarification of requirements and commitments in the licensing basis for concrete and structural steel used in the nuclear island structures will not have an adverse impact on the strength of the nuclear island structures or the response of the structure to internal and external loads, including seismic loads. The nuclear island structures, with the clarification of requirements and commitments in the licensing basis, remains in compliance with ACI 349. The clarification of requirements and commitments in the licensing basis has no impact on design, analysis, or operation of safety related systems and components. The clarification of requirements and commitments in the licensing basis has no impact on plant operating procedures or on the control of the reactions in the core. The clarification of requirements and commitments in the licensing basis has no impact on the finite element analysis methods used to analyze the nuclear island structures. The analysis of the reactor coolant system and core to normal operation and postulated accident conditions is not impacted by the clarification of requirements and commitments in the licensing basis. The clarification of requirements and commitments in the licensing basis for concrete and structural steel used in the nuclear island structures does not alter the assumptions or results of the ex-vessel severe accident assessment. The clarification of the requirements and commitments in the licensing basis for concrete and structural steel used in the nuclear Island structures does not result in modification, addition to, or removal of a structure, system, or component (SSC) such that a design function is adversely affected, has no impact on plant operating procedures or on the control of the reactions in the core design function, does not result in an adverse change to a method of evaluation or use of an alternate method of evaluation, does not represent a test or experiment outside the reference bounds of the design basis, and does not alter the assumptions or results of the ex-vessel severe accident assessment. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.

LDCR / Departure Number: 2012-036

Title: DCD Tier 2 Chapter 3 Not Identified as Impacted in DCP-332

Activity Description:

The principal construction code of the WGS Gas Cooler is categorized as ASME VIII/TEMA in plant-specific DCD, Tier 2, Table 3.2-3. As a result of a previous design change, the WGS Gas Cooler was changed from a shell and tube heat exchanger to an off-the-shelf, dual tube coil heat exchanger. When this design change was originally incorporated into the generic AP1000 DCD, Tier 2 Table 3.2-3 was not updated to reflect the principal construction code of the new heat exchanger. The correct principal construction code for the new heat exchanger is "Manufacturer Std."

Summary of Evaluation:

This change involves modifying plant-specific DCD Tier 2 Table 3.2-3 to accurately reflect the principal construction code of the WGS Gas Cooler. The design function of the WGS remains unchanged and the quality and construction/quality standards are not adversely affected. Therefore, this change does not adversely impact the design function of the WGS. This change does not affect any procedure, method of evaluation, or test and experiment. This activity does not impact a design feature credited in the ex-vessel severe accident assessment. A 10 CFR 50.59/10 CFR 52 Appendix D Section VIII review determined that no prior NRC approval is required.