Omaha Public Power District

444 South 16th Street Mall
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June 17, 2014
LIC-14-0077

10 CFR 50.71(e)
10 CFR 50.59
10 CFR 54.37(b)

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

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40
NRC Docket No. 50-285

Subject: 10 CFR 50.59 Report, Quality Assurance (QA) Program Changes, Technical Specification Basis Changes, and Updated Safety Analysis Report (USAR) Revision for Fort Calhoun Station (FCS), Unit No. 1

Reference: 1. Letter from OPPD (J. B. Herman) to NRC (Document Control Desk), *10 CFR 50.59 Report, Quality Assurance (QA) Program Changes, Technical Specification Basis Changes, and Updated Safety Analysis Report (USAR) Revision for Fort Calhoun Station (FCS), Unit No. 1*, dated June 18, 2012 (ML12226A171) (LIC-12-0076)

In accordance with 10 CFR 50.59(d)(2), the Omaha Public Power District (OPPD) submits Attachment 1 as the report of changes, tests, and experiments performed pursuant to 10 CFR 50.59 for Fort Calhoun Station (FCS), Unit No. 1. Attachment 2 is provided to describe Quality Assurance (QA) Program (USAR Appendix A) changes, as required by 10 CFR 50.54(a)(4)(i). Attachment 2 also contains a description of revised regulatory commitments that require Commission notification in accordance with NEI 99-04, "Guidelines for Managing NRC Commitment Changes," and modifications to the USAR made in accordance with NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports." In accordance with FCS Technical Specification 5.20.d, Attachment 3 provides a brief summary of the Technical Specification Basis Changes (TSBCs) made since the previous submittal (Reference 1) and Attachment 4 includes a copy of the revised pages. This information covers the period of June 19, 2012 through June 13, 2014.

The USAR is reissued in electronic format. Pursuant to 10 CFR 50.71(e) and 10 CFR 50.4(b)(6), enclosed is one (1) original CD-ROM of the FCS USAR, which incorporates changes to the USAR made since the previous submittal (Reference 1) and includes changes made under the provisions of 10 CFR 50.59 but not previously submitted to the Commission. Attachment 5 contains a list of the files on the CD-ROM. The Senior Resident Inspector is provided with an updated copy of the USAR by the FCS distribution process.

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In accordance with 10 CFR 54.37(b), a review of structures, systems, and components (SSCs) was performed; the summary report is contained in Attachment 6.

I certify that the information in this submittal accurately presents changes made since the previous submittal necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirements, and identifies changes made under the provisions of 10 CFR 50.59 but not previously submitted to the Commission.

No commitments to the NRC are made in this letter.

If you should have any questions, please contact Mr. Bill Hansher at (402) 533-6894.

Sincerely,

Louis P. Cortopassi
Site Vice President and CNO

Attachments:

1. Changes, Tests, and Experiments Performed Pursuant to 10 CFR 50.59
2. Quality Assurance Program Changes and Regulatory Commitments Revised in Accordance with NEI 99-04
3. Summary of Technical Specification Basis Changes (TSBC)
4. TSBC Pages
5. List of Files on CD-ROM
6. Summary Report Pursuant to 10 CFR 54.37(b)

Enclosure: CD-ROM (1) of USAR Sections and Figures

LPC/MLE/mle

c: M. L. Dapas, NRC Regional Administrator, Region IV
J. K. Rankin, NRC Project Manager
J. C. Kirkland, NRC Senior Resident Inspector (w/o Enclosure)

Changes, Tests, and Experiments Performed Pursuant to 10 CFR 50.59

Abbreviations and Acronyms:

AFW – Auxiliary Feedwater	LCO – Limiting Conditions for Operation
ANSI – American National Standards Institute	LOCA – Loss of Coolant Accident
AOP – Abnormal Operating Procedure	LPSI – Low Pressure Safety Injection
AOR – Analysis of Record	LTOP – Low Temperature Overpressure Protection
AR – Action Request	MCC – Motor Control Center
BAST – Boric Acid Storage Tank	MFW – Main Feedwater
BTP – Branch Technical Position	MH – Manhole
CCW – Component Cooling Water	msl – Mean Sea Level
CD-ROM – Compact Disk Read-Only Memory	NEI – Nuclear Energy Institute
CEA – Control Element Assembly	NLI - Nuclear Logistics Incorporated
CEAPIS – CEA Position Indication System	NRC – Nuclear Regulatory Commission
CFR – Code of Federal Regulations	OI – Operating Instruction
CIV – Containment Isolation Valve	OPPD – Omaha Public Power District
COLR – Core Operating Limits Report	PDIL – Power Dependent Insertion Limit
CQE – Critical Quality Element	PRC – Plant Review Committee
CR – Condition Report	PSAR – Preliminary Safety Evaluation Report
CRS – Control Room Supervisor	QA – Quality Assurance
CS – Containment Spray	QATR - Quality Assurance Topical Report
CW – Circulating Water	QR – Qualified Reviewer
DCS – Distributed Control System	RCA – Root Cause Analysis
DG – Diesel Generator	RCS – Reactor Coolant System
EA – Engineering Analysis	RFO – Refueling Outage
EC – Engineering Change	RPS – Reactor Protective System
EOP – Emergency Operating Procedure	RSG – Replacement Steam Generators
ERFCS – Emergency Response Facility Computer System	RTD – Resistance Temperature Detector
FCS – Fort Calhoun Station, Unit No. 1	RW – Raw Water
FCSG – Fort Calhoun Station Guideline	SAO – Safety Analysis for Operability
FSAR – Final Safety Analysis Report	SARC – Safety Audit and Review Committee
HEPA – High Efficiency Particulate Air	SDC – Shutdown Cooling
HZP – Hot Zero Power	SER – Safety Evaluation Report
I&C –Instrumentation & Control	SFP – Spent Fuel Pool
INPO – Institute of Nuclear Power Operation	SG – Steam Generator
IST – In-service Testing	SGBD – Steam Generator Blowdown
LBLOCA – Large Break Loss of Coolant Accident	SI – Safety Injection
SIRWT – Safety Injection Refueling Water Tank	SM – Shift Manager

SO – Standing Order	
SR – Surveillance Requirement	
SRP – Standard Review Plan	
SSC – Structures, Systems and Components	
ST – Surveillance Test	
TM – Temporary Modification	
TS – Technical Specification	
TSBC – Technical Specification Basis Change	
UFSAR – Updated Final Safety Analysis Report	
USAR – Updated Safety Analysis Report	
VCT – Volume Control Tank	
WO – Work Order	

2012 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
EC 56618	One-Time Change to CCW Base Line Curve Procedure SE-ST-CCW-3002 for Vendor Testing	**Activity Description:** SE-ST-CCW-3002 is being revised as a one-time change (not to be incorporated) in order to conduct Vendor pump performance testing. This testing will enable the Vendor to generate pump curves that more accurately represent pump performance. The procedure change will add steps that close the discharge valves of the offline pumps and will place the offline pump control switches in pull to lock. The procedure change also includes a provision to credit a dedicated operator to ensure the offline pumps remain available per SO-O-21. Closing the discharge valves will ensure that no unmeasured flow is circulating back through the offline pumps. Guidance is also being added that will allow Operations to adjust CCW Pump test flow above 5500 gpm. 5500 gpm is where the current certified pump performance curve ends. This guidance is necessary as the Vendor pump testing will evaluate pump performance in the region beyond the present certified pump performance curve. This testing is necessary as the pumps have been operated beyond their existing pump performance curves (CR 2012-03254) and because historical and current pump surveillance data is highly unreliable, non-repeatable and cannot be used to determine if the CCW pumps are degraded (CR 2012-05846). The testing of the pumps will be performed with the on-site assistance of the CCW pump manufacturer (Flowserve). Flowserve will be providing the engineering services during the testing and will use the test data to develop the new pump performance curves for each of the CCW pumps. The goal of this test is to obtain multiple test points with the final point at around 6000 gpm (±5%). In the screening process, it was determined that an evaluation was necessary due to testing the pumps beyond their existing pump curve flows, and also because closing the offline CCW Pump discharge valves will affect their autostart capability.

2012 Evaluations

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		Summary of Evaluation: There is no USAR requirement for maximum CCW pump flow for a single pump. However, in order to create new pump curves, the CCW pumps will be operated beyond their existing pump curves which are part of the design basis. The pump manufacturer, Flowserve, will be onsite to take the necessary data to create the new CCW pump performance curves. Detailed work instructions will enable the craft to properly set up Flowserve equipment. Once the equipment has been set up, the procedure SE-ST-CCW-3002 will be performed to take the necessary data for the new pump performance curves. This procedure will be performed three times over the course of three days. One test performance for each pump. There are no CCW system restrictions for the CCW flow in the 6000 gpm range, since the CCW system is designed for a simultaneous operation of 3 pumps. Also, the CCW pumps have been demonstrated via operating experience to operate at higher flows, e.g. 6000 gpm. The review of surveillance tests identified that a determination of pumps' degradation cannot be made without additional testing (CR 2012-05846) which will performed by Flowserve. However, recent maintenance activities (WO 420940) found no indication of impeller or pump casing damage and past surveillances do not indicate any abnormal vibration or abnormal noise. During the performances of SE-ST-CCW-3002 as revised by EC 56618, the plant staff with assistance of the Flowserve specialized test personnel will be evaluating pump performance at the lower flow rates prior to making a decision to advance to each higher flow test point. The decision will be based on considerations such as: relative performance (head and flow) of individual test points vs. the original certified curve, measured pump and motor bearing vibrations, motor power consumption and motor temperature. Past performance of the CCW pumps beyond their existing pump performance curves was evaluated by several industry pump experts (Healy Engineering, Inc., May 24, 2012,

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		OPPD Fort Calhoun Component Cooling Water (CCW) Pumps, and Exelon's pump expert who provided acceptability of operation of the CCW pumps beyond their existing pump performance curves. Additionally, during the test it will be necessary to close the discharge valves of the other two stand-by pumps and to place their control switches in pull-to-lock. These actions will be performed by the plant Operations personnel that will be present during the test both locally at the pump and in the control room. If needed Operations would be able to put the stand-by pumps back in service within minutes if an unexpected pump failure occurred. Since the current safety related plant heat loads are very low, this short interruption of flow will have a negligible impact on plant safety. As documented above, obtaining of the CCW pump performance beyond their existing pump performance curves will be done under controlled conditions based on the review of pump performance at lower flow rates which are within their existing pump performance curves and utilizing the input from the Flowserve test experts who will be on-site during the test.
EC 56864 FCDRs 56996, 57507	Intake Structure Medium Voltage Termination (10 CFR 50.59 description & evaluation summary is from Rev. 1)	**Activity Description:** As a requirement for cable testing for medium voltage motors AC-10A, AC-10B-M, AC-10C and AC-10D-M in the Intake Structure, the motor terminations were required to be lifted and reterminated. In order to properly reterminate the motor connections, the use of Raychem connection kits is the preferred method to insulate the connection. The alternate method would be to use tape. Since FCS does not currently have a taping procedure, a new taping procedure would need to be written and approved for use by the station. The use of Raychem connection kits is not permitted to be installed over existing tape; therefore the existing tape that was used on the cable lugs was required to be removed. When the existing tape was removed, it was discovered that extensive damage was present on the cable insulation and excessive bare conductor was exposed. Due to the

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		location of the insulation damage, the cables are not long enough to allow the cables to be cut back and new lugs installed. Additionally, the cable shields were found to be not properly connected at the motor terminal box. In order to address the cable damage in the motor termination boxes, new cables will be installed from the existing junction boxes in the circulating water pump room back to the motor termination boxes. The new cables will be terminated to the existing cables in the junction boxes. In order to install these new cable extensions, the existing EYS fittings, located near the wall in the raw water pump room, have to be removed. If the fittings cannot be uncoupled from the conduit running through the wall, the conduit will have to be cut off. In either case, new EYS fittings are to be installed on the existing conduit with new 3" flex conduit installed from the EYS fittings to the motor termination boxes. All of the new cables are #2 AWG single conductor. In addition, the 90 degree fittings and rigid steel conduit located near the wall in the circulating water pump room will be removed and replaced with 2" flex conduit from the wall penetration coupling to the existing junction boxes to address bending radius issues. The cable connections in the junction boxes will be addressed by having both the existing and new cables terminated using Burndy lugs and Raychem termination kits. After the lugs are bolted together, the entire connection will be insulated using Raychem connection kits. The shield wires on the new cables (Unishield) will have a #12 lug applied to the drain wires which will be used to terminate the shield ground to the ground clamp of the existing cables in the junction box. The cable connections in the motor terminal boxes will be completed by using a Raychem termination kit on the new cable (Unishield) and a 90 degree lug to connect to the existing motor leads. The bolted connection made between the existing motor leads and new cables will then be insulated using a safety-related Raychem kit as seen on drawing SK-EC-56864-01 and SK-EC-65864-02. The shield wires on the new cable (Unishield) in the motor termination box will be terminated to the ground lug in the motor termination box.

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		This evaluation is based on the conservative "Yes" answer to screening question 1 "Does the proposed activity involve a change to an SSC that ADVERSELY affects a UFSAR Described DESIGN FUNCTION?" Therefore, the answers to the questions are focused on that aspect of the change. **Summary of Evaluation:** This activity can be implemented without obtaining a License Amendment. The new cable terminations do not introduce the possibility of a change in the frequency of occurrence of an accident or likelihood of a malfunction because the failure of a cable termination is not an initiator of an accident or malfunction. The new cable terminations do not introduce a change in the consequences of an accident or malfunction. A cable termination failure would result in the same loss of Raw Water Pump A, B, C or D as in the current design. The failure of the cable termination is considered no different than the failure of the original cable. The cable separation criteria are maintained in accordance with the criteria in USAR 8.5. Safety-related cables are used. These terminations are housed in seismically qualified boxes and the change of weight was evaluated to show that the Junction box and motor terminal box qualification is maintained. The new cable terminations do not affect any parameters that are outside the reference bounds for the design of a fission product barrier. The new cable terminations modify cable to Raw Water Pumps A, B, C & D motors but do not involve a method of evaluation as defined in the USAR. USAR Section. 8.5, Figure 8.5.1, note #19 states "Cable splicing in cable trays is used only for connection of incoming and outgoing containment electrical penetration conductors." The applicable drawings identify the cable terminations and their locations. None of these cable terminations are within a cable tray. Therefore, this USAR requirement is not applicable.

2012 Evaluations

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EC 55358	CCW Pump Motor Cable & Conduit Upgrade	**Activity Description:** CR 2012-03544 identifies insulation and jacket cracking on the feeder cables to the Component Cooling Water pump motors AC-3A-M, AC-3B-M and AC-3C-M at the motor terminal boxes. This cracking was due to training radius violations exerting excessive stresses on the cables. Additionally, further training radius violations were discovered at the conduit fittings above the terminal boxes. The purpose of this modification is to remove all the sections of the cables that have been subjected to the training radius violations and install new splices to reconnect to the motors. The cables will be restored by removing all portions of the cable that are cracked or have been subjected to the training radius violations. The cables are being cut upstream of the existing LB conduit fittings where the other training radius violations are observed. New junction boxes are being installed in-line with the conduit upstream of the LB fittings in order to house the cable splices. This evaluation is based on the conservative "Yes" answer to screening question 1 "Does the proposed activity involve a change to an SSC that ADVERSEL Y affects a UFSAR Described DESIGN FUNCTION?" Therefore, the answers to the questions are focused on that aspect of the change. "Since the activity adds a new connection, the new connection is conservatively considered a potential failure point that did not exist before and is thus assumed to be 'adverse' as compared to the original configuration." **Summary of Evaluation:** This activity can be implemented without obtaining a License Amendment. The new cable and cable terminations do not introduce the possibility of a change in the frequency of occurrence of an accident because the failure of a cable termination is not an initiator of an accident. The new cable and cable terminations do not introduce the possibility of a change in

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		likelihood of a malfunction. Safety related cables are used. The cables are sized for their application per station-approved procedures. The cable separation criteria are maintained in accordance with the criteria in USAR 8.5. These terminations are housed in seismically qualified boxes. The new cable and cable terminations do not introduce a change in the consequences of an accident or malfunction. A cable termination failure would result in the same loss of Component Cooling Water Pumps A, B, or C as in the current design. The failure of the cable termination is considered no different from the failure of the original cable. The new cable and cable terminations do not affect any parameters that are outside the reference bounds for the design of a fission product barrier. The new cable and cable terminations modify cable to Component Cooling Water Pumps A, B, & C motors but do not involve a method of evaluation as defined in the USAR. USAR Section 8.5, Figure 8.5.1, note #19 states "Cable splicing in cable trays is used only for connection of incoming and outgoing containment electrical penetration conductors." The applicable drawings identify the cable terminations and their locations. None of these cable terminations are within a cable tray. Therefore, this USAR requirement is not applicable.
EC35300	Diesel Generator Voltage Regulator Replacement	**Activity Description:** In 2008, OPPD commissioned Sargent & Lundy to evaluate the Fort Calhoun Emergency Diesel Generator (EDG) control circuits and voltage regulators for a potential modification or replacement. This study was done as a result of several condition reports (CR 200700725, CR200700756, CR 2007-3969 and CR 2007-4224,) that were written due to various EDG issues causing startup failures. The problems involved failure of diodes and control relays in the EDG control and voltage regulator circuits. The study proposed several solutions to make the system more robust and reliable. OPPD selected the static exciter regulator replacement option, in addition to replacing the field flash panel and the associated magnetics. This modification replaces the EDG excitation equipment, including the voltage regulator, current forcing,

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		and field flash panels, in addition to the transformers which provide the inputs to these panels. The generator field breaker (GFB) is also being replaced by this modification. FDCR 57789 makes an additional change to the droop circuit by installing a reactive current compensator in place of the droop resistor provided by NLI. Also, a switch is being installed on the droop CT secondary to enable droop functionality during testing, as is standard industry practice when the generator is paralleled to other sources. In order to be more fully in compliance with NRC Regulatory Guide 1.9, which references IEEE 387, Fort Calhoun has decided to perform a full rated load rejection test on the EDGs as part of the post-modification testing. This evaluation is based on the conservative "Yes" answer to screening question 1 "Does the proposed activity involve a change to an SSC that ADVERSELY affects a UFSAR Described DESIGN FUNCTION?" Therefore, the answers to the questions are focused on that aspect of the change. The specific adverse functions are: a. Since the replacement excitation equipment and GFBs are different from the existing components and are not a one for one replacement, they are introducing potential new malfunctions, this change is considered potentially adverse and evaluated in FC-154B. b. The installation of the reactive current compensator into the NLI voltage regulator is performed to match the original General Electric voltage regulator design as the provided NLI design provided inadequate function. The installation of the hand switch functions to enable the droop circuit during generator testing. The reactive current compensator and hand switch are new components that are not currently used and potential new malfunctions are introduced, these components are considered potentially adverse to the design of the voltage regulator.

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		Summary of Evaluation: This activity can be implemented without obtaining a License Amendment. The new excitation equipment, GFB, and reactive current compensators do not introduce the possibility of a change in the frequency of occurrence of an accident because the EDG is not an initiator of an accident. The new excitation equipment, GFB, and reactive current compensators do not introduce the possibility of a change in likelihood of a malfunction. The new equipment will have the same functional characteristics as the existing equipment as evaluated in the transient analyses. A failure modes and effects analysis (FEMA) concludes that the new components provided by NLI improve reliability over the existing components. The components are procured as safety related and are qualified for seismic event, elevated temperatures and EMI/RFI, as applicable. They are functionally tested prior to shipment and/or as part of site testing. The new excitation equipment, GFB, and reactive current compensators do not introduce a change in the consequences of an accident or malfunction. The new excitation equipment, GFB, or reactive current compensator circuit failure would result in the same loss of EDG functionality as a failure of the original voltage regulator. The effect of failure of the new excitation equipment, GFB, and reactive current compensators is considered no different from the effect of failure of the original voltage regulator. The new excitation equipment, GFB, and reactive current compensator circuit and subsequently the EDG do not affect any parameters that are reference bounds for the design of a fission product barrier. The new excitation equipment, GFB, and reactive current compensator circuit modifies the EDG but does not involve a method of evaluation as defined in the UFSAR.

2012 Evaluations

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EC 56556	USAR 14.16, "Containment Pressure Analysis"	**Activity Description:** CR 2011-6185 identified an error in the Main Steam Line Break (MSLB) Mass and Energy Release (MER) analysis used as input into the peak containment pressure and temperature calculation (FC07054) for a MSLB in USAR 14.16. Due to the use of an erroneously fast closure time for the feedwater regulating valves, too little feedwater is entering the steam generator, which non-conservatively minimizes the mass and energy release input into the containment response calculation. The limiting cases from these calculations were re-run, which changes the containment pressure and temperature results. In revising the analysis, the following changes were made that the FC-154A screening identified as needing further 10 CFR 50.59 evaluation: • The error in the feedwater regulating valve closure time was corrected, which increases the amount of feedwater to the ruptured steam generator. This adversely affects a UFSAR described design function. • The feedwater temperature was increased to reflect the final RSG values for the HZP and 70% power cases. This adversely affects a UFSAR described design function. • The containment analysis was performed using GOTHIC Version 7.2b, which is a change from the AOR, which uses GOTHIC Version 7.0. This is a revision to a UFSAR described evaluation methodology used in the safety analysis. The SER for Technical Specification Amendment 222 specifically approved the use of GOTHIC Version 7.0. **Summary of Evaluation:** Based on the results of this evaluation, the activity does not require prior NRC approval. Principally, the Feedwater Regulating Valve (FRV) closure time and feedwater temperature are not initiators of any accidents or malfunctions, so they cannot increase the frequency or likelihood thereof. The changes to these inputs in the

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		containment pressure analysis do not affect the radiological consequences analyses, which are separate from the containment pressure analyses and do not use these inputs. The two changes do not create any new failure mechanisms or failure modes. The revised containment pressure analysis shows that the peak containment pressure remains below the design pressure of 60 psig, and therefore does not exceed a Design Basis Limit for a Fission Product Barrier (DBLFPB). Lastly, the benchmarking analysis in FC07054 Revision 2 shows that the use of GOTHIC Version 7.2b is essentially the same as GOTHIC Version 7.0 for the Fort Calhoun containment pressure analysis.
EC 57976	EC 57976, OI-EG-2, ARP-ERFCS, ARP-CB1,2,3/A6, AOP-31	**Activity Description:** Operating procedures are being revised to provide guidance to operators to monitor plant bus voltages and inspect transmission line conductors supplying power to the plant so that the failure of a single phase conductor would be identified. These changes will also provide interim guidance to shut down the plant and isolate the degraded source of power if it has been determined that one of the three phases providing plant power has become open-circuited. These actions are being taken because of an industry event that revealed a vulnerability in the plant protection systems that permits an open circuited condition to exist on the 161 kV power supply without adequate detection and automatic protection. The development and implementation of interim actions to address this vulnerability are identified in the OPPD response to NRC Bulletin 2012-01 and INPO IER 12-14. SSCs impacted by this activity include the 161 kV power supply, house service transformers T1A-3 and T1A-4 and emergency diesel generators DG-1 and DG-2. The 10 CFR 50.59 screening identified three possible adverse conditions which are addressed in this evaluation. The three possible adverse conditions are: 1) A potential adverse condition is the reliance on manual operator action that did not previously exist to respond to an unbalanced voltage condition. Revisions to OI-EG-2 and AOP-31 will direct operators to monitor 4160 V buses 1 A3 and 1 A4 for an unbalanced condition and to manually trip the reactor, isolate the buses from offsite

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		power, one at a time, and verify that DG-1 and DG-2 automatically respond to provide power to associated buses. 2) The proposed revision to AOP-31 will result in an intentional loss of 4160V Buses 1A1 and 1A2 when the reactor is tripped. The loss of buses 1A1 and 1A2 in addition to the isolation of offsite power from buses 1A3 and 1A4 places the station in a loss of offsite power scenario requiring mitigation of a Loss of Forced Circulation per EOP-2, (Loss of Off-Site Power, and Loss of Forced Circulation). This constitutes a different method for controlling SSCs during 161 KV instabilities than is currently described in the USFAR section 8.3.1 regarding station electrical distribution supply to 4.16-kV during plant shutdown. 3) A third potentially adverse activity is to trip the reactor with an unbalanced voltage condition on the safety related buses. The evaluation will address the impact of the proposed activity on the capability of degraded buses to provide power during a reactor trip. **Summary of Evaluation:** The proposed activity to rely on manual operator action that did not previously exist to monitor for unbalanced voltage conditions and then to trip the reactor with a degraded power supply to the safety related buses does not increase the frequency, or consequences of any evaluated accidents or SSC malfunctions, nor does it introduce any new accidents or SSC malfunctions. These activities will initiate a loss of forced cooling event and a loss of main feedwater flow event; however these events have previously been evaluated in USAR 14.6 and USAR 14.10, respectively. In addition, the loss of forced cooling event will occur with a reactor trip prior to loss of flow which is more conservative due to the reduction in decay heat that will occur by the time that forced cooling is lost. Also, the actions are taken to prevent SSC malfunctions due to operators being unaware of an unbalanced voltage condition on the safety related buses that could result in subsequent motor malfunctions. These activities will be performed to reduce the likelihood of malfunctions, and any consequences of the trip

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		have already been evaluated. Backup pumps are available in the event of a SSC malfunction concurrent with or following the trip. This is an acceptable approach due to the fact that the precursor to this activity is outside of the plant's current licensing basis.
EC 55394	EC 55394, "Raw Water Pump Operation and Safety Classification of Components During a Flood"	**Activity Description:** This activity involves a modification to the plant and its operations which includes revision to AOP-01 for control of the intake structure cell water level during flood conditions and throughout the flood event. This is provided by the operation of four (4) new 10" valves located on the trash rack grid backwash header in the Intake Structure. The previous method for controlling cell level required that the exterior sluice gates be positioned prior to the flood and operation of the raw water pumps was relied upon for controlling cell level. This activity includes the addition of manual valves which will be used as needed in AOP-01 to control in flow to the intake cells and ultimately provide water to the raw water pumps and protection against flooding of the intake structure. This activity also removes OI-CW-1, Attachment 18 which provided a method of controlling intake cell level by backflow from the discharge tunnel. This was developed as a contingency if the partially open sluice gate was blocked. Because of the enhanced reliability using the new manual valves installed under this activity, OI-CW-1, Attachment 18 is being removed. The above changes were determined to be potentially adverse due to the fundamental change of the existing means of performing the design function of maintaining intake cell level during AOP-01 flood conditions and therefore are included in this 50.59 evaluation. **Summary of Evaluation:** EC 55394 proposes to revise AOP-01 to provide Raw Water Pump intake cell level control during flooding conditions from (current) five sluice gates closed with the sixth sluice gate prepositioned with a small opening to (new) all sluice gates closed and

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Change Number	Activity Title	50.59 Evaluation Summary
		control level by operating 4 manual valves. The proposed configuration is more reliable and more flexible, offering additional redundancy than the current configuration. The new configuration also allows for adjustment of flow and control of the intake cell level throughout the duration of the flood because the new valves are accessible from inside the intake structure. The design accommodates a single active failure, either failure of a valve to close or failure of a valve to open (or plugging of a valve). Therefore, the proposed configuration ensures the Raw Water Pumps remain available during a flood. The new configuration does not result in a malfunction or change to a currently evaluated malfunction that affects the likelihood or consequences of an accident and therefore does not require prior NRC approval.
EC 33099 (CR 2012-05793)	Original Steam Generator Storage Facility (10 CFR 50.59 Rev. 1)	**Activity Description:** The Original Steam Generator Storage Facility (OSGSF) is a reinforced concrete building designed to provide shielding and storage for the two Original Steam Generators (OSGs), the Original Pressurizer (OPZR). the Original Reactor Vessel Head (ORVH), and four concrete RVH missile shield blocks cut into twelve pieces. The building consists of three storage bays: one bay for each of the OSGs, one bay for both the OPZR and ORVH, and the RVH missile shield block pieces are placed in the west end of these bays. The building is located on the west side of the railroad tracks in the borrow-pit area (outside the plant's Protected Area, but within the Owner Controlled Area). It is classified as Non-CQE and Seismic Class II. Grading and drainage work is designed to accommodate the revised site conditions. Periodic testing and sampling of the conditions inside the OSGSF is performed. A hydraulic transporter will be used to transport the components to the OSGSF. The transporter can drive directly into the bays to place the components in their storage position. After the components are placed, precast reinforced concrete closure blocks are installed to close the building.

2012 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
		Summary of Evaluation: The OSGSF does not contain any active systems of its own, nor does it interact with any of the SSCs important to safety within the plant. The principal area of concern would be damage to stored components as a result of natural phenomena (such as fire, tornadoes, earthquakes, or floods) or transportation accident which leads to the release of radioactive contamination from the component internals. The OSGSF is constructed from concrete and steel and has limited presence of combustible materials. Therefore there is an insignificant fire risk. The substantial structure will provide a Significant level of protection from winds and tornado borne missiles. The facility is designed for seismic loads per the 2003 International Building Code. The facility was designed for a beyond design basis flood elevation, including dynamic effects of moving water. The OSGSF has a robust design. However, in the event of a collapse of the structure or a component drop from their support saddle, there is a possibility of a radiological release from the stored components to the environment. This would require a failure of the component cover plates, a release of radioactive contamination that is adhered to the inside of the components, and a breach of the OSGSF roof and/or walls. The calculated doses for the Control Room, Exclusion Area Boundary (EAB), and Low Population Zone (LPZ) are within acceptance criteria as described in the USAR. In addition, a failure of the fixative coating system for the components' external surfaces due to a flood does not result in an isotope concentration in excess of the 10 CFR 20 limit for effluents in water. Based on the robust construction of the OSGSF and the minimal potential release of radioactive contamination even if the structure and components experience a failure due to natural phenomena or transportation accident, there is an insignificant increase in the consequences of an accident, and those consequences are bounded by those accidents currently evaluated in the USAR.

2012 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
EC 55032	EA 12-002, Determination of safety classification of HCV-335-O and associated components	**Activity Description:** The safety classification of the HCV-335-O pneumatic sub-components is being changed to L-CQE from Non-CQE. This change is being made as a result of Safety Classification Analysis EA 12-002, which clarifies the safety classification of the valve actuator parts and accessories of HCV-335, "Shutdown Cooling Heat Exchangers AC-4A&B Inlet Header Isolation Valve." The SCA performed under EA 12-002 was created in response to CR 2011-7478 and CR 2011-7131. These two CRs document that the change in safety classification of the pneumatic subcomponents of HCV-335-O from CQE to Non-CQE, performed under EC 36173, was not screened as adverse. Therefore, although the plant documentation is changing from Non-CQE to L-CQE, the change being considered for the purposes of this 50.59 Evaluation is from the original licensing basis of CQE (prior to implementation of EC 36173) to the final classification of L-CQE. **Summary of Evaluation:** The proposed activity involves downgrading the safety classification of the HCV-335-O pneumatic subcomponents from CQE to L-CQE. The air actuator is supplied with Non-CQE instrument air and is not credited for operation in the accident analyses. The credited operation is locally via the operator handwheel, per AOP-19. It has been determined there are no credible failure modes of the pneumatic subcomponents that would prevent the handwheel from operating. Additionally, changing the safety classification (Quality Assurance requirements) of the pneumatic subcomponents does not alter the valve and operator failure modes and effects. Therefore, the proposed activity does not increase the frequency of accidents or malfunctions, does not introduce new accidents or malfunctions, does not increase the effects of existing accidents or malfunctions, and does not result in Design Basis Limits for Fission Product Barriers being exceeded or altered.

2012 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
EC 56808	Determination of Safety Classification of HCV-341 and Associated Components	**Activity Description:** The HCV-341-O pneumatic subcomponents are being reclassified as L-CQE from CQE, as a result of Safety Classification Analysis EA 12-014. Safety Classification Analysis EA12-014 clarifies the safety classification of HCV-341 "Shutdown Cooling Heat Exchangers AC-4A&B Outlet Temperature Control Valve" actuator parts and accessories. Specifically it concludes that, based on required safety functions, HCV-341 and HCV-341-O are CQE and the pneumatic subcomponents for HCV-341-O are L-CQE. SCA EA12-014 is being conducted in response to CR 2011-9937. **Summary of Evaluation:** The proposed activity involves downgrading the safety classification of the HCV-341-O pneumatic subcomponents from CQE to L-CQE. The air actuator is supplied with Non-CQE instrument air and is not credited for operation in the accident analyses. The credited operation is locally via the operator handwheel, per AOP-19. It has been determined there are no credible failure modes of the pneumatic subcomponents that would prevent the handwheel from operating. Additionally, changing the safety classification (Quality Assurance requirements) of the pneumatic subcomponents does not alter the valve and operator failure modes and effects. Therefore the proposed activity does not increase the frequency of accidents or malfunctions, does not introduce new accidents or malfunctions, does not increase the effects of existing accidents or malfunctions, and does not result in Design Basis Limits for Fission Product Barriers being exceeded or altered.

2013 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
CR 2013-00273	AOP-1 and OI-RW-1 Compensatory Measures for AC-10C Cable Damage	**Activity Description:** Moisture was found in cable EC68, which is the feeder cable to AC-10C-M (Raw Water Pump C Motor), while attempting to create a new bolted termination in intake structure junction box JB-95T. Jacket damage somewhere along the underground duct has opened a path for water to permeate into the cable and reach the termination in JB-95T. As a result, if MH-5 or MH-31 was to flood, pressure head could induce enough internal pressure for water to, once again, migrate thru the cable and exit at the termination in JB-95T. This condition may introduce a potential fault if the cable was energized at the time. Therefore, as a precaution, Fort Calhoun Station (FCS) will impose a restriction on AC-10C operability to declare the pump inoperable, and physically remove its power (e.g. open the breaker) in an event that either MH-5 or MH-31 was to flood with water. This compensatory measure will remain in place until the degraded condition is resolved. **Summary of Evaluation:** The evaluation of the activity concludes that prior NRC approval is not required. Operating procedures are being revised to de-energize one raw water pump during flooding conditions. The current FCS design basis does not require all four raw water pumps to operate in any scenario; including flood mitigation. In addition, the FCS design basis currently accounts for the loss of a single raw water pump and Technical Specification 2.4 implements the operating restrictions for this condition. Therefore, the existing design basis encompasses the new procedural requirements and the existing Technical Specification will continue to ensure that FCS does not operate outside of the current licensing basis.
EC 62506	EA 13-042 FCS MSLB Containment Analysis with Spray Pump Failure (AREVA Report 86-9213200-000 and Calculation 32-9213201-001)	**Activity Description:** The change being evaluated is the reduction in minimum guaranteed flow from the containment spray (CS) system to account for the need to throttle the containment spray pumps to prevent runout. The reduction is for the main steam line break (MSLB) event

2013 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
		from 1885 gpm to 1500 gpm for a 1 pump and 1 header configuration and a 1 pump and 2 headers configuration. For 2 pumps and 2 headers, the flow changes from 3770 gpm to 2800 gpm. The Loss of Coolant Analysis (LOCA) is unaffected as containment spray is not credited. **Summary of Evaluation:** The change in the credited minimum containment spray pump flows do not require a license amendment prior to implementation based on the justifications of the eight questions, since the impacts from the bounding accident scenarios have not been increased, the potential for creation of a new type of event not previously evaluated was not impacted, there was no impact on fission product barriers and no impact on the evaluation methodologies described in the UFSAR.
EA-FC-92-072	Revised Technical Data Book Figure TDB III.26.A, Diesel Generator Loading Curve	**Activity Description:** Background EA-FC-92-072 evaluates three areas of diesel generator performance: 1) It determines diesel fuel inventory requirements for 7 days of emergency operation, 2) calculates diesel generator peak power draw for comparison to the 2000 hour rating as required by Technical Specification 3.7, and 3) demonstrates the capability of the diesels to pick up required load groups during sequencing in response to a design basis accident concurrent With a loss of offsite power. The proposed revision 7 to EA-FC-92-072 will address deficiencies In the computer model used to analyze diesel performance. These deficiencies are identified in various condition reports and include: CR 2012-15727: The DG fuel model assumes that only one CCW pump operates after 24 hours. The current LOCA analysis requires two CCW pumps in operation indefinitely.

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		CR 2012-15725: The DG fuel model assumes a LOCA containment pressure profile which is less conservative than the current analysis is of record as documented in USAR section 14.16. CR 2012-00546: Non-CQE contactors associated with turbine building loads are credited to drop out during design basis events coincident with a loss of offsite power. This is inconsistent with QA requirements for credited equipment. (DG-2 only; in the revision to EA-FC-92-072. the non-CQE contactors were not credited for isolation from plant buses requiring associated loads to be added to the diesel load.) CR 2011-4725: A spent fuel pool pump was not included in the diesel fuel model. The previous revision of EA-FC-92-072 was based on a model which assumed that the bounding scenario for diesel performance was a large break loss of coolant accident (LBLOCA). However, with the implementation of EC 40700 in 2008, containment spray pumps no longer operate in response to a LOCA which leads to the possibility that another accident type, i.e., a main steam line break (MSLB) could be the bounding scenario for some aspects of diesel performance. Therefore, revision 7 to the analysis includes the development of a second model based on the diesel loading required to mitigate a MSLB in addition to the model used to address performance requirements during a LOCA. The use of two models will ensure that the bounding cases are identified for each accident type. Each model will be consistent with loads resulting from EOP-directed operator actions taken in response to the two accident types. Proposed Activities The proposed activity consists of four sub-activities which will be evaluated separately as follows. Activity 1: Revise the method of evaluation used to evaluate diesel generator performance as follows: 1) add two CCW pumps in operation indefinitely to replace one CCW pump in operation after 24 hours for LOCA, 2) adjust DG fuel model for a LOCA

2013 Evaluations

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		containment pressure profile to be consistent with the current analysis of record as documented in USAR section 14.16, 3) Non-CQE contactors associated with turbine building loads (DG-2 only) will not be credited to drop out during design basis events coincident with a loss of offsite power, and 4) add a spent fuel pool pump in the diesel fuel model. Activity 2: Revise the method of evaluation used to evaluate diesel generator performance as follows: add a second model based on the diesel loading required to mitigate a MSLB in addition to the model used to address performance requirements during a LOCA. Activity 3: Revision 7 of EA-FC-92-072 involves a change to the peak diesel loading conditions as specified in Technical Data Book (TDB) Figure TDB III.26A which compares calculated peak loading to the 2000 hour diesel rating. The new revision of the EA provides two separate peak diesel loading figures for each diesel generator for TDB III.26.A. One set of Figures for the MSLB and one set of Figures for LOCA. Revision 6 of the EA only provides peak load Figures for LOCA for each diesel generator. The change imposes an operability temperature limit of 105 degrees F for diesel generator DG-2 in contrast to the previous limit of 114 degrees F. Activity 4: The following compensatory measure will be implemented: a change to the administrative limits is being made for diesel fuel inventory to ensure that adequate fuel supplies are present on site to meet the UFSAR-required 7 days of diesel generator performance under worst case loading conditions. The additional fuel required to provide for 7 days of operation is within the available volume in the fuel storage lank of FO-1 or FO-10. **Summary of Evaluation:** The proposed activity is to restrict operation of diesel generator DG-2 to ambient temperatures of less than 105 degrees F from a previous limit of greater than 114

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		degrees F. This proposed activity does not require a license amendment for the following reasons: No configuration changes or changes to the operation or control of plant systems are being made so that no new failures modes are introduced.
EC 29642	OI-AFW-4, Revision 43: Elimination of FW-6 and FW-10 as startup or shutdown pumps during normal plant operations	**Activity Description:** Revision 43 to operating instruction OI-AFW-4 was approved in March 2002 under Engineering Change (EC) 29642. EC 29642 added a restriction to OI-AFW-4 to prevent opening of the auxiliary feedwater (AFW) to main feedwater (MFW) isolation valve HCV-1384 when the reactor coolant system (RCS) is at a temperature greater than 300 degrees F. The purpose of this procedure change was to ensure operability of the AFW system without reliance on operator action to close HCV-1384 if it were to become necessary to isolate AFW from MFW during plant startup. At the time, AFW was considered Inoperable with HCV-1384 open because this configuration cross-tied the safety grade AFW system to the non-safety grade MFW system. Prior to revision 43, credit was taken for operator action to maintain AFW operability. EC 29642 was implemented without consideration of the fact that requiring HCV-1384 to be closed with the RCS greater than 300 degrees F effectively eliminates the use of the motor-driven AFW pump (FW-6) and the steam driven AFW pump (FW-10) as feedwater injection pumps during normal startup or shutdown evolutions. Since the original FSAR and the current USAR both identify the AFW system as capable of providing feedwater for normal plant startup operation, the procedure change should have been evaluated under 10 CFR 50.59 as identified in the screening performed for the procedure change. (Note: The original procedure change under EC 29642 did not have a 10 CFR 50.59 screening performed as part of the procedure change package. A 50.59 screening was performed for revision 43 in January 2012 as part of the actions addressing condition report (CR) 2011-9416. The activity being evaluated under this 50.59 evaluation Is the elimination of FW-6 and FW-10 as startup or shutdown pumps during normal plant operations, relying solely on the use of diesel-driven AFW pump FW-54 as a startup

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Change Number	Activity Title	50.59 Evaluation Summary
		pump. **Summary of Evaluation:** The proposed activity to use the diesel-driven AFW pump FW-54 as the sole startup (or shutdown) pump does not adversely impact the capability of the AFW system to meet its design function to provide decay heat removal capability either during plant startup or In the event of a loss of the running startup pump. The USAR described analyses for a loss of feedwater or a main steam line break remain bounding and the Technical Specification operability requirements for the AFW system will be maintained.
EC 34435	FW-10 pneumatic speed control removal	**Activity Description:** The reason for this modification was to remove the FW-10, Turbine Driven Auxiliary Feedwater Pump, speed control loop. EAR / AR 28528/01 responded to a concern about the reliability of FW-10's pneumatic speed control loop and recommended that the pneumatic components be removed permanently because they were not required for correct operation of the pump. Once removed, the existing mechanical speed-limiting governor provides the necessary speed regulation to allow the pump to be operated as a constant speed machine. The governor's performance has proven to be reliable. **Summary of Evaluation:** The proposed change does not create any accident initiators, nor does it create any new malfunctions not presently described in the UFSAR. The change has no impact on any radiological consequences as the only accident directly associated with this design function of the pump is the loss of feedwater, which is assumed to have no radiological consequences. Therefore, the change to remove the pneumatic control components from FW-10 has identified that this change may be implemented without obtaining a License Amendment.

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Change Number	Activity Title	50.59 Evaluation Summary
EC 50465	EC 50465 LPSI Recirculation Piping and Valve Upgrade in Support of AHLI	**Activity Description:** EC 50465 consists of the installation of new low pressure safety injection (LPSI) pump recirculation piping – including manual valves and orifices - which will connect the LPSI pump discharge header to the containment sump recirculation lines and which will be used for the alternate hot leg injection (AHLI) alignment during the long term core cooling phase of a loss of coolant accident (LOCA) and during testing of that alignment. The new recirculation line will be placed in service only during AHLI operation and during testing of that alignment. Otherwise, flow through the line is prevented by means of manual 3-inch globe valves (SI-511 and SI-512); one for each of the two containment recirculation sump lines, which will normally be locked closed. To place the LPSI recirculation piping into service for AHLI, SI-186 and SI-511 or SI-512 must be manually opened with a "reach" rod through penetrations in Corridor 4. A separate activity, temporary modification EC 58972, is installing pipe stubs and manual isolation valves to facilitate the subsequent installation of the new recirculation piping under EC 50465. The temporary modification also replaces the existing manually operated shutdown cooling header warm-up isolation valve, SI-186, with a similar valve having a higher flow coefficient (Cv). The final configuration resulting from the combination of EC 50465 and EC 58972 (hereafter, referred to as "the proposed activity"), which will provide a new method for alternate hot leg injection during the long term core cooling phase of a LOCA, is the subject of this 50.59 evaluation. EC 50465 makes the required parts installed under EC 58972 permanent plant equipment. The proposed activity is being implemented because the existing plant configuration may not provide the flow rates needed (i.e., LPSI flow controller FIC-326 may not provide accurate flow indication) during AHLI operations. The new configuration will provide the capability to achieve the required flow rates.

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		The scope of this screening includes a change to USAR Section 6.2, which identifies that a new pump recirculation line is added for use during AHLI. USAR Figures 9.3-1 and 9.3-2 are also revised to show the location of the alternate hot leg recirculation line with respect to the current piping arrangement. The safety injection and containment spray system drawings (P&ID file numbers 44353, 10479, and 56027) are revised to show the functional arrangement for alternate hot leg injection. New LPSI maximum /minimum curves are used to for the hydraulic analyses associated with AHLI. The scope of this screening also includes the operating procedure changes associated with the proposed activity. These include emergency operating procedures, system valve lineups, and containment leakage rate test procedures identified in Section II. Currently, there is a conduit interference created by the new remote valve operator rods needed for the recirculation line isolation valves. The cable in these conduits is associated with the spent resin pump (WD-14B). As a result, the existing cable is removed and replaced with a new cable of the same size. In addition, two lighting fixtures are relocated (raised up or down) to allow for installation of the recirculation piping, and are field relocated. Two (2) existing abandoned and plugged penetrations (04-F-51 and 04-F-53) are used for manual valve operation in Corridor 4. The penetrations were electrical penetrations and are re-purposed as mechanical penetrations. Manual operator actions will now take place in Corridor 4 to open and close SI-511 and/or SI-512. The opening of either SI-511 or SI-512 is necessary post-LOCA to provide LPSI minimum flow for long term AHLI operation. The 50.59 screening for the proposed activity concluded that the following were adverse effects of the proposed activity; they are, therefore, the subject of this 50.59 evaluation:

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		• A failure of an isolation valve in the new recirculation piping to perform its isolation function or a failure of a root valve to remain full-open during the different phases of a LOCA • The increase in LPSI and high pressure safety injection (HPSI) pump suction temperature and the corresponding reduction in pump net positive suction head available (NPSHa) • The installation of new components with the potential to produce external leakage in piping and compartments where fluid from the containment sump is recirculated during the recirculation phase of a LOCA • The connection of the higher pressure LPSI pump discharge to the lower pressure containment sump recirculation lines through a single valve (SI-511 or SI-512) • The reduction in shutdown cooling system flow available to perform the core cooling, reactor coolant system (RCS) cooldown, spent fuel pool cooling, and alternate shutdown cooling utilizing containment spray pumps, boron dilution and LPSI pump NPSH functions as a result of the additional flow diversion through the replacement SI-186. • The replacement of the existing method of performing the AHLI function with a method based on hydraulically balanced fixed resistances results in changes to the operator actions required to mitigate the consequences of a design basis accident constitutes a fundamental change in the method of performing or controlling the alternate hot leg injection, LPSI pump minimum flow and sump strainer flow design functions. • The potential impact of the installation of new valves on other (i.e., non-AHLI) operating configurations, where a component malfunction (e.g., valve mispositioning) could produce adverse effects. • Gas accumulation as recirculated fluid passes through the orifices and/or valves during AHLI operation is a design issue. • Dose to operating personnel during AHLI initiation. (Function 9)

2013 Evaluations

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		Summary of Evaluation: This activity can be implemented per plant procedures without obtaining a License Amendment. It does not involve the initiation of any accident previously evaluated in the Updated Final Safety Analysis (UFSAR) (Also known as Updated Safety Analysis (USAR)). Potential structure, system, component (SSC) malfunctions including operator errors have been evaluated. It is determined that this activity will not result in more than a minimal increase in the likelihood of occurrence of a malfunction. This activity provides an enhanced capability for the operators to respond long term to a LOCA (5.5 hours) using the alternate hot leg injection path. It eliminates the potential for errors due to faulty indication. Since the additional local operator action is performed in the same area (Corridor 4) as two other local operation actions, the proposed activity does not affect the ingress and egress paths used in performing the local operator actions. The dose for this activity has been conservatively calculated to be less than 5 rem acceptance criteria in NUREG 0737. The changes to the AHLI operator actions will be implemented by means of a revision to the emergency operating procedures. Therefore, procedural guidance will be available. The additional local operator action – which involves unlocking and opening an isolation valve - is the same as an existing operation performed in the same area. It is, therefore, within the skill of the local operator. Since the proposed activity involves overall changes to the existing method of aligning AHLI, the changes introduced by the proposed activity will be evaluated using the existing systematic approach to training (SAT) and communicated to Operations and other appropriate personnel at the proper time and manner as determined by the SAT process. The proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR. The permissible leakage as specified in Technical Specification 3.16 is not changed. The safety analysis conservatively uses two times the Technical Specification limit for the postulated leakage.

2013 Evaluations

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		Periodic testing ensures that the leakage is maintained below the Technical Specification limits. Therefore, the proposed activity does not result in an increase in the radioactivity released as a result of engineered safety feature system leakage. There is no change in the consequences of a malfunction (single failure) of an SSC important to safety previously evaluated in the USAR. The evaluation of radiological consequences evaluated in the USAR assumes a single active failure of an emergency core cooling system (ECCS) component. The alternate hot leg injection path will be utilized only if there is a single failure that results in a loss of the preferred (HPSI) hot leg injection path. There are no changes to this preferred hot leg injection path in this activity. The changes to the AHLI simplify the operator actions by removing the need to throttle flow. The AHLI path components are designed CQE and seismic and independent of the preferred HLI path. Neither the method of performing the alternate hot leg injection function nor the revised operator actions associated with this change will produce malfunctions of SSCs important to safety. This activity does not create a possibility for an accident of a different type or result than any previously evaluated in the UFSAR. The existing inherent design features along with non-time critical operator actions ensure that a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR is not created. There is no change on ECCS performance and subsequently no change in a design basis limit for a fission product barrier as described in the UFSAR. The items being addressed in this evaluation do not involve a method of evaluation described in the USAR or used in establishing the design bases or in the safety analyses.
EC 52009	Summary of Design Basis Reconstitution for High Energy Line Break (HELB) Outside of Containment in Response to CR 2007-3407	**Activity Description:** The activity is the use of the GOTHIC™ (Generation of Thermal-Hydraulic Information for Containments) computer program for determining the consequences of High Energy Line Breaks (HELBs) outside containment. GOTHIC is used for determining the mass and

2013 Evaluations

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		energy release from line breaks in the Main Steam, Feedwater, CVCS (Letdown and Charging) and FW-BD Systems, and is used for determining the resultant environmental conditions, including pressure, temperature and humidity for each of the above line breaks. • GOTHIC Calculation FC07889 (NAI Calculation NAI-1479-004) is to be issued using GOTHIC Version 7.2b to model the Auxiliary Building Structure and to determine the environmental consequences of a main steam line break in Room 81. • GOTHIC Calculation FC07863 was issued using GOTHIC Version 7.2b to perform mass and energy release calculations for HELBs outside containment in the Letdown line (containment wall to PCV-210), Charging Line, Steam Generator Blowdown, and Main Steam Lines to FW-10. • GOTHIC Calculations FC07862 and FC07864 were issued using GOTHIC Version 7.2b to model the Auxiliary Building Structure and to determine the environmental consequences of the postulated HELBs. The computer program GOTHIC was used because: • The program or method previously used for mass and energy release for the Main Steam and Feedwater Systems is a combination of hand calculations combined with the use of a Gibbs and Hill TPD program. The approach used is described in Letters LIC-73-0006, dated March 14, 1973 and LIC-73-0012, dated May 15, 1973, in USAR Appendix M, and more briefly and with less detail in the NRC Safety Evaluation Report (SER) NRC-73-0015, Supplement 1 to Safety Evaluation by the Directorate, dated April 23, 1973. The code and support for this program is no longer available. The original Architect / Engineer for FCS/OPPD (Gibbs & Hill) is no longer in the nuclear design business, and no longer supports the use of this legacy program. This change will use the Extended Henry-Fauske critical flow model for sub-cooled liquid conditions, and the Moody critical flow model for saturated steam and liquid conditions to determine the applicable mass

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		and energy. • GOTHIC is a general use thermal-hydraulic program, which is technically acceptable to use for the listed applications. GOTHIC has been approved for performing mass and energy release calculations at FCS for line breaks inside containment. • GOTHIC is a general use thermal-hydraulic program that produces conservative results and is technically acceptable to use for the listed application. The use of GOTHIC for outside the containment was identified as a change in methodologies in the 50.59 screen that required a full 50.59 evaluation. **Summary of Evaluation:** The version of GOTHIC used for FCS for the evaluation of breaks outside of the containment has been approved at Point Beach Nuclear Plants Unit 1 and 2 (Reference Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 241 to Renewed Facility Operating License No. DPR-24 and Amendment No. 245 to Renewed Facility Operating License No. DPR-27, NextEra Energy Point Beach LLC, Point Beach Nuclear Plants Unit 1 and 2, Docket No 50-266 and 50-301, Ascension No. ML 110450159) for the same purpose. The application of GOTHIC for high energy line break mass and energy release is part of what the tool is designed for, and will use the approved methodology which requires the use of the Extended Henry-Fauske critical flow model for sub-cooled liquid conditions, and the Moody critical flow model for saturated steam and liquid conditions. The use of GOTHIC for the determination of environmental conditions has been approved by the NRC as noted above at Point Beach Nuclear Plant. The use of GOTHIC for FCS HELBs outside containment is, therefore, acceptable without prior NRC approval as FCS will use GOTHIC as allowed by the Point Beach Nuclear Plant SER for use in the evaluation of HELBs outside of the Containment.
EC 57888	Fire Pump FP-1A Motor Cable Terminations	**Activity Description:** As a requirement for cable testing for medium voltage motor FP-1A in the Intake

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Change Number	Activity Title	50.59 Evaluation Summary
		Structure, the motor terminations are required to be lifted and reterminated. In order to properly reterminate the motor connections, the use of Raychem connection kits and a Raychem tape is the preferred method to insulate the connection. The use of Raychem connection kits is not permitted to be installed over existing tape; therefore the existing tape that was used on the cable lugs is required to be removed. When the existing tape was removed, it was discovered that extensive damage was present on the cable insulation. Due to the location of the insulation damage, the cables were not long enough to allow the cables to be cut back and new lugs installed. In order to address the cable damage in the motor termination box, new cables were installed from existing junction box to the motor termination box. The new cables were then terminated to the existing cables in the junction box. A new junction box is to be installed near the motor and connected to the existing conduit feed to the motor. New flexible conduit will be installed from the junction box to the motor terminal box. The new cable will be the same size and voltage rating as the existing cable, 1/C #2 AWG 5kV. A new ground conductor will be run from the new junction box, along the outside of the new conduit and will terminate at the motor terminal box. The cable connections in the junction boxes will be addressed by having both the existing and new cables terminated using Burndy lugs and Raychem termination kits. After the lugs are bolted together, the entire connection will be insulated using Raychem connection kits. The shield wires on the new cables (Unishield) will have a #12 lug applied to the drain wires which will be used to terminate the shield ground to the ground clamp of the existing cables in the junction box. The cable connections in the motor termination boxes will be completed by using a Raychem termination kit on the new cable (Unishield) and a 90 degree lug to connect to the existing motor leads. The bolted connection made between the existing motor leads and new cables will then be insulated using a safety-related Raychem kit as seen on drawing SK-EC-57888-01. The shield wires on the new cable (Unishield) in the

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Change Number	Activity Title	50.59 Evaluation Summary
		motor termination box will be terminated to the ground lug in the motor termination box. This evaluation is based on the conservative "Yes' answer to screening question 1 "Does the proposed activity involve a change to an SSC that ADVERSELY affects a UFSAR Described DESIGN FUNCTION?" Therefore, the answers to the questions are focused on that aspect of the change. **Summary of Evaluation:** This activity can be implemented without obtaining a License Amendment. The new cable terminations do not introduce the possibility of a change in the frequency of occurrence of an accident or likelihood of a malfunction because the failure of a cable termination is not an initiator of an accident or malfunction. The new cable terminations do not introduce a change in the consequences of an accident or malfunction. A cable termination failure would result in the same loss of Fire Pump FP-1A as in the current design. The results of a failure of the cable termination are no different than the failure of the original cable. The cable separation criteria as specified in the engineering change package are in accordance with the criteria in USAR 8.5. These terminations are housed in a seismically qualified box due to its location near Class I Fire Protection Piping. The new cable terminations do not affect any parameters that are outside the reference bounds for the design of a fission product barrier. The new cable terminations modify cable to Fire Pump FP-1A motor but do not involve a method of evaluation as defined in the USAR. USAR Section 8.5, Figure 8.51, note #19 states "Cable splicing in cable trays is used only for connection of incoming and outgoing containment electrical penetration conductors." The applicable drawings identify the cable terminations and their locations. None of these cable terminations are within a

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Change Number	Activity Title	50.59 Evaluation Summary
		cable tray. Therefore, this USAR requirement is not applicable.
EC 57889	Circulating Water Pumps CW-1 A. 1 Band 1 C Motor Cable Terminations	**Activity Description:** As a requirement for cable testing for medium voltage motors CW-1A, CW-1B and CW-1C in the Intake Structure, the motor terminations are required to be lifted and reterminated. In order to properly reterminate the motor connections, the use of Raychem connection kits and a Raychem tape is the preferred method to insulate the connection. The use of Raychem connection kits is not permitted to be installed over existing tape; therefore the existing tape that was used on the cable lugs is required to be removed. When the existing tape was removed. it was discovered that extensive damage was present on the cable insulation. Due to the location of the insulation damage, the cables were not long enough to allow the cables to be cut back and new lugs installed. In order to address the cable damage in the motor termination boxes, new cables were installed from existing junction boxes to the motor termination boxes. The new cables were then terminated to the existing cables in the junction boxes. For each motor, a new junction box is to be installed near the motor and connected to the existing conduit feed to the motor. New flexible conduit will be installed from the junction box to the motor terminal box. The new cable will be the same size and voltage rating as the existing cable, 1/C #4/0 AWG 5kV. A new ground conductor will be run from the new junction box. along the outside of the new conduit and will terminate at the motor terminal box. The cable connections in the junction boxes will be addressed by having both the existing and new cables terminated using Burndy lugs and Raychem termination kits. After the lugs are bolted together, the entire connection will be insulated using Raychem connection kits. The shield wires on the new cables (Unishield) will have a #12 lug applied to the drain wires which will be used to terminate the shield ground to the ground

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Change Number	Activity Title	50.59 Evaluation Summary
		damp of the existing cables in the junction box. The cable connections in the motor termination boxes will be completed by using a Raychem termination kit on the new cable (Unishield) and a 90 degree lug to connect to the existing motor leads. The bolted connection made between the existing motor leads and new cables will then be insulated using a safety-related Raychem kit as seen on drawing SK-EC-57889-01. The shield wires on the new cable (Unishield) in the motor termination box will be terminated to the ground lug in the motor termination box. This evaluation is based on the conservative "Yes" answer to screening question 1 "Does the proposed activity involve a change to an SSC that ADVERSELY affects a UFSAR Described DESIGN FUNCTION?" Therefore, the answers to the questions are focused on that aspect of the change. **Summary of Evaluation:** This activity can be implemented without obtaining a License Amendment. The new cable terminations do not introduce the possibility of a change in the frequency of occurrence of an accident or likelihood of a malfunction because the failure of a cable termination is not an initiator of an accident or malfunction. The new cable terminations do not introduce a change in the consequences of an accident or malfunction. A cable termination failure would result in the same loss of a Circulating Water Pump as in the current design. The results of a failure of the cable termination are no different than the failure of the original cable. The cable separation criteria as specified in the engineering change package are in accordance with the criteria in USAR 8 5. These terminations are housed in seismically qualified boxes due to their location near Safety-Related power feeds.

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Change Number	Activity Title	50.59 Evaluation Summary
		The new cable terminations do not affect any parameters that are outside the reference bounds for the design of a fission product barrier. The new cable terminations modify cables to the Circulating Water motors but do not involve a method of evaluation as defined in the USAR. USAR Section 8.5, Figure 8.5.1, note #19 states "Cable splicing in cable trays is used only for connection of incoming and outgoing containment electrical penetration conductors." The applicable drawings identify the cable terminations and their locations. None of these cable terminations are within a cable tray. Therefore, this USAR requirement is not applicable.
EC 58972	Alternate Hot Leg Injection (AHLI) SI-186 Change and Taps for LPSI Min Flow Line	**Activity Description:** It has been previously identified that the Alternate Hot Leg Injection (AHLI) mode of LPSI is not functional with the currently installed flow instrumentation (CR 2010-5042). This temporary modification (EC 58972) will install a new SI-186 valve, (LPSI Pumps SI-1A & B Shutdown Cooling Warm Up Isolation Valve). The new SI-186 valve will have a higher Cv than the currently installed valve. EC 58972 will also install stub pipes with manual isolation valves, each on the LPSI suction and discharge lines in Room 23 of the Auxiliary Building. The final modification (EC 50465) will later install a LPSI Pump recirculation path (with orifice(s)) on these pipe stubs. The LPSI pressure retaining components installed by EC 58972 are required to support the final modification (EC 50465) and the purpose of EC 58972 is to minimize the impact of this project on refueling outage activities and schedule. Until it is incorporated into the final modification. (EC 50465), the temporary modification can be credited for function only during Mode 5. Plant mode change cannot occur until Operations has accepted (I.e., OPSAC) the final modification (EC 50465), whereby the temporary modification (EC 58972) will be "removed" through incorporation into EC 50465. **Summary of Evaluation:** Until it is incorporated into the final modification, the temporary modification can be credited for function only during Mode 5. With the plant restricted to Mode 5 operation

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Change Number	Activity Title	50.59 Evaluation Summary
		while the TM is installed, the relevant UFSAR described SSCS and associated design functions are those associated with shutdown cooling operation. For the purposes of this discussion, the applicable accidents are a Loss of Coolant Accident (LOCA) During Shutdown (USAR 14.15.7.1) and a Boron Dilution Incident (USAR 14.3). EC 58972 will install a new SI-186 valve, (LPSI Pumps SI-1A & B Shutdown Cooling Warm Up Isolation Valve). The new SI-186 valve will have a higher Cv than the currently installed valve. The EC will also install stub pipes with manual isolation valves, on the LPSI suction and discharge lines in Room 23 of the Auxiliary Building. These changes potentially impact a boron dilution event and LOCA during shutdown. The changes made by EC 58972 as discussed in this evaluation, show that the SDC system response to the applicable USAR specified accidents is unchanged and does not change their frequency of occurrence, the likelihood of occurrence of a malfunction, an increase in the consequences, does not create a possibility for an accident of a different type, or with a different result or use different methodology or impact fission product barrier.
EC 58972	Alternate Hot Leg Injection (AHLI) SI-186 Change and Taps for LPSI Min Flow Line (10 CFR 50.59 Evaluation Rev. 1 - The intent and scope of EC 58972 and associated 10 CFR 50.59 evaluation were not changed.)	**Activity Description:** It has been previously identified that the Alternate Hot Leg Injection (AHLI) mode of LPSI is not functional with the currently Installed flow instrumentation (CR 2010-5042). This temporary modification (EC 58972) will install a new SI-186 valve, (LPSI Pumps SI-1A & B Shutdown Cooling Warm Up Isolation Valve). The new SI-186 valve will have a higher Cv than the currently Installed valve. EC 58972 will also install stub pipes with manual Isolation valves, each on the LPSI suction and discharge lines in Room 23 of the Auxiliary Building. The final modification (EC 50465) will later install a LPSI Pump recirculation path (with orifice(s)) on these pipe stubs. The LPSI pressure retaining components Installed by EC 58972 are required to support the final modification (EC 50465) and the purpose of EC 58972 Is to minimize the impact of this project on refueling outage activities and schedule. Until it is Incorporated Into the final modification, (EC 50465), the temporary modification can be credited for function only during Mode 5.

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Change Number	Activity Title	50.59 Evaluation Summary
		Plant mode change cannot occur until Operations has accepted (i.e., OPSAC) the final modification (EC 50465), whereby the temporary modification (EC 58972) will be "removed" through Incorporation Into EC 50465. **Summary of Evaluation:** Until it is incorporated into the final modification, the temporary modification can be credited for function only during Mode 5. With the plant restricted to Mode 5 operation while the TM is installed, the relevant UFSAR described SSCs and associated design functions are those associated with shutdown cooling operation. For the purposes of this discussion, the applicable accidents are a Loss of Coolant Accident (LOCA) During Shutdown (USAR 14.15.7.1) and a Boron Dilution Incident (USAR 14.3). EC 58972 will install a new SI-186 valve, (LPSI Pumps SI-1A & B Shutdown Cooling Warm Up Isolation Valve). The new SI-186 valve will have a higher Cv than the currently installed valve. The EC will also Install stub pipes with manual isolation valves on the LPSI suction and discharge lines In Room 23 of the Auxiliary Building. These changes potentially impact a boron dilution event and LOCA during shutdown. The changes made by EC 58972 as discussed in this evaluation show that the SDC system response to the applicable USAR specified accidents is unchanged and does not change their frequency of occurrence, the likelihood of occurrence of a malfunction, an increase in the consequences, does not create a possibility for an accident of a different type, or with a different result or use different methodology or impact fission product barrier.
EC 59874	HPSI Pump Runout, SI-2A, SI-2B and SI-2C - Part 1 of 3 (Safety analyses with reduced flow)	**Activity Description:** Activity #1 - A revised delivered flow vs. RCS pressure curve was developed from newly performed hydraulic analyses. In order to maintain sufficient margin, the credited flows used in the LOCA and other Chapter 14 safety analyses will be reduced.

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<table>
<tr><th>Change Number</th><th>Activity Title</th><th>50.59 Evaluation Summary</th></tr>
<tr><td></td><td></td><td>Activity #2 - Currently, it is assumed that at 30 minutes into a LBLOCA the spillage increases due to additional leakage in the CVCS system caused by opening the cross-connect line, which maintains M-3 penetration pressurized. The activity will apply the additional leakage in the CVCS system from time T=0.

Activity #3 - A minimum required HPSI flow value at 30 minutes and during recirculation is established for the LBLOCA (with RCS pressure <20 psia) to match the most recent analysis.

The 50.59 screening for the proposed activity concluded that the activity has an adverse effect on the following design basis functions and are therefore subject to this evaluation:

1. Decrease in the credited flow from a HPSI pump during the safety injection phase as described in USAR Section 6.2.1 for a LOCA (Both SBLOCA and LBLOCA) and as used in USAR Chapter 14.15
2. Change in description of spillage at 30 minutes to reflect results of new analyses (USAR 6.2.3.3)
3. Decrease in the credited HPSI flow rate during a MSLB accident. (USAR 14.12)

Since all of the proposed activities are linked, they will be addressed "en-toto" within this 50.59 Evaluation.

Summary of Evaluation:

This activity can be implemented per plant procedures without obtaining a License Amendment. The HPSI system has no impact on the frequency of occurrence of any accident previously evaluated in the USAR since the HPSI system is not an initiator of any analyzed accident.

The activity does not involve any physical changes to the system and therefore cannot result in an increase in the likelihood of a malfunction of an SSC.</td></tr>
</table>

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		The proposed activity does not result in an increase in the consequences of an accident previously evaluated in the USAR. Affected safety analyses and their radiological consequences were evaluated to determine if any input parameters are affected by this activity. The results determined that there is no effect on the consequences of an accident previously evaluated in the USAR. Since the activity does not involve any physical changes to the system and does not result in an increase in the likelihood of a malfunction of an SSC, the consequences of a malfunction previously evaluated in the USAR are not affected by this activity. The activity does not introduce the possibility of a new accident because the activity does not involve any physical changes to the system or components, thus it does not introduce create the potential for a different type of accident than currently evaluated in the USAR. The activity involves re-evaluation of affected Chapter 14 of the USAR Safety Analyses that have been identified as potentially impacted by the reduced HPSI flow. There is no physical change to the system that can create a malfunction with a different result than previously evaluated in the USAR. As determined in the FC-154A, "50.59 Screen" form, the activity does not involve a change in the evaluation methodologies as described in the USAR. Therefore, the proposed activity does not result in a departure from a method of evaluation described in the USAR used in establishing the design bases or in the safety analyses.
EC 59874	HPSI Pump Runout, SI-2A, SI-2B and SI-2C - Part 2 of 3 (Orifice plate installation)	**Activity Description:** 1. The activity will install orifice plates in the HPSI pump discharge headers. Recent hydraulic analyses have shown that in certain post-LOCA alignments with a low system resistance, the HPSI pump could be operating in a runout condition that could damage the pump. Although the potential for runout has been previously identified, it was evaluated as being an acceptable condition for a limited duration. A recent analysis performed by the pump OEM (Bingham-Sulzer) recommended

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Change Number	Activity Title	50.59 Evaluation Summary
		that pump operation be restricted to flows at or below 450 gpm at all times in order to avoid accelerated wear. The orifice plates will increase the system resistance and therefore change the operating point of the pump to deliver ≤ 450 GPM. The design of the orifice plates must ensure that minimum flows used in the safety analyses are still achieved while limiting the maximum flow in low system resistance alignments. 2. The SI spillage is reduced from 35% to 25% at 30 minutes into a LOCA. A minimum required HPSI flow value at 30 minutes and during recirculation is established for the LBLOCA (with RCS pressure <20 psia) to match the most recent analysis. The adequacy of the delivered flow to the core at 30 minutes into a LBLOCA and during the recirculation phase is also evaluated. The 50.59 screening for the proposed activity concluded that the following aspects of the modification have an adverse effect on the design functions of the system and are therefore subject to this evaluation: 1. Decrease in the flow delivered by a HPSI pump during the safety injection phase as described in USAR Section 6.2.1 and USAR Section 14.15 for a LOCA (Both SBLOCA and LBLOCA) and as used in USAR Chapter 14.15. 2. Decrease in HPSI flow rate during a MSLB accident. (USAR Section 14.12) 3. Decrease in HPSI flow at 30 minutes with 35% spillage. (USAR 6.2.3.3) **Summary of Evaluation:** This activity can be implemented per plant procedures without obtaining a License Amendment. The HPSI system has no impact on the frequency of occurrence of any accident previously evaluated in the USAR since the HPSI system is not an initiator of any analyzed accident. The proposed activity does not result in an increase in the consequences of an accident previously evaluated in the USAR. Any malfunction in the components installed by this modification is bounded by the previously evaluated malfunctions described in the USAR. Therefore, this activity will not result in more than a

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Change Number	Activity Title	50.59 Evaluation Summary
		minimal increase in the likelihood of a malfunction previously evaluated in the USAR or the consequences of a malfunction previously evaluated in the USAR nor does it create the possibility of a malfunction with a different result than previously evaluated in the USAR. The installation of the orifice plates and flanges does not introduce the possibility of any credible new accident because the HPSI system is not an initiator of any accident and no new failure modes are introduced. The proposed change will not affect ECCS performance and subsequently there is no change in the design basis limits for the fission product barrier as described in the USAR. The proposed activity is a physical modification to the HPSI piping and does not revise or replace a method of evaluation described in the USAR or used in establishing the design bases or in the safety analyses.
EC 59874	HPSI Pump Runout, SI-2A, SI-2B and SI-2C Part 3 of 3 - (HPSI Loop Flow Balancing)	**Activity Description:** Modify the method of establishing HPSI flow through the HPSI injection valves from having the valves go full open to having the valves open to a pre-determined position defined by limit switch settings. The HPSI loop injection valves position will be pre-set to ensure balanced flow through each of the HPSI lines. Setting of the loop injection valves will be achieved by adjusting the limit switches to stop the valve at the established open position. Testing performed to support EC 59874 indicated that flow through three of the HPSI loop injection valves (HCV-312, HCV-317 and HCV-315) was higher than expected, resulting in unbalanced flows in the injection loops. The test results and investigation indicated that these three valves have a higher flow coefficient (Cv) than specified in the design documents. The modification will allow these valves (as well as the other five loop injection valves) to be throttled to a position that will provide the required flow balance between the loops. The design documents will be updated to reflect this change. The

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		FCS LOCA analyses assume approximately equal flow through each injection line. The flow in each injection line will be verified to meet or exceed the requirements of the current safety analyses. **Summary of Evaluation:** This activity can be implemented per plant procedures without obtaining a License Amendment.[1] The HPSI system has no impact on the frequency of occurrence of any accident previously evaluated in the USAR since the HPSI system is not an initiator of any analyzed accident. Potential SSC malfunctions have been evaluated and it was determined that this activity will not change the likelihood of occurrence of a malfunction. The proposed activity does not impact the consequences of an accident or a malfunction previously evaluated in the USAR. The proposed change to the HPSI loop injection valves open position setting will ensure that the flow characteristics assumed in the safety analyses are achieved. There is no possibility of an accident of a different type than is currently analyzed in the USAR. Malfunctions of SSCs were evaluated and it was determined that they will not have a different result than previously evaluated in the USAR. The proposed change will not affect ECCS performance and subsequently there is no change in the design basis limits for the fission product barrier as described in the USAR. The proposed activity is a physical modification to the HPSI loop injection valves and does not revise or replace a method of evaluation described in the USAR or used in establishing the design bases or in the safety analyses.

[1] It was later established that a License Amendment Request (LAR) was necessary to add a surveillance requirement to verify the correct position of the valves required to restrict flow in the HPSI system. LAR 14-03 was submitted by OPPD letter LIC-14-0041 dated April 30, 2014.

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Change Number	Activity Title	50.59 Evaluation Summary
EC 60330	Revise Intake Flood Control Procedures To Support Op Eval CR 2011-10302-018	**Activity Description:** The proposed activities are compensatory measures to support an Operability Determination. The compensatory measures are being reviewed to determine the impact of the compensatory actions on other aspects of the facility as described in the UFSAR, not the degraded and non-conforming condition per Section 4.2.5 of FCSG-23, 10 CFR 50.59 Resource Manual. The activities do not make permanent changes to the facility to correct the degraded and non-conforming condition. This is done under the corrective action program. NOD-QP-31.1 "Operability Determination" performed for CR 2011-10302-018 determined that the traveling screen sluice gates are unable to perform their safety related function to re-position during a flooding event due to the submergence of the non-submersible MOV's. This can challenge the raw water path and cell level control in the event the gates need to be re-positioned during a flood. The Operability Determination determined that the gates are degraded and non-conforming and requires compensatory measures for Operability. The compensatory measures are: 1. Install 4 control and 4 isolation manual valves on the 18" Circulating Water (CW) backwash piping inside the Intake Structure. 2. Install passive stem indicators on sluice gates. The stem indicators consist of steel rods attached to the end of the sluice gate stems. When the gates are fully closed, the extension is lined up with a predetermined marking on clear PVC piping which allow operators to see the position of the stem location and determine if the gate is fully closed. 3. Modify the method of technically controlling (mechanical and hydraulic flow function of new valves installed in activity 1) the method of cell level control in the intake cells during a flooding event. This is done by removing the existing method of closing 5 of 6 sluice gates and throttling the remaining gate for a flood up to

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		1004' and then pre-positioning the throttled gate when water rises past 1004' and using at least 1 raw water pump to remove the inflow while maintaining a cell level below 1007.5'. Currently, in the event the primary path is unable to supply sufficient water to the intake structure cells, water passes through the discharge tunnels, into the condenser, through the circulating water pumps and into the cells. The new method closes all 6 sluice gates prior to river level of 1004' and water enters through the new control valves installed in activity 1. 4. Modify the method of procedurally controlling the technical method described in Activity 3 for controlling cell level in the intake structure cells during a flooding event. This is done by using the proposed technical method described in activity 3 and procedurally implementing it, which modifies the sequence of SSC manipulations necessary to control water level in the intake cells. The 50.59 screening preformed determined that further evaluation is required for the following adverse conditions: 1. The addition of the new flood control valves (CW-323 through CW-326) and isolation valves (CW-327 through CW-330) introduce new failure modes that are adverse. 2. Procedure changes that fundamentally alter the existing means of performing or controlling design functions are adverse. These procedure changes introduce new failure modes because of the operator interaction with the new components and the performance of the new methods to control level in the Raw Water pump intake cell. These activities are linked because both activities are directly dependent of each other. Without the valves, the method of flood control cannot take place as proposed for compensatory actions. The final corrective action In the Operability Determination to address the degraded and

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		non-conforming equipment to control intake cell level during a flooding event under AOP-01 "Acts of Nature." (Receive NRC Approval and Implement EC55394 – CR 2011-10302-010 requires NRC approval and Is separate from these interim activities.)[2] **Summary of Evaluation:** Sluice Gate Operability Determination (CR2011-10302-018) proposes to revise AOP-01 to provide Raw Water Pump Intake cell level control during flooding conditions from (current) five sluice gates closed with the sixth sluice gate prepositioned with a small opening to (new) all sluice gates closed. Operations will control level by operating 4 manual control valves. Four redundant (one for each manual control valve) manual isolation valves are also installed. The manual isolation valve can be closed to stop the unwanted flow of water into the Raw Water pump intake cell if the manual control valve falls open. The proposed compensatory action Is more reliable and more flexible, offering additional redundancy than the current configuration. The new compensatory action also allows for adjustment of flow and control of the intake cell level throughout the duration of the flood because the new valves are accessible from inside the intake structure. Therefore, the operability evaluation and associated compensatory actions ensure the Raw Water Pumps remain operable during a flood. The compensatory actions do not result in a malfunction or change to a currently evaluated malfunction that affects the likelihood or consequences of an accident and therefore does not require prior NRC approval.
EC 60974	Tornado Missile Protection Methodology Change	**Activity Description:** This is a change to implement a new design basis methodology for protection of FCS structures, systems, and components using NRC Regulatory Guide 1.76, Revision 1, Design Basis Tornado and Tornado Missiles for Nuclear Power Plants. The change herein implements the design basis tornado characteristics defined in Table 1 for Region I in RG 1.76 Revision 1, as follows:

[2] LAR 13-03 submitted August 16, 2013 by OPPD letter LIC-13-0105.

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		• tornado wind speed of 230 mph • translational speed of 46 mph • maximum rotational speed of 184 mph • radius of maximum rotational speed of 150 ft. • pressure drop of 1.2 psi • rate of pressure drop of 0.5 psi/sec. The change also implements the tornado-generated missile spectrum and maximum missile speeds (V^{max}) from Table 2 for Region I in RG 1.76 Revision 1 as follows: Missile Type / Schedule 40 Pipe / Automobile / Solid Steel Sphere Dimensions / 6.625" dia x 15' lg / 16.4' x 6.6' x 4.3' / 1" dia Mass[3] / 287 lb. / 4000 lb. / 0.147 lb. C_DA/m / 0.0212 ft^2/lb / 0.0343 ft^2/lb / 0.0166 ft^2/lb V_{Mh}^{max} / 135 ft/s / 135 ft/s / 26 ft/s The missiles are considered capable to strike in all directions with horizontal velocities of V_{Mh}^{max} and vertical velocities equal to 67 percent of V_{Mh}^{max}. Barriers are designed for normal impact to the surface of the schedule 40 pipe and automobile missiles. The automobile missile is considered to impact at all altitudes less than 30 ft. above all grade levels within 0.5 mile of the plant structures. Revision of the USAR sections 5.4, 5.8, 5.11, 5.13 and Criterion 2 of Appendix G to adopt RG 1.76, Revision 1 as the new design methodology for Fort Calhoun Station represents adoption of a replacement methodology. RG 1.76, Revision 1 is adopted in its entirety, without exception, for future design changes to SSCs, to new plant systems and to evaluate the existing structures, systems, and components required to withstand tornado

[3] The values for mass in Table 2 of RG 1.76, Revision 1 are given in Kg and lb. For clarification, the mass in pounds (lb) should be construed to mean pounds-mass (lbm).

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Change Number	Activity Title	50.59 Evaluation Summary
		loadings and missiles. The scope of this 50.59 is to evaluate the USAR updates required to change the tornado and tornado missile protection methodology from NAV Docks P-51, Design of Protective Structures by A. Amirikian, Bureau of Yards and Docks, Department of Navy, August 1950 to RG 1.76, Revision 1, Design Basis Tornado and Tornado Missiles for Nuclear Power Plants. This 50.59 does not authorize any future physical plant modifications. **Summary of Evaluation:** This is a change to implement a new design basis methodology for protection of FCS systems and components using guidance from NRC Regulatory Guide 1.76, Revision 1, Design Basis Tornado and Tornado Missiles for Nuclear Power Plants. In accordance with the FCS licensing basis, equipment that is relied on to shutdown the plant shall be designed to withstand the effects of natural phenomena, such as tornadoes, without loss of capability to perform their required functions. Regulatory Guide (RG) 1.76, Revision 1 issued in March 2007 indicates, "this regulatory guide provides licensees and applicants with new guidance that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in selecting the design-basis tornado and design-basis tornado-generated missiles that a nuclear power plant should be designed to withstand to prevent undue risk to the health and safety of the public." As such, the NRC has established that the generic methods and guidance provided in the Regulatory Guide are acceptable for use to satisfy the licensing requirements. In addition, RG 1.76, Revision 1 has been approved for use at nuclear power facilities in the same Tornado Intensity Region as Fort Calhoun Station. Use of RG 1.76 Revision 1 has been incorporated by Watts Bar Nuclear Plant (WBN) and has been formally accepted for use at WBN via a SER.

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Change Number	Activity Title	50.59 Evaluation Summary
		NEI 96-07, Revision 1 and the FCS 10 CFR 50.59 Resource Manual were consulted to evaluate if RG 1.76, Revision 1 meets the definition of a replacement methodology and whether it can be used without prior NRC approval. The evaluation supports the conclusion that RG 1.76, Revision 1 can be implemented at FCS without prior NRC approval.[4]
EC 61354	VA-71A & B Battery Room Ventilation Tornado Missile Protection	**Activity Description:** The Omaha Public Power District Fort Calhoun Station (OPPD / FCS) is installing a tornado missile barrier on the VA-71A Battery Room Number 1 Exhaust Fan and VA-71B Battery Room Number 2 Exhaust Fan. The purpose of these new tornado missile proof exhaust assemblies is to provide an acceptable method to protect the louvers from potential tornado missiles. The new louvers are constructed of plating and steel, and will increase the structural loadings on the Auxiliary Building wall, to which they are attached. Calc FC00577 Rev 1 has been performed, and has determined that there are no adverse effects on system operation as a result of this change. Calc FC08272 Revision 0 has been created, and has evaluated the barrier design as acceptable to protect against the selected missile spectrum. A review of the Preliminary Safety Analysis Report (PSAR), Final Safety Analysis Report (FSAR), FSAR Questions and Answers, and Updated Safety Analysis Report (USAR) concluded that the original design of OPPD / FCS for tornado missile protection was based on the validation of structures and their ability to survive impacts by tornado missiles. This is documented in Appendix G of the USAR, which provides documentation of how the design of OPPD / FCS complies with the Draft General Design Criteria (GDC) that were released by the Atomic Energy Commission (AEC) as OPPD / FCS was being constructed. In Appendix G, Draft Criterion 2, Performance Standards, addresses tornado missile protection for systems and components by stating:

[4] Contrary to the conclusion of the 10 CFR 50.59 evaluation, OPPD submitted a license amendment request (LAR) to revise the licensing basis and permit use of RG 1.76, Revision 1. The NRC approved the LAR in Amendment No. 272.

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Change Number	Activity Title	50.59 Evaluation Summary
		"Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by the natural phenomena such as earthquakes, tornados, flooding conditions, winds, ice, and other local site effects. The design basis so established shall reflect: (a) Appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and surrounding areas, and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design." FCS documented how it met Criterion 2 in Appendix G of the FSAR, and it shows the focus of those efforts. This focus was consistent with how licensees, who were mid-completion in their construction products, addressed the requirements of GDC 2. Notes are added in braces [ ] to indicate whether each statement is addressing a system, component, or a structure. "This criterion is met. The systems and components of the Fort Calhoun Station, Unit No.1 reactor facility that are essential to the prevention or mitigation of accidents that could affect public health and safety are designed, fabricated, and erected to withstand without loss of capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, floods, winds, ice and other local site effects. The containment [Structure] will be designed for simultaneous stresses produced by the dead load, by 60 psig internal pressure at the associated design temperature, and by the application of forces resulting from an earthquake whose ground motion is 0.08g horizontally and 0.053g vertically. Further, the containment structure [Structure] will be designed to withstand a sustained wind velocity of 90 mph in combination with the dead load and design internal pressure and temperature conditions. The wind load is based on the highest velocity wind at the site location for 100-year period of recurrence: 90 mph

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Change Number	Activity Title	50.59 Evaluation Summary
		base wind at 30 feet above ground level. Other Class I structures [Structure] will be designed similarly except that no internal pressure loading is applicable. Class I systems will be designed for their normal operating loads acting concurrently with the earthquake described above. [System, but earthquake loads only addressed, not tornado missile] The containment structure [Structure] is predicted to withstand without loss of function the simultaneous stresses produced by the dead load, by 75 psig internal pressure and temperature associated with this pressure and by an earthquake whose ground motion is 0.10g horizontally and 0.07g vertically. The containment structure [Structure] is predicted to withstand without loss of function 125% of the force corresponding to a 90 mph wind impinging on the building concurrently with the stresses associated with the dead load and 75 psig internal pressure. With no earthquake or wind acting, the structure [Structure] is predicted to withstand 90 psig internal pressure without loss of function. Under each of these conditions, stresses in the structural [Structure] members will not exceed 0.95 yield. The facility is designed so that the plant can be safely shutdown and maintained in a safe shutdown condition during a tornado. Design considerations associated with tornados are further explained in Section 5.4.7 of the USAR. . ." The focus on structures is further highlighted in various licensing documents: 5.11.2 Design of Structures - Class I (FSAR) Class I structures are designed to ensure that their functional integrity under the most extreme environmental loadings, such as tornado or maximum hypothetical earthquake, will not be impaired and thereby prevent a safe shutdown of the plant.

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		6.2.4.1 General . . . The SIRW tank is located at the southwest side of the auxiliary building basement. The safety injection system pumps are located in two watertight rooms at the lowest level of the auxiliary building in a tornado proof area. This location assures adequate pump suction head when recirculating from the containment sump. Sufficient space is provided around equipment in these rooms to permit installation of temporary shielding for maintenance. Valves required to isolate equipment are provided with remote operators. From Supplement 8 of the PSAR, in response to Question 13.1: All Class I equipment, namely raw water pumps and associated piping are mounted in individual cubicles located below the operating floor. Entrance to the cubicles is from the operating floor, eliminating flooding of the cubicles. The operating slab above these cubicles will be designed to provide the necessary protection for tornado induced loads and falling debris. Figures 13.1-1 and 13.1-2 (Revised 10-26-67) delineate, schematically, the raw water piping layout. As will be noted all raw water piping in the pump house is below the operating floor and thus protected from tornado induced loads and missiles and falling debris. The raw water piping between the pump house and the reactor auxiliary building (Class I structure) will be run in separate trenches and will use a concrete cradle bedding (Class A). In addition, Draft GDC 40 addressed Missile Protection to assess Engineered Safety Features against the dynamic effects and missiles that might result from plant equipment failures. There was no GDC Criterion 4, Environmental and Dynamic Effects Design Bases, as it currently exists in 10CFR 50 Appendix A, in the Draft GDCs. Similarly, there is no version of Draft GDC 40 in the current version of 10CFR 50 Appendix A.

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Change Number	Activity Title	50.59 Evaluation Summary
		From the above, the OPPD / FCS approach, as well as the approach of other facilities constructed in the same time period, addressed missile impacts on structures, and in doing so, provided protection for systems and components. As part of the physical upgrade to improve plant safety margins, OPPD / FCS adopting contemporary guidance to expanding the performance requirements for systems and components in its design and licensing basis. These include the following products, which will be used to address the impact on tornado missiles on OPPD / FCS systems, structures and components: 1. Creation of EA 13-014, Tornado Safety Shutdown Analysis, Mode 5 Operation. This will provide the target selection criteria, based on safe shutdown requirements to support this expansion of the design and performance requirements to additional SSCs. This document will be issued by this design change. 2. Adoption of Reg Guide 1.76 Revision 1, Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants, which will be used to provide the regionalized selection of the Design-Basis Tornado and Tornado Missiles, based on tornado inputs from across the country. OPPD/FCS is in Region I of the Tornado Characteristics. 3. Use of Bechtel BC-TOP-9A Revision 2, (September, 1974) AEC approved Topical Report, Design of Structures for Missile Impacts (Reference letter from R.W. Klecker, Technical Coordinator for Light Water Reactors Group 1, Directorate of licensing, Atomic Energy Commission, to John V. Morowski, Bechtel Power Corporation, dated November 25, 1974), to provide the AEC approved methodology for evaluating the effect of missiles on concrete and steel barriers. 4. Use of Bechtel Design Guide Number C-2.45, Design of Structures for Tornado Missile Impact, Revision 0, to provide a verification of the results using a later Bechtel approach. 5. Use of NUREG-0800, Standard Review Plan (SRP) 3.5.3 Revision 3, to provide appropriate NRC endorsed acceptance criteria for the Design-Basis Tornado

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		Missile impact, e.g., verification that OPPD / FCS provides adequate barrier thickness to prevent perforation and to prevent spalling or scabbing when protection from spalling or scabbing is considered necessary. The intent of this change is to improve and upgrade FCS by using the design and licensing guidance above for developing new defenses and increasing safety margins for systems and components. This will improve and upgrade the FCS tornado protection and analysis for structures, systems and components to conform with the latest available industry and NRC guidance, and in doing so, meet the requirements of the Draft General Design Criterion 2 of the FCS USAR, Appendix G. Our review of Figure 1, "Tornado intensity regions for the contiguous United States for Exceedance Probabilities of 10^{-7} per year" of Reg Guide 1.76 Revision 1, FCS falls within Region I. The change to the USAR herein implements the design-basis tornado characteristics defined in Table 1 for Region I in RG 1.76 Revision 1, as follows: • Maximum (tornado) wind speed: 230 mph • Translational speed: 46 mph • Maximum rotational speed: 184 mph • Radius of maximum rotational speed: 150 ft. • Pressure drop: 1.2 psi • Rate of pressure drop: 0.5 psi/s The change also implements the Design Basis Tornado Missile Spectrum for Region I, documented in Table 2, Design-Basis Tornado Missile Spectrum and Maximum Horizontal Speeds, from Reg Guide 1.76 Revision 1 as follows:

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		(see nested table below) As required by Reg Guide 1.76 Revision 1, the missiles are considered capable to strike in all directions with horizontal velocities of V_{Mh}^{max} and vertical velocities equal to 67 percent of V_{Mh}^{max}. Barrier designs will be evaluated assuming a normal impact to the surface for the schedule 40 pipe and automobile missiles. The automobile missile is considered to impact at all altitudes less than 30 ft. above all grade levels within 0.5 mile of the plant structures. The USAR change will document that FCS is adopting the design-basis tornado from the Table 1 regional values as required, and that a "less-conservative parameter tornado than the regional values in Table 1" will not be used, as required by the Regulatory Position, Section 1, and that the FCS tornado-generated missile spectrum will comply with Table 2. Adoption of Regulatory Guide 1.76 Revision 1, Bechtel BC-TOP-9A Revision 2, and SRP 3.5.3 Revision 3 Acceptance Criteria as the new design methodology for Fort Calhoun Station, and adding this methodology to the USAR for those systems, which are determined to require expansion of their design and performance requirements to improve the OPPD/FCS safety margins, represents a replacement methodology. These documents are adopted in their entirety in the case of Reg Guide 1.76 Revision 1 and BC-TOP-9A Revision 2, and in the case of SRP 3.5.3 Revision 3, for the Acceptance

Missile Type	Schedule 40 Pipe	Automobile	Solid Steel Sphere
Dimensions	6.625" dia x 15' lg	16.4' x 6.6' x 4.3'	1" dia
Mass[5]	287 lb.	4000 lb.	0.147 lb.
C_DA/m	0.0212 ft^2/lb	0.0343 ft^2/lb	0.0166 ft^2/lb
V_{Mh}^{max}	135 ft/s	135 ft/s	26 ft/s

[5] The values for mass in Table 2 of RG 1.76, Revision 1 are given in Kg and lb. For clarification, the mass in pounds (lb) should be construed to mean pounds-mass (lbm).

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		Criteria, for future design changes to SSCs, to new plant systems and to evaluate the existing structures, systems and components that were previously not evaluated and are now being improved to increase safety margins for the Design-Basis tornado and tornado missiles. The original analysis method (Nav Docks P-51, missile sets, and wind velocities) will remain in place for the previously analyzed structures of OPPD/FCS. OPPD/FCS is adopting Reg Guide 1.76 Revision 1, Bechtel BC-TOP-9A Revision 2, and SRP 3.5.3 Revision 3 because the Fort Calhoun Station (FCS) licensing basis is not clear with regards to Design Basis tornado and tornado missile protection in all aspects of its design and licensing basis. The adoption of the guidance provided by Reg Guide 1.76 Revision 1, Bechtel BC-TOP-9A Revision 2, and SRP 3.5.3 Revision 3 provides an improvement in the FCS overall design, as well as this replacement methodology providing a clearer, definitive, and much less ambiguous standard for use when considering future changes. The use of NUREG-0800, SRP 3.5.3 for the acceptance criteria is acceptable, as the document clearly states that it is providing acceptance criteria necessary to meet the relevant requirements of GDC 2 and 4. A comparison of the Draft GDC 2 and the current 10CFR50 Appendix A GDC 2 was conducted, and found that the two GDCs are similar enough to apply the acceptance criteria from this to a Draft GDC 2 plant, and still be within the use as approved for the intended application. Creation of EA13-014 to evaluate the need to protect additional SSCs from the Design-Basis Tornado and Tornado missiles is within the scope of the licensee's authority when voluntarily expanding equipment design and performance requirements to increase safety.

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		The use of an industry methodology such as Bechtel C-2.45 for validation of the results of the approved method is a reasonable use of such an industry standard. This is not being used for primary design activities, so prior NRC approval for use is not required.[6] **Summary of Evaluation:** This is a change to implement a new design methodology for protection of FCS systems and components using guidance from NUREG-0800 SRP 3.5.3 Revision 3, Reg Guide 1.76 Revision 1, and Bechtel BC-TOP-9A Revision 2. The development and use of EA13-014, Tornado Safety Shutdown Analysis to provide the target selection criteria, based on safe shutdown requirements to which Design-Basis Tornado Missiles will be applied is appropriate, and may be implemented without prior NRC approval. The proposed changes are increasing the margins of safety and protection by expanding the design and performance requirements for these new SSCs, and protecting SSCs that were previously unprotected from the design-basis tornado missiles. The determination of the expanded scope of systems is an improvement to the facility. In accordance with the FCS licensing basis, equipment that is relied on to shut down the plant shall be designed to withstand the effects of natural phenomena, such as tornadoes, without loss of capability to perform their required functions. Regulatory Guide 1.76 Revision 1 issued in March 2007 indicates "this regulatory guide provides licensees and applicants with new guidance that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in selecting the design-basis tornado and design-basis tornado-generated missiles that a nuclear power plant should be designed to withstand to prevent undue risk to the health and safety of the public." As such the NRC has established that the generic methods and guidance provided in the Regulatory Guide are acceptable for use to satisfy the licensing requirements. In addition,

[6] Contrary to the conclusion of the 10 CFR 50.59 evaluation, OPPD submitted a license amendment request (LAR) to revise the licensing basis and permit use of RG 1.76, Revision 1. The NRC approved the LAR in Amendment No. 272.

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Change Number	Activity Title	50.59 Evaluation Summary
		Reg Guide 1.76 Revision 1 has been approved for use at nuclear power facilities in the same Tornado Intensity Region as Fort Calhoun Station, as documented in NUREG-0847, Supplement 22, Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Unit 2. The use of Bechtel BC-TOP-9A Revision 2, (September 1974) and it's AEC approved Topical Report, Design of Structures for Missile Impacts (Reference letter from R.W. Klecker, Technical Coordinator for Light Water Reactors Group 1, Directorate of Licensing, Atomic Energy Commission, to John V. Morowski, Bechtel Power Corporation, dated November 25,1974), provide the AEC approved methodology for evaluating the effect of missiles on concrete and steel barriers, and is appropriate. The use of NUREG-0800 Standard Review Plan (SRP) 3.5.3 Revision 3, to provide acceptance criteria for the analysis of the results of the impact of the design basis tornado missiles on systems and components was prepared to establish criteria that the NRC staff uses in evaluating whether a licensee meets the NRC's regulations. The acceptance criteria within 3.5.3 states "Specific criteria necessary to meet the relevant requirements of GDC 2 and 4 are as follows." The acceptance criteria are being used *"en toto"* which is acceptable as it will be used as approved for the intended application. The use of Use of Bechtel Design Guide Number C-2.45, Design of Structures for Tornado Missile Impact, Revision 0, to provide an alternate method for verification of results using a later Bechtel approach to evaluate the effect of missiles on concrete and steel barriers is appropriate and conservative given the use of the above methodology, and is within the licensee's authority to implement without prior NRC approval, as we are expanding the scope of our protected components beyond what was originally described to in our response to how OPPD / FCS is in compliance with Draft GDC 2, in Appendix G of the USAR. This is not being used for primary design activities, so prior NRC approval for use is not required. NEI 96-07 Revision 1 and the FCSG-23, 10 CFR 50.59 Resource Manual, Revision 8 were consulted to evaluate if the use of Reg Guide 1.76 Revision 1, Bechtel BC-TOP-9A Revision 2, and SRP 3.5.3 Revision 3 acceptance criteria meet the definition of a replacement methodology, and whether it can be used without prior NRC approval. The evaluation

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		supports the conclusion that Reg Guide 1.76 Revision 1, Bechtel BC-TOP-9A Revision 2, and SRP 3.5.3 Revision 3 Acceptance Criteria can be implemented at FCS without prior NRC approval as together they provide an acceptable methodology that has been approved by the NRC for the intended application. As noted in the associated 50.59 screening, the new louvers have been evaluated as capable of withstanding the Reg Guide 1.76 Revision 1 missiles as required. The evaluation included the additional weight of the missiles on the Aux Building structural loading, and determined that the additional weight was acceptable, and did not affect the seismic capability of the structure. Calc FC00577 Rev 1 has determined that there are no adverse effects on system operation as a result of this change. Calc FC08272 Revision 0 has evaluated the barrier design as acceptable to protect against the selected missile spectrum. Based on the above, this activity can be implemented without prior NRC approval.
EC 61770	Change to HELB Design Requirements to allow crediting the break exclusion zone at Penetrations M-2, M-3, M-10 and M-13	**Activity Description:** Currently, FCS considers high energy line breaks (HELBs) to occur between the outside Containment Isolation Valves (CIVs) and the Containment wall for the following penetrations: a. Penetration M-2 to HCV-204 - Letdown b. Penetration M-3 to CH-198 - Charging c. Penetration M-10 to HCV-1387B - Steam Generator Blowdown from SG 2B d. Penetration M-13 to HCV-1388B - Steam Generator Blowdown from SG 2A The current HELB consideration extends the harsh environment in the Auxiliary Building to areas beyond Room 13 (the Mechanical Penetration Room) which are not currently considered harsh. The proposed activity is to eliminate the consideration of HELBs between the outside CIVs and the Containment wall for the four penetrations identified above, by adopting BTP MEB 3-1.

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		The reason for this proposed activity is to not have to consider a harsh environment in the Auxiliary Building areas beyond Room 13 that were not previously considered harsh. **Summary of Evaluation:** This change introduces a more than minimal increase in the likelihood of occurrence of malfunction of an SSC important to safety and the possibility of an equipment malfunction with a different result and therefore requires prior NRC approval.[7] A License Amendment Request is required to obtain NRC review of the changes crediting the exclusion of the piping between the containment isolation valves from consideration of breaks. EC 61770 proposes modified criteria for determining break locations that includes exclusion of the piping between containment penetrations. USAR Appendix M and licensing letter LIC-73-0012 specifically include consideration of breaks at the containment wall. The addition of this exclusion is a change that requires prior NRC review and approval of the piping and penetration configuration, design and inspection criteria to ascertain conformance to the NRC's criteria for excluding breaks.
EC 62416	Temporary Modification - Throttle Discharge Valves HCV-2958, HCV-2968 and HCV-2978	**Activity Description:** The Containment Spray Pump discharge gate valves, HCV-2958, HCV-2968, and HCV-2978, are being throttled to increase system hydraulic resistance and limit flow to prevent run out of the Containment Spray Pumps SI-3A, SI-3B, and SI-3C. The Temporary Modification will include: • Throttling and locking of the discharge valves to provide adequate flow to the system and remain within the design envelope (e.g., pump flow, motor amps), of the pump in service. • Installation of "Throttle Position Indicator" collars on the actuator shafts of the discharge valves. The collars will provide a visual stop indication to assure the valves are set to their required throttled position or can be reset, to their proper

[7] LAR 13-08 submitted on October 6, 2013 by OPPD letter LIC-13-0146. Amendment No. 273 was issued on October 25, 2013.

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		post-Temp Mod configuration without additional flow testing. • Adjustment of the "Open" limit switches on the discharge valves to indicate "Open" in the new temporary position of the valve. The function of the Containment Spray (CS) System is to limit the Containment pressure rise by providing a means for cooling the Containment following a Main Steam Line Break (MSLB). The 50.59 screen has determined this function is adversely affected. Specifically, throttling the Containment Spray pump discharge gate valves will potentially affect the valves and piping due to flow induced cavitation, erosion, and vibration. Therefore, further evaluation is required. **Summary of Evaluation:** The Containment Spray Pump Discharge Valves, HCV-2958, HCV-2968 and HCV-2978, are being throttled to increase hydraulic resistance and prevent run out of the Containment Spray Pumps. Containment Spray pumps can operate beyond the CS pump motor nameplate service factor when the gate valves are full open. The worst-case condition identified is when 2 CS pumps start providing flow to 2 spray header configuration and a single failure of one CS pump (sheared shaft or failed coupling or a CS pump discharge check valve failing to open) occurs. This results in one CS pump providing flow to two CS headers. Due to these conditions, the CS operating CS pump motor can draw more amps than analyzed for, which could adversely affect motor operation. Throttling of the CS Pump discharge gate valves will result in the CS pumps operating within their analyzed design flow rate and motor ratings. Containment Spray Pump SI-3C is tagged out of service. This prevents testing of the pump and the throttle setting of HCV-2978 until SI-3C is repaired and plant conditions / Technical Specifications allow.

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		Throttling the CS pump discharge gate valves was evaluated for a potentially adverse effect on the valves, piping and system components due to flow induced cavitation, erosion and vibration. An effect on the valves or piping could potentially affect the ability of the Containment Spray system to perform its design function. Evaluation of the valves and piping system components has determined that this affect will not result In the inability for the CS system to perform Its design basis function. This conclusion Is supported by the very short duration for performing the CS system function in addition to the Inherent ruggedness of the valve design and materials (stainless steel being a highly resistant material for erosion and cavitation) and post modification testing to ensure flow rate requirements and piping vibration are acceptable. The Containment Spray Pump and motor have been tested and meet the IST Program vibration testing acceptance criteria. Therefore, the design basis function of the Containment Spray System as described In the USAR will be met with the Installed temporary modification. A license amendment request Is not required.
EC No.: 32387 Rev. 0 / FDCR 60365	EC 32387 - Turbine Controls System Replacement	**Activity Description:** The proposed activity consists of (1) an upgrade of the controls for the main turbine and (2) changes to the steam dump and bypass and reactor regulating systems. It is part of an overall phased project to upgrade select plant instrumentation and control (I&C) systems and integrate them into a single distributed control system (DCS). Control system functions will be performed using digital instead of analog devices. The turbine emergency trip system will use redundant electrical trains, and the mechanical trips will be eliminated. The operator interface currently performed using analog meters, indicating lights, push buttons, and rotary knobs on the main control board-will be performed using touchscreen monitors and associated workstations. The proposed activity is being undertaken for two reasons: (1) to improve plant control

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		system reliability by replacing components that are or will soon become obsolete and by eliminating various single point vulnerabilities and (2) to automate certain tasks currently performed manually by the operators. For the most part, the proposed changes do not alter existing system functions but, rather, modify the way these functions are performed or controlled in one or both of the following ways: 1. The platform used to perform these functions is being changed from analog components to a network of digital devices. 2. The human-machine interface is being altered by the introduction of touchscreen monitors in place of the existing control room devices. These changes are conservatively treated as adversely affecting the relevant USAR-described functions and. therefore, are addressed in the evaluation. The proposed activity also includes changes in the method of performing or controlling a USAR-described function that involve additional considerations: 3. The method of performing the turbine emergency trip and overs peed protection functions. 4. The change from manual to automatic operation of the steam dump and bypass pressure relief and cooldown functions and the sequencing of the dump and bypass valves. 5. The elimination of the manual selector switch (in the reactor regulating system) which allows the operator to select the Tave channel used in the steam dump and bypass control circuits, in favor of the selection logic that determines the Tave signal to be used. These changes are also addressed in the evaluation.

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		Summary of Evaluation: The proposed activity consists of (1) an upgrade of the controls for the main turbine and (2) changes to the steam dump and bypass and reactor regulating systems. It is part of an overall phased project to upgrade select plant instrumentation and control (I&C) systems and integrate them into a single distributed control system (DCS). Control system functions will be performed using digital instead of analog devices. The turbine emergency trip system will use redundant electrical trains, and the mechanical trips will be eliminated. The operator interface - currently performed using analog meters, indicating lights, push buttons, and rotary knobs on the main control board - will be performed using touchscreen monitors and associated workstations. The proposed activity is being undertaken for two reasons: (1) to improve plant control system reliability by replacing components that are or will soon become obsolete and by eliminating various single point vulnerabilities and (2) to automate certain tasks currently performed manually by the operators. Since single point vulnerabilities are reduced and system dependability is improved, there is no increase in the frequency of occurrence of an accident previously evaluated in the USAR or in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the USAR. Because the proposed activity involves changes to plant control systems, it has the potential to affect events classified under the categories "normal operation" or "incidents of moderate frequency". However, these do not have radiological consequences. Therefore, the proposed activity does not increase the consequences of an accident previously evaluated in the USAR and does not increase the consequences of a malfunction of an SSC important to safety previously evaluated in the USAR. Any new failure modes are consistent with or bounded by previously evaluated accidents

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		and malfunctions. Therefore, the proposed activity does not create a possibility for an accident of a different type than any previously evaluated in the USAR or a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR. Since the proposed activity maintains the existing functionality of the turbine control and turbine trip, steam dump and bypass, and reactor regulating systems, control bands and protection system setpoints are not changed. Therefore, the proposed activity does not result in a design basis limit for a fission product barrier as described in the USAR being exceeded or altered. Finally, the proposed activity does not change any method of evaluation described in the USAR used in establishing the design bases or in the safety analyses. Therefore, the proposed activity can be implemented without prior NRC approval.
EC 33464 (FDCR 59413)	EC 33464 - Replace AK-50 480V Main & Bus-Tie Breakers With Molded Case type or Equivalent	**Activity Description:** The 480 VAC load center main and bus tie breakers, GE AK-50 are obsolete and no longer manufactured by the OEM (FCS Tags: 1B3A-1B3A, 1B3B-1B3B, 1B3C-1B3C, 1B4A-1B4A, 1B4B-1B4B, 1B4C-1B4C, BT-1B3A, BT-1B3B, BT-1B3C, BT-1B4A, BT-1B4B, BT-1B4C). These breakers and their trip devices will be replaced with NLI/Square D Masterpact breakers and digital trip devices. Differences between the original GE Breaker and the NLI Breaker Original GE AK-50 The original GE AK-50 Circuit Breaker was designed to be installed within the tolerances of the GE switchgear it was installed in. The breaker had 4 bolts (2 per side) for mounting, which fit within the "drawout" rails of the cubicle. When racked in, these 4 bolts in the "drawout" rails assured that the breaker was installed and would properly connect

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		with the associated cubicle. The primary disconnects between the circuit breaker and the bus bars were large, had 8 fingers (4 top, 4 bottom) for each phase of the connection (6 connections, 3 for the line in, 3 for the load out), and designed to work within the tolerance of the existing GE buswork. The primary disconnects were not sensitive to bus bar angle in any axis. Control signals and position information were transferred into and out of the breaker by the means of spring loaded contacts, which contacted fixed rails in the breaker cubicle. 125 VDC was provided for trip and close operations. Original GE AK-50 Overcurrent Protection The original GE AK-50 circuit breaker was equipped at the start of plant operation with dashpot overcurrent devices, which were located within the frame of the circuit breaker. Due to problems with reliability and repeatability of the dashpot devices, they were changed to the GE RMS-9 Trip Modules, which replaced the electro-mechanical dashpot with a digital trip device, and were mounted on the frame of the circuit breaker inside the door. Issues with spurious tripping of the RMS-9 resulted in their being replaced with the Westinghouse Amptector, which was a similar device that was mounted on the outside of the cubicle door. Replacement NLI Masterpact Note: For clarity, the Square D / NLI breaker cradle, and Micrologic Trip Unit comprise the "assembly" that replaces the GE AK-50 series breaker, and in this document, this "assembly" is hereafter referred to as the "NLI Breaker Package." Individual parts of the assembly are referred to separately when necessary. The new NLI/Square D Circuit Breakers are physically smaller than the existing GE breakers, use pin-and-socket type connectors, and have a different connection system (different shape, orientation) to connect to the bus bars. As a result, the NLI/Square D Circuit Breaker requires an "adapter" to interface it with the existing GE switchgear cubicle. This "cradle" as it is called, is supplied with each new NLI/Square D Circuit

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		Breaker, and adapts the NLI / Square D Circuit Breaker and associated connections for power, control, indication, and circuit breaker functions to the GE switchgear cubicle. The outside dimensions and electrical connections on the cradle are similar to the existing GE switchgear breakers. There are some differences, which include: The NLI cradle primary disconnects (connect the breaker to the electrical bus and load) are designed to exceed the performance of the original GE disconnects. This required a change in the design from the large footprint GE primary disconnects to a different design that uses more and smaller elements. These connection "fingers" number 20 per phase (10 top, 10 bottom), for each of the 6 power-carrying phase connections, to connect to the original GE switchgear bus bars. The "fingers" are narrow and longer than the original GE primary disconnects (which connect the circuit breaker to the line and load), and are not able to consistently accept the unadjusted GE switchgear bus bar construction tolerance without adjustment to the switchgear bus bars. Section 2 of EC 33464 contains guidance on the "normalization" of the switchgear, to allow the NLI primary disconnects to properly connect to the existing bus bars. This includes criterion for: • Cleanliness of the bus bars. If hardened grease is present, some of the NLI disconnect "finger elements" can be displaced, and not make the required contact, • Alignment (normalization), which is bending the existing GE bus bars to meet the installation tolerance of the NLI Breaker Package, which assures the bus bar will connect with the NLI primary disconnects and not be displaced, and • The NLI primary disconnect is properly aligned with the bus bar with all fingers in contact, and all fingers of each phase's connection on the silver plating of the existing bus bar. • Acceptance criteria for Digital Low Resistance Ohmmeter (DLRO) Testing to assure an acceptably resistance level is present in the connection.

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		In addition, the NLI cradle does not automatically line up the way the previous GE circuit breakers would when installed in the "drawout" rails. This requires detailed inspections during cradle installation in the GE switchgear using remote monitoring equipment (e.g., borescope or equivalent) to assure that the NLI breaker cradle primary disconnects land on the approximate 1 inch silver-plated tip of each associated bus bar. (Note: As part of the actions out of RCA 2011-5414, the bus feed and cross tie breaker bars have been entirely silver plated to remove this design vulnerability. The borescope or equivalent inspection is still required, as all of the disconnect "fingers" must be verified to be in contact with the associated bus bar.) Once installed and visually verified as described above, a stop bolt is tightened on the cradle, which locks the cradle into position. As the new breakers have a different faceplate, a new breaker door is supplied each NLI / Square D Breaker Package (Reference FDCR 47687). The inside dimensions and electrical connections on the NLI cradle are designed to accept the new NLI/Square D breaker. Replacement NLI Masterpact Overcurrent Protection The Square D breaker installed in the cradle has a Square D Micrologic Trip Unit, which is in a recess in the face of the Square D circuit breaker. The Micrologic Trip Unit is a digital trip device that can be field tested without current injection testing using a "Full Function Test Kit," (FFTK) which provides simulated current to test-trip the circuit breaker. This is advantageous for routine trip tests, as it avoids damage to the load breaking contacts of the circuit breaker, which would otherwise have to discharge the trip current and incur minor wear in the process. As described in Section 2 of EC 33464, the post-maintenance testing of the new NLI Breaker package requires the breaker to be trip tested using current injection across the input and output circuit breaker connections, and not using the FFTK to assure that proper breaker coordination is verified. The Micrologic Trip Unit is programmed with the required trip curve trip curve to maintain coordination.

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		The Micrologic Trip Unit in the NLI Breaker Package uses a lithium battery to provide indication of a trip. The previous Amptector directly generates this indication without the need of a battery. The Micrologic Trip Unit protects the downstream loads (e.g., motors and motor control centers). The Micrologic Trip Unit consists of a set of current transformers called CTs or sensors) to sense current, a trip unit to evaluate the current, and a tripping solenoid to trip the circuit breaker. The trip unit samples the analog current signal and converts it to a digital signal to allow for more accurately measured magnitudes of a non-sinusoidal waveform. The NLI Breaker Package has a "Zone Selective Interlock" or ZSI capability, which allows these breakers to be interconnected with similar breakers with wiring, and to cause preferential tripping of breakers to protect specific zones. This feature is not utilized in this modification, as it would violate electrical separation requirements that are present in nuclear plant design; however, it must properly disabled by installing jumpers in the correct blocks as called out in Section 2 of EC 33464. The effect of not installing the jumpers correctly is that the NLI Breaker Packages will not coordinate as required by the design. Section 2 calls out the required visual and continuity testing for these ZSI jumpers, which in turn with current injection post-installation testing, assures that breaker coordination is maintained as called out in USAR Section 8.3.2.3. General Change Information Although physical wiring changes in the existing GE switchgear are not anticipated, differences in the breakers internal wiring and internal component arrangement make it necessary to revise the existing plant wiring/schematic drawings. The bell alarm contact for the NLI Breaker Package has a current rating of 0.3A at 125VDC. This contact, when closed, actuates lockout relay BAX associated with each load center main feeder breaker. The relay contacts are all currently spared and the

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		relay does not provide any design function. Since the bell alarm contact rating is not compatible with the relay current rating, this relay will be disconnected and abandoned in place. The 50.59 screening that led to this evaluation was being revised due to concerns identified in Condition Reports CR 2012-02652 and CR 2012-04114 with the adequacy of the 50.59 review for EC 33464, as well as those issues identified in Root Cause Analyses (RCA) 2011-5414 and 2011-6621. The design functions in Section III have been numbered to make them easier to refer to in the justification discussions in Section IV. A Failure Modes and Effects Analysis (FMEA) has been prepared to identify and compare new component failure modes to those failure modes for the existing breakers, and is captured in EC 33464. Adverse results of the FMEA, CRs 2012-02652, 2011-04114, and RCAs 2011-5414 and 2011-6621 are addressed in this 50.59 Evaluation. New Failure Mode Sources Identified in CR 2012-02652, CR 2012-04114, RCA 2011-5414 and RCA 2011-6621: The new NLI Breaker Packages are considered alternate replacements that were equal to or better than the existing GE breakers. However, there are some design differences between the original and replacement breakers that warrant additional discussion as a potential failure mode. While the replacement breakers perform the same function as the original breakers, the following new failure mode sources must be considered for the original failure modes: (CR 2012-02652, CR 2012-04114, RCA 2011-5414, RCA 2011-6621) Breaker Failure to Open (locally, remotely) Breaker Failure to Close (locally, remotely) Breaker Failure to Coordinate

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		Note: If a new failure mode source was considered plausible either by NLI or the corrective action program documentation, it was included. These new sources of failure modes and their impacts are described below: Change 1: Thermal Imaging / Memory - The NLI Breaker Package comes with "Thermal Imaging" or "Thermal Memory." This is a new function that was not part of the original GE breakers and their respective trip devices. According to Square D (the breaker and trip device OEM), this function monitors the current over a rolling period of 15 minutes. If the current approaches the trip setpoint, the thermal memory starts accumulating. If over the course of the next 15 minutes the "Thermal Imaging" reaches the trip setpoint, it will trip the associated circuit breaker. Change 2: Interfacing Connection between the Square D Breaker and NLI Cradle, and the GE Switchgear. This connection is not part the AK-50 design and introduces new failure modes to the circuit breaker. The new failure modes introduced by this interfacing connection include: (New Source of Breaker Failure to Open, Close and Coordinate malfunctions) Connections between Square D Breaker / Cradle: (Reference RCA 2011-5414 / RCA 2011-6621) 2.1.a Displaced connector (no connection on cradle to breaker engagement) 2.1.b Incorrectly wired connector, 2.1.c Damage to Square D to NLI / Square D Cradle arms which lift the Square D breaker into the correct position and make up the electrical connections between the Square D breaker and the top of the cradle, 2.1.d Pinched wire(s) under the WAGO block cover. Bus Bar to NLI Breaker Package Connection (Reference RCA 2011-5414) 2.2.a Incorrect primary disconnect to bus bar alignment (vertical, horizontal, angular, and

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		linear) causing the breaker primary disconnect fingers to potentially not all be in contact with the silver plating of associated bus bar, 2.2.b No over-racking protection as was provided by the original GE design, 2.2.c The potential that inadequate cleaning of the bus bars could interfere and disrupt the primary disconnect connection, 2.2.d The potential failure of the interfacing safety devices (shutters, cradle racking equipment) between the Square D Breaker and NLI Cradle to properly function, and 2.2.e The change in the Primary Disconnect design (from 8 large "fingers" to 20 smaller "fingers" per phase connection). While these issues are broken out to document the potential adverse effects for analysis, they are related by the interdependence rule, and will be evaluated as Change 2 in aggregate. Change 3: ZSI Jumpers - ZSI (or Zone Selective Interlocking) jumpers. The new failure modes introduced by the new breaker requiring jumpers include: (New source of Breaker Failure to Coordinate Malfunction) (Reference RCA 2011-6621) 3.1 ZSI jumpers not inserted properly (must be inserted enough for conductor to catch WAGO "pinch" connector, but not so far as to have the wire insulation captured in the WAGO "pinch" connector.) 3.2 ZSI jumpers in the correct configuration (all 3 jumpers in their proper location to assure the design function is performed) 3.3 ZSI jumper wire failure (no continuity)

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		3.4 ZSI jumper position testing using test leads on the jumpers or pinch clips can displace the jumpers While these issues are broken out to document the potential adverse effects for analysis, they are related by the interdependence rule, and will be evaluated as Change 3 in aggregate. Change 4: Full Function Test Kit (FFTK) - if the FFTK is used to test the breaker, the ZSI jumpers are not in the test circuit, which prevents verification of the design breaker coordination. This represents a potential gap to compliance with FSAR 8.3.2.3 regarding selective fault protection that is credited in the FCS design to assure electrical separation. (Reference RCA 2011-6621). Adverse Findings in 50.59 Screening (FC-154A) With regards to the Changes identified in the Brief Description of the Activity, the following are applicable to this function and were found to be Adverse: Change 2: The interposing connections between the breaker and the original GE cubical add a new connection that was not present in the old circuit breaker, along with new associated failure modes, which is Adverse. This is a new source of Breaker Failure to Open, Close and Coordinate malfunctions. Change 3: The addition of ZSI jumpers, and their ability to affect the response of the breakers with their new failure modes, is Adverse. This is a new source of Breaker Failure to Coordinate Malfunction. Change 4: The ability to use the Full Function Test Kit (FFTK) to trip test the circuit breakers without current injection, and by doing so, potentially impact the ability to detect incorrect breaker coordination is Adverse. This is a new source of Failure to Coordinate Malfunction.

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		Design Function 7. The NLI Breaker Package has the same external connection interface as the existing GE breakers, and is designed to rack into the switchgear and function the same way as the existing GE breakers without the need to make any wiring changes to the plant components. Although wiring changes are not required, differences in the breaker's internal wiring and internal component arrangement makes it necessary to revise the existing plant wiring/schematic drawings. Since these devices are internally different from the existing Safety-Related AK-50 circuit breakers, introduce new interposing connectors, which did not exist in the previous design, and are not a one-for-one replacement, this change is considered adverse to this Design Function. The applicable failure modes are evaluated in this 50.59 Evaluation FC-154B. This is a new source of Breaker Failure to Open, Close and Coordinate malfunctions. Associated SSC Design Functions Design Function 9. Provide a connection between the line and the load for each circuit breaker. The NLI Breaker Package includes a cradle to interface the Square D Circuit Breaker with the original GE Switchgear connections and bus bars. This adds a new set of interposing connections between the circuit breaker and the plant, which were not present in the original design. In addition, the GE cubicle needs to be "normalized" (i.e., bus bars bent) and aligned (i.e., using a borescope to verify that all disconnect fingers land on the associated phase bus bar within the silver plating), which was not required when using the GE AK-50 line. This is Adverse. This is a new source of Breaker Failure to Open, Close, and Coordinate Malfunctions. Design Function 10. Provide indication (local, remote) of breaker position and status. The new interposing connectors could impede or prevent this function. This is Adverse. This is a new source of Breaker Failure to Open and Close malfunctions. Design Function 11. Provide the ability to locally and remotely open and close the circuit breaker, including the same capabilities as the breakers being replaced. The new interposing connectors could impede or prevent this function. This is Adverse. This is a

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		new source of a Breaker Failure to Open and Close malfunction. **Summary of Evaluation:** There are no accidents that were previously evaluated in the UFSAR that are credibly affected by the activity to replace the GE AK-50 breakers with the NLI Breaker Packages. Implicit malfunctions in the UFSAR include the failure of the breaker to supply the desired electrical loads (Failure to Close) the failure of the breaker to trip on a faulted condition or on demand (Failure to Open), and the failure of the circuit breaker to properly coordinate (Failure to Coordinate). The malfunctions of the Masterpact breakers and trip system and their subcomponents were reviewed as documented in NLI FMEA 093-351020032-1. Additional inputs were provided by CRs 2012-02652 and 2012-04114, and from RCAs 2011-5414 and 2011-6621, based on post-event investigations. Notwithstanding the design control failures that led to Root Cause Analyses (RCAs) 2011-5414 and 2011-6621, the FMEA and failure analysis within the referenced RCAs demonstrate that in all cases, the reliability of the NLI Breaker Packages increased, or there was no impact when the NLI Breaker Package was properly installed and tested. All conditions determined to be Adverse have been evaluated. There are no radiological consequences associated with the malfunctions of the NLI Breaker Package. The effect of the failure of the NLI Breaker Package on the electrical distribution system (EDS) as described in the UFSAR is no different from the effect of failure of the original breaker and trip device. Therefore, this activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR. The accidents UFSAR in the remain bounding. The NLI Breaker Package performs the same system function as the existing breakers and trip devices. The NLI Breaker Package is seismically qualified and designed for the environmental conditions in which they are placed. The NLI Breaker Packages have the same external connection interfaces as the existing devices. The interposing

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		connections introduced by the cradle, which is required to interface the Square D Circuit Breaker with the GE switchgear, are acceptable with the administrative controls that are included in Section 2 of EC 33464 which assure the NLI Breaker Packages are properly installed. The addition of ZSI jumpers is acceptable with the post-installation inspections and testing that is performed to assure electrical continuity, and through that test, proper installation, as documented in Section 2 of EC 33464. The new NLI Breaker Package does not reduce system equipment redundancy, separation, or independence. No manual actions are substituted for automatic actions. Likewise, no automatic actions are substituted for manual actions. The new NLI Breaker Package components have been tested for EMI/RFI and seismic influences, and the associated firmware has been "verification and validation" (V&V) tested by the vendor. The NLI Breaker Package is designed to rack into the switchgear and function the same way as the existing GE breakers without the need to make any wiring changes to the plant components. This includes actions to: • Normalize the original GE switchgear bus bars to align with, as well as using remote visual aids to establish the proper cradle stop position to assure the NLI Breaker Package will consistently fully engage the six Primary Disconnect Finger Cluster Assembles with the original GE Bus Bars; • Clean and verify clean of the switchgear bus bars, to assure proper contact is maintained; • Perform of DLRO testing of the bus bar to breaker connections, to assure that the resistance reading is within the established acceptance criteria; • Inspect and Test the ZSI jumpers to assure the design coordination is maintained; • Perform trip testing on the NLI Breaker Package using current injection, to verify the design coordination is maintained, that the new Thermal Imaging / Memory function does not impact the design functions, and all control and trip functions work as designed. This testing eliminates the vulnerability of FFTK testing by

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		providing current injection without the use of the FFTK; • Inspect the breakers to verify that the WAGO blocks are not displaced, wires within the circuit breaker connecting to the WAGO blocks are not pinched, and that all connections are properly made up. The above changes, as implemented by Section 2 of EC 33464, as well as the administrative controls (instructions in the modification, procedures implemented by this modification, established DLRO reading acceptance criteria) will assure the NLI Breaker Package will function and operate as the previous equipment that it replaced performed, is acceptable and will assure that the new sources of failure modes identified in Changes 1, 2, 3, and 4 have been addressed and ameliorated. Therefore, the NLI Breaker Package does not introduce the possibility of a malfunction of a SSC with a different result. The proposed activity does not involve any fission product barrier design basis limit for fuel cladding, reactor coolant system (RCS) boundary or containment. This modification involves a physical change to SSC. It does not involve the revising or replacing any analytical models. Therefore, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.
EC 53257 (FDCR 60650)	EC 53257 - 1B4A Bus Repair / Replacement	**Activity Description:** 480 Vac Bus 1B4A was severely damaged in a fire on June 7, 2011. This EC repairs and replaces components on the 1B4A Bus to restore it to full qualification and function. To facilitate ease of processing and understanding, the following terms are defined: Cradle: This is an interposing connector assembly, which is used to connect the Square D breaker that is part of the NLI NT and NW Breaker Packages to the original switchgear bus bars. This is required as the Square D breaker is physically smaller than the original

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		GE AK-series breakers, and has connections that would otherwise not directly interface with the as-originally constructed GE switchgear. NLI (NT or NW) Breaker Package - The assembly of the Square D Breaker, the Micrologic 3.0A / 5.0A Trip Device, and the associated cradle, all of which together, comprise the assembly which replaces the original GE AK-25 or 50 series circuit breaker. Retro-fill Design: Strips out previous GE slide contacts and racking mechanism, and replaces with a bolted-in cradle assembly. Retro-fit Design: Previous design used in EC 33464 to install the cradle into the existing GE racking mechanism. The following activities are reviewed in this screening: • The new Nuclear Logistics Incorporated (NLI) 480 Vac NT and NW breaker packages have cradle assemblies (which are used to connect the Square D Circuit breaker to the Switchgear Bus Bars) in which the cradles are bolted to the associated bus bars, instead of having a removable cradle. • Change from aluminum welded buswork to bolted copper buswork. • Replacement of the fire-damaged Bus Way (flat pack of conductors, that provides the connection between Bus 1B4A / BT-1B4A and Bus 1B3A-4A) that includes replacement of the original conductors with copper in place of the original aluminum. • Change to "Retro-Fill" rather than "Retro-Fit" where the inside of the original GE breaker cubicles are removed, and the cradle interfaces directly with the associated wiring. This is in contrast to the previous NLI modification performed under EC 33464, in which the cradle had to interface through the original GE "sliding contacts." • Rejection Capability - While not used, the NLI cradles will reject a non-Safety related circuit breaker. All FCS circuit breakers are safety related.

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		• Unused Current Transformers in the Cradle Input and Outputs have been grounded. • Cradle Mounted mechanism operated cell (MOC), in contrast to the AK-25 breakers which had an integral switch with more contacts available. • Change in truck operated cell (TOC) switches with a reduced number of available contacts. • Cradle mounted resistor, to eliminate flickering of light indications identified through OPEX reviews. • Seismic qualification of the associated breaker / cradle assembly. • Reduction in Control Power Circuit Breaker amp capacity from 70 amps to 50 amps tor 8-2/1B4A. • Improved access panels in switchgear, to facilitate ease of inspection. • The new NLI NT Breaker Package has a reduced electrical rating. The original GE AK-series of breakers was rated at 600 Vac. The NLI NT Breaker is rated at 535 Vac. • Wider (improved) degraded voltage range for close and trip operations. • NLI NT and NW Breaker Package Changes which increase complexity, including: o Zone Selective Interlock provisions. o Use of the Full Function Test Kit (FFTK). o Connections between the Cradle and the Square D breaker. o WAGO brand connector use and failure modes. o Micrologic Trip Units 3.0A / 5.0A which replaces previous analog and digital devices (Digital Modification) o Changes in human factors and breaker operation activities. o For breaker 1B4A-6, elimination of the previous manual charging action for the breaker by the use of onboard charging for the new NLI NT Breaker Package.

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		Background: To facilitate an understanding of the activity being performed, the differences between the original GE Breaker and the NLI NT and NW Breaker Packages are discussed below: Original GE AK-25 / 50 Breakers The original GE AK-25 and 50 Circuit Breakers were designed to be installed within the tolerances of the GE switchgear it was installed in. The breaker had 4 bolts (2 per side) for mounting, which fit within the "drawout' rails of the cubicle. When racked in, these 4 bolts in the "drawout' rails assured that the breaker was installed and would properly connect with the associated cubicle. The primary disconnects between the circuit breaker and the bus bars were large, had 8 fingers on the AK-50 (4 top, 4 bottom) and 4 fingers on the AK-25 (2 top, 2 bottom), for each phase of the connection (6 connections, 3 for the line in, 3 for the load out), and were designed to work within the tolerance of the existing GE buswork. The primary disconnects were not sensitive to bus bar angle in any axis. Control signals and position information were transferred into and out of the breaker by the means of spring loaded contacts, which contacted fixed rails in the breaker cubicle. 125 VDC was provided for trip and close operations. Original GE AK-25 / AK-50 Overcurrent Protection The original GE AK-25 and 50 circuit breakers were equipped at the start of plant operation with dashpot overcurrent devices, which were located within the frame of the circuit breaker. Due to problems with reliability and repeatability of the dashpot devices, they were changed to the GE RMS-9 Trip Modules, which replaced the electro-mechanical dashpot with a digital trip device, and were mounted on the frame of the circuit breaker, inside the door. Issues with spurious tripping of the RMS-9 resulted in their being replaced with the Westinghouse Amptector, which was a similar device that was mounted on the outside of the cubicle door.

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		Replacement NLI Masterpact Breakers The new NLI/Square D Circuit Breakers are physically smaller than the existing GE breakers, use pin-and-socket type connectors, and have a different connection system (different shape, orientation) to connect to the bus bars. As a result, the NL|/Square D Circuit Breaker requires an "adapter" to interface it with the existing GE switchgear cubicle. This "cradle" as it is called, is supplied with each new NL|/Square D Circuit Breaker, and adapts the NLI/Square D Circuit Breaker and associated connections for power, control, indication, and circuit breaker functions to the GE switchgear cubicle. The outside dimensions and electrical connections on the cradle are similar to the existing GE switchgear breakers. There are some differences, which include a bolted connection between the cradle line and load disconnects, and the associated line and load bus bars. Section 9 of EC 53257 includes acceptance criteria for Digital Low Resistance Ohmmeter (DLRO) Testing to assure an acceptable resistance level is present in the connection, and the need to adjust the bus bars to interface and bolt to the cradle connection. In contrast to the original EC 33464 which installed the NLI Bus Feed and Bus Tie breakers, the cradle is directly installed via the Retro-Fill process into a stripped out GE breaker cubicle. This eliminates the slip connection that was previously implicated as the most probable cause of the 1B4A Bus Fire (Reference RCA 2011-5414), This included the removal of the racking mechanisms and draw-out rails, as well as the original slide contacts. As the new breakers have a different faceplate, a new breaker door is supplied with each new NLI / Square D Breaker Package. The inside dimensions and electrical connections on the NLI cradle are designed to accept the new NLI/Square D breaker.

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		Replacement NLI Masterpact Overcurrent Protection The Square D breaker installed in the cradle has a Square D Micrologic Trip Unit (either Model 3.0A or 5.0A, depending on the breaker size), which is installed in a recess in the face of the Square D circuit breaker. The Micrologic Trip Unit is a digital trip device that can be field tested without current injection testing using a "Full Function Test Kit," (FFTK) which provides simulated current to test-trip the circuit breaker. This is advantageous for routine trip tests, as it avoids damage to the load breaking contacts of the circuit breaker, which would otherwise have to discharge the trip current and incur minor wear in the process. As described in Section 9 of the EC 53257, the post-maintenance testing of the new NLI NT and NW Breaker packages must be trip tested using current injection across the input and output circuit breaker connections, and not using the FFTK to assure that proper breaker coordination is verified. The Micrologic Trip Unit is programmed with the required trip curve to maintain coordination. The Micrologic Trip Unit in the NLI NT and NW Breaker Packages uses a lithium battery to provide indication of a trip. The previous Amptector directly generates this indication without the need of a battery. The Micrologic Trip Unit protects the downstream loads. The Micrologic Trip Unit consists of a set of current transformers (called CTs or sensors) to sense current, a trip unit to evaluate the current, and a tripping solenoid to trip the circuit breaker. The trip unit samples the analog current signal and converts it to a digital signal to allow for more accurately measured magnitudes of a non-sinusoidal waveform. The NLI NT and NW Breaker Packages has a "Zone Select Interlock" or ZSI capability, which allows these breakers to be interconnected with similar breakers with wiring, and to cause preferential tripping of breakers to protect specific zones. This feature is not utilized in this modification as it would violate electrical separation requirements that are present in nuclear plant design; however, it must be properly disabled by installing jumpers in the correct blocks, as called out in Section 2 of EC 53257. The effect of not

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		installing the jumpers correctly is that the NLI NT and NW Breaker Packages will not coordinate as required by the design. Section 2 calls out the required visual and continuity testing for these ZSI jumpers, which in turn with current injection post-installation testing, assures that breaker coordination is maintained as called out in USAR Section 8.3.2.3. As an Extent of Condition, this 50.59 evaluation is being performed due to concerns identified in Condition Reports CR 2012-02652 and CR 2012-04114 with the adequacy of the 50.59 review for EC 33464, as well as those issues identified in Root Cause Analyses (RCA) 2011-5414 and 2011-6621. The original FC-154A 50.59 screening incorrectly provided the justification for determination that a 50.59 evaluation was not required in Section VI, Conclusion portion of the form. The design functions described in Section III have also been numbered to make them easier to refer to in the justification discussions in Section IV. A Failure Modes and Effects Analysis (FMEA) has been prepared to identify and compare new component failure modes to those failure modes for the existing breakers, and is captured in EC 53257. The results of the FMEA, as well as RCAs 2011-5414 and 2011-6621 are addressed in the 50.59 screening and this evaluation, which were performed for the proposed activity. New Failure Modes identified in CR 2012-02652, CR 2012-04114, RCA 2011-5414 and RCA 2011-6621: The new NLI NT and NW Breaker Packages are considered alternate replacements that were equal to or better than the existing GE breakers. However, there are some design differences between the original and replacement breakers that warrant additional discussion as a potential failure mode. While the replacement breakers perform the same function as the original breakers, the following new failure modes must be considered and were reviewed in the screening. (CR 2012-02652, CR 2012-04114, RCA 2011-5414, RCA 2011-6621)

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Change Number	Activity Title	50.59 Evaluation Summary
		Note: If a new failure mode was considered plausible either by NLI or the corrective action program documentation, it was included. Change 1: Thermal Imaging / Memory - The NLI NT and NW Breaker Packages comes with "Thermal Imaging" or "Thermal Memory." This is a new function that was not part of the original GE breakers and their respective trip devices. According to Square D (the breaker and trip device OEM), this function monitors the current over a rolling period of 15 minutes. If the current approaches the trip setpoint, the thermal memory starts accumulating. If over the course of the next 15 minutes the "Thermal Imaging" reaches the trip setpoint, it will trip the associated circuit breaker. Change 2: Interfacing Connection between the Square D Breaker and NLI Cradle, and the GE Switchgear. This connection is not part of the AK-50 design and introduces new failure modes to the circuit breaker. The new failure modes introduced by this interfacing connection include: Connections between Square D breaker / Cradle: (Reference RCA 2011-5414 / RCA 2011-6621) 2.1.a Displaced connector (no connection on cradle to breaker engagement) 2.1.b Incorrectly wired connector, 2.1.c Damage to Square D to NLI / Square D Cradle arms which lift the Square D breaker into the correct position and make up the electrical connections between the Square D breaker and the top of the cradle, 2.1.d pinched wire(s) under the WAGO block cover. Bus Bar to NLI NT and NW Breaker Packages Connection (Reference RCA 2011-5414) 2.2.aThe potential failure of the interfacing safety devices (shutters, cradle racking equipment) between the Square D Breaker and NLI Cradle to properly function, and 2.2.bThe change in the Primary Disconnect design

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		While these issues are broken out to document the potential adverse effects for analysis, they are related by the interdependence rule, and will be evaluated as Change 2 in aggregate. Change 3: ZSI Jumpers - ZSI (or Zone Selective Interlocking) jumpers. The new failure modes introduced by the new breaker requiring jumpers include: (Reference RCA 2011-6621) 3.1 ZSI jumpers not inserted properly (must be inserted enough for conductor to catch WAGO "pinch" connector, but not so far as to have the wire insulation captured in the WAGO "pinch" connector.) 3.2 ZSI jumpers in the correct configuration (all 3 jumpers in their proper location to assure the design function is performed) 3.3 ZSI jumper wire failure (no continuity) 3.4 ZSI jumper position testing using text leads on the jumpers or pinch clips can displace the jumpers Change 4: Full Function Test Kit (FFTK) - if the FFTK is used to test the breaker, the ZSI jumpers are not in the test circuit, which prevents verification of the design breaker coordination. This represents a gap to compliance with USAR 8.3.2.3 regarding selective fault protection that is credited in the FCS design to assure electrical separation. (Reference RCA 2011-6621) Adverse Findings in 50.59 Screening (FC-154A) With regards to the Changes identified in the Brief Description of the Activity, the following are applicable to this function and were found to be Adverse: Change 2: The addition of interposing connections of a different design between the Micrologic Trip Unit, the Square D Breaker, and the NLI NT and NW Breaker Cradle create the possibility of a malfunction that did not exist in the previous breaker design.

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		This has an Adverse Effect on the Design Function, and will be addressed in the 50.59 Evaluation. Change 3: The addition of ZSI jumpers, and their ability to affect the response of the breakers, has an Adverse Effect on the Design Function, and will be addressed in the 50.59 Evaluation. Change 4: The ability to use the Full Function Test Kit (FFTK), and by doing so, potentially impact the ability to detect proper breaker coordination, has an Adverse Effect on the Design Function, and will be addressed in the 50.59 Evaluation. Design Function 7. The NLI NT and NW Breaker Package has the same external connection interface as the existing GE breakers, are designed to rack into the NLI Cradles which are installed in the switchgear and function the same way as the existing GE breakers without the need to make any wiring changes to the plant components. Since these devices are internally different from the existing Safety-Related AK-25 and 50 circuit breakers, introduce different interposing connectors, which did not exist in the previous design, and are not a one-for-one replacement, this change is considered adverse to this Design Function. The applicable failure modes are evaluated in this 50.59 Evaluation FC-154B. This is a new source of Breaker Failure to Open, Close and Coordinate malfunctions. The applicable failure modes as outlined in Changes 2, 3 and 4 above, are evaluated in this 50.59 Evaluation FC-154B. Associated SSC Design Functions Design Function 9. Provide a connection between the line and the load for each circuit breaker. The NLI NT and NW Breaker Packages include a cradle to interface the Square D Circuit Breaker with the original GE Switchgear connections and bus bars. This changes the interposing connections between the circuit breaker from that present in the original design. This is a new source of Breaker Failure to Open, Close and Coordinate malfunctions. This has an Adverse Effect on the Design Function, and will be addressed

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		in this 50.59 Evaluation. Design Function 10. Provide indication (local, remote) of breaker position and status. The new NLI NT and NW Breaker Packages have local and remote indication. This design function is satisfied and is acceptable as it is functionally identical to the previous function. This function is impacted by Change 2. This is a new source of Breaker Failure to Open and Close malfunctions. The assessment of Design Function 9 covers this interface, and by the rules of interdependence, it will be addressed in Design Function 9. Design Function 11. Provide the ability to locally and remotely open and close the circuit breaker, including the same capabilities as the breakers being replaced. The new NLI NT and NW Breaker Packages have the same capabilities of being tripped and closed both locally and remotely. Local trip buttons are protected, which is an enhancement over the previous GE AK-25 and 50 breakers, which were not protected. The protection is comprised of a button cover, which is not locked, and does not inhibit the use of the buttons. The control room interface for the circuit breakers are unchanged. This design function is satisfied and is acceptable as it is functionally identical to the previous function and methods. This is a new source of Breaker Failure to Open and Close malfunctions. This function is impacted by Change 2. The assessment of Design Function 9 covers this interface, and by the rules of interdependence, it will be addressed in Design Function 9. **Summary of Evaluation:** There are no accidents that were previously evaluated in the UFSAR that are credibly affected by the activity to replace the GE AK-25 breakers with the NLI NT Breaker Packages, nor by the introduction of the modified cradle now utilized for the NLI NW Breaker Packages originally evaluated in EC 33464, and replaced in this EC. Implicit malfunctions in the UFSAR include the failure of the breaker to supply the desired electrical loads (Failure to Close) the failure of the breaker to trip on a faulted condition or

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		on demand (Failure to Open), and the failure of the circuit breaker to properly coordinate (Failure to Coordinate). The malfunctions of the Masterpact breakers and trip system and their subcomponents were reviewed as documented in NLI FMEA 093-351020032-1. Additional inputs were provided by CRs 2012-02652 and 2012-04114, and from RCAs 2011-5414 and 2011-6621, based on post-event investigations. Notwithstanding the design control failures that led to Root Cause Analyses (RCAs) 2011-5414 and 2011-6621, the FMEA and failure analysis within the referenced RCAs demonstrate that in all cases, the reliability of the NLI Breaker Packages increased, or there was no impact when the NLI Breaker Package was properly installed and tested. All conditions determined to be Adverse have been evaluated. There are no radiological consequences associated with the malfunctions of the NLI NT and NW Breaker Packages. The effect of the failure of the NLI NT and NW Breaker Packages on the electrical distribution system (EDS) as described in the UFSAR is no different from the effect of failure of the original breaker and trip device. Therefore, this activity does not create a possibility for an accident of a different type than any previously evaluated in the UFSAR. The accidents in the UFSAR remain bounding. The NLI NT and NW Breaker Packages perform the same system function as the existing breakers and trip devices. The NLI NT and NW Breaker Packages are seismically qualified and designed for the environmental conditions in which they are placed. The NLI NT and NW Breaker Packages have the same external connection interfaces as the existing devices. The different interposing connections introduced by the cradle, which is required to interface the Square D Circuit Breaker with the GE switchgear, are acceptable with the administrative controls that are included in Section 2 or Section 9 of EC 53257, which assure the NLI NT and NW Breaker Packages are properly installed. The addition of ZSI jumpers is acceptable with the post-installation inspections and testing that is performed to assure electrical continuity, and through that test, proper installation, as documented in Section 2 of EC 53257.

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		The new NLI NT and NW Breaker Packages do not reduce system equipment redundancy, separation, or independence. No manual actions are substituted for automatic actions. While an automatic action could be considered included with the change of automatic charging springs in the case of Breaker 1B4A-6, this reduces operator workload, which is not adverse. The new NLI NT and NW Breaker Package components have been tested for EMI/RFI and seismic influences, and the associated firmware has been "verification and validation" (V&V) tested by the vendor. The NLI NT and NW Breaker Packages are designed to rack into the cradle that is bolted into the switchgear and function the same way as the existing GE breakers without the need to make any wiring changes to the plant components. This includes actions to: • Clean and verify clean of the switchgear bus bars, to assure proper contact is maintained; • Perform of DLRO testing of the bus bar to breaker connections, to assure that the resistance reading is within the established acceptance criteria; • Inspect and Test the ZSI jumpers to assure the design coordination is maintained; • Perform trip testing on the NLI Breaker Package using current injection, to verify the design coordination is maintained, that the new Thermal Imaging / Memory function does not impact the design functions, and all control and trip functions work as designed. This testing eliminates the vulnerability of FFTK testing by providing current injection without the use of the FFTK; • Inspect the breakers to verify that the WAGO blocks are not displaced, wires within the circuit breaker connecting to the WAGO blocks are not pinched, and that all connections are properly made up. The above changes, as implemented by Section 2 or Section 9 of EC 53257, as well as the administrative controls (instructions in the modification, procedures implemented by this modification, established DLRO reading acceptance criteria) will assure the NLI NT and NW Breaker Packages will function and operate as the previous equipment that it replaced performed, is acceptable and will assure that the new sources of failure modes

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		identified in Changes 1, 2, 3, and 4 have been addressed and ameliorated. Therefore, the NLINT and NW Breaker Packages do not introduce the possibility of a malfunction of a SSC with a different result. The proposed activity does not involve any fission product barrier design basis limit for fuel cladding, reactor coolant system (RCS) boundary or containment. This modification involves a physical change to SSC. It does not involve the revising or replacing any analytical models. Therefore, the proposed activity does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.
EC 57131	Inverter Synchronization Modification	**Activity Description:** This change implements a time delay on the inverter synchronization by installing a new relay into the sync circuit. The intent of this delay is to monitor bypass voltage into the inverter and upon de-energization (i.e. 480V load shed) and subsequent re-energization (i.e. 480V bus loaded onto the diesel), implement the time delay to prevent synchronization to the bypass source during the diesel load sequencing. This will prevent the inverter from the synch reference cycling that is causing the output transients. This is an adverse impact on the ability of the inverter to maintain "synchronous operation with the 480V system" for two minutes after the bypass source becomes available. **Summary of Evaluation:** The modification does not impact any accident initiators and does not result in any failure modes that differ from those currently evaluated. As a result, there are no new accidents or new malfunctions to consider, and the consequences of the currently evaluated malfunctions are not impacted. Therefore, prior NRC approval is not required for implementation.

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Change Number	Activity Title	50.59 Evaluation Summary
EC 59874	HPSI Pump Runout, SI-2A, SI-2B and SI-2C	**Activity Description:** 1. The activity will install orifice plates in the HPSI pump discharge headers. Recent hydraulic analyses have shown that in certain post-LOCA alignments with a low system resistance, the HPSI pump could be operating in a runout condition that could damage the pump. Although the potential for runout has been previously identified, it was evaluated as being an acceptable condition for a limited duration. A recent analysis performed by the pump OEM (Bingham-Sulzer) recommended that pump operation be restricted to flows at or below 450 gpm at all times in order to avoid pump damage or accelerated wear. The orifice plates will increase the system resistance and therefore change the operating point of the pump to deliver ≤ 450 GPM. The design of the orifice plates must ensure that minimum flows used in the safety analyses are still achieved while limiting the maximum flow in low system resistance alignments. The 50.59 screening for the proposed activity concluded that the following aspects of the modification have an adverse effect on the system and are therefore subject to this evaluation: 1. Decrease in the flow delivered by a HPSI pump during the safety injection phase as described in USAR Section 6.2.1. 2. Decrease in the flow delivered by a HPSI pump during the recirculation phase as described in USAR Section 6.2.1. 3. Decrease in HPSI flow rate during a MSLB accident. **Summary of Evaluation:** This activity can be implemented per plant procedures without obtaining a License Amendment. The HPSI system has no impact on the frequency of occurrence of any accident previously evaluated in the USAR since the HPSI system is not an initiator of any analyzed accident. Potential SSC malfunctions have been evaluated and it was determined that this activity will not result in more than a minimal increase in the

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		likelihood of occurrence of a malfunction. The proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR. Any malfunction in the components installed by this modification is bounded by the previously evaluated malfunctions described in the USAR. Therefore, this activity will not result in more than a minimal increase in the likelihood of a malfunction previously evaluated in the USAR, nor does it create the possibility of a malfunction with a different result than previously evaluated in the USAR. The installation of the orifice plates and flanges does not introduce the possibility of a new accident because the HPSI system is not an initiator of any accident and no new failure modes are introduced. The modification does not affect the input parameters into the LOCA analysis, therefore the radiological consequences of the LOCA analyses will not be adversely impacted by the modification. The proposed change will not affect ECCS performance and subsequently there is no change in the design basis limits for the fission product barrier as described in the USAR. The proposed activity Is a physical modification to the HPSI piping and does not revise or replace a method of evaluation described in the USAR or used in establishing the design bases or in the safety analyses"
Calculation FC08034	Diesel Fuel Usage During a Severe Flooding Event; OP-ST-SHIFT-0001: Operations Technical Specifications Required Shift Surveillance	**Activity Description:** A revision is being made to calculation FC08034 to add a case to determine the required diesel generator fuel inventory for a non-flooding loss of offsite power scenario with the plant in mode 4 or mode 5, i.e. In cold shutdown. The new case identifies the required loads for safe shutdown and determines the amount of fuel that must be stored In diesel fuel storage tanks FO-1 and FO-10 to support 7 days of emergency power capability. The required amount of fuel inventory for FO-1 and FO-10 in mode 4 or mode 5 is determined to be 16,000 gallons (indicated) and 2500 gallons (indicated), respectively. Currently, surveillance procedure OP-ST-SHIFT-0001 contains no diesel fuel storage

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		requirements for either FO-1 or FO-10 when the RCS is less than 300 degrees F. Therefore, a change is proposed to add this inventory requirement to OP-ST-SHIFT-0001 when the plant is in mode 4 or mode 5. The existing requirement to maintain 10,000 gallons in FO-10 when the RCS is above 300 degrees F is being revised to apply when the RCS is above 210 degrees F to ensure that there is no gap in administrative control of FO-10 inventory while transitioning to cold shutdown. These changes will ensure that diesel fuel inventories are adequate for seven days of emergency diesel operation in the event of a loss of offsite power in any plant mode. Specifically, the activity that was determined to require a 50.59 evaluation during the screening process is the addition of a new requirement to specify 2500 gallons of diesel fuel inventory in FO-10 and 16000 gallons in FO-1 when the plant is in mode 4 or mode 5. **Summary of Evaluation:** The proposed calculation revision and change to the amount of fuel required to be stored in storage tank FO-1 to 16000 gallons and FO-10 to 2500 gallons during cold shutdown conditions does not require prior approval because: 1) Diesel fuel inventory Is not an initiator of any analyzed accident, 2) Adding a requirement for diesel fuel in modes 4 and 5 does not introduce any new failure modes 3) Diesel fuel and associated storage tanks are not inputs into any previously analyzed events, 4) No changes In plant configuration are proposed, 5) The operation and control of diesel generators and associated fuel supply is not changed, 6) Diesel fuel inventory does not impact design basis limits for fission product barriers, and 7) The proposed activity does not involve methods of evaluation described in the USAR.

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		Analysis has demonstrated that 16000 gallons in storage tank FO-1 in combination with 2500 gallons in storage tank FO-10 will provide an adequate amount of fuel for 7 days of DG operation. Prior to the proposed change to OP-ST-SHIFT-0001, there was no administrative (i.e., surveillance) requirement to maintain any Inventory in FO-1 or FO-10 with the RCS at a temperature less than 300 degrees F. USAR and Technical Specification discussions in regard to requiring 10,000 gallons in FO-10 are focused on diesel fuel requirements for mitigating a LOCA. Since a LOCA is not a credible event when the plant is in cold shutdown, it is appropriate to calculate required fuel based on safe shutdown equipment necessary for a mode 4 or mode 5 loss of offsite power event. FCS Technical Specifications do not require DG operability when the plant is in mode 4 or mode 5. Consequently, the capability of DGs to provide 7 days of emergency power to safe shutdown loads during a loss of offsite power from a mode 4 or mode 5 condition was not previously evaluated. The addition of an administrative requirement to maintain a minimum inventory where no such requirement previously existed is not inconsistent with the USAR requirement for FO-10 inventory. The addition of an administrative limit of 16000 gallons in FO-1 and 2500 gallons in FO-10 will ensure that the 7 day emergency AC power requirement of Industry Standard IEEE 308 (referenced in USAR Section 1.6.7) Is maintained during a loss of offsite power occurring with the plant in a cold shutdown condition.
EC 61599	Replace Socket Welds with Butt Welds in Room 13 in CVCS and SGBD piping	**Activity Description:** EC 61770 addresses the change in HELB requirements. They are not addressed in this EC. The current piping configuration for the Chemical and Volume Control System (CVCS - charging and letdown), and Steam Generator Blowdown (SGBD) System specifies stronger butt weld connections for 2-1/2" diameter piping and socket welds for small-bore piping (2" diameter and under). To meet HELB weld inspection requirements, socket welds on the 2" diameter piping will be replaced with full penetration butt welds. This change to butt welds will allow the use of conventional inspection and examination

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		techniques with clear acceptance criteria to be used for volumetric examination of welds (i.e., UT or RT). The current weld configurations do not allow for volumetric examination. Butt welded reducing fittings will be altered to connect to the outer valve face on valves (CH-198, HCV-204, HCV-1387B, and HCV-1388B) at the boundaries of the piping sections. For HCV-204, HCV-1387B, and HCV-1388B, the inlet valve end will be altered for butt welding to a reducing fitting, which will be butt, welded to the connected piping. CH-198 is a CVCS check valve at the boundary of the charging piping section. It will be replaced with an identical valve that will be altered so that both ends are butt welded to reducing fittings, which are butt welded to the connected piping. The configuration of the weld fit-up to the ends of containment penetrations M-2 and M-3 are also changed. The charging, letdown, and SGBD piping systems and components were originally designed and analyzed to USAS B31.7 (Draft 1968 Edition, Code of Record). As modified, the materials and system design and analysis conforms to the requirements of ASME Section III, 1980 edition, with Summer 1981 Addenda. This ASME Section III design is consistent with current Fort Calhoun design practices described in PED-MSS-11 "Design Specification for Piping and Pipe Supports" as reconciled by Engineering Analysis EA91-054. **Summary of Evaluation:** EC 61599 changes the socket welds in piping sections of the Charging, Letdown, and SGBD piping in Room 13 to butt welds so that the welds can be volumetrically examined. This modification was designed to ASME Section III 1980 Edition with Summer 1981 Addenda as reconciled to B31.7 (Draft 1968, Code of Record). Previous OPPD reconciliation documented Engineering Analysis EA91-054 along with disposition of applicable items not reconciled in EA91-054 provide the basis for the conclusion that the change in methodology is equivalent. The function and operation of the SSCs and associated USAR analyses remain unchanged. The conclusion of this evaluation is that a License Amendment is not required.

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EC 59711 EC 59712	Install Pipe Spool Pieces in Place of HCV-480/481 Valve Body	**Activity Description:** This Temporary Modification (EC 59711/59712) is needed to support maintenance on HCV-480 and HCV-481. After the CCW System piping is drained, HCV-480 and HCV-481 will be removed from the system and spool pieces/blank flanges will be installed in place of the valve bodies. The spool piece and the blank flanges will be manufactured from CQE, Carbon Steel material HCV-480 will have blank flanges installed; HCV-481 will have a spool piece installed. The spool pieces will be installed into the CCW piping system with a gasket on both sides and will be tightened into place. The open pipe spool piece will pass CCW flow as an open valve would for HCV-481, Inlet CCW Isolation Valve to AC-4B Shutdown Cooling Heat Exchanger. The blind flanges will prevent CCW flow as a closed valve would for HCV-480, Inlet CCW Isolation Valve for AC-4A Shutdown Cooling Heat Exchanger. The CCW system will then be refilled and vented as necessary and the pumps restarted. The flanges of the spool pieces will be visually inspected with an in-service leak test when CCW system has reached Normal Operating Pressure and Normal Operating Temperature. With this Temporary Modification installed, the Plant is limited to the current condition of Mode 5 with no fuel In the reactor core and no refueling activities/operations in progress. No heavy load movement is allowed while this Temporary Modification is installed. (Heavy load as defined by USAR per USAR-14.24) The CCW system will be available, but cannot be made operable while this TM is installed. This operability restraint is based on the non-compliant code design while the TM is installed. **Summary of Evaluation:** The Temporary Modification will be removed prior to exiting the current condition of Mode 5 with no refueling activities/operations. With the plant restricted to Mode 5 with no refueling activities in operation while the Temporary Modification is installed, the relevant

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		UFSAR described SSCs and associated design functions are those associated with spent fuel pool (SFP) cooling. There are no applicable accidents that have been evaluated in the UFSAR that are affected by the Temp Mods because no fuel is located in the reactor vessel and no refueling activities will be in place during the duration of the Temp Mod. The 50.59 Screening associated with this Temp Mod concluded that the activity is an adverse change in the design function of the shutdown cooling system because CCW is not available to two shutdown heat exchangers and potentially adverse change in design and method of closing the shutdown cooling heat exchanger isolation valves to isolate the CCW system when establishing RW direct cooling and code compliance of the spool pieces/blank flanges. The following 50.59 evaluation concludes that these changes are acceptable. (End of Summary)
EC 62065	Temp Mod -Install Block on HCV-2983 Isolation Valve	**Activity Description:** A stem adaptor device is being installed on the Safety Injection system containment isolation valve HCV-2983 to allow the valve to be secured in the closed position for the duration of Operating Cycle 27. The valve stem of HCV-2983 will be decoupled from its operator HCV-2983-O and a stem adaptor device will be installed which will allow the operator to hold the valve in the closed position without applying a lateral load on the valve stem. The instrument air supply to HCV-2983-O will be secured and tagged to prevent repositioning of the valve while this temporary modification is installed. The stem adaptor is being installed on HCV-2983 to allow the valve to be secured in the closed position in accordance with Technical Specification 2.6.(1).a due to the inability of HCV-2983 to meet Local Leakage Rate Testing (LLRT) requirements for valve leakage. Multiple performances of LLRT via IC-ST-AE-3122 have failed, and it has been determined that the appropriate repairs cannot be completed prior to restart of Fort Calhoun Station. Installation of the stem adaptor will allow HCV-2983 to be secured in the closed position in accordance with the requirements of TS 2.6.(1).a until the start of RFO 27 when the valve will be repaired. This secured position for the valve will ensure

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Change Number	Activity Title	50.59 Evaluation Summary
		that containment integrity will not be violated. However, by securing the valve closed, this will adversely affect the ability to provide shutdown cooling purification. **Summary of Evaluation:** It has been determined that implementation of the proposed activity does not result in more than minimal increases in any of the eight categories evaluated here. No aspects of this temporary modification were found to have a more than minimal impact on the outcomes or frequencies of any previously evaluated events, and were determined to not be capable of creating events or malfunctions not previously identified. Based upon the evaluations performed here, it has been determined that implementation of this activity does not require FCS to obtain an operating license amendment.
EC 61206	Update USAR 8.3.5 for Inverter Synchronization	**Activity Description:** USAR 8.3.5 is being revised to more accurately discuss the station inverter operation when the 480V busses are powered from an emergency diesel generator instead of offsite power. The section currently implies that the inverters are always synchronized to their associated bypass transformer (i.e. 480v system); unless the bypass source is unavailable. In actuality, the inverters will not likely be synchronized when the 480 volt bus associated with the inverters bypass source is powered by an emergency diesel generator. This is because the EDGs are set to start, by existing station procedures, in a frequency (speed) range that is primarily above the allowable synchronization setpoint of the inverter. Therefore, the inverters will not likely synchronize to the bypass source unless/until EDG speed is reduced, per existing station procedures, to establish a frequency below the inverter synchronization setpoint. This condition is documented in CR 2013-09409. **Summary of Evaluation:** Prior NRC approval is not required for this activity. The activity has been evaluated and it

2013 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
		has been demonstrated that short term operation of the inverters out of synch with the CEAs inserted does not impact the current assumptions and analysis in the USAR, and does not create any new malfunction or accidents.
EC 50465	LPSI Recirculation Piping and Valve Upgrade in Support of AHLI (10 CFR 50.59 Revision 1 – Neither the Activity Description or Evaluation Summary were affected by the revision.)	**Activity Description:** EC 50465 consists of the installation of new low pressure safety injection (LPSI) pump recirculation piping – including manual valves and orifices - which will connect the LPSI pump discharge header to the containment sump recirculation lines and which will be used for the alternate hot leg injection (AHLI) alignment during the long term core cooling phase of a loss of coolant accident (LOCA) and during testing of that alignment. The new recirculation line will be placed in service only during AHLI operation and during testing of that alignment. Otherwise, flow through the line is prevented by means of manual 3-inch globe valves (SI-511 and SI-512); one for each of the two containment recirculation sump lines, which will normally be locked closed. To place the LPSI recirculation piping into service for AHLI, SI-186 and SI-511 or SI-512 must be manually opened with a "reach" rod through penetrations in Corridor 4. A separate activity, temporary modification EC 58972, is installing pipe stubs and manual isolation valves to facilitate the subsequent installation of the new recirculation piping under EC 50465. The temporary modification also replaces the existing manually operated shutdown cooling header warm-up isolation valve, SI-186, with a similar valve having a higher flow coefficient (Cv). The final configuration resulting from the combination of EC 50465 and EC 58972 (hereafter, referred to as "the proposed activity"), which will provide a new method for alternate hot leg injection during the long term core cooling phase of a LOCA, is the subject of this 50.59 evaluation. EC 50465 makes the required parts installed under EC 58972 permanent plant equipment. The proposed activity is being implemented because the existing plant configuration may not provide the flow rates needed (i.e., LPSI flow controller FIC-326 may not provide

2013 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
		accurate flow indication) during AHLI operations. The new configuration will provide the capability to achieve the required flow rates. The scope of this screening includes a change to USAR Section 6.2, which identifies that a new pump recirculation line is added for use during AHLI. USAR Figures 9.3-1 and 9.3-2 are also revised to show the location of the alternate hot leg recirculation line with respect to the current piping arrangement. The safety injection and containment spray system drawings (P&ID file numbers 44353, 10479, and 56027) are revised to show the functional arrangement for alternate hot leg injection. New LPSI maximum/minimum curves are used to for the hydraulic analyses associated with AHLI. The scope of this screening also includes the operating procedure changes associated with the proposed activity. These include emergency operating procedures, system valve lineups, and containment leakage rate test procedures identified in Section II. Currently, there is a conduit interference created by the new remote valve operator rods needed for the recirculation line isolation valves. The cable in these conduits is associated with the spent resin pump (WD-14B). As a result, the existing cable is removed and replaced with a new cable of the same size. In addition, two lighting fixtures are relocated (raised up or down) to allow for installation of the recirculation piping, and are field relocated. Two (2) existing abandoned and plugged penetrations (04-F-51 and 04-F-53) are used for manual valve operation in Corridor 4. The penetrations were electrical penetrations and are re-purposed as mechanical penetrations. Manual operator actions will now take place in Corridor 4 to open and close SI-511 and/or SI-512. The opening of either SI-511 or SI-512 is necessary post-LOCA to provided LPSI minimum flow for long term AHLI operation. The 50.59 screening for the proposed activity concluded that the following were adverse

2013 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
		effects of the proposed activity; they are, therefore, the subject of this 50.59 evaluation: • A failure of an isolation valve in the new recirculation piping to perform its isolation function or a failure of a root valve to remain full-open during the different phases of a LOCA. • The increase in LPSI and high pressure safety injection (HPSI) pump suction temperature and the corresponding reduction in pump net positive suction head available (NPSHa). • The installation of new components with the potential to produce external leakage in piping and compartments where fluid from the containment sump is recirculated during the recirculation phase of a LOCA. • The connection of the higher pressure LPSI pump discharge to the lower pressure containment sump recirculation lines through a single valve (SI-511 or SI-512). • The reduction in shutdown cooling system flow available to perform the core cooling, reactor coolant system (RCS) cooldown, spent fuel pool cooling, and alternate shutdown cooling utilizing containment spray pumps, boron dilution and LPSI pump NPSH functions as a result of the additional flow diversion through the replacement SI-186. • The replacement of the existing method of performing the AHLI function with a method based on hydraulically balanced fixed resistances results in changes to the operator actions required to mitigate the consequences of a design basis accident constitutes a fundamental change in the method of performing or controlling the alternate hot leg injection, LPSI pump minimum flow and sump strainer flow design functions. • The potential impact of the installation of new valves on other (i.e., non-AHLI) operating configurations, where a component malfunction (e.g., valve mispositioning) could produce adverse effects. • Gas accumulation as recirculated fluid passes through the orifices and/or valves during AHLI operation is a design issue. • Dose to operating personnel during AHLI initiation. (Function 9).

2013 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
		Summary of Evaluation: This activity can be implemented per plant procedures without obtaining a License Amendment. It does not involve the initiation of any accident previously evaluated in the Updated Final Safety Analysis (UFSAR) (Also known as Updated Safety Analysis (USAR)). Potential structure, system, component (SSC) malfunctions including operator errors have been evaluated. It is determined that this activity will not result in more than a minimal increase in the likelihood of occurrence of a malfunction. This activity provides an enhanced capability for the operators to respond long term to a LOCA (5.5 hours) using the alternate hot leg injection path. It eliminates the potential for errors due to faulty indication. Since the additional local operator action is performed in the same area (Corridor 4) as two other local operation actions, the proposed activity does not affect the ingress and egress paths used in performing the local operator actions. The dose for this activity has been conservatively calculated to be less than 5 rem acceptance criteria in NUREG 0737. The changes to the AHLI operator actions will be implemented by means of a revision to the emergency operating procedures. Therefore, procedural guidance will be available. The additional local operator action – which involves unlocking and opening an isolation valve - is the same as an existing operation performed in the same area. It is, therefore, within the skill of the local operator. Since the proposed activity involves overall changes to the existing method of aligning AHLI, the changes introduced by the proposed activity will be evaluated using the existing systematic approach to training (SAT) and communicated to Operations and other appropriate personnel at the proper time and manner as determined by the SAT process. The proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR. The permissible leakage as specified in Technical Specification 3.16 is not changed. The safety analysis conservatively uses two times the Technical Specification limit for the postulated leakage.

2013 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
		Periodic testing ensures that the leakage is maintained below the Technical Specification limits. Therefore, the proposed activity does not result in an increase in the radioactivity released as a result of engineered safety feature system leakage. There is no change in the consequences of a malfunction (single failure) of an SSC important to safety previously evaluated in the USAR. The evaluation of radiological consequences evaluated in the USAR assumes a single active failure of an emergency core cooling system (ECCS) component. The alternate hot leg injection path will be utilized only if there is a single failure that results in a loss of the preferred (HPSI) hot leg injection path. There are no changes to this preferred hot leg injection path in this activity. The changes to the AHLI simplify the operator actions by removing the need to throttle flow. The AHLI path components are designed CQE and seismic and independent of the preferred HLI path. Neither the method of performing the alternate hot leg injection function nor the revised operator actions associated with this change will produce malfunctions of SSCs important to safety. This activity does not create a possibility for an accident of a different type or result than any previously evaluated in the UFSAR. The existing inherent design features along with non-time critical operator actions ensure that a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the USAR is not created. There is no change on ECCS performance and subsequently no change in a design basis limit for a fission product barrier as described in the UFSAR. The items being addressed in this evaluation do not involve a method of evaluation described in the USAR or used in establishing the design bases or in the safety analyses.

2014 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
EC 64102	Place AI-55 Trip Switch in Disabled Position	**Activity Description:** Place Turbine Vibration Trip Bypass Switch (S3002) in the disabled position in the Turbine Supervisory Panel AI-55. The potential for spurious trips due to a single component failure (vibration monitors) can be reduced by the removal of the automatic high vibration trip and allow operators to manually trip the Turbine. Disabling of the automatic turbine trip on high vibration and replacing it with a manual operator trip is adverse to the design function of the emergency trip or protection function (USAR 14.8.5 d.). Replacement of the automatic function with a manual operator action is an adverse change to how the design function is performed. This evaluation considers the identified adverse effects of this activity. **Summary of Evaluation:** The events applicable to the proposed activity are Turbine-Generator Overspeed Incident (USAR 14.8) and Loss of Load (USAR 14.9). The analyses for these events remain bounding. Replacement of the automatic trip on high vibration by the Turbine Supervisory Instrumentation System with a manual operator trip does not increase the likelihood or consequences of an accident or malfunction previously evaluated in the USAR. The identified events remain bounding; there are no new accidents or malfunctions that need to be considered.
EC 49655	USAR-15.2, Programs and Activities for Managing the Effects of Aging, was changed from using FatiguePro to WESTEMS (metal fatigue cycle counting software)	**Activity Description:** USAR-15.2. 10 listed FatiguePro (Structural Integrity Associates' software) as the industry's automated cycle counting software. FCS switched to WESTEMS (Westinghouse's software) cycle counting software in 2010. The switch was done for economic reasons.

2014 Evaluations

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Change Number	Activity Title	50.59 Evaluation Summary
		Summary of Evaluation: Changing from FatiguePro to WESTEMS cycle counting software does not require an LAR to implement the activity. The software is used to count thermal fatigue cycles for the RCS and CVCS systems. The USAR described design functions for counting design cyclic loads for CVCS and RCS (automatically or manually) is not being changed as WESTEMS counts the cyclic loads in the same manner as FatiguePro using ASME design stress and fatigue analyses using NB-3200 criteria.

Quality Assurance Program Changes

and

Regulatory Commitments Revised in Accordance with NEI 99-04

Quality Assurance Program Changes

QA Program Change Number [Date of change]	Description
Revision 41 June 5, 2014	As part of the Exelon Integration Process, USAR Appendix A was converted to an NQA-1-1994 based QA Program details of which are contained in OPPD Letter LIC-14-0037, "Change to the Quality Assurance Program and Quality Assurance Plan," dated April 2, 2014.

Regulatory Commitments Revised in Accordance with NEI 99-04

Commitment Number	Description
AR 13509	This is a Safety Enhancement Program (SEP) commitment made in letter LIC-88-1094 dated December 9, 1988. The commitment stated that OPPD would develop plant and system level design basis documents for safety systems and other selected systems. The commitment has been revised such that as ECs are implemented at Fort Calhoun Station, the pertinent design basis document (DBD) level information will be included in the USAR. This will continue until the Design and Licensing Basis Reconstitution Project is complete when the commitment will no longer be necessary. DBDs will be made historical controlled documents used for information purposes only.

Summary of Technical Specification Basis Changes (TSBC)

TSBC No. TS Page(s) [Date]	Description
08-002-1 1.0 – Page 3	This TSBC corrected a typographical error in Technical Specification (TS) Basis Section 1.0, related to the settings and capacity of the main steam safety valves. Although the information provided to the NRC in license amendment request 07-02 was correct, when Amendment No. 252 was typed for implementation by OPPD, the correct value of 1050 psia was transposed to 1500 psia in Revision 0 of TSBC 08-02. TSBC-08-02, Revision 1 corrected the value to 1050 psia.
12-003-0 2.3 – Page 5	In accordance with CR 2012-04815, the Basis of TS 2.3 and USAR Section 6.2.3.5 were revised to state "The four pressurized safety injection tanks are of the passive type and require no outside power or safety injection actuation signal to operate." The apparent cause concluded that failure to recognize that the passive design of the SITs cannot credit the use of active components for operability. The change is consistent with USAR Section 6.2.2.
13-001-0 1.0 – Page 2 2.10 – Pages 18, 19, and 20	This TSBC made changes to the Bases of Section 1.0 for consistency with the new definition of F_R^T as approved by Amendment No. 269. Changes to the Bases of Section 2.4 were intended primarily to better organize the sequencing of Bases information to the associated LCO and reflect the changes to F_R^T and azimuthal power tilt. In addition, out-of-date information related to the validity of azimuthal tilt assumed by ABB-CE was deleted as OPPD now uses AREVA NP methodology.
13-002-0 2.15 – Pages 5 and 6	TS Bases Section 2.15 was revised to incorporate a description of an RPS logic matrix channel as part of the implementation of Amendment No. 270 and was also revised to reflect changes due to renumbering that occurred throughout the TS LCO.

TSBC No. TS Page(s) [Date]	Description
13-003-0 3.6 – Page 4	TS Bases Section 3.6 was revised due to a reduction in minimum guaranteed flow from the containment spray (CS) system to account for the need to throttle the containment spray pumps to prevent runout. The reduction is for the main steam line break (MSLB) event from 1885 gpm to 1500 gpm for a 1 pump and 1 header configuration and a 1 pump and 2 headers configuration. This change was associated with EC 62506 described above.
14-001-0 2.16 – Pages 1 & 2 3.2 – Page 5	Amendment No. 274 revised LCO 2.16 and TS Surveillance Requirement (SR) 3.2 and made a related change to the Radiological Emergency Response Plan to revise two emergency action levels (EAL) related to high water level in the Missouri River. The changes corrected a non-conservative deficiency in the required actions of the LCO and in the SR. TSBC 14-001-0 incorporated the associated Bases changes.

Technical Specification Basis Change (TSBC) Pages

1.0 **SAFETY LIMITS**

1.1 Safety Limits (SLs) (continued)

Flow maldistribution effects for operation under less than full reactor coolant flow have been evaluated via model test.[2] The flow model data established the maldistribution factors and hot channel inlet temperature for the thermal analyses that were used to establish the safe operating envelopes presented in Figure 1-1. The reactor protective system is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than the minimum DNBR limit.[1]

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

The reactor coolant system serves as a barrier to prevent radionuclides in the reactor coolant from reaching the containment atmosphere.[3] In the event of a fuel cladding failure, the reactor coolant system is the primary barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of the reactor coolant system and fuel cladding. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, section III, is 110% of design pressure. The maximum transient pressure allowable in the reactor coolant system piping, valves and fittings under USAS section B31.1 is 120% of design pressure. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established.

The settings and capacity of the main steam safety valves (1000 - 1050 psia)[5], the reactor high-pressure trip (≤ 2400 psia) and the reactor coolant system safety valves (2500-2545 psia)[6] have been established to assure never reaching the reactor coolant system pressure safety limit. The initial hydrostatic test pressure was conducted at 3125 psia (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by setting the pressurizer power-operated relief valves, consistent with the reactor high pressure trip, and opening the steam system steam dump and bypass valves upon receipt of a turbine trip signal.[7]

References

(1) USAR, Section 3.6.6
(2) USAR, Section 1.4.6
(3) USAR, Section 4
(4) USAR, Section 4.3.3
(5) USAR, Section 4.3.4
(6) USAR, Section 4.3.9.5
(7) USAR, Section 7.4.5.1
(8) XN-NF-82-06(P)(A), Revision 1, Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," October 1986
(9) XN-NF-79-56(P)(A), Revision 1, Supplement 1, "Gadolinia Fuel Properties of LWR Fuel Safety Evaluation," November 1981

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

The USAR Loss of Coolant Accident analysis assumes a minimum SIRW tank inventory of 250,000 gallons has been pumped from the SIRW tank when recirculation begins. Technical Specification 2.3(1) requires that the SIRW tank contains a minimum of 283,000 gallons of usable water. This additional volume over that assumed in the USAR analysis provides sufficient margin to account for the instrument uncertainty. The SIRW tank contains water containing a boron concentration of at least the refueling boron concentration. This is sufficient boron concentration to provide a shutdown margin of 5%, including allowances for uncertainties, with all control rods withdrawn and a new core at a temperature of 54°F.[2]

The four pressurized safety injection tanks are of the passive type and require no outside power or safety injection actuation signal to operate. The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 116.2 inch level corresponds to a volume of 825 ft^3 and the maximum 128.1 inch level corresponds to a volume of 895.5 ft^3. Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

The operable status of the various systems and components is to be demonstrated by periodic tests. A large fraction of these tests will be performed while the reactor is operating in the power range.

If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function. If it develops that the inoperable component is not repaired within the specified allowable time period, or a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) is not corrected, the reactor will be placed in the cold shutdown condition utilizing normal shutdown and cooldown procedures. In the cold shutdown condition, release of fission products or damage of the fuel elements is not considered possible.

The plant operating procedures will require immediate action to effect repairs of an inoperable component and therefore in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are intended to assure that operability of the component will be restored promptly and yet allow sufficient time to effect repairs using safe and proper procedures.

The time allowed to repair a safety injection tank is based on the deterministic and probabilistic analyses of Reference (8). The time allowed to repair a LPSI train is based on the deterministic and probabilistic analysis of Reference (9). These analyses concluded that the overall risk impact of the completion times are either risk-beneficial or risk neutral.

The requirement for core cooling in case of postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition reduces the consequences of a loss-of-coolant accident and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case, the reactor is to be put into the cold shutdown condition.

1.0 SAFETY LIMITS

1.1 Safety Limits (SLs) (continued)

Basis

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB).

At DNB there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperature and the possibility of clad failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of reactor thermal power and reactor coolant flow, temperature and pressure can be related to DNB through a correlation. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients corresponds to a 95% probability at a 95% confidence level that DNB will not occur, which is considered an appropriate margin to DNB for all operating conditions.(1)

The curves of Figure 1-1 represent the loci of points for reactor thermal power (either neutron flux instruments or ΔT instruments), reactor coolant system pressure, and cold leg temperature for which the minimum DNBR is not less than the minimum DNBR limit. The area of safe operation is below these lines.

SL 1.1.1(b) ensures that fuel centerline temperature remains below the fuel melt temperature 5081°F during normal operating conditions or design anticipated operational occurrences (AOOs) with adjustments for burnup and burnable poison. An adjustment of 58°F per 10,000 MWD/MTU has been established in XN-NF-82-06(P)(A), Revision 1, Supplements 2, 4 and 5 (Ref. 8) and adjustments for burnable poisons are established based on XN-NF-79-56(P)(A), Revision 1, Supplement 1 (Ref. 9).

The reactor core safety limits are based on radial peaks limited by the CEA insertion limits in Section 2.10 and axial shapes within the axial power distribution trip limits in the COLR. The Thermal Margin/Low Pressure trip requirements shall be within the limits provided in the COLR. The Thermal Margin/Low Pressure trip is based on the maximum radial peaking factor (F_R^T) that is provided in the COLR.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

(5) DNBR Margin During Power Operation Above 15% of Rated Power

(a) The following limits on DNB-related parameters shall be maintained:

(i)	Cold Leg Temperature (Core Inlet Temperature)	as specified in the COLR
(ii)	Pressurizer Pressure	≥ 2075 psia[(1)]
(iii)	Reactor Coolant Flow rate	≥ 202,500 gpm indicated
(iv)	Axial Shape Index	as specified in the COLR

(b) With any of the above parameters exceeding the limit, restore the parameter to within its limit within 2 hours or reduce power to less than 15% of rated power within the next 8 hours.

Basis

Linear Heat Rate

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System or the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the linear heat rate does not exceed its limit. The Excore Detector Monitoring System performs this function by continuously monitoring the axial shape index (ASI) with the operable quadrant symmetric excore neutron flux detectors. The axial shape index is maintained within the allowable limits of the Limiting Condition for Operation for Excore Monitoring of LHR Figure provided in the COLR. This ASI is adjusted by Specification 2.10.4(1)(c) for the allowed linear heat rate of the Allowable Peak Linear Heat Rate vs. Burnup Figure provided in the COLR and the F_R^T and Core Power Limitations Figure provided in the COLR. In conjunction with the use of the excore monitoring system and in establishing the axial shape index limits, the following assumptions are made: (1) the CEA insertion limits of Specification 2.10.2(6) and long term insertion limits of Specification 2.10.2(7) are satisfied, and (2) the flux peaking augmentation factors are as shown in Figure 2-8.

(1) Limit not applicable during either a thermal power ramp in excess of 5% of rated thermal power per minute or a thermal power step of greater than 10% of rated thermal power.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

The Incore Detector Monitoring system provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be continuously maintained within the allowable limits of the Allowable Peak Linear Heat Rate vs. Burnup Figure provided in the COLR. The setpoints for these alarms include allowances, set in the conservative directions. If the plant computer fails, the incore detector alarms become inoperable. The provisions of Section 2.10.4(1)(b) are intended to address this situation and assure safe operation of the reactor for up to 7 days.

Calibration of the ex-core detector input to the APD calculator is required to eliminate ASI uncertainties due to instrument drift and axially nonuniform detector exposure. If the recalibration is not performed in the period specified, the prescribed steps will assure safe operation of the reactor.

Maximum Radial Peaking Factor (F_R^T)

The limitation on F_R^T is provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If the maximum F_R^T exceeds the COLR limit, operation may continue under the additional restrictions imposed by the action statements since these additional restrictions provide adequate assurance that the assumptions used in establishing the Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid. The surveillance requirements for verifying that the maximum F_R^T is within the COLR limit provides assurance that the actual value of F_R^T does not exceed the assumed values. Verifying F_R^T after each fuel loading prior to exceeding 70% of rated power provides additional assurance that the core was properly loaded.

Azimuthal Power Tilt

Azimuthal Power Tilt is measured using symmetric in-core or ex-core detectors by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at that location is of the form:

$$P_{tilt}(r,\theta)/P_{avg}(r,\theta) - 1 = T_q \bullet g(r) \bullet \cos(\theta - \theta_o)$$

where

$P_{tilt}(r,\theta)$	is the tilted power at radius r and azimuthal angle θ
$P_{avg}(r,\theta)$	is the average or untilted power at that location
T_q	is the azimuthal tilt magnitude
$g(r)$	is the radial normalizing factor, normalized to a maximum value of unity
θ	is the azimuthal core location
θ_o	is the azimuthal core location of maximum tilt

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2.0 **LIMITING CONDITIONS FOR OPERATION**
2.10 Reactor Core (Continued)
2.10.4 Power Distribution Limits (Continued)

T_q represents the maximum fractional increase in power that can occur anywhere in the core because of tilt.

An azimuthal power tilt > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

DNBR Margin During Power Operation Above 15% of Rated Power

The selection of limiting safety system settings and reactor operating limits is such that:

1. No specified acceptable fuel design limits will be exceeded as a result of the design basis anticipated operational occurrences, and

2. The consequences of the design basis postulated accidents will be no more severe than the predicted acceptable consequences of the accident analysis in Section 14.

In order for these objectives to be met, the reactor must be operated consistent with the operating limits specified for margin to DNB.

The parameter limits given in (5) and the COLR along with the parameter limits on azimuthal tilt and control element assembly position (Power Dependent Insertion Limit Figure provided in the COLR) provide a high degree of assurance that margin to DNB will be maintained during steady state operation.

The actions specified assure that the reactor is brought to a safe condition.

The reactor coolant system flow rate of 202,500 gallons per minute is the indicated value. It does not include instrumentation uncertainties.

The calorimetric methodology shall be used to measure the reactor coolant system flow rate.

2.0 LIMITING CONDITIONS FOR OPERATION
2.15 Instrumentation and Control Systems

Basis

During plant operation, the complete instrumentation systems will normally be in service. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor protective system (RPS) and engineered safety features (ESF) system when one or more of the channels are out of service. Reactor safety is provided by RPS, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continued operation with certain instrumentation channels out of service since provisions were made for this in the plant design.

The RPS and most engineered safety feature channels are supplied with sufficient redundancy to provide the capability for channel test at power, except for backup channels such as derived circuits in the ESF logic system.

When one of the four channels is taken out of service for maintenance, RPS logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel. If the bypass is not effected, the out-of-service channel (Power Removed) assumes a tripped condition (except high rate-of-change of power, high power level and high pressurizer pressure),(1) which results in a one-out-of-three channel logic. If in the 2-out-of-4 logic system of the RPS one channel is bypassed and a second channel manually placed in a tripped condition, the resulting logic is 1-out-of-2. At rated power, the minimum OPERABLE high-power level channel is 3 in order to provide adequate power tilt detection. If only 2 channels are OPERABLE, the reactor power level is reduced to 70% rated power which protects the reactor from possibly exceeding design peaking factors due to undetected flux tilts and from exceeding dropped CEA peaking factors.

An RPS Logic matrix channel consists of two matrix power supplies, four matrix relays and their associated contacts as well as all interconnecting wiring. An RPS Trip Initiation Logic channel consists of an M contactor and associated contacts, an interposing relay and all interconnecting wiring. Two RPS Trip Initiation Logic channels associated with the same pair of CEDM clutch power supplies are considered to affect the same trip leg.

Integrated into the trip initiation logic are two RPS Manual Trip channels. Manual Trip #1 operates by directly de-energizing all four M contactors in response to the operation of a manual pushbutton. Manual Trip #2 operates by de-energizing an undervoltage relay which results in the opening of two circuit breakers, CB-AB and CB-CD, which supply power to the CEDM clutch power supplies. Manual Trip channel #1 consists of manual trip pushbutton #1 and interconnecting wiring. Manual Trip channel #2 consists of manual trip pushbutton #2, circuit breakers CB-AB and CB-CD, and associated interconnecting wiring.

With one manual reactor trip channel inoperable, operation may continue until the reactor is shut down for other reasons. No safety analyses assume operation of the Manual trip. Because of this, the Required Action is to restore the inoperable channel to OPERABLE status prior to entering MODE 2 from MODE 3 during the next plant startup.

2.0 LIMITING CONDITIONS FOR OPERATION
2.15 Instrumentation and Control Systems (Continued)

Basis (Continued)

The ESF logic system is a Class 1 protection system designed to satisfy the criteria of IEEE 279, August 1968. Two functionally redundant ESF logic subsystems "A" and "B" are provided to ensure high reliability and effective in-service testing. These logic subsystems are designed for individual reliability and maximum attainable mutual independence both physically and electrically. Either logic subsystem acting alone can automatically actuate engineered safety features and essential supporting systems.

All Engineered Safety Features are initiated by 2-out-of-4 logic matrices except containment high radiation which operates on a 1-out-of-2 basis. The number of installed channels for Containment Radiation High Signal (CRHS) is two. CRHS isolates the containment pressure relief, air sample and purge system valves.

Entry into Technical Specification 2.15.1(3) is made when conditions have caused one logic subsystem ("A" or "B") to become inoperable but the redundant logic subsystem remains operable. The loss of a prime initiation relay (which renders all 4 channels of a logic subsystem inoperable) is the condition most likely to cause entry into Technical Specification 2.15.1(3). In this situation, the remaining ESF logic subsystem still has the capability to automatically actuate engineered safety features equipment and essential supporting systems. The 48-hour completion time is commensurate with the importance of avoiding the vulnerability of a single failure in the remaining ESF logic subsystem. Technical Specification 2.15.1(3) will not be used upon loss of the common channels that affect both "A" and "B" subsystems prime initiators operability unless the permissible bypass condition is met. Upon exiting TS 2.15.1(3) following the restoration of a prime initiation relay to OPERABLE status, if any channel(s) remain inoperable, the appropriate Limiting Conditions for Operation (LCO) (TS 2.15.1(1) or TS 2.15.1(2)) is applicable with the length of inoperability measured from time of discovery of: 1) prime initiation relay inoperable, or 2) channel inoperability, whichever is longer.

The ESF system provides a 2-out-of-4 logic on the signals used to actuate the equipment connected to each of the two emergency diesel generator units.

The rod block system automatically inhibits all CEA motion in the event a LCO on CEA insertion, CEA deviation, CEA overlap or CEA sequencing is approached. The installation of the rod block system ensures that no single failure in the control element drive control system (other than a dropped CEA) can cause the CEAs to move such that the CEA insertion, deviation, sequencing or overlap limits are exceeded. Accordingly, with the rod block system installed, only the dropped CEA event is considered an Anticipated Operational Occurrence (AOO) and factored into the derivation of the Limiting Safety System Settings (LSSS) and LCO. With the rod block function out-of-service, several additional CEA deviation events must be considered as AOOs. Analysis of these incidents indicates that the single CEA withdrawal incident is the most limiting of these events. An analysis of the at-power single CEA withdrawal incident was performed for Fort Calhoun for various initial Group 4 insertions, and it has been concluded that the LCO and LSSS are valid for a Group 4 insertion of less than or equal to 15%.

3.0 **SURVEILLANCE REQUIREMENTS**
3.6 Safety Injection and Containment Cooling Systems Tests (Continued)

Basis

The safety injection system and the containment cooling system are principal plant safeguards that are not operated during normal reactor operation.

Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes containment isolation and a containment spray system test requires the system to be temporarily disabled. The method of assuring operability of these systems is, therefore, to combine systems tests to be performed during refueling shutdowns in addition to more frequent component tests which can be performed during reactor operation.

The refueling shutdown tests demonstrate proper automatic operation of the safety injection and containment spray systems. A test signal is applied to initiate automatic action and verification made that the components receive the safety injection actuation signals in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.(1) (2)

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked daily and the initiating circuits are tested periodically. In addition, the active components (pumps and valves) are to be tested every three months to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of three months is based on the judgment that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent tests would result in increased wear over a long period of time.

Verification that the spray piping and nozzles are open will be made initially by a smoke test or other suitably sensitive method, and at appropriate intervals thereafter. A single containment spray header flow rate of 1500 gpm of atomized spray is required to provide the containment response(3) specified in Section 2.4 of the Technical Specification: To achieve the 1500 gpm flow rate, no greater than ten (10) spray nozzles may be inoperable of which no more than one may be missing. Since the material is all stainless steel, normally in a dry condition, with no plugging mechanism available, retesting at appropriate intervals is considered to be more than adequate.

Other systems that are also important to the emergency cooling function are the SI tanks, the component cooling system, the raw water system and the containment air coolers. The SI tanks are a passive safeguard. In accordance with the specifications, the water volume and pressure in the SI tanks are checked periodically. The other systems mentioned operate when the reactor is in operation and are continuously monitored for satisfactory performance.

2.0 LIMITING CONDITIONS FOR OPERATION

2.16 River Level

Applicability

At all times.

Objective

To specify maximum and minimum Missouri River levels as measured at the intake structure which must be present to assure safe reactor operation.

Specification

The water level of the Missouri River shall remain less than 1004 feet mean sea level and greater than or equal to 976 feet 9 inches mean sea level as measured at the intake structure.

Required Actions

(1) When the Missouri River level reaches elevation 1004 feet mean sea level, the reactor shall be in a HOT SHUTDOWN condition and in COLD SHUTDOWN within 36 hours following entry into HOT SHUTDOWN; and

(2) If the Missouri River level is less than 976 feet 9 inches mean sea level, the reactor shall be placed in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 36 hours; and,

(3) At Missouri River levels less than 980 feet or greater than 1002 feet mean sea level, a continuous watch will be maintained to monitor river levels to assure no sudden loss of water supply occurs on low river level and provide adequate response time for rising river levels.

Basis

At the Fort Calhoun Station (FCS) site, the probable maximum flood that might occur as a result of runoff from a probable maximum rainstorm over the area below the Gavins Point dam coupled with an assumed outflow of 50,000 cubic feet per second from Gavins Point reservoir is 1009.3 feet. In the unlikely event that the Oahe or Fort Randall dams fail at that time, the Corps of Engineers has estimated that the flood level could be as high as 1014 feet[1].

The intake structure can be protected from these Missouri River floods using removable flood gates on doorways and the screen wash discharge trough. The water level inside the intake cells can be controlled by positioning the exterior sluice gates to severely restrict the flow into the cells and then varying the raw water pump output to remove the inlet flow. The position of the exterior sluice gates must be verified by manual actuation, which requires access to the intake structure veranda. Access to the veranda is lost when the east doors to the intake structure are blocked by installing the flood barriers which must be installed prior to a river level of 1004 feet in order to allow egress to the north. This requires the station to be shutdown prior to 1004 feet. The 36-hour allowance to cold shutdown following hot shutdown entry allows for cool down by steaming to atmosphere, if desired. If the station desires to cool down by steaming to the condenser, shutdown cooling must be initiated prior to 1004 feet so that actions verifying sluice gate position can be completed before access to the veranda is lost. A continuous watch will be established at 1002 feet msl to provide adequate response time for rising river levels in accordance with the abnormal operating procedure. The FCS emergency plan will be implemented during these high and low river level conditions to protect the plant. The auxiliary building can be protected to 1014 feet with the installation of removable flood barriers and sandbagging at the 1013 foot elevation of the equipment hatch room (Room 66).

2.0 LIMITING CONDITIONS FOR OPERATION

2.16 River Level (Continued)

Basis (Continued)

The minimum river level of 976 feet 9 inches provides adequate suction to the raw water (RW) pumps for cooling plant components. The minimum elevation of the RW pump suction is 973 feet 9 inches. An intake cell level of 976 feet 9 inches is required for RW pump minimum submergence level (MSL)[(2)]. The partial loss of this supply is considered highly unlikely. However, provisions for low water levels during winter and spring ice conditions are considered necessary. When river level is low, head loss from debris and/or ice on the traveling screens and/or trash racks could reduce intake cell levels such that the required RW pump MSL is not achieved. This could lead to pump degradation from the formation of vortices at the free water surface. Thus, when the continuous watch requirement is in effect, in addition to monitoring river level to assure no sudden loss of water supply occurs, the level of the intake cells is monitored.

Intake cell levels are also adversely affected by the flows associated with the non-safety related circulating water (CW) pumps since the large flow rates associated with the CW pumps create significant head losses even with relatively clean intake cell conditions. However, the CW pumps have a much higher MSL requirement (983 feet 0 inches) and would become unstable and trip or be manually shutdown well before intake cell levels decrease to the RW pump MSL. The head loss associated with CW pump flow would then be recovered and intake cell levels would rise.

References

(1) USAR, Section 2.7.1.2
(2) USAR, Section 9.8

3.0 **SURVEILLANCE REQUIREMENTS**

3.2 Equipment and Sampling Tests (continued)

Table 3-5, Item 8b verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this surveillance requirement is not met, compliance with LCO 3.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of daily is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

Table 3-5, Item 25 verifies adequate measurements are taken to ensure that facility protective actions will be taken (and power operation will be terminated) in the event of high and/or low river level conditions. The high river level limit of less than 1004 feet mean sea level is based on the maximum elevation at which facility flood control measures provide protection to safety related equipment (i.e., due to restricted access/egress to the intake structure veranda once the flood barriers are installed prior to river level reaching 1004 feet msl). A continuous watch will be established at 1002 feet mean sea level to provide adequate response time for rising river levels in accordance with the abnormal operating procedure. The river level surveillance requirement specified also ensures sufficient net positive suction head is available for operating the RW pumps. The minimum river level of 976 feet 9 inches provides adequate suction to the RW pumps for cooling plant components. The surveillance frequency of "Daily" is a reasonable interval and models guidance provided in NUREG-0212, Revision 2, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors," Section 4.7.6. This surveillance requirement verifies that the Missouri River water level is maintained at a level greater than or equal to 976 feet 9 inches mean sea level. A continuous watch is established to monitor the river level when the river level reaches 980 feet mean sea level to assure no sudden loss of water supply occurs.

References

1) USAR, Section 9.10
2) ASTM D4057, ASTM D975, ASTM D4176, ASTM D2622, ASTM D287, ASTM 6217, ASTM D2709
3) ASTM D975, Table 1
4) Regulatory Guide 1.137
5) EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

List of Files on CD-ROM

#	File Name	Size	Sensitivity Level	Location
001	1 - USAR Index.pdf	35 KB	Publicly Available	CD-ROM
002	USAR 01-01.pdf	165 KB	Publicly Available	CD-ROM
003	USAR 01-02.pdf	274 KB	Publicly Available	CD-ROM
004	USAR 01-03.pdf	179 KB	Publicly Available	CD-ROM
005	USAR 01-04.pdf	252 KB	Publicly Available	CD-ROM
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10 CFR 54.37(b)
Aging Management Review Summary

A.4 Newly Identified SSCs (10 CFR 54.37(b))

After the renewed license is issued, the UFSAR update required by 10 CFR 50.71(e) must include any systems, structures, and components newly identified that would have been subject to an aging management review or evaluation of time-limited aging analysis in accordance with 10 CFR 54.21. This UFSAR update must describe how the effects of aging will be managed such that the intended function(s) in 10 CFR 54.4(b) will be effectively maintained during the period of extended operation.

No.	Date Identified	SSC Description	Aging Management Review (AMR) Conclusion	Aging Management Program
1	10/17/2012	The following Control Room Ventilation System valves were installed in the plant prior to the issuance of the renewed license but after the application had been submitted: • VA-1228A, VA-46B Refrigeration Test/Charging Isolation Valve • VA-1228B, VA-46B Refrigeration Testing Isolation Valve • VA-1228C, VA-46B Refrigeration Testing Isolation Valve • VA-1228D, VA-46B Refrigeration Testing Isolation Valve • VA-1228E, VA-46B Refrigeration Gauge Isolation Valve • VA-1228F, VA-46B Refrigeration Gauge Isolation Valve • VA-1228G, VA-46B Refrigeration Gauge Isolation Valve • VA-1228H, VA-46B Refrigeration Gauge Isolation Valve	SSCs that are included in the scope of "Metals in Refrigerants" and "Stainless Steel in Air" do not require management of aging effects but these components are now included in the license renewal scope.	The addition of these SSCs to the scope of "Metals in Refrigerants" and "Stainless Steel in Air" do not require management of aging effects but these components are now included in the license renewal scope.

LIC-14-0077
Enclosure

CD-ROM of USAR Sections and Figures

Fort Calhoun Station
Updated Safety Analysis Report
June 2014