DUKE ENERGY.

David H. Corlett
Manager, Regulatory Affairs
Harris Nuclear Plant
5413 Shearon Harris Road
New Hill NC 27562-9300

919.362.3137

10 CFR 50.59(d)(2)

April 29, 2013
Serial: HNP-13-046

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

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400

Subject: Report of Changes Pursuant to 10 CFR 50.59

Ladies and Gentlemen:

In accordance with 10 CFR 50.59(d)(2), Duke Energy Progress, Inc., formerly known as Carolina Power & Light Company, submits the attached report for the Harris Nuclear Plant (HNP). The report provides a brief description of changes to the facility and a summary of the evaluations required per 10 CFR 50.59 for those items, regardless of implementation status, between May 3, 2011, and April 15, 2013.

This letter also informs the NRC that there have been no unreported changes in commitments made during the period from May 3, 2011, through April 15, 2013.

This letter contains no new regulatory commitments.

If you have any questions regarding this submittal, please contact me at (919) 362-3137.

Sincerely,

David H. Corlett

Attachment: Report of Changes Pursuant to 10 CFR 50.59

cc: Mr. J. D. Austin, NRC Sr. Resident Inspector, HNP
Ms. A. T. Billoch Colón, NRC Project Manager, HNP
Mr. V. M. McCree, NRC Regional Administrator, Region II

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
440377 Engineering Change (EC) 76961, Rev. 1	This REG AR 440377 revises REG AR 426203 which revised REG AR405669. The scope of the EC76961 activity has now been significantly increased as described below. Part A: The FSAR currently commits HNP to comply with Regulatory Guide 1.76 Revision 0. Regulatory Guide 1.76 Revision 1 “Design Basis Tornado and Tornado Missiles for Nuclear Power Plants” was issued by the NRC in March 2007 and is available for use as design basis for future plant modifications that have a clear nexus with the subject. This activity will adopt the tornado design basis of Regulatory Guide 1.76 Revision 1 and revise the FSAR. The revised design basis will be incorporated into the design of tornado missile/pressure barriers per EC76961 Rev 1 as discussed in Part B. The revised design basis will also be available for future plant modifications. Part B: The Central Alarm Station (CAS) will be relocated from its current location in the Security Building to its new location in RAB el 286’ per EC76961 Rev 1. Prior to this relocation there are several prerequisite activities that must be completed. These activities are evaluated in Part B.	To date the Harris Nuclear Plant was designed and constructed to comply with design basis tornado guidance provided in Regulatory Guide 1.76 Revision 0. FSAR Sections 3.3 and 3.5 are specifically written to address requirements of Regulatory Guide 1.76 Revision 0. The Nuclear Regulatory Commission revised Regulatory Guide 1.76 in March 2007. The NRC states of Reg Guide 1.76 Revision 1, “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.” The NRC further states, “The NRC staff will use the methods described in this guide to evaluate… (2) submittals from operating reactor licensees who voluntarily propose to initiate system modifications that have a clear nexus with the subject for which guidance is provided herein.” This activity installs new reinforced concrete missile barrier walls and tornado pressure door 1FP-D1215 using the approved design basis tornado guidance provided in Regulatory Guide 1.76 Revision 1. Guidance provided in Revision 1 is less conservative than that provided in Revision 0 of Reg Guide 1.76 which results in a departure from the method of evaluation *currently* described in the FSAR used in establishing the design basis or in the safety analysis. However, Reg Guide 1.76 Revision 1 provides an alternate method of evaluation that was previously approved by the NRC in March 2007 for use by operating reactor licensees. Per REG-NGGC-0010 Attachment 12 instructions this departure from the method of evaluation *currently* described in the FSAR is acceptable because that method has been previously approved by the NRC for the intended application. Use of previously approved NRC design basis tornado guidance provided in Reg Guide 1.76 Revision 1 satisfies all 10 CFR 50.59 criteria. Reg Guide 1.76 Revision 1 will provide the design basis of the reinforced concrete missile barrier walls and tornado pressure door 1FP-D1215 for this activity. No License Amendment is required and no further prior NRC approval is required for implementation. This EC76961 activity will revise the FSAR to electively adopt the guidelines of Reg Guide 1.76 Revision 1 for any system modifications that have a clear nexus with the subject occurring after March 2007.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
467465 Engineering Change (EC) 70350, Rev. 15	This REG AR is a revision to REG AR 440069 (for EC 70350 Rev 12), REG AR 428915 (for EC 70350, Rev 10), REG AR 4259914 (for EC 70350, Rev 8) and REG AR 360889 (for EC 70350 Rev 0). Revision 15 of the EC adds a spot cooler to the Alternate Seal Injection (ASI) room. The purpose of the cooler is to prevent the ASI room temperature from exceeding 95 deg F during normal operations. This maximum allowable temperature in the ASI room was determined through a transient analysis of a fire in the adjacent area 1-A-BAL-B as performed in calculation HNP-M/HVAC-0004. The analysis showed that the ASI room temperature through the 24-hour ASI pump run and fire transient will not exceed the equipment tolerance value of 104 deg F provided the temperatures in the ASI room and surrounding areas are below 95 deg F prior to the start of event.	Per this 50.59 evaluation, it has been determined that there is no significant adverse impact to the design bases or safety analysis as described in the FSAR as a result of these activities. The modification will not adversely affect the ability of any SSC to perform their intended design function, either during normal plant operation or during accident mitigation. This activity will not adversely affect the probability or likelihood of occurrence, or the consequences, of a SSC malfunction beyond a minimal amount. Therefore, this activity does not constitute an unreviewed safety concern and may be implemented without prior NRC approval.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
492942 LDCR 3202 (Licensing Document Change Request) Rev. 0	This evaluation is a revision to REG AR 442400. The text was revised to address comments from the independent review of 442400 by NOS (AR 489635). LDCR 3202 revises the HNP FSAR to incorporate revised margin to overfill results. One sentence was added to the screen question 11a justification as follows: The reduction of the margin to overfill causes the answer to 11a to be “Yes”.	The proposed activity involves modifying the FSAR to report revised sensitivity analyses of the SGTR overfill event. The event was reanalyzed using the existing methodology. The Steam Generator Tube Rupture (SGTR) overfill cases are analyzed to confirm there is no flow of liquid water from the Steam Generator (SG) into the steam lines. Since there is no predicted flow of liquid into the steam lines, the activity does not create the possibility of an accident of a different type or the malfunction of an structure, system, or component (SSC) important to safety that will not cause the event to progress to an accident of a different type. The dose bases for SGTR is from a different scenario (main steam pressure operated relief valve (PORV) fails open) which bounds the overfill cases. The dose bases are not affected by the re-analyzed overfill cases, therefore the consequences of a failure of an SSC or in the consequences of the SGTR accident. The proposed activity is strictly analytical and does not involve changes to SSC or to operating procedures for SSC; therefore the proposed activity does not impact the likelihood of SSC failure or frequency of the event. The SGTR does not challenge the design basis of fission product barriers.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
514240 MST-E0013, Tech Spec Basis 3/4.8.1, 3/4.8.2, and 3/4.8.3, FSAR 8.3	Revise Technical Specification Bases and FSAR and subsequently MST-E0013 to incorporate new station battery testing methods. The current test is a constant amperage discharge test as specified in IEEE 450-1980. This does not completely bound the Station Blackout Load Profile. The proposed change is to allow the usage of the testing method specified in IEEE-450-2010, which, in addition to the constant amperage discharge test, allows the usage of a modified performance test that allows varying the current to encompass every portion of the battery duty cycle.	Tech Spec Basis 3/4.8.1, 3/4.8.2, AND 3/4.8.3, FSAR 8.3 and subsequently MST-E0013 can be revised to allow usage of IEEE 450 2010 for meeting the requirements of the Performance Test.
517267 MST-E0013, Tech Spec Basis 3/4.8.1, 3/4.8.2, and 3/4.8.3, FSAR 8.3	This is a revision to AR 514240 to change the Load Profile chart. Revise Technical Specification Bases and FSAR and subsequently MST-E0013 to incorporate new station battery testing methods. The current test is a constant amperage discharge test as specified in IEEE 450-1980. This does not completely bound the Station Blackout Load Profile. The proposed change is to allow the usage of the testing method specified in IEEE-450-2010, which, in addition to the constant amperage discharge test, allows the usage of a modified performance test that allows varying the current to encompass every portion of the battery duty cycle.	Tech Spec Basis 3/4.8.1, 3/4.8.2, AND 3/4.8.3, FSAR 8.3, and subsequently MST-E0013 can be revised to allow usage of IEEE 450 2010 for meeting the requirements of the Performance Test.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
501629 Engineering Change (EC) 74907 Rev. 1, EPT-444 Rev. 1	This is a revision to 464834. The HNP turbine train includes a double flow BB296PA (partial arc) HP turbine which will be upgraded with a new BB296FA (full arc) retrofit. The steam turbine will be designed for an uprate to 2961.7 MWt with throttle steam conditions of approximately 982 psia and 542.5°F. This EC does not authorize operation of the plant at greater than 2900 MWt although the HP Turbine has been designed for higher power conditions. Operation above 2900 MWt reactor power requires a license change and is being addressed by EC74914 and EC78541. Therefore, this 50.59 Screen will state that no Tech Spec or Operating License Change is required to implement this activity.	The new HP turbine rotor to be installed under EC74907 has been evaluated in two areas; 1) sulfur printing that was apparently done for the original rotor was not done for the new rotor, and 2) a reduction in yield strength from 90 ksi to 84 ksi. <u>Sulfur printing</u> Sulfur printing is a surface inspection technique to determine the extent of non-metallic inclusions in the metal product. Such inclusions, if too large of too numerous can weaken the metal. It is believed this inspection would have been performed primarily in the bore area as this is the most likely point for inclusions and the highest stressed area. The new rotor has only a short bore, therefore modern volumetric ultrasonic inspection is effective at determining internal non-metallic inclusions and any other discontinuities that could affect the quality of the metal. There is no increase in the likelihood of a malfunction or consequences of a malfunction of the HP turbine rotor due to lack of sulfur printing. <u>Yield strength reduction</u> The reduction in yield strength from 90 ksi to 84 ksi is not significant since the shaft stress margins are substantial. There is no increase in the likelihood of a malfunction or consequences of a malfunction of the HP turbine rotor due to a reduction in yield strength. Sulfur printing and yield strength reduction have no impact on evaluation questions 1 and 3 thru 8. <u>Conclusion</u> The new rotor does not require sulfur printing and the yield strength reduction is acceptable. No licensing change is required to install this rotor. However, the FSAR will be revised.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
500282 Engineering Change (EC) 75840, Rev. 0	The scope of the “Implementing activity” (EC75840) is the collection of items required to support Cycle 18 operation with the loading pattern and other plant configuration changes defined within this activity. The fresh fuel for Cycle 18 has a manufacturing designation of SHA1-18 from AREVA the fuel vendor. The reactor core shall be designed such that it will operate safely within the bounds defined in Technical Specifications. The installed safety systems, and Plant Operating Manual (POM) procedures shall remain effective at mitigating the consequences of the events and accidents described in Chapter 15 of the FSAR. Reload licensing analysis performed by AREVA shall be based upon the inputs contained in the Plant Parameters Document (PPD, Calculation HNP-F/NFSA-0215). The procedures and tools (POWERTRAX) used to perform Technical Specification required surveillance of the core shall remain capable of performing their functions.	HNP was analyzed for Cycle 18 operation with a reload batch of 72 feed natural uranium axial blanket (NUAB) fuel assemblies. The SHA1-18 (Region 19) NUAB fuel assemblies contain gadolinia-bearing fuel rods. All 72 assemblies contain High Thermal Performance (HTP) spacers, Intermediate Flow Mixer (IFM) grids, and FUELGUARD™ debris-resistant lower tie plates. Cycle 18 is the thirteenth HNP core reload of NUAB assemblies and containing gadolinia-bearing fuel. The feed fuel assembly design is modified to use M5 as the fuel rod cladding tube and end plug material and the design of the fuel pellet geometry is modified to a “chamfer” design versus the previous pellet shape. M5 is a proprietary AREVA zirconium alloy that uses niobium as the primary alloying element versus tin which is the primary alloying element in the Zircaloy 4 in the balance of the fuel. The changes to the fuel assembly design uses proven manufacturing techniques and previously approved NRC methodologies. The change in the cladding material required NRC approvals for an exemption from certain requirements of 10CFR Part 50, the addition of new LOCA methodology and addition of a methodology topical report. As noted in the Special Conditions Section, the approvals requested are still pending NRC approval; these approvals are expected before the affected RFO17 milestones. The safety analyses support Cycle 18 operation at a nominal core power level of 2948 MWt for up to 530 effective full-power days (EFPD). The increase in rated power to 2948 from 2900 MW is covered by a separate engineering effort (see EC74914) and a separate license amendment to the NRC. The Cycle 18 safety analyses are based on a Cycle 17 shutdown between 492 EFPD and 522 EFPD. The safety analyses do not introduce any new methodology for Cycle 18. The following areas were evaluated to support Cycle 18: mechanical evaluation, neutronics evaluation, thermal hydraulic evaluation, setpoints verification, and Chapter 15 safety analyses. The fuel mechanical design was evaluated and the results support operation to a maximum assembly exposure of 59.8 Gigawatt-day per metric ton uranium (GWd/MTU) and a maximum rod exposure of 60.0 GWd/MTU. The characteristics of the fuel and the reload core were verified to be in conformance with the current Technical Specification limits.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
500282 Engineering Change (EC) 75840, Rev. 0 (Cont.)	See above.	(Cont.) The thermal hydraulic compatibility of all the core assemblies is ensured because the Cycle 18 core will consist of only HTP/IFM fuel. The cladding material change does not result in any changes to the fuel rod diameter or pitch; therefore no mixed core penalty will be applied to the Minimum Departure from Nucleate Boiling Ratio (MDNBR) limit for Cycle 18. The potential for rod bow effects was evaluated and it was determined that no rod bow penalty on MDNBR or peak Linear Heat Generation Rate (LHGR) is required. The effect of DNB propagation on fuel failure is reflected in the results for Cycle 18. The OTΔT and OPΔT trip functions were re-analyzed. The results verified that the trip functions and setpoints provide sufficient protection for Cycle 18. Based on the analysis performed, the proposed activity will not create the possibility of an accident of a different type and does not create the possibility of a malfunction of a SSC important to safety with a different result. The analysis of record Alternative Source Term (AST) LOCA doses are bounding for Cycle 18. For the radiological consequences of the non-LOCA events, it was verified that the parameters applied in the AST analysis are bounding of the same parameters for Cycle 18. Therefore the proposed activity does not increase the consequences of analyzed accidents or increase the consequences of the malfunction of SSCs important to safety. Several Chapter 15 were re-analyzed for Cycle 18: FSAR 15.4.2 and FSAR 15.4.3.1 (Uncontrolled RCCA Bank Withdrawal at Power and Withdrawal of a Single RCCA at Power, respectively) were re-analyzed to increase the MDNBR margin. The Power Range Nuclear Instrument High Flux setpoint was lowered commensurate with a TS change submitted with the MUR license application. The analysis was performed using NRC-approved AREVA methodology EMF-2310. This was the first use of EMF-2310. FSAR 15.4.7 (Boron dilution) is re-analyzed every cycle to account for core reactivity changes. However for Cycle 18, the “dilution front” method from EMF-2310 was used in modes where only RHR is in service.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
500282 Engineering Change (EC) 75840, Rev. 0 (Cont.)	See above.	(Cont.) FSAR 15.6.5 (both Large and Small Break LOCA) involved re-analysis with new methodologies because of the use of M5 cladding. The SBLOCA analysis was re-analyzed with EMF-2328. LBLOCA was re-analyzed using a "best estimate" LOCA methodology. A plant specific method, ANP-3011, was used for LBLOCA. ANP-3011 is a variant of the generic AREVA methodology (EMF-2103). The inputs for the LBLOCA required an ESF response time for the RHR system that was previously not specified. This requires changes to Attachment 2 of PLP-106. As part of the review of RHR inputs, the RHR system response time was increased to replace a previous qualitative assessment of the impact of the RHR min-flow valve response. The proposed activity has changes to the respective surveillance tests to enforce the revised analysis inputs. The balance of the Chapter 15 events were reviewed with respect to Cycle 18 plant configuration/operation and neutronics changes. The event review indicated that due to changes in neutronic characteristics some events required a partial (e.g., MDNBR or fuel centerline melt) re-analysis for Cycle 18. The re-analysis and evaluation confirmed that all applicable acceptance criteria continue to be met for each event. Therefore the proposed activity will not increase the frequency of occurrence of an accident previously evaluated in the FSAR, will not increase the likelihood of the malfunction of an SSC important to safety previously evaluated and the proposed activity does not result in the design basis limit for a fission product barrier to be exceeded. The analyses that were performed to support the Cycle 18 design were performed using methods that were previously approved by the NRC. As described in the above discussion of the acceptability of the analysis results, implementation of the Cycle 18 core design and supporting safety analyses demonstrated that the requirements and acceptance criteria defined in the FSAR are satisfied for Cycle 18 operation. Therefore, the Cycle 18 reload design will continue to meet the plant licensing basis.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	**Description of Change**	**Evaluation Summary**
505256 Engineering Change (EC) 76961, Rev. 3	This REG AR revises the following REG AR's listed chronologically: REG AR440377 REG AR426203 REG AR405669 The scope of the EC76961 activity has now been significantly increased as described below. Part A: The FSAR currently commits HNP to comply with Regulatory Guide 1.76 Revision 0. Regulatory Guide 1.76 Revision 1 "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants" was issued by the NRC in March 2007 and is available for use as design basis for future plant modifications that have a clear nexus with the subject. This activity will adopt the tornado design basis of Regulatory Guide 1.76 Revision 1 and revise the FSAR. The revised design basis will be incorporated into the design of tornado missile/pressure barriers per EC76961 Rev 1 as discussed in Part B. The revised design basis will also be available for future plant modifications. Part B: The Central Alarm Station (CAS) will be relocated from its current location in the Security Building to its new location in RAB el 286' per EC76961 Rev 1. Prior to this relocation there are several prerequisite activities that must be completed.	This evaluation addresses only Parts A and C. Part B screened out and did not require further evaluation. Part A: To date the Harris Nuclear Plant was designed and constructed to comply with design basis tornado guidance provided in Regulatory Guide 1.76 Revision 0. FSAR Sections 3.3 and 3.5 are specifically written to address requirements of Regulatory Guide 1.76 Revision 0. The Nuclear Regulatory Commission revised Regulatory Guide 1.76 in March 2007. The NRC states of Reg Guide 1.76 Revision 1, "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." The NRC further states, "The NRC staff will use the methods described in this guide to evaluate… (2) submittals from operating reactor licensees who voluntarily propose to initiate system modifications that have a clear nexus with the subject for which guidance is provided herein." This activity installs new reinforced concrete missile barrier walls and tornado pressure door 1FP-D1215 using the approved design basis tornado guidance provided in Regulatory Guide 1.76 Revision 1. Guidance provided in Revision 1 is less conservative than that provided in Revision 0 of Reg Guide 1.76 which results in a departure from the method of evaluation *currently* described in the FSAR used in establishing the design basis or in the safety analysis. However, Reg Guide 1.76 Revision 1 provides an alternate method of evaluation that was previously approved by the NRC in March 2007 for use by operating reactor licensees. Per REG-NGGC-0010 Attachment 12 instructions this departure from the method of evaluation *currently* described in the FSAR is acceptable because that method has been previously approved by the NRC for the intended application.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
505256 Engineering Change (EC) 76961, Rev. 3 (Cont.)	(Cont.) Part C: The HNP licensing basis requires design of concrete masonry unit (CMU) reinforced block walls per FSAR section 3.8.4.8 in accordance with loads and load combinations indicated in Tables 3.8.4-1 and 3.8.4-2. During development of this EC it was discovered that the design methodology used in calculation CAR-C-0091 is non-conservative compared to the requirements in the FSAR.	(Cont.) Use of previously approved NRC design basis tornado guidance provided in Reg Guide 1.76 Revision 1 satisfies all 10 CFR 50.59 criteria. Reg Guide 1.76 Revision 1 will provide the design basis of the reinforced concrete missile barrier walls and tornado pressure door 1FP-D1215 for this activity. No License Amendment is required and no further prior NRC approval is required for implementation. This EC76961 activity will revise the FSAR to electively adopt the guidelines of Reg Guide 1.76 Revision 1 for any system modifications that have a clear nexus with the subject occurring after March 2007. Part C: The design of all plant masonry block walls is recorded in calculation CAR-C-0091. The method of determining effective shear area per CAR-C-0091 is non-conservative with respect to our licensing basis commitments shown in FSAR Table 3.8.4-2 Note 2. Use of the FSAR methodology will cause some CMU block walls to be stressed beyond shear allowable when the wall is subjected to design basis loading. Specifically the FSAR requires that effective shear area be calculated as the product of the CMU block face shell thickness (approximately 1-1/2") multiplied by 3 times the CMU wall thickness. Calculation CAR-C-0091 uses a much greater effective shear area which is the product of the grouted core length multiplied by the effective depth to tensile reinforcement. Design of CMU reinforced block walls is described in FSAR section 3.8.4.8 in accordance with loads and load combinations indicated in Tables 3.8.4-1 and 3.8.4-2. Standard Review Plan sections 3.7 and 3.8 are complied with to the extent applicable. Standard Review Plan Section 3.5 was not used as no credit is taken for these walls for missile protection. Also per FSAR section 3.8.4.8, the design criteria utilized in the design of masonry walls located in Seismic category 1 structures complies with the NRC Structural Engineering Branch Criteria for Safety Related Wall Evaluation dated July 1981. Masonry walls have been addressed in CP&L's June 30, 1982, response to Enclosure 5, Item No. 17, of NRC Acceptance Review of OL Application, dated November 25, 1981.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
505256 Engineering Change (EC) 76961, Rev. 3 (Cont.)	See above.	(Cont.) Per FSAR section 3.8.4.8, the following codes are used for the analysis and design on masonry block walls and any associated steel framing: ACI 531-79 American Concrete Institute "Building Code Requirements for Concrete Masonry Structures" UBC-79 Uniform Building Code, by International Conference of Building Officials AISC 7th Ed. American Institute of Steel Construction "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings" ACI 318-71 American Concrete Institute "Building Code Requirements for Reinforced Concrete" The design methodology used in CAR-C-0091 is entirely consistent with the guidelines of ACI 531-79 "Building Code Requirements for Concrete Masonry Structures". It is also consistent with a report titled "Safety Related Concrete Masonry Walls" dated October 6, 1980 prepared by Owners and Engineering Firms Informal Group on Concrete Masonry Wall (See EC76961 Z22 Attachment W). This was a final report prepared by the informal AE and Owners Group established to provide a uniform industry response to the NRC regarding IE Bulletin 80-11. Projects performing masonry wall re-evaluations were instructed to conform to the recommendations and guidelines of this report. The report describes how stresses are to be calculated and what limits they should be compared to. This report includes a table that is nearly identical to FSAR Table 3.8.4-2. The report pre-dates the FSAR submittal and it is apparent that FSAR Table 3.8.4-2 was entirely based on the report. However Ebasco erroneously substituted a different sketch to be used for determination of effective shear area and submitted that for incorporation in FSAR Amendment 3 in 1982. While this is a departure from the method of evaluation currently described in FSAR Table 3.8.4-2, the design methodology used in CAR-C-0091 is entirely consistent with the guidelines of ACI 531-79 "Building Code Requirements for Concrete Masonry Structures". Per FSAR section 3.8.4.8, ACI 531-79 is the standard used for the analysis and design on masonry block walls. No License Amendment is required and no prior NRC approval is required for implementation. Use of approved design basis guidance provided in ACI 531-79 satisfies all regulatory considerations. FSAR Table 3.8.4-2 will be revised allowing HNP to use effective shear area based on guidance provided in ACI 531-79.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
536835 SPP-0685T, Rev. 0	EC79281 requires extensive yard area soil excavation to install new duct bank. This excavation will pass over existing buried safety related SSCs that credit the soil overburden to provide missile protection for the buried safety related SSCs. With the soil overburden removed, the safety related SSCs are at risk of sustaining damage from the design basis missile. The soil overburden is credited to provide protection of safety-related SSC's from design basis tornado scenarios postulated per Regulatory Guide 1.76 R0. Removal of the passive soil overburden barrier results in an associated increase in risk to SSC's being protected from adverse effects of design basis tornado missile. Damage to these emergency service water (ESW) SSCs would result in declaring the Aux Reservoir loop inoperable and entering either Tech Spec 3.7.4. or Tech Spec 3.0.3 depending on whether only one train or both trains were declared inoperable respectively. EC79281 evaluates the risk of missile damage and specifies that compensatory protective measures be taken at any location where the new duct bank crosses safety-related pipe with less than 4.7 ft of soil cover remaining to top of pipe after excavation, and at any location where the new duct bank crosses the existing safety related duct bank that runs between manholes M86 and M87.	Procedure SPP-0685T will administratively control the missile shield such that it may only be removed when the possibility of the design basis hazard does not exist. The procedure implements the administrative control of the 10 CFR 50.65(a)(4) risk assessment requirements as evaluated per EC79281. The intent is to allow only intermittent and temporary opening of the excavation such that implementation of the modification may proceed. The missile shield will be temporarily opened as allowed per provisions of 10 CFR 50.65(a)(4). This requires that this Maintenance Activity be treated as a "Temporary Alteration in support of Maintenance" and subject to the 90 day limitation described in NEI 96-07 Section 4.1.2, Revision 1. The 90 day clock resets each time the missile shield is installed and the buried safety related SSCs restored to their as-designed, operable condition. A new 90 day clock is initiated when the missile shield is removed in support of a Maintenance Activity per 10 CFR 50.65(a)(4).

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
536835 SPP-0685T, Rev. 0 (Cont.)	(Cont.) If there is a threat of tornado generated missiles during construction, the compensatory protective measures (hereinafter referred to as "missile shield") require that the trench shall be filled with #57 stone and the stone covered with a steel plate 1"x 48" x 8'-0" minimum on top at grade level. These protective compensatory measures (via installation of the missile shield) restore the credited protection of safety related SSCs from the adverse effects of the design basis missile. The replacement missile shield will be administratively controlled by special procedure SPP-0685T "Control of Missile Shield for Excavations of Auto Pumps Modifications Modes 1-6". The purpose of this 50.59 screen is to evaluate special procedure SPP-0685T. Procedure SPP-0685T will administratively control the replacement missile shield such that it may only be absent when the possibility of the design basis missile hazard does not exist. The procedure implements administrative control of the missile shield utilizing 10 CFR 50.65(a)(4) risk assessment requirements for temporary alterations in support of maintenance activities. The intent is to allow only temporary opening of the excavation such that implementation of the modification may proceed.	See above.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
537906 Engineering Change (EC) 86877, Rev. 0	During RFO17, one coil in AH-37 and AH-38 and two coils in AH-39 – were found leaking. These are the three non-safety Containment Fan Coolers (CFCs). The coil leaks are from normal service water (NSW). Small leaks inside containment, even from non-safety sources such as NSW, are not acceptable as they will contribute to containment sump in-leakage and can mask leaks from other safety-critical equipment during the operating cycle. Leaks can adversely impact RCS leak detection and may require at-power entries into containment. Time and resource limitations drove a decision not to replace these coils during RFO17. Therefore, the leaking coils should be isolated before exiting RFO17. Temporary blanks (pancake flanges) will be installed in the supply and return flanged connections of the leaking coils in AH-37, AH-38 and AH-39. The blanks will be removed after the coil bundle is replaced or repaired. This is currently expected to occur during the next scheduled refueling outage (RFO18).	AH-37, AH-38, and AH-39 are the non-safety containment cooling units served by Normal Service Water. Cooling coils in each unit is to be temporarily isolated in order to stop leaks. These air handling units (AHUs) serve the three RCP and SG cubicles as well as containment as a whole. The leaking coils are to be isolated by installing blank (pancake) flanges in the supply and return flanges of the coil assemblies. EC 86877 references prior plant operating experience to show that this activity will not cause normal operating limits on containment average temperature, reactor coolant pump (RCP) motor stator winding temperatures, or RCP vibrations to be exceeded. No procedures, or design or operating limits are being revised. If existing operating limits were to be reached, existing guidance would be followed to ensure SSCs important to safety were operated within the current design and operating limits. AH-37, AH-38, and AH-39 are not credited for accident mitigation in any Chapter 15 event, so HNP's response to any design-basis accident remains unchanged. The reduction in the cooling capacity of the non-safety air handling units does not affect accident response, accident severity, accident consequences, equipment interactions, or require any new evaluation methodology.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
543117 SPP-0685T, Rev.1	This is a revision to the original 50.59 evaluation for SPP-0685T Rev. 0 which was performed by REG AR 536835-011. Two sentences were added to the activity section of REG 536835. They are as follows: As an alternative to the compensatory measure of using #57 stone, the compensatory protective measures can be fulfilled by filling the trench with freshly poured reinforced concrete per EC 79281 design, and then covered with a steel plate 1”x 48” x 8’-0” minimum on top at grade level. In addition, the missile shield protection is provided after the reinforced concrete installed by EC 79281 has cured for a minimum of 12 hours, after which time any steel plate installed can be removed.	Revision to this evaluation did not change the summary or conclusions described in Rev. 0, see above page 12 of 24.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
547691 LDCR 3248 (Licensing Document Change Request) Rev. 0	As documented in CR 547630, FSAR Section 12.3.2.16.2 states that the atmospheric dispersion factors used for the Technical Support Center (TSC) dose analysis are determined based on the methodology in NUREG/CR-6331 (1995). This statement was correct for the dose analysis performed prior to Steam Generator Replacement - Power Up Rate (SGR-PUR) (Ref. calculation CPL-VII-009B). However, after SGR-PUR, the dose analysis of record became HNP-F/NFSA-0072 and it reverted back to atmospheric dispersion factors based not on NUREG/CR-6331 but on the Murphy and Campe methodology from “Nuclear Power Plant CR Ventilation System Design For Meeting GDC 19 (1974)” . Therefore, the FSAR description no longer matches the current analysis of record for the TSC. In addition, a recent Numerical Applications (NAI) analysis of TSC in-leakage has been performed. It uses improved dispersion factors based on NUREG/CR-6331 (1997) methodology for all containment releases and existing Murphy and Campe dispersion factors for all other releases (those from ECCS and RWST). The analysis credits the benefit of the improved containment dispersion factors to produce an increased unfiltered TSC in-leakage limit that will maintain existing analyzed TSC doses within the limits of the current analysis of record.	As documented in CR 547630, FSAR Section 12.3.2.16.2 states that the atmospheric dispersion factors used for the Technical Support Center (TSC) dose analysis are determined based on the methodology in NUREG/CR-6331 (1995). This statement was correct for the dose analysis performed prior to SGR-PUR (Ref. calculation CPL-VII-009B). However, after SGR-PUR, the dose analysis of record became HNP-F/NFSA-0072 and it reverted back to atmospheric dispersion factors based not on NUREG/CR-6331 but on the Murphy and Campe methodology from “Nuclear Power Plant CR Ventilation System Design For Meeting GDC 19 (1974).” Therefore, the FSAR description no longer matches the current analysis of record for the TSC. In addition, a recent Numerical Applications (NAI) analysis of TSC in-leakage has been performed. It uses improved dispersion factors based on NUREG/CR-6331 (1997) methodology for all containment releases and existing Murphy and Campe dispersion factors for all other releases (those from ECCS and RWST). The analysis credits the benefit of the improved containment dispersion factors to produce an increased unfiltered TSC in-leakage limit that will maintain existing analyzed TSC doses within the limits of the current analysis of record. This NAI analysis is being incorporated into HNP-F/NFSA-0072 and will become the new analytical basis for TSC in-leakage. FSAR Section 12.3.2.16.2 is being revised to accurately describe how atmospheric dispersion factors are calculated for the TSC dose analysis. The revised description will consider the most recent NAI analysis. It will note that current dispersion factors are based on both NUREG/CR-6331 (1997) and Murphy and Campe. Due to the difference in NUREG publication date, as well as the mixed used of both NUREG/CR-6331 and Murphy and Campe dispersion factors, this is considered a change to methodology as described in the FSAR. Although the Analysis of Record (AOR) values for allowable unfiltered TSC in-leakage is are being increased (AOR in-leakage varies with the event), the values are not currently given in the FSAR and it will not be added by this revision. Likewise, calculated TSC doses are not currently stated in the FSAR and are not being added or changed by this FSAR revision. The existing 5 REM limit for the TSC remains unaffected.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
547691 LDCR 3248 (Licensing Document Change Request) Rev. 0 (Cont.)	(Cont.) This NAI analysis is being incorporated into HNP-F/NFSA-0072 and will become the new analytical basis for TSC in-leakage FSAR Section 12.3.2.16.2 is being revised to accurately describe how atmospheric dispersion factors are calculated for the TSC dose analysis. The revised description will consider the most recent NAI analysis. It will note that current dispersion factors are based on both NUREG/CR-6331 (1997) and Murphy and Campe. Due to the difference in NUREG publication date, as well as the mixed used of both NUREG/CR-6331 and Murphy and Campe dispersion factors, this is considered a change to methodology as described in the FSAR. Although the analysis of record (AOR) values for allowable unfiltered TSC in-leakage in HNP-F/NFSA-0072 are being increased (AOR in-leakage varies by event), the values are not currently given in the FSAR and they will not be added by this revision. Likewise, calculated TSC doses are not currently stated in the FSAR and are not being added or changed by this FSAR revision. The existing 5 REM limit for the TSC, which is in the FSAR, remains unaffected.	(Cont). The proposed activity does not result in an increase in frequency of an accident or increase the likelihood of a malfunction of an SSC important to safety as previously evaluated in the FSAR. The activity does not result in an increase in the consequence of an accident or malfunction of equipment as previously evaluated in the FSAR. The activity does not create the possibility of an accident of a different type or malfunction of an SSC with a different result and does not affect the fission product barriers. The proposed activity uses a different method than that described in the FSAR in the calculation of atmospheric dispersion factors (X/Q); however, comparable analyses at other plants (H. B. Robinson, Unit 2; Bellefonte; San Onofre Units 2 and 3; St. Lucie Plant, Unit 2; Seabrook Station, Unit 1; Oconee Units 1,2 and 3; Catawba and Millstone Unit 2) using the new method (ARCON96) have been accepted by the NRC. NEI 96-07 (Reference 4) provides considerations for determining if new methods may be considered “Approved by the NRC for the Intended Application.” It has been determined that the use of ARCON96 to calculate atmospheric dispersion factors (X/Q) meets those considerations and may be used as approved by the NRC for this application. The new evaluation for TSC unfiltered in-leakage also relies on the RADTRAD-NAI code to develop a relationship between changes in dispersion factors and changes in dose (and, subsequently, in-leakage). The evaluation does not use RADTRAD-NAI to calculate new dose values. The existing analysis-of-record values are left unchanged as it uses this code only to establish the relative in-leakage sensitivities. Therefore, since the original Westinghouse TITAN5 code remains the source of the calculated TSC dose values, there is no change in methodology associated with the dose calculations.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
536669 Condition Report CR 501117 LDCR 3249 (Licensing Document Change Request) Rev. 0	In response to NCR 446515, the ultimate heat sink (UHS) heat-up analysis (calculation SW-0085) was revised to incorporate the effects of Auxiliary Reservoir leakage on post-accident reservoir temperatures and levels. The new analysis also included improved inputs and mass-and-energy balance equations resulting in an improved (lower) peak reservoir temperature even with valve leakage considered. Based on the above, FSAR Section 2.4.11.7 is being revised to delete the existing sentence which reads, "The Auxiliary Reservoir is completely isolated from the Main Reservoir". Now that leakage through 1SW-3 and 1SW-4 is permitted within the bounds of calculation SW-0085, complete isolation is not required. FSAR Section 2.4.11.7 is also being revised to change the final, 30-day, post-LOCA Auxiliary Reservoir temperature from 95.33 ºF to 95.17 ºF per Revision 3 of SW-0085.	The evaluation finds that allowing limited leakage through 1SW-3 and 1SW-4, rather than considering them as completely isolated, does not require a license amendment. Calculation SW-0085 Rev.3 provides conservative limitations on the amount of leakage allowed through 1SW-3 and 1SW-4 (or, more specifically, the size of the seat-to-body gaps in these valves) such that Auxiliary Reservoir temperatures, ESW Pump submergence, and ESW Pump capacities remain unaffected. The gap limits calculated by SW-0085 are substantially larger than the limits used by CM-M0274 during valve maintenance. So, over time, the valves will be maintained with large operating margins. Because the seat-to-body gap limits are based on inputs designed to ensure the function of associated safety-related SSCs, removing the "completely isolated" description from the FSAR does not result in any increases in accident probabilities, effects, or consequences. The method of evaluation used in the revised UHS heat-up calculation is the same used in previous revisions, and the details of this calculation are not described in the FSAR. All existing Tech Spec limits associated with the Auxiliary Reservoir will continue to apply. FSAR Section 2.4.11.7 is also being revised to change the final, 30-day, post-LOCA Auxiliary Reservoir temperature from 95.33 ºF to 95.17 ºF per Revision 3 of SW-0085.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
561057 LDCR 3248 (Licensing Document Change Request) Rev. 0	REG AR 561057 is Revision 1 of REG AR 547691. This revision affects only the Attachment 11 evaluation and does not impact the Attachment 1 screen. One sentence was clarified to the evaluation question 8 justification as follows: Has industry experience with the computer code been appropriately considered? Yes. This application was performed by NAI and relies on their previous dose analysis experience, including the appropriate use of ARCON96 and RADTRAD-NAI. The following are NRC SERs including the use of ARCON96: - H. B. Robinson, Unit 2 - Bellefonte - San Onofre Units 2 and 3 - St. Lucie Plant, Unit 2 - Seabrook Station, Unit 1 - Oconee Alternative Source Term - Catawba Alternative Source Term - Millstone Power Station Unit 2 [Alternate] Alternative Source Term (AST)	Revision to this evaluation did not change the summary or conclusions described in Rev. 0, see above page 16 of 24.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
575728 LDCR 3259 (Licensing Document Change Request) Rev. 0	Review the changes being made to the FSAR as a result of the ASME Section XI Inservice Inspection (ISI) programs transition to the 3rd Inspection Interval and subsequent move to the 2001 edition with 2003 addenda of ASME Section XI. In addition, this screen will review the changes being made to the FSAR as a result of HNP adopting risk informed programs for ISI weld examinations and Break Exclusion Region (BER) piping weld examinations. This screen does not include specific procedure changes made, only FSAR changes made per licensing document change request (LDCR) 3259 as listed below: Section 1.8 Section 5.2.4.5 Section 5.2.4.7 Section 6.6.1 Section 6.6.6 Section 6.6.7 Section 6.6.8	The changes to the FSAR evaluated under this 10 CFR 50.59 evaluation can be implemented without prior NRC approval. All of the aforementioned changes have been previously approved for use by the NRC for the intended applications through either Safety Evaluations or Federal Regulation. As such, the changes do not result in a departure from a method of evaluation and can be implemented.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
572396 Engineering Change (EC) 86545 Rev. 0	Engineering Change EC 86545 will provide the details required to fabricate and install an elevated head stand and a shield wall for the IRVH (Integrated Reactor Vessel Head) inspection. The head stand will be an extension of the existing head stand and the shield wall will be installed on the existing concrete pedestal. The head stand and shield wall will stay permanently in the containment. These will have Service Level I (SLI) – Protective Coatings applied. SLI coatings shall be in accordance with CPL-XXXX-W-005 and applied per MNT-NGGC-0009. The head stand extensions and shield wall are designed for self weight and seismic loads for remaining in containment during the operating cycle. The head stand extension will be qualified for the deadweight load of IVRH in addition to shielding weight during outages. The shield wall is designed for the shielding loads during outage conditions. The stands and shield wall are non-safety related. Calculation HNP-C/STRU-1097 addresses these loading conditions. The elevated head stand and the Shield wall will be mounted above the existing head support in the lay down area on concrete slab on elevation 286'– not above grating – and will therefore not interfere with HVAC flow patterns inside containment.	The reactor head will be transported to the lay down area during outages. The head will be lifted higher from the administratively controlled height to be placed on the extended head stand designed to provide access for head inspections. The increase in height is bounded by the same limited amount of safe shutdown equipment that could be damaged directly below the lay down area currently described in FSAR section 9.1.4.2.2.8.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
572396 Engineering Change (EC) 86545 Rev. 0 (Cont.)	(Cont.) Also, the change in containment free volume is so small that the number of air changes per hour calculated in 9-RAB-001A (4.04 changes per hour for Space #6) will not be affected. Calculation HNP-C/STRU-1097 evaluation address the following:. 1. The reactor head stand assembly installed permanently is qualified for deadweight and seismic conditions to ensure that the stand will maintain its structural integrity and there are no II over I concerns. 2. The reactor head stand is qualified for deadweight condition during the outages, for the weight of the reactor head and lead shielding. 3. The Shield Wall with two doors installed 180 degrees apart, is qualified for deadweight and seismic conditions so that it can be left in the containment during plant operation. 4. The shield wall is also qualified for lead shielding that can be hung during the outages. The head stand and Shield Wall will be fabricated pre-outage and will be installed in location during RFO-18.	See above.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
560647 Engineering Change (EC) 84155 Rev. 0	Install 230 kV overhead lines to replace the existing underground oil-filled cables that provide offsite power from the Switchyard to the Start-up Transformers, SUT-A and SUT-B. Two three-phase overhead 230 kV lines will be installed on different routes to connect each SUT to the Switchyard. The overhead lines will be installed on new self supporting structures (no guyed structure) anchored to new concrete foundations. The new overhead lines for SUT-A will be routed towards the switchyard along the parking lot east of the Security Building and Admin Buildings. The new overhead lines for SUT-B will be routed eastward, south of the EDG Building and link to the existing (abandoned) 230 kV lines that connect to the switchyard. This route will require the removal of (2) high mast security light poles that interfere with the new lines. New replacement lights will be attached to the south wall of the Diesel Generator Building. Another high mast Security light is modified to prevent potential interaction with the new lines. This EC will resolve the problem of the oil-filled cables that have sprung leaks over the past several years. This modification also eliminates issues associated with aging of the oil filled cables, including testing program described in FSAR 18.1.1.40.	The proposed activity to be implemented by EC 84155 to install 230 kV overhead lines to replace the existing underground oil-filled cables that provide offsite power from the Switchyard to the Start-up Transformers, SUT-1A and SUT-1B, can be implemented without NRC approval or License Amendment.

SHEARON HARRIS NUCLEAR POWER PLANT
REPORT OF CHANGES PURSUANT TO 10 CFR 50.59

Log Number / Implementing Document	Description of Change	Evaluation Summary
560647 Engineering Change (EC) 84155 Rev. 0 (Cont.)	(Cont.) Sections of the FSAR that describes the underground 230kV low-pressure oil-filled cables will be updated to incorporate the changes required by the EC.	See above.