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10 CFR 50.59(d)(2)

HNP-18-023

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

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400/Renewed License No. NPF-63

Subject: Report of Changes Pursuant to 10 CFR 50.59 and Summary of Commitment Changes

Ladies and Gentlemen:

In accordance with 10 CFR 50.59(d)(2), Duke Energy Progress, LLC, submits the attached report for Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The enclosure 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 April 12, 2016, and April 5, 2018.

This letter also informs the NRC that there have been no unreported changes in commitments made during the period from April 12, 2016, through April 5, 2018.

This letter contains no new regulatory commitments.

If you have any questions regarding this submittal, please contact Jeff Robertson, Manager – Regulatory Affairs, at (919) 362-3137.

Sincerely,

Bentley K Jones

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Enclosure: Report of Changes Pursuant to 10 CFR 50.59

cc: J. Zeiler, NRC Sr. Resident Inspector, HNP
M. Barillas, NRC Project Manager, HNP
NRC Regional Administrator, Region II

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ENCLOSURE

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63

REPORT OF CHANGES PURSUANT TO 10 CFR 50.59
(27 pages plus cover)

Log Number / Implementing Document	Description of Change	Evaluation Summary
01962459/ Engineering Change (EC) 284102, Revision 0	While engineering personnel were evaluating operational experience associated with the Susquehanna Plant, it was discovered that the Harris Nuclear Power Plant, Unit 1 (HNP), Technical Specifications (TS) contained non-conservative surveillance test acceptance criteria for determining Emergency Diesel Generator (EDG) operability. The associated surveillance test procedures utilize EDG transient ranges for voltage and frequency, i.e. 6900 volts ± 10% and 60 Hz ± 2%. These values are applicable when the EDG is operating in the “isochronous mode” (i.e. isolated from the offsite source) and only when the generator is coming up to speed or is being loaded (i.e. transient). Steady state frequency and voltage conditions were not identified. Therefore, the existing TS ranges for frequency and voltage are too wide for steady-state conditions. This condition was entered into the Corrective Action Program as Nuclear Condition Report (NCR) 461896. The HNP EDG voltage regulators are set at 6900 volts alternating current (VAC) ±120 volts. ECs 69609 and 82877 replaced the originally-supplied EDG Woodward analog speed control system with a new Woodward 2301A electronic speed control governor. The steady state speed band of the governor is ± 0.25%, which results in a steady state frequency range between 60.15 hertz (Hz) and 59.85 Hz. EC 284102 provides the basis for changes to new voltage (±4%) and frequency (±0.8%) tolerances, which are more restrictive than current limits, and	EC 284102 results in an increase to the maximum required RWST switchover volume and decreases in CT pump and RHR pump NPSH margins. These components are used for accident mitigation and are not potential accident initiators. Therefore, this activity has no impact on the frequency of occurrence of any accident previously evaluated in the FSAR. Built-in margin exists to compensate for the increase in the maximum required RWST switchover volume. Part of the volume between the RWST Lo-Lo and Empty set points is margin that is uncredited by analysis. The currently available switchover margin is approximately 20,600 gallons. This will decrease to approximately 19,300 gallons. Since some of the RWST margin can be credited to compensate for the increase in analytical outflow, none of the existing RWST set points are affected; all associated automatic and procedural actions remain unchanged. Therefore, the increase in maximum required switchover volume and corresponding decrease in switchover margin have no impact on the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the FSAR. EC 284102 results in a reduction to the injection and recirculation-mode NPSH margin for the CT pump. Insufficient NPSH margin can result in pump cavitation and performance degradation. Although there is a reduction in the available NPSH for the CT pumps, the available NPSH is still greater than the required NPSH in both modes of operation. Therefore, this has a minimal

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	updates all affected documents accordingly. The TS revision process will take place independently from EC 284102. The steady-state frequency tolerance change was incorporated into plant design-basis documents (calculations) in order to account for the effect on EDG-driven safety-related components such as pumps, fans, and motor-operated valves and updates all affected plant documents accordingly. When +0.8% frequency was incorporated into calculations for maximum Containment Spray (CT), Residual Heat Removal (RHR), and Charging/Safety Injection (CSIP) flow rates during switchover, it resulted in an increase to the maximum required Refueling Water Storage Tank (RWST) switchover volume from 63,360 gallons to 64,688 gallons. Since the HNP Final Safety Analysis Report (FSAR), Section 6.3.2 cites this volume, and since an increase in this volume is non-conservative, this was identified as an adverse effect to an FSAR-described design function. Also, the increase in the CT pump flow rate resulted in a decrease in CT pump net positive suction head (NPSH) margin during injection and recirculation modes. In injection mode, the net positive suction head available (NPSHA) decreased from 92.3 feet to 92.0 feet and net positive suction head required (NPSHR) increased from 12.5 feet to 13.0 feet. In recirculation mode, NPSHA decreased from 27.1 feet to 25.5 feet and NPSHR increased from 12.0	impact on the likelihood of CT pump malfunction. EC 284102 results in a reduction to the RHR pump NPSHA in recirculation mode to be consistent with the existing design basis. Insufficient NPSH margin can result in pump cavitation and performance degradation. Although there is a reduction in the available NPSH for the RHR pumps as shown in the FSAR, the available NPSH is still greater than the required NPSH. Therefore, this has a minimal impact on the likelihood of an RHR pump malfunction. The reductions in RWST switchover margin and CT and RHR NPSH margin do not have an impact on the ability of any equipment to perform their accident and dose mitigating functions. Although RWST switchover margin was reduced, significant positive margin still remains. Likewise, the NPSH margins for the CT and RHR pumps also remain positive, so there is no impact on pump performance or their ability to mitigate an accident. Therefore, this activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR. The consequences of a failure of a CT pump or an RHR pump do not change as a result of this activity. Therefore, this activity does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR. There are no new failure modes established by EC 284102 and no new equipment added to the plant. Reductions in the RWST switchover margin and the CT

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	feet to 12.4 feet. Since FSAR Section 6.2.2.3.2.1 cites these values, and since this reduction in NPSH margin is non-conservative, this was identified as an adverse effect to an FSAR-described design function. During EC 284102 development, it was noted that the minimum NPSHA for the RHR pumps immediately following switchover to recirculation is cited as 22.14 feet in FSAR Table 6.3.2-1. This does not agree with the value of 20.85 feet shown in the existing plant calculation, SI-0043. So, the value in the FSAR will be corrected by EC 284102 to match SI-0043. This reduction in RHR pump NPSHA as shown in the FSAR is considered an adverse effect to an FSAR-described design function.	and RHR pump NPSH margins do not result in new accident types. Therefore, this activity does not create a possibility for an accident of a different type than any previously evaluated in the FSAR. This activity does not affect the design function of the RWST, the CT pumps, or the RHR pumps. Therefore, this activity does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in FSAR. This activity does not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered. The method of calculating these margins was unchanged – only the flow-rate inputs were revised to account for increased pump speeds associated with +0.8% frequency tolerance. Therefore, this activity does not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.
01981672/ EC 296193, Revision 0	In order to resolve the non-conforming condition described in Nuclear Condition Report (NCR) 626242, EC 296193 amends the HNP steam generator tube rupture (SGTR) margin-to-overfill (MTO) analysis of record and, subsequently, revises the FSAR, Section 15.6.3, and affected plant procedures. As identified in NCR 626242, a credible failure in the turbine-driven auxiliary feedwater pump (TDAFWP) speed control system could cause the TDAFWP to run at the upper end of its speed-control range of 4,100 revolutions per minute (RPM), rather	This activity concerns the HNP's response to a SGTR event, regardless of its frequency of occurrence. The MTO analysis is based on existing plant design features and existing emergency operating procedures. The activity does not add, delete, or modify any plant components. Therefore, this activity has no impact on the frequency of occurrence of a SGTR event or any other accident previously evaluated in the FSAR. The evaluation also concludes that the reduction in required AFW isolation time for a SGTR event from 10 minutes to 8.8 minutes does not result in more than a minimal

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	than at the normal steady-state speed of approximately 3,500 RPM as is implicitly assumed by the current FSAR Section 15 SGTR MTO analysis. Such a failure would result in additional feedwater delivery to a faulted steam generator and could adversely impact the calculated margin to overfill. This single failure scenario involving the TDAFWP speed controller is not new. It was originally considered in calculation HNP-M/MECH-1049, Revision 0, in 2001, prior to the performance of the current MTO analysis. However, when the MTO analysis was supplemented in 2010, the supplement failed to incorporate or consider this credible failure. The maximum allowable SGTR AFW isolation time is being reduced. This action is credited and described in the SGTR FSAR Chapter 15 analysis and is a design function. Reducing this time has an adverse impact on a design function and on the control of this design function.	increase in the likelihood of occurrence of a malfunction of an SSC important to safety as evaluated in the FSAR. This activity does not revise the SGTR dose analysis; it is limited to the MTO analysis and the single-failure assumptions within the MTO analysis. This activity has determined that the TDAFWP speed controller failure is the most limiting single failure with respect to MTO following a SGTR and has shown that acceptable MTO is maintained with this limiting equipment malfunction. Other plausible equipment malfunctions associated with MTO result in greater MTO. Therefore, positive MTO is maintained regardless of the equipment malfunction and this basic assumption in the dose analysis remains valid. None of the results of this activity impact the SGTR dose analysis. Therefore, this activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR. This activity also does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR. This activity does not add, delete, or modify components within the plant. This activity is limited in scope to the re-evaluation of an existing accident, a SGTR event, given a different input value for AFW delivery. No new accident types are considered or can be introduced. Therefore, this activity does not create the possibility of an accident of a different type not previously evaluated in the FSAR. This activity also does not create the possibility for a malfunction of an SSC important to safety with different results than any previously evaluated in the FSAR.

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		A SGTR event assumes the failure of a portion of the reactor coolant system (RCS) pressure boundary (a steam generator tube). This is an existing FSAR Chapter 15 analysis. The AFW delivery value modified in the most recent version of the SGTR MTO analysis in HNP-M/MECH-1049, Revision 1, does not affect safety injection inputs, operator action types, or other parameters that would adversely impact fuel cladding integrity. Therefore, this activity does not result in a design basis limit for an FSAR-described fission product barrier being exceeded or altered. For this activity, the SGTR MTO supplemental analysis in HNP-M/MECH-1049, Revision 1, uses the existing analysis of record (AOR), which is contained in calculation CN-CRA-10-31, as its basis. CN-CRA-10-31 was based on, and supplements, the previous AOR identified in calculation CN-CRA-99-80, which was based on the methodology of WCAP-10698, as described in FSAR Section 15.6.3. The evaluation performed in HNP-M/MECH-1049, Revision 1, is a disposition of a single failure not previously considered in calculation CN-CRA-10-31. The evaluation and calculation both follow the NRC approved methodology of WCAP-10698. Although the evaluation performed in HNP-M/MECH-1049 is not a mechanistic code run as are the cases in calculation CN-CRA-10-31, the evaluation presents the expected results should the mechanistic run be performed. The evaluation determines whether various input changes that affect the ruptured SG mass yield a

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		different margin to overfill than the prior calculation in CN-CRA-10-31. The evaluation is based on the previous results of the mechanistic calculation which follows the methodology. Because the only changes being made are inputs and are not elements of the method, these changes are not considered a departure from the method of evaluation described in the FSAR.
02053968/ EC 296136, Revision 0	EC 296136 resolves a non-conforming condition identified by the Westinghouse Nuclear Safety Advisory Letter (NSAL) -14-2, "Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties." In NSAL-14-2, Westinghouse identifies an error in their calculation of mass and energy (M&E) release histories for large-break loss-of-coolant accidents (LOCAs) applicable to the HNP, among other nuclear power plants. Specifically, NSAL-14-2 notes that LOCA M&E analyses are sensitive to the energy stored in the RCS metal mass, which includes the mass of the steam generator (SG) tubes. The Westinghouse M&E analysis for the HNP has historically assumed the SG tubes to be stainless steel. The HNP SG tubes are Alloy 690. As a result of this NSAL, peak post-LOCA containment pressure and temperature at HNP have increased by small amounts in order to compensate for the error discovered by Westinghouse in the M&E analysis. Specifically, the peak post-accident containment pressure for the LOCA double-ended	The pressure and temperature changes for the EC apply to the analysis of a post-LOCA containment atmosphere, after an accident has already occurred. There are no additions, deletions, or modifications to any SSCs as a result of this activity. Therefore, there is no increase in the frequency of occurrence of an accident previously evaluated in the FSAR. The peak containment pressure is increased from 41.8 psig to 42.0 psig. 42.0 psig is less than the design pressure of 45 psig specified in TS 5.2.2 and in HNP-M/MECH-1008. The pressure margin, as shown in FSAR Table 6.2.1-3, decreases from 7.1% to 6.7% at 42.0 psig. 42.0 psig is less than the initial containment pressure used during EST-210, the integrated leak rate test for containment, which pressurizes containment to 44-45 psig. Based upon this, the new maximum calculated post-LOCA containment pressure of 42.0 psig is acceptable and does not represent more than a minimum increase in the likelihood of occurrence of a malfunction of the containment pressure boundary. The new maximum calculated post-LOCA sump and spray pH values remain within the 7.0 to 11.0 range

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	hot-leg break case increases from 41.8 pounds per square inch gauge (psig) to 42.0 psig. This bounds all cases (LOCA and Main Steam Line Break) and is the most- limiting containment pressure case. The maximum LOCA-based containment atmospheric temperature increases from 270.2 degrees Fahrenheit (^{o}F) to 270.4^{o}F. However, this is not the bounding containment temperature case, as the bounding case is based upon the Main Steam Line Break and remains unchanged. EC 296136 incorporates pressure and temperature penalties into the HNP containment analysis and evaluates the impact on plant documents. The peak post-LOCA containment pressure for HNP is identified in HNP TS, Section 6.8.4.k. EC 296136 provides the basis for the change to the peak post-LOCA containment pressure value identified in TS. A TS change is necessary to implement EC 296136. Containment integrated leak rate testing is controlled through Engineering Surveillance Tests (ESTs), which will be revised to reflect the pressure change as a result of the TS Change. Specifically, EST-209, EST-210, EST-212, EST-219, EST-220, EST-221, and EST-222. EPT-221 will be revised. These procedures are used to ensure containment integrity and to ensure that the structure continues to perform its pressure-boundary design function. In each of these procedures, peak accident pressure (P_a) is used as an acceptance criteria for the measured end-of-test pressure or as the minimum pressure to be maintained during testing. The existing P_a value of	specified in Design Basis Document (DBD) -106, TS Bases, Section 3/4.1.2, and the FSAR, Sections 3.11.5.1, 6.1.1.2, and 6.5.2.1.2. The upper pH limit is intended to preclude excessive corrosion of equipment inside containment. Increases of the magnitudes noted, combined with the remaining pH margin (pH remains less than 11.0), indicates that there will be no practical or unacceptable change in the amount of corrosion expected inside containment. Based upon this, the new maximum calculated post-LOCA sump and spray pH values are acceptable and do not represent more than a minimal increase in the likelihood of occurrence of a malfunction of equipment inside containment due to excessive post-LOCA corrosion. HNP Dose Analysis is independent of peak containment pressure and relies instead on the leak rate limit from TS. The dose analysis in HNP-F/NFSA-0072 was not revised for this activity. Also, the containment leak rate assumed in the dose analysis remains bounding since completed integrated leak rate tests and local leak rate tests have used test pressures higher than new analytical limit. The changes to the sump and spray pH profiles are a factor in the calculation of chemical precipitate formation in the recirculation pool. The quantities of these precipitates affect the pressure drop across the strainers and, consequently, core cooling through the RHR Pumps. However, EC 296136 shows that the quantities of precipitates used during strainer testing remain bounding compared to the revised calculated amounts. Therefore, there is no impact to the analyzed strainer pressure drop or core cooling. Based on the above, this activity does

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	41.8 psig will be revised to 42.0 psig. EC 296136 has shown that existing (most recent) test results will still meet the revised criteria. The pressure penalty used to compensate for the M&E error described in NSAL-14-2 is 0.2 psi. When added to the existing containment analysis results, the peak containment pressure increases from 41.8 psig to 42.0 psig. While this remains less than the 45 psig design pressure for the containment structure, the increase represents an adverse effect on a design function described in the FSAR. The post-accident pH analysis for the containment sump is a function of containment pressure. Specifically, containment pressure affects the calculated rates of injection of sodium hydroxide (NaOH) solution from the Containment Spray Additive Tank and borated water from the RWST. When the pressure profile was adjusted for NSAL-14-2, and when a latent non-conservatism in the pH analysis was corrected, the resulting maximum pH values in the sump and in the containment spray system went up slightly. For the sump, maximum pH went from 9.420 to 9.422. For the spray, maximum pH when up from 10.578 to 10.606. Although the final pH values are within the design range of 7.0 to 11.0 from FSAR Section 6.5.2, the change represents an adverse effect on a design function described in the FSAR.	not result in an increase in the consequences of an accident previously evaluated in the FSAR. The increases in calculated post-LOCA containment pressure and sump and spray pH do not have any impact on any SSC failure effects. Therefore, this activity does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety as previously evaluated in the FSAR. This activity does not make any physical changes to the plant. The analyses revised for this activity involve post-LOCA conditions where an accident has already been assumed to occur. Slight increases in post-LOCA containment pressure and sump and spray pH do not result in any new accident types. Therefore, this activity does not create the possibility of an accident different from any previously evaluated in the FSAR, nor does it create the possibility for a malfunction of an important SSC with a result that is different from that previously evaluated in the FSAR. In addition, this activity does not result in a design basis limit for a fission product barrier being exceeded. This activity does not alter the existing containment design pressure of 45 psig. This activity revises the containment analysis in HNP-M/MECH-1008 to note that the NSAL-14-2 pressure and temperature penalties are to be applied to the existing results. This revision is an amendment to the existing analysis to require the manual addition of pressure and

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		temperature penalties; thus there is no reanalysis performed. As a result of the containment analysis changes, a revision is also required to the containment sump and spray pH analysis in calculation 14.06.000-021. The change to this calculation involves an input (containment pressure). The existing analytical method is not modified, only rerun. Based on the above, this activity does not result in a departure from a method of evaluation described in the FSAR that is used to establish a design basis or used in a safety analysis.
02055046/ HNP-F/NFSA-0264, Revision 1, HNP Cycle 21 Loading Pattern and Core Models	The cycle-specific thermal-hydraulic analysis results for the HNP Cycle 21 reload core show that for the rod ejection accident documented in the FSAR under Section 15.4.8, there is a change from no fuel assemblies failing to all the rods in one fuel assembly having failed cladding as a result of a departure from nucleate boiling (DNB). This is an American Nuclear Society (ANS) Condition IV accident and the cycle-specific results are within the fuel failure assumptions specified by the dose analysis.	The activity does not modify or remove any SSC other than fuel. The results of an accident analysis do not affect the frequency of its occurrence. Therefore, this activity does not affect the frequency any accident. This change to the rod ejection accident remains less than the number of failed fuel assemblies evaluated in the dose analysis documented in the FSAR. This activity does not involve change to the RCS pressure boundary, fuel, or any other SSC which would affect the likelihood of a rod ejection. Further, this activity does not modify, add, or remove any SSC (other than fuel) nor change how SSCs are used during normal operation or to mitigate an accident. Therefore, this activity does not result in more than a minimal increase in the likelihood of a malfunction of an SSC important to safety. Since the dose analysis assumed cladding failures bound the estimated failures from the safety analysis, the

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		predicted dose consequences remain the same. Therefore, the proposed activity does not result in a more than minimal increase in the consequences of an accident, nor does it result in a more than minimal increase in the consequences of a malfunction of an SSC. A change to analysis results in FSAR, Section 15.4.8, does not constitute a new or different type of accident. Thus, the proposed activity does not create the possibility of an accident of a different type. The change to analysis results in FSAR 15.4.8 does not constitute a new malfunction. Thus, the proposed activity does not create the possibility for a malfunction of an SSC with a different result. The change from no fuel assemblies failing to all the rods in one fuel assembly having failed cladding as a result of DNB in the rod ejection accident involves an ANS Condition IV event where fuel failures are allowed. Evaluations have been performed as necessary to ensure the fission product barrier (fuel cladding, RCS boundary, and containment) limits are not compromised except where allowed in the design basis accident dose analyses. In addition, this activity does not represent a departure in a method of evaluation.
02079940/ EC 402237, Revision 0	NRC Bulletin 2012-01, "Design Vulnerability In Electric Power System," dated July 27, 2012, requires licensees to install open-phase protection on station transformers supplying offsite power to essential plant safety equipment. An open-phase	Installation of the OPP system was evaluated under 10 CFR 50.59 in accordance with guidance provided in Nuclear Energy Institute (NEI) 96-07, Revision 1, Guidelines for 10 CFR 50.59 Implementation, and NEI 01-01 (EPRI TR-102348, Revision 1), Guideline on

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	condition (OPC) present in the offsite power system may not be detectable on the low side of the startup transformer (SUT) when the transformer load is not large enough to cause the emergency bus voltage to drop below the under-voltage protective relay setting. The transformers at HNP that may be susceptible to an OPC are limited to the startup transformers, SUT-A and SUT-B, as they are the primary transformers used to supply offsite power to the 6.9 kilovolt (kV) essential buses. A previous EC 296261 installed a digital-based Open Phase Protection (OPP) system on the high-voltage side of SUT-1A and SUT-1B. The OPP system installed under EC 296261 provides OPC monitoring only and is not capable of locking-out (i.e., tripping) a SUT. The intent of the new OPP system is to enhance protection of the Class 1E (safety-related) power system from a potential degraded condition caused by an OPC that could adversely affect both Class 1E and non-Class 1E systems. The OPP system consists of four separate cabinets per SUT with each cabinet housing one of four separate OPC sensing/trip channels. Of the four channels per SUT, two channels employ one type of controller platform and central processing unit (CPU) architecture while the other two channels employ a different type of controller platform and CPU architecture. A SUT lockout command is generated when any two of the four channels detects an OPC. Thus, satisfying the SUT trip logic when a valid OPC	Licensing Digital Upgrades. The OPP system provides SUT protection similar to the existing SUT protective relays. Inadvertent operation of the existing SUT protective relays can lead to lockout of a SUT resulting in partial or complete loss of non-emergency AC power and subsequently partial or complete loss of forced reactor coolant flow. Inadvertent operation of the new OPP system can also result in lockout of a SUT, but to no more extent than the existing SUT protective relaying. The OPP system has been subjected to analyses, tests, and requirements typically applied to equipment used in safety related applications (demonstrating a high quality threshold). Additionally, the OPP system employs two-out-of-four coincidence trip logic, which provides added reliability and further assures that a valid trip signal will be processed while an invalid trip signal will be disregarded. Existing SUT protective relaying employs one-out-of-one coincidence trip logic, which is less reliable. With the existing protection scheme, failure (malfunction) of a single protective relay could result in SUT lockout while a single failure within the OPP system will not result in a SUT lockout. Trip setpoints for the new OPP system were established to maintain coordination with other protective relay schemes and to accommodate normal plant operation (such as equipment starts and stops) to ensure spurious actuation of the OPP system does not occur. The setpoint values will be monitored and validated during the

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	is detected does not rely on a single type of controller platform or CPU architecture. This activity (EC 402237) will physically connect the OPP system installed under EC 296261 to the corresponding SUT lockout relay.	OPP system monitoring phase of operation. Based on the robust design, considerable testing, and analyses performed on the OPP system to be installed as part of this activity, it can be reasonably concluded that the quality and reliability of the OPP system is at least as good as the existing SUT protective relays. Therefore, implementation of the proposed activity will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR, nor will it result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR. The new OPP system will not increase operator burden or place constraints on an operator's ability to adequately respond to an accident. The initial accident assessments contained in the FSAR remain valid and unchanged as a result of the implementing activity. The new equipment installed by this activity will have no adverse impact on its installed environment or another plant SSC. Therefore, the proposed activity will not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR, nor will this activity result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR. Failure of the new OPP system can result in loss of SUT, but to no more extent than failure of an existing SUT protective relay. Since only the SUTs are affected by the

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		proposed activity, and the types of accidents resulting from a loss of SUT have already been analyzed in the safety analysis, the proposed activity cannot create the possibility for an accident of a different type than previously evaluated in the FSAR. No new outcomes have been introduced and the proposed activity to provide SUT open-phase protection cannot create the possibility for a malfunction with a different result than previously evaluated in the FSAR. The proposed activity will not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered. The proposed activity does not involve a change to any element of the analytical methods described in the FSAR used to demonstrate the design meets the design basis or that the safety analysis is acceptable, nor does this change involve use of a method or evaluation not already approved by the NRC. Therefore, the proposed activity will not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.
02093286/ EC 296136, Revision 1	This evaluation is a revision to an evaluation completed under Log Number 02053968, which is described in this Enclosure. The revision adds EPT-222 to the list of impacted procedures. The initial issuance of this procedure occurred during EC 296136, Revision 0, development and was not identified in the first revision of the EC. EC 296136, Revision 1, includes EPT-222 as an impacted procedure.	There is no impact to the 10 CFR 50.59 evaluation conclusions presented under Log Number 02053968 as a result of the revised evaluation, which is described in this Enclosure.

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02092509/ EC 284243, Revision 0	This evaluation is prepared for EC 284243, HNP Turbine Control System Upgrade (TCSU) Integration. The existing Westinghouse Digital Electro-Hydraulic Control (DEH) system for the turbine-generator is being replaced with an Invensys Triconex Digital Electro-Hydraulic Turbine Control System (TCS) utilizing triple modular redundant (TMR) digital controllers, redundant input sensors and output actuators to control and protect the turbine. The controls and electro-hydraulic interface include stand-alone, fault-tolerant, and online maintainable trip block assemblies that will hydraulically trip the turbine on overspeed conditions sensed by either the Turbine Controller or the diverse Secondary Overspeed Protection System (SOPS) system, or will act to slow down the turbine speed by closing the control valves during certain scenarios (load rejection). This modification is being implemented to improve plant reliability. The existing DEH control system reflects a relatively old design provided by Westinghouse. In addition to obsolescence issues with the existing system, a number of single failure points exist since fault tolerance was not a significant consideration during its development. The new TCS design provides a state-of-the-art, fault-tolerant control system. The new TCS Triconex network is composed of the Turbine and Valve Control System (TVCS), Turbine Protection System (TPS), SOPS, Human-System	The new digital TCS was evaluated under 10 CFR 50.59 in accordance with guidance provided in NEI 96-07, Revision 1, and NEI 01-01 (EPRI TR-102348, Revision 1). The new digital TCS performs the same turbine speed and load control functions and interfaces with the same components and systems as the existing analog DEH system. The new TCS design assures that a single component failure within the system will not result in a loss of steam or load control, or prevent a valid trip response. A Failure Modes and Effects Analysis (FMEA) was conducted, which concludes that the TCS contains no single points of vulnerability and that there is no single failure, which on its own, could result in a turbine trip. Therefore, this activity does not result in more than a minimal increase in the frequency of occurrence of any accident previously evaluated in the FSAR. The existing turbine mechanical and electrical hydraulic trip components are replaced in the new TCS design with equipment of equal or greater reliability, and will be controlled through redundant components and single failure proof voting logic. The new system improves the reliability of the entire TCS system, will not result in a system-level failure, and either will not affect or will reduce the likelihood of occurrence of a malfunction as postulated in the FSAR. The consequences of a failure of the new TCS are bounded by the consequences of a failure of the existing TCS. Failure of the new TCS could result in an increase or decrease in heat removal from the secondary system, but no more than the failure of the existing TCS. Thus,

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	Interface (HSI) cabinet, Distributed Control System (DCS) equipment, Engineering Workstation (EWS), and Main Control Room (MCR) Operator HSIs. The TVCS is a subsystem of the TCS that includes all critical control functions and turbine protection: speed and load control, turbine protection trips (other than diverse SOPS and any other trips performed by the TPS that are external to the TVCS) and valve management. The TPS is a subsystem of the TCS that includes the hydraulic trip functions and the diverse and independent SOPS. The SOPS is a digital trip system that replaces the mechanical overspeed trip system and is diverse and independent from the other control and protective features of the TCS. Both overspeed trip systems rely on independent triple speed sensing inputs and voting logic, including sensor health monitoring and fault notification alarms and warnings. Both systems will trip the turbine on loss of or diverging speed signals. The design of the new turbine control provides the plant operators with a better graphical interface on a common set of monitors using a trackpad, keypad, and pointing devices rather than discrete switches and indication. The TCS upgrade includes the following functional differences that are conservatively treated as adverse: (1) The change from functionally diverse mechanical and electrical overspeed turbine trip mechanisms to redundant and electrically diverse overspeed trip	replacement of the TCS will not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR. The new TCS also does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR. This activity does not introduce any components with new failure modes and effects that are not bounded by the accidents evaluated in the FSAR. This activity does not alter the failure modes and effects of the existing components. The overspeed trip portion of the new system remains independent from the control portion of the system. System-level failure modes for the equipment are immediate or result in initiation of a turbine trip. The FMEA for the new TCS also concludes that at the system level the most severe effects of failures are a turbine trip and loss of some functionality at the HSI. No new failure modes are introduced by replacement of the existing equipment with the TCS, and this replacement makes no change to the most limiting scenario of the turbine trip previously evaluated in the FSAR. Therefore, the new TCS does not create the possibility of an accident of a different type than previously evaluated in the FSAR. The TCS upgrade does not introduce any new failure or operating modes, functions, interfaces, or operating parameters that would create a possibility of a malfunction for any SSC important to safety with a different result than any previously analyzed. Thus, the failure effects of the new digital TCS are consistent with

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	mechanisms. (2) Conversion from hard controls to soft controls because it involves more than minimal differences in the HSI.	the failure effects of the existing DEH system and the results of these malfunctions are the same as previously evaluated in the FSAR. Therefore, the new TCS does not create the possibility of a malfunction for any SSC important to safety with a different result than any previously analyzed in the FSAR. The new TCS also does not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered. The proposed activity does not necessitate revision nor replacement of any evaluation methodology used in establishing any design basis or in the safety analysis. No new methods of evaluation are required to assess the new equipment installed as part of this activity. No alternative or new methods of evaluation are required or employed for this activity. Therefore, the TCS upgrade does not impact existing evaluation methodology used in establishing any design basis or in the safety analysis.
02100628/ EC 298102, Revision 1	This Evaluation addresses a revised post-LOCA Emergency Core Cooling System (ECCS) RWST backleakage dose assessment based upon EC 298102, Revision 1. The implementation of EC 298102, Revision 1, involves revising or replacing certain evaluation methodologies described in the FSAR, which were used in the design basis post-LOCA ECCS RWST backleakage dose analysis. This methodology change required a 10 CFR 50.59 evaluation; no other aspect of this activity required 10 CFR 50.59 evaluation. The methodology changes are:	From engineering evaluation, the RADTRAD-NAI code implements the same methods as TITAN5 and generates essentially the same results as TITAN5 for the RWST dose component of the total LOCA dose. From engineering evaluation, the IODEX-NAI code implements the same NUREG/CR-5950 methods as the Duke IODEX code, generates the same results as IODEX for the RWST iodine releases, and there are no restraints or restrictions imposed on IODEX’s use for iodine release from review of NRC safety evaluation considerations. Based on criteria established by NEI 96-

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	(1) The Westinghouse TITAN5 dose analysis computer code results for the RWST component of total LOCA dose will be replaced with RADTRAD-NAI dose analysis computer code results. (2) The Westinghouse approximations for iodine gas conversion and iodine gas partitioning between the RWST liquid inventory and the RWST air inventory will be replaced with IODEX-NAI computer code calculations. This code is derived from Duke developed IODEX, which has been used for NRC approved applications at other Duke nuclear sites.	07, Revision 1, it was concluded that the use of both the RADTRAD-NAI code and the IODEX-NAI codes as alternate methodologies is acceptable to implement without prior NRC review and approval. The EC 298102, Revision 1, update to the AOR dose assessment demonstrates that the ECCS backleakage allowable value, considering both onsite and offsite dose criteria, can be increased from its current evaluation basis limit of 3.0 gallons per minute (gpm). The change in the ECCS backleakage allowable value is evaluated under Log Number 02128760, which is described in this Enclosure. The revised post-LOCA ECCS RWST backleakage dose assessment ensures that the allowable backleakage flows to the RWST remain below the dose consequence results in the FSAR, Table 15.6.5-16.
02118552/ EC 405128, Revision 0	This evaluation addresses the Security Information and Event Manager (SIEM), which will collect logging data through the existing Plant Process Network (PNET) infrastructure. It was determined that the SIEM has the potential to fail or malfunction in a manner that results in a multicast/broadcast data transmission (data storm) which could adversely affect the reliability of PNET and interfacing SSCs, such as the Emergency Response Facility Information System (ERFIS) and Leading Edge Flowmeter (LEFM). Therefore, the scope of this evaluation is limited to SSCs with FSAR described design functions that depend on PNET because only those functions were identified to be adversely	The SIEM was evaluated under 10 CFR 50.59 in accordance with guidance provided in NEI 96-07, Revision 1, and NEI 01-01 (EPRI TR-102348, Revision 1). The SIEM performs a monitoring function through a network that performs no control functions and cannot initiate any plant transients or FSAR-described accidents. Therefore, the proposed activity will result in no increase in the frequency of occurrence of any accident previously evaluated in the FSAR. The SIEM is qualitatively determined to be at least as dependable as the SSCs to which it is connected. The failure modes of the SIEM, and the likelihood of malfunction, are indistinguishable from those of the

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	effected by EC 405128.	existing equipment. Since there is no clear trend toward increasing the likelihood of failure, the proposed change is considered to have a negligible effect on the likelihood of malfunction. As a result, there is no credible malfunction of the SIEM that can increase the dose consequences of any FSAR-described accident. Based on the above, the proposed activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR. In addition, there is no credible malfunction of the SIEM that can increase the dose consequences of the malfunction of any SSC. Based on the above, the proposed activity does not result in more than a minimal increase in the consequences of malfunction of an SSC important to safety. The SIEM does not have its own computing network; rather, it uses the same network as the components that it is monitoring, to collect the log data. A FMEA was performed for the SIEM. It concludes that there are no new failure modes or failure modes with a different result. Therefore, the proposed activity does not create a possibility for an accident of a different type than previously evaluated in the FSAR, nor does it create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR. The proposed activity does not directly or indirectly involve the fuel, the RCS pressure boundary, the containment, or any of the design basis limits associated with these fission product barriers. Consequently, the

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		activity cannot result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered. The proposed activity neither involves a change to any element of the analytical methods described in the FSAR used to demonstrate the design meets the design bases or that the safety analyses are acceptable, nor involves use of a method or evaluation not already approved by the NRC. Therefore, the proposed activity will not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.
02128760/ EC 298102, Revision 1	This evaluation addresses a revised post-LOCA recirculation sump water level decrease due to increasing the allowable backleakage to the RWST during post-LOCA recirculation. For above RWST water line backleakage, allowable seat leakage has been increased for the Containment Spray (CT) system and Charging/Safety Injection System (CS) boundary isolation valves. This increase in allowable seat leakage supports initiation of Category A seat leakage testing of the subject valves. For below RWST water line backleakage, the allowable value has also been increased to support future leakage assessments. No change is being made to the allowed ECCS leakage within the Reactor Auxiliary Building (RAB). The minimum sump water level calculation, SD-0022, previously assumed an ECCS leakage rate back to the RWST of 520 cubic centimeters per hour (cc/hr) or 0.0023 gpm. This has been increased to 17.30	The proposed activity addresses the outcome of accidents previously evaluated in FSAR, Chapter 15. The relevant accidents include LOCAs. After the water level of the RWST reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching from the injection mode to the cold leg recirculation mode of operation in which spilled borated water is drawn from the containment sump by the low head safety injection (RHR) pump and returned to the RCS cold legs. The CT System continues to operate to further reduce containment pressure. It is during the recirculation mode of operation where the activities associated with EC 298102, Revision 1, are applicable. The potential sump water level decrease due to increasing the allowable backleakage to the RWST during post-LOCA recirculation occurs during accident mitigation (not initiation). Therefore, these changes are limited to accident mitigation (not initiation) and do not increase the frequency of occurrence of an accident previously evaluated in the FSAR. Component

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	gpm, and the previously assumed ECCS leakage to the RAB of 5643 cc/hr (0.025 gpm) has been increased to 1.0 gpm, as documented in SD-0022, Revision 2. SD-0022, Revision 1, concluded that after 30 days of ECCS leakage at 0.027 gpm, the recirculation sump water level remained above the recirculation sump strainer ECCS strainer vortex suppressor. With the newly established allowable ECCS leakage of 18.30 gpm, sump water level will drop below the vortex suppressor before 30 days (31.9 hours is the minimum duration based upon Calculation SD-0022, Revision 2). This requires re-injection from the RWST to the containment sump (using the water that had leaked back to the RWST) using emergency operations procedure guidance, to ensure the vortex suppressor remains covered. EC 298102, Revision 1, may require periodic re-injection from the RWST to the recirculation sump during recirculation to accommodate higher allowed ECCS leakage out of containment. As determined in Calculation SD-0022, Revision 2, the minimum time following start of recirculation, until re-injection is needed ranges from approximately 31 to 41 hours depending on break size. If re-injection is not initiated, sump water level would eventually decrease to the point where vortexing and/or incomplete submergence of the ECCS strainers would result. Emergency operations procedure guidance for the transfer to cold leg recirculation, EOP-ES-1.3, currently supports re-injection from the RWST to the	manipulations needed to support re- injection would not increase the likelihood of a malfunction of any components. As shown in the FSAR, Table 15.6.5-2, the safety analysis for a large break LOCA is terminated at 829.7 seconds (0.23 hours). As shown in the FSAR, Figure 15.6.5-32, the safety analysis for a small break LOCA is terminated at 6000 seconds (1.7 hours). The earliest potential need for re-injection from the RWST to the recirculation sump is 31.9 hours. This is well past termination of LOCA accident analyses and therefore there is no impact to accident analyses. Since accident analyses remain valid, there can be no increase in accident dose consequences. The dose consequences resulting from increased RWST backleakage allowed under EC 298102, Revision 1, remain bounded by the dose consequences identified in the FSAR, Table 15.6.5-16. The revision to EOP-ES-1.3, to utilize a CT pump to re-inject from the RWST to the recirculation sump to maintain sump inventory in the event of significant backleakage to the RWST, ensures that adequate sump level is maintained to support ECCS and CT pump operation to mitigate the accident. In the event of failure of a CT train (pump, valve, etc.), the redundant CT train would be available to support re-injection, should it be required due to significant RWST backleakage. Containment sump level is provided with redundant safety-related instrumentation, so the redundant instrument can be used by control room personnel to

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	recirculation sump using a CSIP and is being revised to support re-injection using a CT pump as the preferred method. However, this is an adverse change (i.e., method used to perform post-LOCA recirculation design function is adversely affected) from the current post-LOCA leakage assessment in SD-0022, Revision 1, where no re-injection is shown to be necessary for the 30-day LOCA. FSAR, Section 6.3.2.8, will be revised to acknowledge that re-injection from the RWST may be required following switchover to sump recirculation, to compensate for significant ECCS leakage outside containment in order to maintain adequate sump inventory.	monitor sump level and assess the need for re-injection. Equipment malfunctions will therefore not increase accident doses since the ability to maintain long-term core and containment cooling is assured by maintaining adequate sump inventory. The changes being addressed in this evaluation are associated with LOCA consequence mitigation. So the changes are applicable only after the accident has occurred. There is no credible mechanism for another accident to occur following a LOCA. Therefore, the proposed activity does not create an accident of a different type than previously evaluated. The CT system is designed for single failure. In the event of a failure that renders one train non-functional, the other train is capable of providing adequate injection flow from the RWST or recirculation flow from the sump. Re-alignment of a CT pump from recirculation back to injection (if needed to compensate for significant RWST backleakage), does not change this capability. The consequences of failure of a CT train are not changed by incorporating the ability to swap back to RWST injection if required. The remaining train remains adequate to perform its design basis function of containment cooling to ensure that containment integrity is maintained as assumed in the dose calculation. The remaining train is also available to be realigned for RWST re-injection if necessary. Therefore, the proposed change does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR.

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		The changes being addressed in this evaluation are associated with LOCA consequence mitigation. So the changes are applicable only after the LOCA has occurred, at which point, two of the three fission product barriers (RCS and fuel cladding) are assumed to have been breached. The third barrier, containment, remains intact, experiencing only design basis leakage. The changes associated with this evaluation will not affect the ability of ECCS and CT to remove post-accident decay heat from containment. Therefore, containment pressure will remain within design limits. Thus, these changes do not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered. In addition, the proposed activity does not result in departure form a method of evaluation described in the FSAR.
02153055/ EC 409889, Revision 0	EC 409889 implements changes to the PNET configuration such that some of the PNET devices will be reconfigured to separate the primary ERFIS related devices, including the Multiplexor Fiber Ring, from the remaining portion of the PNET using a Firewall and Intrusion Detection System. Additionally, one of the ERFIS workstations has additional functionality as a "QNX" workstation used to manage the Waste Processing Building computer.	The activity has been evaluated under 10 CFR 50.59 in accordance with guidance provided in NEI 96-07, Revision 1, and NEI 01-01 (EPRI TR-102348, Revision 1). ERFIS provides monitoring, alarming, displaying, reporting and archiving capabilities to the Control Room operators, the Technical Support Center and the Emergency Operations Facility through a network and performs no control functions. ERFIS is not an initiator of any FSAR-described accidents. Therefore, the proposed activity will not result in an increase in the frequency of occurrence of any accident previously evaluated in the FSAR. The failure modes of ERFIS, and the likelihood of malfunction, are indistinguishable from those of the existing equipment. Since there is no clear trend toward increasing the likelihood of malfunction, the proposed

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		change is considered to have a negligible effect on the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR. ERFIS is not credited for mitigating the consequences of an accident. The changes to ERFIS per EC 409889 are not visible to the end user. Post implementation of EC 409888, ERFIS will still not be credited for mitigating the consequences of an accident. Therefore, the proposed activity has no impact on the consequences of an accident previously evaluated in the FSAR. ERFIS is not credited for mitigating the consequences of an accident. Based on the above, the proposed activity does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR. A FMEA was performed for the ERFIS related network changes under EC 409889. It concludes that there are no new failure modes and existing failure modes are not accident initiators. Consequently, the proposed activity does not create a possibility for an accident of a different type than previously evaluated in the FSAR and there is no possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR. The proposed activity does not directly or indirectly involve the fuel, the RCS pressure boundary, the containment, or any of the design basis limits associated with these fission product barriers. Consequently, the activity cannot result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered. The proposed activity neither involves a change to any element of the analytical methods described in the FSAR used to demonstrate the

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		design meets the design bases or that the safety analyses are acceptable, nor involves use of a method or evaluation not already approved by the NRC. Therefore, the proposed activity will not result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.
02183576/ Revision to FSAR, Section 15.8	The proposed activity is the revision to the FSAR Section 15.8, Anticipated Transients without Scram (ATWS), as a result of a revision to the respective underlying calculation. Revision 0 of the Westinghouse ATWS analysis predicted the SG low-low water level ATWS Mitigation System Actuation Circuitry (AMSAC) setpoint utilizing SG mass. This may not have adequately addressed the transient behavior for HNP, as identified in NSAL-11-01, “Calculation of the Steam Generator Mass for the Low-Low Water Level Setpoint in LOFTRAN Analyses.” Based upon NSAL-11-01, the trip mass at the low-low SG level setpoint needs to be reanalyzed. Thus, the ATWS analysis has been recalculated by Westinghouse to accommodate the issue discussed above for HNP. From NEI 96-07, Revision 1, this activity required a 10 CFR 50.59 evaluation because the change to the peak RCS pressure in the reanalysis for the Loss of Normal Feedwater ATWS adversely affects the design function of the RCS, as described in the FSAR. One of the design functions of the RCS is to provide a second barrier against fission product release in the event of fuel cladding failure. Another	The changes that require a revision to the FSAR analysis are a result of the response to NSAL-11-01 to address potentially inadequate prediction of the SG low-low water level ATWS AMSAC setpoint utilizing SG mass. The changes are best characterized as changes in input to the ATWS analysis. It is not possible to characterize the proposed activity as an accident initiator. There are also no SSCs involved in this analysis related activity. Thus, no SSCs could initiate an accident. Without changing the frequency of occurrence of any accident initiators, no change in the classification of the accidents can occur. Since there is no impact on the frequency of occurrence of an accident, it can be concluded that the proposed activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR. The revision to the FSAR ATWS analysis for peak RCS pressure is not related to the performance of SSCs. The proposed activity does not affect any aspect of SSC design, including changing material or construction standards of SSCs. The proposed activity is solely focused on updating the FSAR analysis to reflect changes and the associated impacts on the ATWS accident peak RCS pressure analysis result. Results are

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	design function of the RCS is to help ensure that coolant (water) is available to remove heat from the fuel. The peak RCS pressure must not exceed the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Service Level C, stress limit criterion of 3,215 pounds per square inch absolute or 3,200 psig. In order for the RCS to perform its design function, RCS pressure needs to stay within the acceptance criteria.	still within the design limit, which does not change. Therefore, it can be concluded the proposed activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the FSAR. The net effect of this change was an increase in the peak RCS pressure. DNB is not a concern for the ATWS analysis. Thus, there is no fuel failure associated with the ATWS analysis. Therefore, there will be no impact on dose analyses. Also, since the RCS pressure increase does not exceed the ASME B&PV Code, Service Level C stress criterion of 3,200 psig, the RCS would not have been breached and any potential fuel failures would not have been released from the RCS. Thus, the proposed activity will have no impact on any dose analyses of record. Factors which influence dose calculations and environmental consequences remain consistent with the FSAR analyses. Therefore, the proposed activity does not result in more than a minimal increase in the consequences of an accident evaluated in the FSAR. There is no fuel failure associated with the ATWS analysis; however, even if fuel failure was present, the RCS pressure increase will not cause a breach to the RCS. Thus, any potential fuel failure will be contained in the RCS. Therefore, the proposed activity will have no impact on any dose analyses of record and the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR will not be increased. The proposed activity does not create a possibility for an accident of a different type than

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		previously evaluated in the FSAR, does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR, does not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered, and does not result in a departure from a method of evaluation described in the FSAR.
02189627/ HNP-F/NFSA-0284, Revision 0, HNP Cycle 22 Loading Pattern and Core Models	The HNP Cycle 22 reload core analysis results show that for the single control rod withdrawal accident, documented in the FSAR under Section 15.4.3, there is a change such that the predicted minimum departure from nucleate boiling ratio (MDNBR) is less than the 95/95 safety limit. For HNP Cycle 22, less than 4% of the fuel is predicted to fail based on the DNB criteria whereas previously for HNP Cycle 21, no fuel was predicted to fail because the MDNBR was greater than the 95/95 safety limit. The single control rod withdrawal accident is an ANS Condition III accident and the cycle-specific results are within the fuel failure assumptions specified by the dose analysis.	The activity does not modify or remove any SSC other than fuel. The results of an accident analysis do not affect the frequency of its occurrence. Therefore, this activity does not affect the frequency any accident. This change to the single control rod withdrawal accident remains less than the number of failed fuel assemblies evaluated in the dose analysis documented in the FSAR. This activity does not involve change to the RCS pressure boundary, fuel, or any other SSC which would affect the likelihood of a single control rod withdrawal. Further, this activity does not modify, add, or remove any SSC (other than fuel) nor change how SSCs are used during normal operation or to mitigate an accident. Therefore, this activity does not result in more than a minimal increase in the likelihood of a malfunction of an SSC important to safety. Since the dose analysis assumed cladding failures bound the estimated failures from the safety analysis, the predicted dose consequences remain the same. Therefore, the proposed activity does not result in a more than minimal increase in the consequences of an accident, nor does it result in a more than minimal

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		increase in the consequences of a malfunction of an SSC. A change to analysis results in FSAR, Section 15.4.3, does not constitute a new or different type of accident. Thus, the proposed activity does not create the possibility of an accident of a different type. The change to analysis results in FSAR 15.4.3 does not constitute a new malfunction. Thus, the proposed activity does not create the possibility for a malfunction of an SSC with a different result. The change from no fuel assemblies failing to one assembly exceeding the DNB cladding failure criteria for the single rod withdrawal accident involves an ANS Condition III event where a small fraction of fuel rod failures are acceptable. Evaluations have been performed as necessary to ensure the fission product barrier (fuel cladding, RCS boundary, and containment) limits are not compromised except where assumed in the design basis accident dose analyses. In addition, this activity does not represent a departure in a method of evaluation.