Southern Nuclear

Regulatory Affairs

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U. S. Nuclear Regulatory Commission
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Vogtle Electric Generating Plant - Units 1 & 2
Revision 24 to the Updated Final Safety Analysis Report, Technical Specification Bases Changes, Technical Requirements Manual Changes, 10 CFR 50.59 Summary Report, and Revised NRC Commitments Report

Ladies and Gentlemen:

In accordance with 10 CFR 50.4(b) and 50.71(e), Southern Nuclear Operating Company (SNC) hereby submits Revision 24 to the Vogtle Electric Generating Plant (VEGP) Units 1 and 2 Updated Final Safety Analysis Report (UFSAR). The revised VEGP Units 1 and 2 UFSAR pages, indicated as Revision 24, reflect changes through October 31, 2022.

The VEGP Units 1 and 2 Technical Specifications, Section 5.5.14, "Technical Specifications (TS) Bases Control Program," provides for changes to the Bases without prior NRC approval. In addition, TS Section 5.5.14 requires that Bases changes made without prior NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Pursuant to TS 5.5.14, SNC hereby submits a complete copy of the VEGP TS Bases. The revised VEGP TS Bases pages, indicated as Revision 82, reflect changes to the TS Bases through October 31, 2022.

In accordance with Regulatory Issue Summary (RIS) 2001-05, "Guidance on Submitting Documents to the NRC by Electronic Information Exchange or on CD-ROM," all of the current pages of the VEGP Units 1 and 2 UFSAR, the VEGP Units 1 and 2 UFSAR reference drawings, the TS Bases, and the Technical Requirements Manual (TRM) are being submitted on CD-ROM in portable document format (PDF). The revised VEGP Units 1 and 2 TRM pages, indicated as Revision 62, reflect changes to the TRM through October 31, 2022.

In accordance with 10 CFR 50.59(d)(2), SNC hereby submits the 10 CFR 50.59 Summary Report containing a brief description of any changes, tests, or experiments, including a summary of the safety evaluation of each. This report is based on the same time period as Revision 24 of the UFSAR.

In accordance with NEI 99-04, "Guidelines for Managing NRC Commitment Changes," Revision 0, SNC reviewed its Commitment Database and identified no commitment changes for the applicable reporting period (March 1, 2021 to October 31, 2022).

Enclosure 1 provides a table of contents with associated file names for the CD-ROM (Enclosure 2). Enclosure 3 provides the 10 CFR 50.59 Summary Report.

A053
NRR

This letter contains no NRC commitments. If you have any questions, please contact Amy Chamberlain at (205) 992-6361.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 31st day of October 2022.

Respectfully submitted,

Cheryl Gayheart
Regulatory Affairs Director

CAG/kgl/cg

Enclosures:
1. CD-ROM Table of Contents
2. CD-ROM (1 disc) containing Files 001 – 029
3. 10 CFR 50.59 Summary Report

cc: Regional Administrator, Region II (w/o enclosures)
Senior NRR Project Manager – VEGP Units 1 and 2 (w/o enclosures)
Senior Resident Inspector – VEGP Units 1 and 2 (w/o enclosures)
RType: CVC7000

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**Enclosure 1**
**CD-ROM Table of Contents**

| SEQ | CONTENT | FILENAME EXTENSION |
|---|---|---|
| | **DISC 1** | |
| | VEGP-File Nomenclature | .pdf |
| 001 | VEGP FSAR_CH1, CH2 (PRT 1)<br>Chapter 1<br>Chapter 2 (PRT 1)<br>Section 2.1 to 2.3 to<br>Figure 2.3.5-4 | .pdf |
| 002 | VEGP FSAR_CH2 (PRT 2) & CH3 (PRT 1)<br>Chapter 2 (PRT 2)<br>Figure 2.3.5-5 to 2.3.5-6<br>Section 2.4<br>Table 2.4.1-1 to Table 2.4.13-1<br>Figure 2.4.1-1 to Figure 2.4.11-6<br>Section 2.5<br>Appendix 2A<br>Appendix 2B<br>Chapter 3 (PRT 1)<br>Section 3.1 to 3.6<br>Section 3.7 to Figure 3.7.B.2-24 | .pdf |
| 003 | VEGP FSAR_CH3 (PRT 2)<br>Chapter 3 (PRT 2)<br>Figures 3.7.B.2-25 to 3.7.4-1<br>Section 3.8 to 3.11<br>Appendix 3A to 3C<br>Appendix 3D to Figure 3D-17 | .pdf |
| 004 | VEGP FSAR_CH3 (PRT 3)<br>Chapter 3 (PRT 3)<br>Appendix 3D Figures 3D-18 to 3D-40 | .pdf |
| 005 | VEGP FSAR_CH3 (PRT 4)<br>Chapter 3 (PRT 4)<br>Appendix 3D Figures 3D-41 to 3D-65 | .pdf |
| 006 | VEGP FSAR_CH3 (PRT 5)<br>Chapter 3 (PRT 5)<br>Appendix 3D Figure 3D-66 to 3D-92 | .pdf |
| 007 | VEGP FSAR_CH 3 (PRT 6)<br>Chapter 3 (PRT 6)<br>Appendix 3D-93 to 3D-136 | .pdf |

| SEQ | FILENAME CONTENT | EXTENSION |
|---|---|---|
| 008 | VEGP FSAR_CH 3 (PRT 7) & CH 4 (PRT 1)<br>Appendix 3D-137 to 3D-148<br>Appendix 3E<br>Appendix 3F<br>Chapter 4 (PRT 1)<br>Section 4.1 to 4.3 | .pdf |
| 009 | VEGP FSAR_CH 4 (PRT 2), CH5, & CH6 (PRT 1)<br>Chapter 4 (PRT 2)<br>Tables 4.3-1 to 4.3-12<br>Figures 4.3-1 to 4.3-57<br>Section 4.4 to 4.6<br>Appendix 4A<br>Chapter 5<br>Chapter 6 (PRT 1)<br>Section 6.1<br>Section 6.2 to Figure 6.2.1-15(SH 14) | .pdf |
| 010 | VEGP FSAR_CH6 (PRT 2)<br>Chapter 6 (PRT 2)<br>Figure 6.2.1-15 (SH 15) to 6.2-21 (SH 74) | .pdf |
| 011 | VEGP FSAR_CH6 (PRT 3) & CH7 (PRT 1)<br>Chapter 6 (PRT 3)<br>Figure 6.2.1-22 (SH 1) to 6.2.5-7<br>Section 6.3 to 6.6<br>Appendix 6A<br>Chapter 7 (PRT 1)<br>Section 7.1 to 7.5<br>Section 7.6 to Figure 7.6.4-1 | .pdf |
| 012 | VEGP FSAR_CH7 (PRT 2), CH8 (PRT 1)<br>Chapter 7 (PRT 2)<br>Figure 7.6.5-1<br>Section 7.7 to Figure 7.7.2-2<br>Chapter 8 (PRT 1)<br>Section 8.1 to 8.2<br>Section 8.3 to Figure 8.3.1-1 (SH 4) | .pdf |
| 013 | VEGP FSAR_CH8 (PRT 2)<br>Chapter 8 (PRT 2)<br>Figure 8.3.1-1 (SHs 5 thru 24) | .pdf |

| SEQ | FILENAME CONTENT | EXTENSION |
|---|---|---|
| 014 | VEGP FSAR_CH 8 (PRT 3), CH 9, CH 10 (PRT 1) | .pdf |
| | Chapter 8 (PRT 3) | |
| | Figure 8.3.1-1 (SHs 25 thru 35) | |
| | Section 8.4 | |
| | Chapter 9 | |
| | Chapter 10 (PRT 1) | |
| | Section 10. 1 | |
| | Section 10.2 to Figure 10.2.2-2b | |
| 015 | VEGP FSAR CH10 (PRT2), CH11, CH12, CH13, CH14, CH 15 (PRT 1) | .pdf |
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| | Chapter 11 | |
| | Chapter 12 | |
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| | Chapter 15 to Figure 15.1.5-11 | |
| 016 | VEGP FSAR CH15 (PRT2), CH16, CH17, CH18, CH19, TOC, REV 24 EPL | .pdf |
| 017 | VEGP TECHNICAL REQUIREMENTS MANUAL | .pdf |
| 018 | VEGP BASES | .pdf |
| 019 | VEGP FSAR REF DWGS PART 1 (1K5-1305-058-01 thru 1X3D-BD-J02D) | .pdf |
| 020 | VEGP FSAR REF DWGS PART 2 (1X3DG001 thru 1X4DB150-1) | .pdf |
| 021 | VEGP FSAR REF DWGS PART 3 (1X4DB151-1 thru 1X4DB203) | .pdf |
| 022 | VEGP FSAR REF DWGS PART 4 (1X4DB205-1 thru 1X4DE507) | .pdf |
| 023 | VEGP FSAR REF DWGS PART 5 (1X4DE508 thru 1X6AA02-00238) | .pdf |

| SEQ | FILENAME<br>CONTENT | EXTENSION |
|---|---|---|
| 024 | VEGP FSAR REF DWGS PART 6 (1X6AA02-00239 thru 2X4DB170-2) | .pdf |
| 025 | VEGP FSAR REF DWGS PART 7 (2X4DB174-2 thru AX1D11A04-1) | .pdf |
| 026 | VEGP FSAR REF DWGS PART 8 (AX1D11A04-3 thru AX4DB241) | .pdf |
| 027 | VEGP FSAR REF DWGS PART 9 (AX4DB242-1 thru AX4DJ8047) | .pdf |
| 028 | VEGP FSAR REF DWGS PART 10 (AX4DJ8048 thru AX6DD303) | .pdf |
| 029 | VEGP FSAR REF DWGS PART 11 (AX6DD304 thru CX5DT101-58) | .pdf |

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**CD-ROM (1 disc) containing Files 001 – 029**

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**Enclosure 3**
**10 CFR 50.59 Summary Report**

**Activity:** 17019-1/2, 18040-1/2

**Title:** Major procedure revision to 17019-1/2 regarding operator response to ALB19-B04 Turbine Condenser Low Vacuum / High Rate Of Change Lowering and revision to 18040 1/2 Subsequent Operator Actions for "Partial Loss of Condenser Vacuum"

**10 CFR 50.59 Evaluation Summary:**

This change modifies the immediate operator action contained within 17019-1/2 relative to ALB19-B04 "Turbine Condenser Low Vacuum / High Rate Of Change Lowering" and first note in Subsequent Operator Actions in 18040 1/2, "Partial Loss of Condenser Vacuum". Current guidance states that it is desired that operators immediately lower turbine load in response to the alarm in order to maintain the C-9 Interlock. The C-9 interlock arms the steam dumps when condenser vacuum is above 24.7" Hg. This change gives operations the discretion to allow the plant to operate below the C-9 setpoint which would render the steam dumps unavailable for the duration of operation below C-9. UFSAR Section 15.2.2.1 describes the function of the "automatic turbine bypass system" or steam dumps relative to a generator load rejection response. The purpose of the steam dumps is to reject excess heat from the secondary loop in the event of a primary to secondary power mismatch in order to prevent primary loop transients. This procedural change allows for operations to run without this automatic action in order to eliminate the risk associated with moving reactor power around due to environmental conditions when the low vacuum can be attributed to the "Circulating Water System" conditions.

This change does not increase the frequency of any accident previously analyzed. The function of the steam dumps are described in the UFSAR however they are not credited. The steam dumps are non-safety related and are not required to safety shutdown and cooldown the plant. Per UFSAR section 10.4.4.1 "The turbine bypass system serves no safety function and has no safety design basis."

This change allows operations to operate the plant below the C-9 interlock which arms the steam dumps. The steam dumps would automatically arm and disarm based on condenser vacuum. No action needs to be taken to allow this to happen as it is a designed function of the C-9 interlock. Additionally, the steam dumps are listed and discussed in Section 7.7 which contains equipment NOT important to safety therefore this change does not impact equipment important to safety in any way.

This change does not represent an increase in consequence relative to any previously analyzed accident. Even in the presence of a known fuel leak coincident with a known primary to secondary tube leak the consequence would not increase provided secondary activity is maintained below the tech spec limit.

All equipment affected by this change is non-safety related non-credited equipment. This change does not impact any piece of equipment credited to shut down or cooldown the plant.

This change does not create any new accident possibilities. Per UFSAR 15.2.2.1 for loss of external load event "If the condenser is not available, the excess steam generation is relieved to the atmosphere." It also states "A loss-of-external-load event results in a nuclear steam supply system transient that is bounded by the turbine trip event analyzed in subsection 15.2.3." Section 15.2.3 discusses a turbine trip coincident with the condenser being unavailable. This scenario is already discussed and analyzed in the UFSAR.

This does not create the possibility of any new malfunction results. The UFSAR already contains the exact scenario that this change allows. The absence of steam dumps during a turbine trip or load rejection is already covered in chapter 15 Therefore, this change does not create the possibility for a malfunction of an SSC important to safety.

This change only impacts secondary non-crediting SSCs and therefore has not impact on any of the three fission product barriers. This change impacts SSCs listed specifically as NOT important to safety in section 7.7. This change does not introduce any scenario that is not already contained within Chapter 15.

UFSAR Chapter 15 clearly documents the plant response associated with a turbine trip or loss of load event with or without the steam dumps therefore this is not an adverse change requiring prior approval.

**Activity:** TE 1102038

**Title:** Change to Allow Movement of Loads Over the Reactor Vessel While the Head is in Place

**10 CFR 50.59 Evaluation Summary:**

The activity being implemented provides an extended safe load path for moving loads over the reactor vessel during outages while the integrated head package is still in place on the reactor vessel during Modes 5 and 6. DOC SNC1222466 and associated calc.X2CJ04.02.10 determined max lift height and weight of the loads to ensure that a load drop will not penetrate the missile shield on the integrated head package. This will ensure that a load drop will not affect any fuel in the reactor vessel. In addition, it was determined that in the unlikely event a load drop would occur over the extended load path, no decay heat removal equipment or other equipment needed to maintain the reactor in safe shutdown condition in Modes 5 and 6 would be impacted.

The changes being made to allow an extended load path over the reactor vessel will introduce a slight, but not more than minimal, increase in the opportunity for a load drop inside containment since it expands the area where a drop could occur However, the change does not degrade the functionality of the polar crane or rigging used to move loads and the maximum weight of the load permitted by the being made is well within the capacity of the polar crane.

Even though there is a small, but not more than minimal, increase in the opportunity for a load drop inside containment, there are no conditions being introduced that would lead to the conclusion that the frequency of a heavy load drop inside containment has increased. Therefore, the proposed activity will NOT result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the Updated Vogtle FSAR.

This change would not be more than a minimal increase in the likelihood of a malfunction of an SSC important to safety for the following reasons: the extended load path will not cause an incident where the reactor vessel could be penetrated by a dropped load, resulting in water leakage that could uncover fuel, and it has been determined there is no equipment in the immediate vicinity around the reactor vessel that is needed for decay heat removal or to keep the reactor in a safe shutdown condition.

The malfunction associated with the activity is the failure of the polar crane to carry the load across the new load path and result in a load drop. The activity does not change the operating characteristics or functions of the polar crane or the rigging used for miscellaneous loads. Additionally, there are no changes to fuel loading that may be in the core. Since no other parameters associated with the activity will be changed, there will be no impact to the radiological consequences of the malfunction of the equipment, i.e., the malfunction and its consequences will be the same. Therefore, this activity will not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the Updated FSAR.

There is no new failure mechanism associated with the proposed activity since there are no new or modified SSCs being implemented as a result of the proposed activity. This activity does not create the possibility for an accident of a different type than any previously evaluated in the Updated Vogtle FSAR.

The bounding conditions for the drop analysis are contained as restrictions in the proposed revision to the polar crane operating procedure, 93247-C. There are no other new type of accident conditions, or equipment or component malfunctions that are introduced as a result of the changes. There are no changes to the inputs for or results of the other load analyses described in the Updated FSAR. There are no changes being made that would impact any design basis limit Accordingly, the changes being made do not have any impact on the integrity of the fuel cladding, reactor coolant pressure boundary, or the containment.

**Activity:** DCP SNC922254

**Title:** Vogtle Unit 1 Auxiliary Feedwater Orifice Modifications

**10 CFR 50.59 Evaluation Summary:**

The installation of the orifices in the AFW system piping for SNC's Vogtle Unit 1 plant does not constitute a need to notify or seek permission from the Nuclear Regulatory Commission as specified in 10CFR50.59 using the guidance obtained from supporting document NEI 96-07 Revision 1, "Guidelines For 10CFR50.59 Implementation". These installed components cannot cause any condition I, III or IV event and thus their frequency and severity of occurrences are not changed. The analyses have documented that the revised design basis flows, temperatures and pressures can be maintained and are acceptable to meet the revised accident analyses. All potential failures of the new orifices along with the existing orifices (resized and reinstalled) are bounded within the revised accident analyses. The increase in off-site doses due to the SGTR event do not result in more than a minimal increase in the consequences of an accident. The changes to the AFW pumphouse internal flooding does not create the possibility of a malfunction of a SSC important to safety with a different result than previously evaluated in the Updated UFSAR. The updated HZP SLB analysis including the AFW flow requirement change does not create the possibility of a malfunction of a SSC important to safety with a different result than previously evaluated in the Updated UFSAR or have any impact on the integrity of the fuel cladding, the reactor pressure boundary, or containment.

**Activity:** SNC1042051

**Title:** U1 & U2 Low Condenser Vacuum Alarm and C9 Interlock Setpoint Change

**10 CFR 50.59 Evaluation Summary:**

This activity involves a design change to lower the low condenser vacuum alarm and C9 setpoints. This changes both the low condenser vacuum alarm and C9 from fixed setpoints to adjustable setpoints based on turbine load (%). The C9 interlock arms the steam dumps when condenser vacuum is above 24.7" Hg and the low condenser vacuum alarm alerts the Main Control Room when Main Condenser vacuum degrades below 24.7" Hg. DCP SNC1042051 will lower the C9 setpoint from 24.7" Hg to a new minimum of 23.7" Hg Vac, allowing for steam dump availability at a higher condenser backpressure. This allows for more operator flexibility during times when the condenser vacuum is challenged.

The Main Condensers are outlined in section 10.4.1 of the UFSAR. The turbine bypass system is outlined in section 10.4.4 of the UFSAR. Neither system has any safety function, but they do have power generation design bases. Both the Main Condenser and the turbine bypass systems are described in Chapter 15 of the UFSAR as being available as needed in the accident scenarios described in that Chapter. The vacuum used to provide the low vacuum alarm and C9 interlock are not documented there. This change only slightly lowers this vacuum level to bring in the alarm and allow for the turbine bypass system to operate at a lower level, but does not increase the frequency of any accident scenarios described in Chapter 15 of the UFSAR.

The turbine bypass system is not a safety-related system, and the failure of the system will not compromise other safety-related systems or prevent a safe shutdown. The scope of work associated with the feedwater and condensate system (Main Condenser low vacuum alarm) is not safety-related and will not compromise a safe shutdown or affect containment design pressure or the Departure from Nucleate Boiling Ratio (DNBR). Therefore, this change does not increase the likelihood of a malfunction of an SSC important to safety.

This activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR. This activity only provides guidance to allow lowering the low condenser vacuum alarm and C9 setpoints associated with the turbine bypass system. The Main Condensers and turbine bypass system will operate as designed in all other aspects. By maintaining the design functions of the Main Condensers and the turbine bypass system, the consequences of an accident will remain unchanged as described in Chapter 15 of the UFSAR.

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 UFSAR. There are no Technical Specifications associated with the Main Condenser vacuum or the turbine bypass system. In addition, neither system provides a safety-related function. The small change in vacuum setpoint associated with this DCP will not compromise the power generation function of the turbine bypass system or its function as described in Chapter 15 of the UFSAR.

This activity does not create the possibility for an accident of a different type than any previously evaluated in the UFSAR. This activity only provides design and guidance for lowering of the low condenser vacuum alarm and C9 interlock setpoints. The design and guidance to do this will be provided in the associated Design Change Package.

This activity does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR. This activity only provides design and guidance to lower the low condenser vacuum alarm and C9 setpoints. This small change cannot cause a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

This activity will not have any impact on the integrity of the fuel cladding, reactor coolant pressure boundary, or containment. This activity only provides design and guidance design and guidance to lower the low condenser vacuum alarm and the C9 setpoints.

**Activity:** DCP SNC1046656

**Title:** U2 A-Train MSIV Actuator Replacements

**10 CFR 50.59 Evaluation Summary:**

Design Change Package SNC1046656 will replace the existing Unit 2 Train A main steam isolation valve (MSIV) actuators with system media actuated type actuators. System media actuators use the system media (steam) as the motive force for stroking the valve. As part of the actuator replacement, the existing electro-hydraulic system and nitrogen supply to the existing actuators will be removed, which will support the site's single point of vulnerability reduction initiative. Two aspects of the design activity represent a modification to an SSC such that a design function as described in the UFSAR is adversely affected. These are (1) the inability to close the MSIV in less than the required 5 seconds upon receipt of an isolation signal, and (2) the introduction of a vent line that creates a potential flow path that requires isolation to prevent the release of iodine activity accumulating in the secondary side of the steam generator following an accident that is not considered in the dose analysis. The replacement of the MSIV actuators cannot contribute to the initiation of any accident previously evaluated in the UFSAR. Therefore, the activity has no impact on the frequency of occurrence of an accident previously evaluated in the UFSAR. The activity does not result in an increase in likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. To address the inability to close the MSIV in less than 5 seconds upon receipt of an isolation signal, a new MSIV closure time of 7 seconds is established for accidents initiated above 325 psia, which is within the time assumed in the accident analyses. An analysis of the applicable events for the longest MSIV closure time expected for accidents initiated at 325 psia and below demonstrate that the current analysis of record is bounding and substantiates the acceptability of a longer valve closure initiated at lower system pressures. To address the introduction of a vent line that creates a potential flow path that requires isolation to prevent the release of iodine activity, the vent line is equipped with a solenoid valve that automatically closes after MSIV closure, isolating the flow path. To address a possible single failure of the solenoid valve, a new operator action is established to close a manual isolation valve on the vent line. This new operator action is considered to be a new manual operator action in support of a design function credited in the safety analyses. The new operator action is reflected in plant procedures and operator training programs, and can be completed in the time required considering the aggregate affects, such as workload or environmental conditions, expected to exist when the action is required. The evaluation of the change considers the ability to recover from credible errors in performance of the action and the expected time required to make such a recover, and also considers the effect of the change on plant systems. Therefore, this activity does not result in an increase in likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR. For a main steam line break accident there is no impact to the offsite doses if the Unit 2 train A MSIV actuator vent pathway remains unisolated during the 20-hour accident duration. With regard to a steam generator tube rupture accident, the vent line for the SMA actuators contains a single safety-related valve solenoid valve that is open during the closure of the MSIV and automatically closes after a delay period following the closure of the MSIV. During the time when the solenoid valve is open, steam can be released through the new vent line to the environment. This results in a new single failure case where MV3 fails to automatically close after a delay period following an SGTR. A supplemental calculation evaluates this new case, in which the postulated single failure is a failure of MV3 to automatically close rather than a failure of an ARV to close. The calculation determines the time when operations will need to close the manual valve on the SMA actuator vent line so that the dose consequences in the design basis case remain bounding. An isolation time of 172 minutes after event initiation results in an offsite dose that is within what was previously calculated in the ARV single failure case. There are redundant MSIVs in each steamline. Any single failure (i.e., malfunction) of an MSIV to close upon receipt of an isolation signal will still result in blowdown of only one steam generator at most. As discussed above, isolation of the new vent line will be achieved such that the current licensing basis cases for offsite doses remain bounding. Therefore, this activity does not result in more than a minimal increase in the consequences of a malfunction previously evaluated in the UFSAR. The change in MSIV closure time has been evaluated and determined to be acceptable. There are no new accident types introduced by the change in actuator. Therefore, this activity does not create the possibility for an accident of a different type than any previously evaluated in the UFSAR. The new system media actuators will fail to the closed position, have redundant MSIVs in each main steam line, and are seismically designed and qualified for normal and post-accident environmental conditions. Therefore, this activity does not create the possibility for a malfunction of an SSC important to safety with a different result than previously evaluated in the UFSAR. Applicable safety analyses have been analyzed for variable valve closure times with acceptable results, and therefore this activity has no impact on the integrity of the fuel cladding and integrity of containment. This activity does not modify or affect the reactor coolant pressure boundary and therefore does not impact the integrity of the reactor coolant pressure boundary.

**Activity:** DCP SNC1168303

**Title:** U1 Main Turbine Vibration System Trip Function Removal

**10 CFR 50.59 Evaluation Summary:**

DCP SNC1168303, provides a justification for permanently defeating the automatic turbine trip from excessive bearing vibration. The automatic trip previously associated with excessive bearing vibrations was for asset protection only and has no impact on the UFSAR safety analysis. No safety related SSCs, nor those important to safety, will be impacted by the proposed modification. Tripping on excessive bearing vibration was one of the protective trips listed in Section 10.2.2.4 of the UFSAR, prior to implementation of LDCR 2020069 (part of Temporary Modification SNC1125090). UFSAR section 3.5.1.3 does not credit the automatic trip on excessive bearing vibration with prevention of turbine induced missiles. In addition, Section 15.2.3.1 analyzes the impact of a turbine trip without a reactor trip (most limiting scenario), but does not cite the reason for the turbine trip itself is bounding with respect to safety analysis. As such, the automatic trip on excessive bearing vibration functionality was not credited as a safety mitigation function.

This change did not result in a more than a minimal increase in the frequency of occurrence of an accident, likelihood of the occurrence of a malfunction of a structure, system or component (SSC) important to safety, consequences of an accident or malfunction; create the possibility of an accident of a different type or malfunction with a different result; result in a design basis limit for a fission product barrier being exceeded or altered; or result in a departure from a method of evaluation.

**Activity:** DCP SNC1185737

**Title:** U2 Main Turbine Vibration System Trip Function Removal

**10 CFR 50.59 Evaluation Summary:**

DCP SNC1185737, provides a justification for permanently defeating the automatic turbine trip from excessive bearing vibration. The automatic trip previously associated with excessive bearing vibrations was for asset protection only and has no impact on the UFSAR safety analysis. No safety related SSCs, nor those important to safety, will be impacted by the proposed modification. Tripping on excessive bearing vibration was one of the protective trips listed in Section 10.2.2.4 of the UFSAR, prior to implementation of LDCR 2020050 (part of Temporary Modification SNC1111310). UFSAR section 3.5.1.3 does not credit the automatic trip on excessive bearing vibration with prevention of turbine induced missiles. In addition, Section 15.2.3.1 analyzes the impact of a turbine trip without a reactor trip (most limiting scenario) but does not cite the reason for the turbine trip itself is bounding with respect to safety analysis. As such, the automatic trip on excessive bearing vibration functionality was not credited as a safety mitigation function.

This change did not result in a more than a minimal increase in the frequency of occurrence of an accident, likelihood of the occurrence of a malfunction of a structure, system or component (SSC) important to safety, consequences of an accident or malfunction; create the possibility of an accident of a different type or malfunction with a different result; result in a design basis limit for a fission product barrier being exceeded or altered; or result in a departure from a method of evaluation.

**Activity:** SNC1091488

**Title:** Vogtle Turbine Valve Frequency Analysis

**10 CFR 50.59 Evaluation Summary:**

This activity involves an evaluation to extend turbine valve maintenance and testing frequencies on Vogtle Units 1 & 2. The Vogtle UFSAR, as written, details the inspection and testing requirements for the turbine valves stating that they are readily accessible for inspection and are available for testing during normal plant operation. The UFSAR also states the TRM details the testing requirements for the turbine valves. TRM specifies that while in mode 1 and mode 2 with turbine operating to cycle (test) the valves in accordance with GET-8039 but not to exceed 3 months. UFSAR sections 10.2.3.5 and 10.2.4.6 also states that each valve Is disassembled approximately every 120 months during scheduled refueling or maintenance shutdowns.

REA SNC1091488 evaluates the extension of these frequencies and the impact to turbine missile protection as described in the UFSAR. This evaluation documents the turbine missile probabilities will remain within the NRC acceptance criterion. This activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated In the UFSAR provided that the plant-specific threshold is not exceeded as a result of the change. Therefore, this activity does not result in more than a minimal Increase the frequency of an accident previously evaluated In the UFSAR.

Appendix B Table 6-4 of MPR report 1380-0029-RPT-001 Revision 1, provides change in the reliability for each type of turbine valve based on the increase in test or maintenance interval for several cases. Note that since the CIVs contain two valves. The unreliability of each valve is the square root of the total CIV reliability. This data shows that the average unreliability of the valves will increase by less than a factor of two due to the increase in the test interval from 3 months to 6 months or the increase in the maintenance interval from 108 months to 126 months. The change to the maintenance interval and the test interval is treated separately.

Therefore, this activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the UFSAR. RER SNC1091468 evaluates the failure modes associated with these components and the operating and inspection history. This data was used to determine that the missile probability as documented in the UFSAR will not be negatively affected. Therefore. extending the testing frequency to 6 months and the maintenance frequency to 126 months will not result in more than a minimal increase the likelihood of occurrence of a malfunction of an SSC Important to safety previously evaluated in the UFSAR.

This activity does not result in more than a minimal Increase in the consequences of an accident previously evaluated in the UFSAR. This activity only provides guidance to extend turbine valve maintenance and testing frequencies as evaluated in RER SNC1091468. The Turbine-Generator will operate as designed in all other aspects. There are no radiological dose consequences associated with turbine missile events, therefore, this activity will not cause an increase in radiological dose to the public or control room operators.

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 UFSAR. There are no Technical Specifications associated with the turbine valves. In addition, the turbine-generator system does not provide a safety-related function. The turbine valve maintenance and testing frequency changes will not compromise the power generation function of the turbine-generator or its function as described in the UFSAR. The consequences of a HP turbine failure would be the same as previously evaluated.

There are no radiological dose consequences associated with turbine missile events, therefore, this activity does not cause an increase in radiological dose to the public or control room operators.

This activity does not create the possibility for an accident of a different type than any previously evaluated in the UFSAR. This activity only provides guidance for extension of the maintenance and testing Intervals on Unit and 2 Turbine-Generator valves. The evaluation to do this is provided In REA SNC1091488.

This activity does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR. This activity only provides design and guidance to extend maintenance and testing frequencies on the turbine valves. This change has been evaluated and cannot cause a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

This activity will not have any impact on the integrity of the fuel cladding. reactor coolant pressure boundary, or containment. This activity only provides guidance to extend maintenance and testing Intervals on the turbine valves associated with the turbine-generator system.

**Activity:** DCP SNC545365

**Title:** U2 Modifications for GSI-191

**10 CFR 50.59 Evaluation Summary:**

The removal of disks from the RHR sump strainers results in reduced strainer surface area, therefore, the strainer may have a slight increase in debris loading and approach velocity that may result in higher flow resistance and head loss through the debris bed. However, this change has an offsetting improvement by resulting in the strainers being fully submerged for additional LOCA break scenarios. Full submergence can prevent RHR pump failures due to vortexing and air ingestion through the strainers and thus provide protection for the RHR pumps that take suction from the strainers in the recirculation mode. Deterministic analyses were performed that verified the modified RHR strainer with two disks removed will maintain structural and functional performance when structurally loaded, including seismic and hydrodynamic forces, and debris loading. The change will not have an adverse effect on containment heat sink evaluations and will not exceed maximum strainer gap size requirements which will prevent entry of larger sized debris into the ECCS.

ECCS pump performance analyses were performed with the modified strainer design which proved the strainer design will support resolution of GSI-191 for VEGP using an NRC-approved risk-informed resolution strategy. The station is currently operating under an industry Justification for Continued Operation (JCO) as documented in GL 2004-02. Under this JCO, continued plant operation is justified until GL 2004-02 is resolved for each station. This JCO is based, in part, on the extremely low probability of the most severe initiating LOCA event, existing PWR design features that prevent blockage of the ECCS sumps, NRC's approval for leak-before-break (LBB) credit on the largest RCS primary coolant piping, margins in PWR designs that are not credited in the plant licensing basis, and various interim compensatory measures implemented in response to Bulletin 2003-01.

The physical modifications to the RHR sump strainers do not pose a more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety and does not create a malfunction of an SSC important to safety with a different result. These changes have no effect on the frequency of an accident and the radiological consequences of an accident. The changes do not create the possibility for an accident of a different type or create a possibility for a malfunction of an SSC important to safety with a different result and has no effect on design basis limits for fission product barriers.

Consequently, the physical modifications to the RHR sump strainers do not require prior NRC approval to implement the physical changes.

**Activity:** DCP SNC545351

**Title:** U1 Modifications for GSI-191

**10 CFR 50.59 Evaluation Summary:**

Physical Modifications to the RHR Sump Strainers The removal of disks from the RHR sump strainers results in reduced strainer surface area, therefore, the strainer may have a slight increase in debris loading and approach velocity that may result in higher flow resistance and head loss through the debris bed. However, this change has an offsetting improvement by resulting in the strainers being fully submerged for additional LOCA break scenarios. Full submergence can prevent RHR pump failures due to vortexing and air ingestion through the strainers and thus provide protection for the RHR pumps that take suction from the strainers in the recirculation mode. Deterministic analyses were performed that verified the modified RHR strainer with two disks removed will maintain structural and functional performance when structurally loaded, including seismic and hydrodynamic forces, and debris loading. The change will not have an adverse effect on containment heat sink evaluations and will not exceed maximum strainer gap size requirements which will prevent entry of larger sized debris into the ECCS.

ECCS pump performance analyses were performed with the modified strainer design which proved the strainer design will support resolution of GSI-191 for VEGP using an NRC-approved risk-informed resolution strategy. The station is currently operating under an industry Justification for Continued Operation (JCO) as documented in GL 2004-02. Under this JCO, continued plant operation is justified until GL 2004-02 is resolved for each station. This JCO is based, in part, on the extremely low probability of the most severe initiating LOCA event, existing PWR design features that prevent blockage of the ECCS sumps, NRC's approval for leak-before-break (LBB) credit on the largest RCS primary coolant piping, margins in PWR designs that are not credited in the plant licensing basis, and various interim compensatory measures implemented in response to Bulletin 2003-01.

The physical modifications to the RHR sump strainers do not pose a more than minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety and does not create a malfunction of an SSC important to safety with a different result. These changes have no effect on the frequency of an accident and the radiological consequences of an accident. The changes do not create the possibility for an accident of a different type or create a possibility for a malfunction of an SSC important to safety with a different result and has no effect on design basis limits for fission product barriers. Consequently, the physical modifications to the RHR sump strainers do not require prior NRC approval to implement the physical changes.

**Activity:** SNC1172254

**Title:** Head Stand Modification to Support Peening In 1R24

**10 CFR 50.59 Evaluation Summary:**

The reactor head lift height is being increased to 32 feet (six feet above the operating deck at El. 220') from 31 feet (five feet above the operating deck).

The revised reactor vessel head drop conditions remain consistent with NUREG-0612, Appendix A for Vogtle Units 1 and 2 with respect to possible effects on the reactor vessel and shield wall concrete. The results of the analysis demonstrate that previous conclusions are unimpacted by the change.

The postulated head drop can only occur during reactor disassembly/reassembly. A change to the height and weight of the reactor head lift does not change how often the head is assembled/disassembled and therefore does not increase the frequency of occurrence of an accident previously evaluated in the UFSAR. An evaluation has been performed to document that the revised design combination of 425 kips and 32 feet (having been altered from 450 kips and 31 feet) does not result in a change to previously documented effects on the core cooling and vessel integrity. The cooling system and vessel were previously evaluated for impacts due to a combination of 450 kips and 31 feet which combine in a resulting kinetic energy which causes deflection/stresses upon impact to the vessel. When the parameters are updated to 425 kips and 32 feet, the resulting kinetic energy is bounded by the previous design combination Therefore, there is not change to the likelihood of occurrence of malfunction of a structure, system, or component important to safety previously evaluated in the UFSAR.

Actual reactor head weight is identified in existing plant documentation as 398 kips (approximately 400 kips). The weight of the load block as identified by Multipage Drawing AX4AL01-00195 is 18 kips resulting in a total weight of 418 kips. A value of 425 kips is used for evaluation and is conservative.

The postulated head drop can only occur during reactor disassembly or reassembly. A change to the height and weight of the reactor head lift does not change how often the head is assembled/disassembled and therefore does not increase the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the FSAR.

An evaluation has been performed to document that increased lift height for the reactor head does not result in a change to previously documented effects on the core cooling and vessel integrity Operation of required safety systems is unaffected

An evaluation has been performed to document that increased lift height for the reactor head does not result in a change to previously documented effects on the core cooling and vessel integrity. Operation of required safety systems is unaffected. The increased lift height does not result in an increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

UFSAR Section 9.1.5.3.1.1.2 identifies that the polar crane is postulated to fail resulting in a direct, concentric drop of the head (including polar crane load block) onto the reactor vessel, which analysis has shown is the most limiting scenario for the head impact on the vessel. This type of accident is unchanged by the increased lift height.

An evaluation has been performed to document that increased lift height for the reactor head does not result in a change to previously documented effects on the core cooling and vessel integrity. Therefore, the proposed change does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated.

An evaluation has been performed to document that the revised design combination of 425 kips and 32 feet (having been altered from 450 kips and 31 feet) does not result in a change to previously documented effects on the core cooling and vessel integrity. The cooling system and vessel were previously evaluated for impacts due to a combination of 450 kips and 31 feet which combine in a resulting kinetic energy which causes deflection/stresses upon impact to the vessel. When the parameters are updated to 425 kips and 32 feet. the resulting kinetic energy is bounded by the previous design combination. Therefore, there is no impact to the integrity of the fuel cladding, reactor coolant pressure boundary, or containment as previously identified in UFSAR section 9.1.5.3.1.1.2.

Actual reactor head weight is identified in existing plant documentation as 398 kips (approximately 400 kips). The weight of the load block as identified by Multipage Drawing AX4AL01-00195 is 18 kips resulting in a total weight of 418 kips. A value of 425 kips is used for evaluation and is conservative.

It is therefore concluded that the postulated head drop of 32 feet through air does not represent an unreviewed safety question and will not involve a change to plant technical specifications.