From: Billy Gleaves
Sent: Friday, April 7, 2023 7:57 AM
To: Vogtle PEmails
Cc: Cayetano Santos
Subject: FW: Pre-Submittal Meetings
Attachments: NL-23-0135 PCIT+LAR [PSM DRAFT].pdf

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From: Dorsey, Keith A. <kadorsey@southernco.com>
Sent: Thursday, April 6, 2023 8:10 PM
To: Cayetano Santos <Cayetano.Santos@nrc.gov>
Cc: Billy Gleaves <Bill.Gleaves@nrc.gov>
Subject: [External_Sender] Pre-Submittal Meetings

Tanny,

Attached is the draft LAR that will used during the Tech Spec Exceptions Prior to Initial Criticality LAR pre-submittal meeting to be held next Thursday, April 13th. Also, we are targeting an April 20th pre-submittal date for the integrated leak rate testing exception LAR. If either of these dates need to be moved to accommodate the staff's schedule, please let me know.

Please contact me if you have any questions.

Keith Dorsey P.E.
Lead Engineer
Nuclear Development
Regulatory Affairs

Bin N-226-EC
3535 Colonnade Parkway
Birmingham, Al 35243
205-992-7480

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Recipients:
"Cayetano Santos" <Cayetano.Santos@nrc.gov>
Tracking Status: None
"Vogtle PEmails" <Vogtle.PEmails@usnrc.onmicrosoft.com>
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ENCLOSURE

Evaluation of the Proposed Change

Subject: License Amendment Request: Timing of Unit 4 Technical Specifications Effectiveness Prior to Initial Criticality (LAR-23-005)

ATTACHMENTS:

1. Combined License and Technical Specifications Marked-up Pages
2. Combined License and Technical Specifications Revised Pages
3. Technical Specifications Bases Marked-up Pages (For Information Only)

1. SUMMARY DESCRIPTION

Southern Nuclear Operating Company (SNC) is proposing this license amendment request (LAR) to revise the Combined License (COL) for Vogtle Electric Generating Plant (VEGP) Unit 4, to modify the current COL Condition 2.D(9), Technical Specifications (TS), to limit the scope of TS that become effective upon a Commission finding in accordance with 10 CFR 52.103(g). The proposed revision to the COL Condition would provide temporary exceptions prior to initial criticality of the reactor core for certain TS while operating in Modes 4, 5, and 6. The COL Appendix A TS are proposed to be fully effective at Unit 4 initial criticality of the reactor core. The proposed change also includes revision to COL Appendix A TS Limiting Condition for Operation (LCO) 3.0.7 to coordinate the TS compliance provisions proposed in the COL Condition.

The 10 CFR 52.103(g) finding determines that inspections, tests, analyses, and acceptance criteria (ITAAC) have been met, and authorizes the licensee to operate the facility (i.e., commence loading fuel into the reactor core). Currently the COL condition makes all TS requirements effective at this time. The full scope of the current TS LCOs require the lowest functional capability and performance levels of equipment based on worse-case assumptions reflecting the full range of the 40 year operating license period, which includes maximum fission product inventory, maximum decay heat, and maximum approved operating power levels. However, prior to initial criticality, safe operation of the facility is met with reduced functional capability and performance levels of equipment, since there is no fission product inventory, no decay heat, and no nuclear fission power being generated.

2. DETAILED DESCRIPTION

2.1 System Design and Operation

2.1.1 Engineered Safety Features (ESF)

As described in Updated Final Safety Analysis Report (UFSAR) Section 6.0, ESF protect the public in the event of an accidental release of radioactive fission products from the reactor coolant system. The engineered safety features function to localize, control, mitigate, and terminate such accidents and to maintain radiation exposure levels to the public below applicable limits and guidelines. The following are defined as engineered safety features:

<u>Containment</u>

The function of the containment vessel, as part of the overall containment system, is to contain the release of radioactivity following postulated design basis accidents (DBAs). The containment system is designed such that for all break sizes, up to and including the double-ended severance of a reactor coolant pipe or secondary side pipe, the containment peak pressure is below the design pressure.

Design basis events of concern for containment analysis are steamline break (SLB) inside containment with the failure of the associated main steam isolation valve (MSIV) to close, or a main feedline break with the associated failure of a feedline isolation or control valve to close. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers, downstream from the other MSIV, contribute to the total mass and energy release. The non-safety related turbine stop or control valves, in combination with the turbine bypass, and moisture separator reheater 2nd stage steam isolation valves, are assumed as a backup to isolate the steam flow path given a single failure of an MSIV. The design precludes the blowdown of more than one steam generator (SG).

Similarly, closure of the main feedwater isolation valves (MFIVs), main feedwater control valves (MFCVs), and startup feedwater system, effectively terminates the addition of main feedwater to an affected SG, limiting the mass and energy release for steam or feedwater line breaks inside containment. The design precludes the blowdown of more than one SG.

The containment vessel also functions as the safety-related ultimate heat sink by transferring the heat associated with accident sources to the surrounding environment.

Passive Containment Cooling System (PCS)

The function of the passive containment cooling system is to maintain the containment temperature below a maximum value and to reduce the containment temperature and pressure following a postulated design-basis event. The passive containment cooling system removes thermal energy from the containment atmosphere. The passive containment cooling system also serves as the safety-related ultimate heat sink. The passive containment cooling system limits the release of radioactive material to the environment by reducing the pressure differential between the containment atmosphere and the external environment. This diminishes the driving force for leakage of fission products from the containment to the atmosphere.

Containment Isolation System

The major function of the containment isolation system is to provide containment isolation to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary, if required. This prevents or limits the escape of fission products that may result from postulated accidents. Containment isolation provisions are designed so that fluid lines penetrating the primary containment boundary are isolated in the event of an accident. This minimizes the release of radioactivity to the environment.

Passive Core Cooling System (PXS)

The primary function of the passive core cooling system is to provide emergency core cooling following postulated design-basis events. The passive core cooling system provides reactor coolant system makeup and boration during transients or accidents where the normal reactor coolant system makeup supply from the chemical and volume control system is lost or is insufficient. The passive core cooling system provides safety injection to the reactor coolant system to provide adequate core cooling for the complete range of loss of coolant accident (LOCA) events up to, and including, the double ended rupture of the largest primary loop reactor coolant system piping. The passive core cooling system provides core decay heat removal during transients, accidents, or whenever the normal heat removal paths are lost.

PXS accumulators provide a very high flow for a limited duration of several minutes. The Core Makeup Tanks (CMTs) provide a relatively high flow for a longer duration. The In-containment Refueling Water Storage Tank (IRWST) provides a lower flow, but for a much longer time.

The Automatic Depressurization System (ADS) is designed to assure that core cooling and injection can be achieved for DBAs. For non-LOCA events, use of the ADS is not required and is not anticipated. For these events, injection of borated water into the core from the CMTs may be required for makeup or boration. However, the amount of water necessary will not reduce the level in the CMTs to the point of ADS actuation. For events which involve a loss of primary coolant inventory, such as a LOCA, the ADS will be actuated, allowing for injection from the accumulators, the IRWST.

Main Control Room (MCR) Emergency Habitability System (VES)

VES is designed so that the main control room remains habitable following a postulated design basis event that require protection from the release of radioactivity. The VES design basis also limits the temperature increase of the Main Control Room Envelope (MCRE) equipment and facilities that must remain functional during an accident, via de-energizing (load shedding) nonessential, non-safety MCR electrical equipment (e.g., wall panel information system displays, office equipment, water heater, kitchen appliances, and non-emergency lighting) and the heat absorption of passive heat sinks. VES actuation on a Class 1E 24-hour battery charger undervoltage signal, also de-energizes MCR air supply radiation monitor sample pumps to conserve battery capacity.

Fission Product Control

Post-accident safety-related fission product control is provided by natural removal processes inside containment, the containment boundary, and the containment isolation system. The natural removal processes, including various aerosol removal processes and pool scrubbing, remove airborne particulates and elemental iodine from the containment atmosphere following a postulated design basis event.

Control of the pH in the containment sump water post-accident is achieved through the use of pH adjustment baskets containing granulated trisodium phosphate (TSP). The TSP is designed to maintain the pH of the containment sump water to reduce radiolytic formation of elemental iodine in the containment sump, consequently reducing the aqueous production of organic iodine, and ultimately reducing the airborne iodine in containment and offsite doses.

2.1.2 Accident Analyses

As described in UFSAR Chapter 15.0, accident analyses begin with classifying plant conditions into four categories according to anticipated frequency of occurrence and potential radiological consequences to the public. Initial conditions assume a variety of core power, power distributions, Protection and Safety Monitoring System (PMS) functions, fission product inventories, and core decay heat.

As presented in UFSAR 15.6.5.1, the acceptance criteria for the LOCA are described in 10 CFR 50.46 and are summarized as follows:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- Localized cladding oxidation shall not exceed 17 percent of the total cladding thickness before oxidation.
- The amount of hydrogen generated from fuel element cladding reacting chemically with water or steam shall not exceed 1 percent of the total amount if all metal cladding were to react.
- The core remains amenable to cooling for any calculated change in core geometry.
- The core temperature is maintained at a low value, and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

During anticipated operational occurrences (AOOs), which are those events expected to occur one or more times during the unit life, the acceptable limits are:

- The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);

- Fuel centerline melt shall not occur; and
- The RCS pressure SL of 2750 psia shall not be exceeded.

DBAs are events that are analyzed even though they are not expected to occur during the unit life. The acceptance limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of the limits. Different accident categories are allowed a different fraction of these limits, based on the probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The accident analysis of the main steam line break assumes a secondary coolant specific activity radioactive isotope concentration. This assumption is used in the analysis for determining the radiological consequences of the postulated accident.

2.1.3 Reactor Vessel

Heatup and cooldown pressure-temperature limit curves are required as a means of protecting the reactor vessel during startup and shut down to minimize the possibility of fast fracture. The operating pressure and temperature curves are developed with methodology in accordance with 10 CFR 50, Appendix G.

As a safety precaution, there are no reactor vessel penetrations below the top of the core. This eliminates the possibility of a loss of coolant accident by leakage from the reactor vessel, which could allow the core to be uncovered. The core is positioned as low as possible in the vessel to limit reflood time in an accident.

The reactor vessel and reactor coolant system (RCS) is located in the reactor containment.

2.1.4 Steam Generator (SG) Isolation Valves

The PORV and PORV block valves are isolated following a Steam Generator Tube Rupture (SGTR) to minimize radiological releases into the atmosphere. In addition, the PORV block valves are containment isolation valves and support the assumptions related to establishing the containment boundary during major accidents.

The blowdown flow path on each SG must be isolated following Loss of Feedwater and Feedwater Line Break events to retain the steam generator water inventory for use in RCS heat removal via the SGs. RCS heat removal for these events is, primarily, provided by the Passive Residual Heat Removal Heat Exchanger (PRHR HX); however, the SG heat removal is assumed.

2.1.5 Protection and Safety Monitoring System

Reactor Trip System (RTS) Instrumentation initiates a unit shutdown to protect against violating the core fuel design limits and RCS pressure boundary during AOOs and to assist the Engineered Safety Feature Actuation System (ESFAS) in mitigating accidents.

The purpose of the required channels of ESFAS instrumentation and controls is to sense accident situations and initiate ESF systems to provide plant protection in the event of any of the analyzed accidents or transients where the actuated function is credited to perform its safety function. The purpose of the Diverse Actuation System (DAS) manual controls is to provide non-Class 1E backup in case of common-mode failure of PMS. DAS manual controls are not credited for mitigating accidents in the FSAR Chapter 15 analyses.

2.2 Current Requirements

COL Condition 2.D(9) requires that technical specifications in Appendix A to this license become effective upon a Commission finding that the acceptance criteria in this license (ITAAC) are met in accordance with 10 CFR 52.103(g).

Initial criticality reflects a plant transition into TS Mode 2, and eventually Mode 1. The TS described in Section 2.4 are those that have Applicability requirements for one or more of Modes 4, 5, and 6 (or portions thereof) and are proposed for modification of the timing of their effectiveness. These TS are currently required to become effective at the 10 CFR 52.103(g) finding and are included in the LAR Section 3 evaluation of the appropriate TS requirements to be temporarily excluded from being permanently effective between the 10 CFR 52.103(g) finding (i.e., authorization for initial loading of fuel assemblies into the reactor core) and up to, but not including, while operating in plant operational Mode 3 and initial criticality.

The provision of TS LCO 3.0.7 allows specified TS requirements to be changed to permit performance of stated special tests and operations, which otherwise could not be performed if required to comply with the requirements of those TS.

2.3 Reason for the Proposed Change

TS LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. The TS LCOs that become effective based on COL Condition 2.D(9) reflect the functional capability and performance levels of equipment that are based on worse-case assumptions reflecting the full range of the 40-year operating licensed period, which includes maximum fission product inventory, maximum decay heat, and maximum approved operating power levels. However, while operating in Modes 4, 5, and 6, and prior to initial criticality, safe operation of the facility is met with reduced functional capability and performance levels of equipment, since there is no fission product inventory, no decay heat, and no nuclear fission power being generated.

The requested change to COL Condition 2.D(9) and TS LCO 3.0.7 will allow flexibility for various unplanned contingencies during the startup test phase (i.e., commencing with initial fuel loading into the reactor vessel, while operating in Modes 4, 5, and 6, and prior to initial criticality). The startup tests are performed to demonstrate the capability of individual systems, as well as the integrated plant, to meet performance requirements; however, during this time, it is anticipated that equipment repairs, and subsequent retesting, may be necessary. These repairs and retests may have TS LCO constraints imposed which are not necessary for safe operation of the facility and impose hardships to efficiently conduct the repairs and efficiently complete startup testing.

During the VEGP Unit 3 startup testing phase, there has been experience with unnecessary hardships imposed by the TS. In order to avoid unnecessary delays, VEGP Unit 3 TS amendments were requested based on both an emergency situation and exigent circumstances resulting in Amendment Nos. 189 and 190, respectively (refer to Precedent Section).

This proposed Unit 4 LAR is intended to avoid future unnecessary startup test phase maintenance constraints and schedule impacts imposed by excessively restrictive TS and avoid the potential for additional LAR(s) for emergency situation and/or exigent circumstances.

2.4 Description of the Proposed Change

The COL Condition 2.D(9) is proposed to provide temporary Modes 4, 5 and 6 effectiveness timing exceptions prior to initial criticality for those TS that are justified in Section 3 as unnecessary to provide for safe operation of the facility. The specific TS exceptions are identified in the proposed markup of the COL Condition provided in the Attachment 1 as well as the revised pages in Attachment 2.

Along with the change to the COL Condition, the COL Appendix A TS LCO 3.0.7 is proposed to be revised to add a paragraph identifying that, for Unit 4 only, Combined License Condition 2.D(9) provides the milestone effectiveness for specified TS requirements prior to becoming fully effective at initial criticality of the reactor core, and that compliance with TS requirements that are excluded from becoming effective while operating in MODES 4, 5, and 6 in accordance with the COL Condition, is optional prior to operation in Mode 3. Attachment 1 also shows the TS LCO 3.0.7 proposed markup reflecting this additional paragraph and Attachment 2 shows the revised page.

Attachment 3 provides a marked-up TS Bases for information only.

3. TECHNICAL EVALUATION

The proposed COL Condition and TS LCO 3.0.7 changes do not represent changes to the design or construction of the plant. The proposed changes to COL Condition 2.D(9) and COL Appendix A TS LCO 3.0.7 would allow excluding certain TS requirements where the Applicability is stated as including one or more of Modes 4, 5, and 6 (or portions thereof) until the unit achieves initial criticality. The requirements that become effective at the 10 CFR 52.103(g) finding and not proposed to be excluded include those providing protections related to core reactivity (i.e., maintaining the reactor covered with borated water sufficient to maintain shutdown margin (SDM) as required by TS 3.1.1, “SDM") and reactor vessel integrity, as well as requirements for unirradiated fuel assembly storage.

TS LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. The currently approved TS LCOs reflect the functional capability and performance levels of equipment that are based on worse-case assumptions reflecting the full range of the 40-year operating licensed period, which includes maximum fission product inventory, maximum decay heat, and maximum approved operating power levels. However, while operating in Modes 4, 5, and 6, and prior to initial criticality, safe operation of the facility is met with reduced functional capability and performance levels of equipment.

The term “initial criticality” is a commonly used term in the nuclear industry to refer to the time at which the reactor is first made critical. A reactor achieves criticality (and is said to be critical) when each fission event releases a sufficient number of neutrons to sustain an ongoing series of reactions. Initial criticality is an important milestone in the construction and commissioning of a nuclear power plant. Initial criticality is referred to repeatedly throughout the licensing basis documents, including the Combined License and UFSAR, and its meaning is unambiguous, as there is a single defined point at which the reactor first reaches criticality.

ESF systems, and the associated supporting PMS and DAS actuation instrumentation and controls, are designed to provide core cooling and injection for DBAs in order to protect the public in the event of an accidental release of radioactive fission products from the reactor coolant system. The ESF systems function to localize, control, mitigate, and terminate such accidents and to maintain radiation exposure levels to the public below applicable limits and guidelines.

Plant-specific design performance criteria (e.g., to meet 10 CFR 50.46) involves protections from potential impacts that are associated with excessive core heating or decay heat or potential release of fission products. However, prior to initial criticality, with unirradiated assemblies, there is no core heat or decay heat present. Further, as the fuel is unirradiated, no fission products are available, so there would be no radiological consequences if certain ESF systems, and associated PMS and DAS actuations, were to be unavailable while operating in Modes 4, 5, and 6, and prior to achieving initial criticality. As such, no DBA or AOO acceptance criteria are challenged even when certain ESF systems are not credited. No exclusions are proposed for TS requirements applicable in Mode 3.

To continue to provide the appropriate lowest functional capability or performance levels of equipment required for safe operation of the facility, the COL Condition 2.D(9) proposed exceptions for the effectiveness of TS LCOs while operating in Modes 4, 5, and 6 (or portions thereof) do not exclude requirements related to core reactivity, reactor vessel integrity protections, and protections for maintaining the containment vessel from exceeding its design pressure, as well as not excluding requirements for unirradiated fuel assembly storage. These protections provide appropriate defense in depth for safe operation of the facility prior to initial criticality.

The specific TS LCOs that are proposed for delayed effectiveness while operating in Modes 4, 5, and 6, and prior to initial criticality, as well as the supporting PMS and DAS instrumentation and controls, are justified in sections below.

3.1 Modes 5 and 6 Exclusions

During Mode 5 and 6 with no irradiated fuel in the core, maintaining the reactor core covered with borated water provides adequate shutdown margin (SDM) in accordance with TS 3.1.1, "SDM." With RCS water level at the lowest reactor vessel penetration (i.e., direct vessel injection [DVI] nozzle) adequate core coverage is provided without using ESF passive injection systems. There are no penetrations below the level of the DVI nozzle that could lead to RCS water level dropping below the DVI nozzle. The elevation of the top of the active core region is approximately 4 ft below the bottom of the DVI nozzle.

SNC has determined that additional makeup from the IRWST is not necessary to replace the potential reactor coolant volume lost from a loss of RCS inventory event to continue to maintain the reactor core adequately covered with borated water and provide continued compliance with the required SDM. As such, IRWST requirements of TS 3.5.7 and TS 3.5.8 are proposed to be excluded in Modes 5 and 6, and, therefore, with no need for passive IRWST injection, there is also no need for ADS to vent the RCS to allow passive IRWST injection. Similarly, the IRWST support function of TS 3.7.13 is also not required to perform a safety function in Mode 6.

Furthermore, with no core decay heat the decay heat removal safety function of PRHR HX for Mode 5 (TS 3.5.5) is not required.

3.1.1 Mode 5 with the RCS "Vented" and Mode 6 Exclusions

In accordance with the TS definition, "vented" is the condition when all ADS stage 1, 2, and 3 (or equivalent vent area) are open per the requirements of LCO 3.4.13, "Automatic Depressurization System (ADS) - Shutdown, RCS Open." In Mode 5 with the RCS "vented" and in Mode 6 the RCS coolant is maintained under subcooled conditions. This provides assurance that the evaporative losses in the RCS would not be sufficient to uncover the core. The low evaporation rate also assures that the boron concentration in the core would not increase to the precipitation limit.

3.1.2 Mode 5 with the RCS Not "Vented" Exclusions

In Mode 5 with the RCS not "vented," RCPs could be running (adding heat) and pressurizer heaters could be energized. To account for these potential sources of heat, and potential for reduction of reactor coolant inventory from flashing, the CMT requirements of TS 3.5.3 for Mode 5 not "vented," as well as the Pressurizer Level – Low 2 actuation signal, are not proposed for exclusion. The Pressurizer Level – Low 2 actuation will trip the RCPs if running and the CMT actuation will trip the pressurizer heaters if energized. CMT actuation will also conservatively provide additional RCS inventory.

3.2 Mode 4 Exclusions

The potential for steamline break (SLB) mass and energy release inside containment could result in increased containment pressures that need consideration for maintaining containment pressure below its design pressure to provide continued protection of the reactor coolant pressure boundary (RCPB) and/or do not violate the existing environmental qualification (EQ) envelopes. The only SLB accident mitigation function proposed for exclusion in Mode 4 that is credited in the current license basis analyses is the PCS (TS 3.6.6). In the absence of decay heat and with adequate SDM (TS 3.1.1), the availability of PCS is not required to mitigate a Mode 4 SLB. PRHR HX (TS 3.5.4 and TS 3.5.5 for Mode 4) remains required, and once actuated on Containment Pressure – High 2, the RCS temperature would rapidly decrease, limiting the release of energy into containment to the faulted SG. In this event, even without PCS cooling the containment pressure and temperature remain bounded by current analysis (which is based on conditions for Modes 1 and 2). Therefore, the containment design pressure and EQ envelopes would not be challenged.

The requirement for PORV and PORV block valves (TS 3.7.10) are not required to perform the safety function to isolate following a SGTR to minimize radiological releases into the atmosphere since there are no fission products in Mode 4 prior to initial criticality.

In Mode 4 there are various TS LCO Applicabilities that have break-points (i.e., different requirements above or below the break-point) that either involve whether RCS cooling is being provided by the Normal Residual Heat Removal System (RNS) or not, or involve whether all four cold leg temperatures are > 275°F or any cold leg temperature is ≤ 275°F. For this proposed amendment, certain Mode 4 TS requirements are proposed to be excluded with any cold leg temperature ≤ 275°F and when all four cold leg temperatures are > 275°F the exclusion is not proposed. This approach is followed even when the specific TS Applicability might utilize whether RCS cooling is being provided by RNS or not.

When any cold leg temperature is ≤ 275°F the requirements of TS 3.4.14, LTOP, are imposed requiring two RNS suction relief valves to be operable (or alternatively the RCS depressurized). Aligning RNS to support this requirement for operable RNS relief valves is effectively considered sufficient to meet the TS Applicability of "RCS cooling provided by RNS" during the prior-to-initial-criticality period where there is no decay heat and where there is the availability of relief valves to accommodate any overheating. Therefore, while RNS could continue to be aligned when all four cold leg temperatures are > 275°F, certain exclusions would not be proposed to be applicable based on the LTOP applicability being exited. Utilizing the break-point consistent with the Mode 4 LTOP requirements provides a more uniform and conservative application.

TS Applicabilities that use the transition point of RCS not being cooled by the RNS use that transition as indicative of the RCS being at a high pressure condition when accidents such as LOCAs or SGTRs would be postulated. Conversely, the required alignment of LTOP protections when any cold leg temperature is ≤ 275°F ensures that a high pressure condition would not exist.

In addition, some TS Applicabilities that use RCS not being cooled by the RNS are intended to address conditions in which the SGs are used as heat sinks. The ≤ 275°F cold leg temperature threshold also conservatively addresses that condition since at or below 275°F, the SGs would not be effective heat sinks due to the low saturation pressure.

Therefore, while RNS could continue to be aligned when all four cold leg temperatures are > 275°F, certain exclusions would not be proposed to be applicable based on the LTOP applicability being exited. Utilizing the break-point consistent with the Mode 4 LTOP requirements provides a more uniform and conservative application.

While operating in MODE 4 where the average reactor coolant temperature could be between 200°F and 420°F, a sudden breach of the RCPB could lead to loss of reactor coolant from flashing. Specific evaluations presented below were conducted to address the necessary requirements to maintain the reactor core adequately covered with borated water and provide continued compliance with the required SDM.

3.2.1 Mode 4 When Any Cold Leg Temperature is ≤ 275°F Exclusions

A sudden breach of the RCPB could lead to loss of reactor coolant from flashing, which could lead to the need for a makeup source to provide continued reactor coolant coverage of the reactor fuel with borated water sufficient to maintain SDM. To address this potential, the requirement for CMT operability, in accordance with TS 3.5.2 and TS 3.5.3, will not be excluded. SNC has determined that when any cold leg temperature is ≤ 275°F the injection volume from one CMT (as required by TS 3.5.3) is sufficient to replace the potential reactor coolant volume lost from flashing and evaporation in the event of a worst case breach of the RCPB to continue to maintain the reactor core adequately covered with borated water and provide continued compliance with the required SDM. Since no LOCAs are postulated MODE 4 with the RCS cooling provided by RNS, the possibility of a break in the direct vessel injection line or CMT injection line is not considered. As a result, only one CMT is required to be available to provide core coverage in response to postulated events. The two parallel CMT outlet isolation valves ensure that injection from one CMT occurs in the event of a single active failure.

When any cold leg temperature is ≤ 275°F SNC has determined that additional makeup from the Accumulators and IRWST is not necessary to replace the potential reactor coolant volume lost from flashing and evaporation in the event of a worst case breach of the RCPB to continue to maintain the reactor core adequately covered with borated water and provide continued compliance with the required SDM. As such, Accumulator requirements of TS 3.5.1, and IRWST requirements of TS 3.5.6 for Mode 4 when any cold leg temperature is ≤ 275°F, are proposed to be excluded. Therefore, with no need for passive IRWST injection, there is also no need for ADS to vent the RCS to allow passive injection. Note that while the IRWST injection safety function is proposed to be excluded in Mode 4 when any cold leg temperature is ≤ 275°F, the PRHR HX function continues to be required to be operable. As such, maintaining PRHR HX operability would continue to implicitly require the availability of IRWST volume and temperature requirements as found in Surveillance Requirements 3.5.6.1 and 3.5.6.2.

3.2.2 Mode 4 With All Four Cold Leg Temperatures > 275°F Exclusions

When all four cold leg temperatures are > 275°F (i.e., above the LTOP Applicability) the injection volume from one CMT and the IRWST is available for the potential reactor coolant volume lost from flashing and evaporation in the event of a worst case breach of the RCPB. When all four cold leg temperatures are > 275°F, consideration for potential single failure to preclude injection from one CMT is accounted for by requiring both CMTs be operable in accordance with TS 3.5.2.

SNC has determined that additional makeup from the Accumulators is not necessary to replace the potential reactor coolant volume lost from flashing and evaporation in the event of a worst case breach of the RCPB to continue to maintain the reactor core adequately covered with borated water and provide continued compliance with the required SDM.

Additionally, when all four cold leg temperatures are > 275°F the injection volume from the IRWST will continued to also be required. Together, the CMT and IRWST volumes will continue to maintain the reactor core adequately covered with borated water and provide continued compliance with the required SDM. However, the events that lead to maintaining the makeup capability of the IRWST in Mode 4 when all four cold leg temperatures are > 275°F (TS 3.5.6) are depressurization events and as such, do not rely on the need for ADS to vent the RCS to allow passive injection. While the requirements for ADS are excluded, since the IRWST actuation depends on ADS actuation signals, those ADS actuation signals are not proposed to be excluded in Mode 4 when all four cold leg temperatures are > 275°F.

3.3 Other Modes 4, 5, and 6 Exclusions

Certain TS LCO where the Applicability is stated as one or more of Modes 4, 5, and 6 (or portions thereof) do not perform a safety function at any time prior to achieving initial criticality and are evaluated below.

<u>Protection and Safety Monitoring System (PMS) and Diverse Actuation System (DAS)</u>

For ESF functions proposed for exclusion one or more of Modes 4, 5, and 6 (or portions thereof) prior to initial criticality the associated supporting actuation signals are similarly proposed to be excluded.

The actuation of IRWST depends on ADS actuations (which in turn depend on CMT level actuations following CMT injection). While the mechanical ADS functions are not required to perform a safety function (as discussed previously), the signals to actuate ADS will remain required in Mode 4 with all four cold leg temperatures > 275°F to support IRWST actuation.

Where automatic actuations are proposed to be excluded, the corresponding manual actuations are also proposed to be excluded. However, in the case of ADS, where manual IRWST actuations are provided (and remain required), the corresponding ADS manual actuations (and interlocks) are proposed to be excluded.

Therefore, the following PMS and DAS support functions are proposed to be excluded in one or more of Modes 4, 5, and 6 (or portions thereof) as described in Section 2.4, Description of the Proposed Change, and shown in Attachment 1 and Attachment 2:

- TS 3.3.8, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation,” Table 3.3.8-1,
    - Function 3, Containment Radioactivity – High
    - Function 14, RCS Wide Range Pressure – Low
    - Function 15, Core Makeup Tank (CMT) Level – Low 3
    - Function 16, CMT Level – Low 6
    - Function 18, IRWST Lower Narrow Range Level – Low 3
    - Function 19, Reactor Coolant Pump Bearing Water Temperature – High 2

- TS 3.3.9, "ESFAS Manual Initiation," Table 3.3.9-1
    - Function 3, Containment Isolation - Manual Initiation
    - Function 6, ADS Stages 1, 2 & 3 Actuation – Manual Initiation
    - Function 7, ADS Stage 4 Actuation – Manual Initiation
    - Function 8, Passive Containment Cooling Actuation – Manual Initiation
    - Function 9, Passive Residual Heat Removal Heat Exchanger Actuation - Manual Initiation
    - Function 12, IRWST Injection Line Valve Actuation – Manual Initiation
    - Function 13, IRWST Containment Recirculation Valve Actuation – Manual Initiation
    - Function 14, SG Power Operated Relief Valve and Block Valve Isolation – Manual Initiation
- TS 3.3.10, "ESFAS Reactor Coolant System (RCS) Hot Leg Level Instrumentation"
- TS 3.3.13, ESFAS Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization, Table 3.3.13-1 Function 1, Main Control Room Air Supply Iodine or Particulate Radiation – High 2
- TS 3.3.14, ESFAS In-containment Refueling Water Storage Tank (IRWST) and Spent Fuel Pool Level Instrumentation
    - Function 1, Spent Fuel Pool Level – Low 2
- TS 3.3.19, "DAS Manual Controls"
    - Function 2, Passive Residual Heat Removal Heat Exchanger (PRHR HX) control and In-Containment Refueling Water Storage Tank (IRWST) gutter control valves
    - Function 4, Automatic Depressurization System (ADS) stage 1 valves
    - Function 5, ADS stage 2 valves
    - Function 6, ADS stage 3 valves
    - Function 7, ADS stage 4 valves,
    - Function 8, IRWST injection squib valves
    - Function 9, Containment recirculation valves
    - Function 10, Passive containment cooling drain valves
    - Function 11, Selected containment isolation valves
- TS 3.3.20, "ADS and IRWST Injection Blocking Device" Table 3.3.20-1 Function 2, ADS and IRWST Injection Block Switches for Manual Unblocking

<u>Containment</u>

In the absence of fission products, the containment does not perform a safety function as a fission product barrier. Furthermore, containment closure for the purpose of maintain the cooling water inventory within containment to support long term passive core cooling recirculation is determined to not be required. For the specific Modes as described above, the requirements for CMT injection and IRWST injection have been shown to be adequate to maintain the reactor core covered with borated water to provide adequate shutdown margin (SDM) in accordance with TS 3.1.1, “SDM,” without the need for containment to collect spilling inventory and provide recirculation.

Therefore, the containment barrier requirements of TSs 3.6.1, 3.6.2, 3.6.3, and 3.6.7, and the supporting PMS and DAS containment isolation functions, can be excluded from becoming effective while operating in one or more of Modes 4, 5, and 6 (or portions thereof).

Main Control Room (MCR) Emergency Habitability System (VES) Radiological Controls

The VES requirements that involve protection from the release of radioactivity (and support actuation) would not be required while operating in Mode 4 until initial criticality since there are no fission products to be released that would challenge MCR habitability. The COL proposed exclusions for TS 3.7.6, “Main Control Room Habitability System (VES)” for the radiological protection functions involve excluding:

- TS 3.3.13, “ESFAS Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization,” Table 3.3.13-1, Function 1, Main Control Room Air Supply Iodine or Particulate Radiation – High 2
- Surveillance Requirements (SR) 3.7.6.9, “Perform required MCRE unfiltered air inleakage testing in accordance with the Main Control Room Envelope Habitability Program” and SR 3.7.6.10, “Perform required VES Passive Filtration system filter testing in accordance with the Ventilation Filter Testing Program (VFTP)”
- TS 5.5.12, “Main Control Room Envelope Habitability Program”
- TS 5.5.13, “Ventilation Filter Testing Program (VFTP)”

Fission Product Monitoring and Control Not Required Prior to Criticality

The following TS LCOs with Applicability including Mode 4 would not be required until initial criticality. With no fission products in the RCS or secondary system, monitoring for them provides no safety benefit. With no fission products, pH adjustment to enhance iodine retention in the containment water provides no safety benefit.

- TS 3.6.8, “pH Adjustment”
- TS 3.7.4, “Secondary Specific Activity”

TS Administrative Controls Not Required Prior to Criticality

The following TS Administrative Controls would not be required until initial criticality. These Programs and Reports address mitigation or monitoring of doses produced as a result of plant operation after criticality. Deferring these Programs and reports until initial criticality (i.e., prior to generating fission products) is consistent with the intended basis for these Programs and Reports not being required prior to the 10 CFR 52.103(g) finding. Requiring them prior to initial criticality provides no safety benefit.

- TS 5.5.1, “Offsite Dose Calculation Manual (ODCM)”
- TS 5.5.2, “Radioactive Effluent Control Program”
- TS 5.6.1, “Annual Radiological Environmental Operating Report”
- TS 5.6.2, “Radioactive Effluent Release Report”
- TS 5.7, “High Radiation Area”

3.4 Conclusions

The proposed changes to COL Condition 2.D(9) and TS LCO 3.0.7 provide appropriate continued protections for core reactivity and for reactor vessel integrity to provide for continued safe operation of the facility.

In conclusion, if there are no irradiated assemblies in the core, there would be no decay heat generated by the fuel in the core and no fission products that could be released. Therefore, while operating in Modes 4, 5, and 6, and prior to initial criticality, the proposed deferred effectiveness for certain TS LCOs at the 10 CFR 52.103(g) finding are requirements that do not perform a required safety function. The proposed changes do not represent changes to the design or construction of the plant. The proposed change has no adverse effect on the UFSAR accident analysis prior to initial criticality and will continue to provide reasonable assurance that the health and safety of the public will not be endangered.

The proposed TS changes continue to satisfy the requirements of 10 CFR 50.36(c)(2)(i) because they continue to define the lowest functional capability or performance levels of equipment required for safe operation of the facility.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.36(c)(2)(i), requires that the “Limiting conditions for operation [LCOs] are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications.”

10 CFR 52.98(c) requires NRC approval for any modification to, addition to, or deletion from the terms and conditions of a COL. This activity involves changes to a Condition of the COL and to COL Appendix A Technical Specifications; therefore, this activity requires an amendment to the COL. Accordingly, NRC approval is required prior to making the plant-specific changes in this license amendment request.

10 CFR Part 52, Appendix D, VIII.C.6, states that after issuance of a license, “Changes to the plant-specific TS will be treated as license amendments under 10 CFR 50.90.” 10 CFR 50.90 addresses the application for amendment of a license, including a combined license. The proposed LAR requires changes in the TS, and therefore an LAR is required to be submitted for NRC approval.

10 CFR 50, Appendix A, General Design Criterion (GDC) 34 requires the plant design to include a system to remove residual heat from the reactor core so specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. As the requested amendment revises the initial effectiveness of various Technical Specifications for conditions while operating in Modes 4, 5, and 6, and prior to initial criticality, when there is no decay heat present, the change adequately satisfies the requirements of GDC 34.

10 CFR 50, Appendix A, GDC 20, Protection system functions, requires the protection system to be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. The Protection and Safety Monitoring System (PMS) continues to satisfy this design criteria since the PMS changes align

with the change to the Actuated systems, such that PMS continues to sense accident conditions and to initiate the operation of systems and components important to safety.

4.2 Precedent

1. NRC Issuance of Amendment Regarding Technical Specification Operability Requirements for Automatic Depressurization System Stage 4, Amendment No. 189 to Combined License (COL) No. NPF-91 for the Vogtle Electric Generating Plant, Unit 3, dated January 13, 2023 (ML23013A214).
2. NRC Issuance of Amendment Regarding Technical Specification Operability Requirements for IRWST Operability Prior to Initial Criticality, Amendment No. 190 to Combined License (COL) No. NPF-91 for the Vogtle Electric Generating Plant, Unit 3, dated February 8, 2023 (ML23037A082).

4.3 No Significant Hazards Consideration Determination Analysis

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC) requests an amendment to the combined license (COL) for Vogtle Electric Generating Plant (VEGP) Unit 4 (COL Number NPF-92). The requested amendment would modify the current COL Condition 2.D(9), Technical Specifications (TS), to limit the scope of TS that become effective upon a Commission finding in accordance with 10 CFR 52.103(g), while operating in Modes 4, 5, and 6 prior to initial criticality. At Unit 4 initial criticality the Technical Specifications in Appendix A to the COL are proposed become fully effective. The proposed change also includes revision to COL Appendix A TS Limiting Condition for Operation (LCO) 3.0.7 to coordinate the TS compliance provisions proposed in the COL Condition.

The 10 CFR 52.103(g) finding determines that inspections, tests, analyses, and acceptance criteria (ITAAC) have been met, and authorizes the licensee to operate the facility (i.e., commence loading fuel into the reactor core). However, currently COL Condition 2.D(9) makes all TS requirements effective at this time. The full scope of the current TS Limiting Conditions for Operability (LCOs) require the lowest functional capability and performance levels of equipment based on worse-case assumptions reflecting the full range of the 40 year operating licensed period, which includes maximum fission product inventory, maximum decay heat, and maximum approved operating power levels. However, while operating in Modes 4, 5, and 6, and prior to initial criticality, safe operation of the facility is met with reduced functional capability and performance levels of equipment, since there is no fission product inventory, no decay heat, and no nuclear fission power being generated.

SNC has evaluated whether a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes do not adversely affect the operation of any structures, systems, or components (SSCs) associated with an accident initiator or initiating sequence of events. The proposed changes do not affect the design of Engineered Safety Systems (ESF), or the associated Protection and Safety Monitoring System (PMS) and Diverse Actuation System (DAS) instrumentation and controls.

The proposed amendment does not affect accident initiators or precursors nor adversely alter the design assumptions, conditions, and configuration of the facility. The proposed amendment does not alter any plant equipment or operating practices with respect to such initiators or precursors in a manner that the probability of an accident is increased. The proposed amendment will not alter assumptions relative to the mitigation of an accident or transient event, as these assumptions are based upon irradiated fuel for the associated accident or transient. The proposed amendment does not increase the likelihood of the malfunction of a system, subsystem, or component (SSC) or impact analyzed accidents.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment does not introduce any new or unanalyzed modes of operation. The proposed changes do not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the assumptions made in the safety analysis, as these assumptions are based upon irradiated fuel for the associated accident or transient.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The margin of safety is related to the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system, and the containment. The performance of these fission product barriers is not affected by the proposed amendment as the proposed changes apply only when there are no fission products in the reactor core; therefore, the margins to the onsite and offsite radiological dose limits are not significantly reduced.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATION

The proposed changes to the Technical Specifications (TS) are described in Section 2.5 of this Enclosure.

A review has determined that the proposed changes require an amendment to the COL. A review of the anticipated construction and operational effects of the requested amendment has determined that the requested amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), in that:

(i) There is no significant hazards consideration.

As documented in Section 4.3, No Significant Hazards Consideration Determination Analysis, of this license amendment request, an evaluation was completed to determine whether or not a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment.” The Significant Hazards Consideration evaluation determined that (1) the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) the proposed amendment does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of “no significant hazards consideration” is justified.

(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed changes are unrelated to any aspect of plant construction or operation that would introduce any change to effluent types (e.g., effluents containing chemicals or biocides, sanitary system effluents, and other effluents) or affect any plant radiological or non-radiological effluent release quantities. Furthermore, the proposed changes do not affect any effluent release path or diminish the functionality of any design or operational features that are credited with controlling the release of effluents during plant operation. Therefore, it is concluded that the proposed amendment does not involve a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change in the requested amendment does not affect the shielding capability of, or alter any walls, floors, or other structures that provide shielding. Plant radiation zones and controls under 10 CFR 20 preclude a significant increase in occupational radiation exposure. Therefore, the proposed amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above review of the proposed amendment, it has been determined that anticipated construction and operational effects of the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in

10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

1. NRC Issuance of Amendment Regarding Technical Specification Operability Requirements for Automatic Depressurization System Stage 4, Amendment No. 189 to Combined License (COL) No. NPF-91 for the Vogtle Electric Generating Plant, Unit 3, dated January 13, 2023 (ML23013A214).
2. NRC Issuance of Amendment Regarding Technical Specification Exceptions for In-containment Refueling Water Storage Tank (IRWST) Operability Prior to Initial Criticality, Amendment No. 190 to Combined License (COL) No. NPF-91 for the Vogtle Electric Generating Plant, Unit 3, dated February 8, 2023 (ML23037A082).

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Attachment 1
to the Enclosure of NL-23-0135

COL Condition 2.D(9) and Technical Specification Markups

Insertions denoted by underlined Blue text.
Deletions denoted by ~~Red Strikeout text~~
Omitted text is identified by three asterisks (* * *)

(This Attachment consists of seven pages, including this cover page)

Combined License (COL) Condition 2.D

D. The license is subject to, and SNC shall comply with, the conditions specified and incorporated below:

* * *

(8) Incorporation

The Technical Specifications, Environmental Protection Plan, and ITAAC in Appendices A, B, and C, respectively of this license, as revised through Amendment No. XXX, are hereby incorporated into this license.

(9) The technical specifications in Appendix A to this license (TS) become effective upon a Commission finding that the acceptance criteria in this license (ITAAC) are met in accordance with 10 CFR 52.103(g) with the following exceptions:

(a) Prior to initial criticality of the reactor core while operating in plant operational Mode 5 (Cold Shutdown) or Mode 6 (Refueling) the following TS are temporarily excluded from becoming effective:

- TS 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," Table 3.3.8-1,
  - Function 14, RCS Wide Range Pressure – Low
  - Function 15, Core Makeup Tank (CMT) Level – Low 3
  - Function 16, CMT Level – Low 6
  - Function 18, IRWST Lower Narrow Range Level – Low 3
- TS 3.3.9, "ESFAS Manual Initiation," Table 3.3.9-1
  - Function 6, ADS Stages 1, 2 & 3 Actuation – Manual Initiation
  - Function 7, ADS Stage 4 Actuation – Manual Initiation
  - Function 8, Passive Containment Cooling Actuation – Manual Initiation
  - Function 9, Passive Residual Heat Removal Heat Exchanger Actuation - Manual Initiation
  - Function 12, IRWST Injection Line Valve Actuation – Manual Initiation
  - Function 13, IRWST Containment Recirculation Valve Actuation – Manual Initiation
- TS 3.3.10, "ESFAS Reactor Coolant System (RCS) Hot Leg Level Instrumentation"
- TS 3.3.14, "ESFAS In-containment Refueling Water Storage Tank (IRWST) and Spent Fuel Pool Level Instrumentation," Table 3.3.14-1
  - Function 1, Spent Fuel Pool Level – Low 2

- TS 3.3.19, "DAS Manual Controls," Table 3.3.19-1
    - Function 2, Passive Residual Heat Removal Heat Exchanger (PRHR HX) control and In-Containment Refueling Water Storage Tank (IRWST) gutter control valves
    - Function 4, Automatic Depressurization System (ADS) stage 1 valves
    - Function 5, ADS stage 2 valves
    - Function 6, ADS stage 3 valves
    - Function 7, ADS stage 4 valves,
    - Function 8, IRWST injection squib valves
    - Function 9, Containment recirculation valves
    - Function 10, Passive containment cooling drain valves
    - Function 11, Selected containment isolation valves
- TS 3.3.20, "ADS and IRWST Injection Blocking Device," Table 3.3.20-1
    - Function 2, ADS and IRWST Injection Block Switches for Manual Unblocking
- TS 3.4.12, "ADS – Shutdown, RCS Intact"
- TS 3.4.13, "ADS – Shutdown, RCS Open"
- TS 3.5.5, "PRHR HX – Shutdown, RCS Intact"
- TS 3.5.7, "IRWST – Shutdown, MODE 5"
- TS 3.5.8, "IRWST – Shutdown, MODE 6"
- TS 3.6.7, "Containment Penetrations"
- TS 3.7.13, "Spent Fuel Pool Cooling System (SFS) Containment Isolation Valves"
- TS 5.5, "Programs and Manuals"
    - TS 5.5.1, "Offsite Dose Calculation Manual (ODCM)"
    - TS 5.5.2, "Radioactive Effluent Control Program"
    - TS 5.5.12, "Main Control Room Envelope Habitability Program"
    - TS 5.5.13, "Ventilation Filter Testing Program (VFTP)"
- TS 5.6, "Reporting Requirements"
    - TS 5.6.1, "Annual Radiological Environmental Operating Report"
    - TS 5.6.2, "Radioactive Effluent Release Report"
- TS 5.7, "High Radiation Area"

(b) Prior to initial criticality of the reactor core while operating in plant operational Mode 4 (Safe Shutdown) when any cold leg temperature is ≤ 275°F the following TS are temporarily excluded from becoming effective:

- TS 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," Table 3.3.8-1,
  - Function 14, RCS Wide Range Pressure – Low
  - Function 15, Core Makeup Tank (CMT) Level – Low 3
  - Function 16, CMT Level – Low 6
  - Function 18, IRWST Lower Narrow Range Level – Low 3
  - Function 19, Reactor Coolant Pump Bearing Water Temperature – High 2
- TS 3.3.9, "ESFAS Manual Initiation," Table 3.3.9-1
  - Function 3, Containment Isolation - Manual Initiation
  - Function 6, ADS Stages 1, 2 & 3 Actuation – Manual Initiation
  - Function 7, ADS Stage 4 Actuation – Manual Initiation
  - Function 8, Passive Containment Cooling Actuation – Manual Initiation
  - Function 12, IRWST Injection Line Valve Actuation – Manual Initiation
  - Function 13, IRWST Containment Recirculation Valve Actuation – Manual Initiation
- TS 3.3.13, "ESFAS Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization," Table 3.3.13-1
  - Function 1, Main Control Room Air Supply Iodine or Particulate Radiation – High 2
- TS 3.3.19, "DAS Manual Controls," Table 3.3.19-1
  - Function 4, Automatic Depressurization System (ADS) stage 1 valves
  - Function 5, ADS stage 2 valves
  - Function 6, ADS stage 3 valves
  - Function 7, ADS stage 4 valves,
  - Function 8, IRWST injection squib valves
  - Function 9, Containment recirculation valves
  - Function 10, Passive containment cooling drain valves
  - Function 11, Selected containment isolation valves
- TS 3.3.20, "ADS and IRWST Injection Blocking Device," Table 3.3.20-1
  - Function 2, ADS and IRWST Injection Block Switches for Manual Unblocking
- TS 3.4.11, "Automatic Depressurization System (ADS) – Operating"
- TS 3.5.1, "Accumulators"

- TS 3.5.6, "IRWST – Operating"
- TS 3.6.1, "Containment"
- TS 3.6.2, "Containment Air Locks"
- TS 3.6.3, "Containment Isolation Valves"
- TS 3.6.6, "Passive Containment Cooling System (PCS)"
- TS 3.6.8, "pH Adjustment"
- TS 3.7.4, "Secondary Specific Activity"
- TS 3.7.6, "Main Control Room Habitability System (VES)" for the radiological protection functions
- TS 3.7.10, "Steam Generator (SG) Isolation Valves" only for PORV and PORV block valves (SG blowdown isolation valve is not excluded)
- TS 5.5, "Programs and Manuals"
    - TS 5.5.1, "Offsite Dose Calculation Manual (ODCM)"
    - TS 5.5.2, "Radioactive Effluent Control Program"
    - TS 5.5.12, "Main Control Room Envelope Habitability Program"
    - TS 5.5.13, "Ventilation Filter Testing Program (VFTP)"
- TS 5.6, "Reporting Requirements"
    - TS 5.6.1, "Annual Radiological Environmental Operating Report"
    - TS 5.6.2, "Radioactive Effluent Release Report"
- TS 5.7, "High Radiation Area"

(c) Prior to initial criticality of the reactor core while operating in plant operational Mode 4 (Safe Shutdown) with all four cold leg temperatures > 275°F the following TS are temporarily excluded from becoming effective:

- TS 3.3.8, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," Table 3.3.8-1,
    - Function 3, Containment Radioactivity – High
    - Function 18, IRWST Lower Narrow Range Level – Low 3
    - Function 19, Reactor Coolant Pump Bearing Water Temperature – High 2
- TS 3.3.9, "ESFAS Manual Initiation," Table 3.3.9-1
    - Function 3, Containment Isolation - Manual Initiation
    - Function 6, ADS Stages 1, 2 & 3 Actuation – Manual Initiation
    - Function 7, ADS Stage 4 Actuation – Manual Initiation
    - Function 8, Passive Containment Cooling Actuation – Manual Initiation
    - Function 13, IRWST Containment Recirculation Valve Actuation – Manual Initiation

  - Function 14, SG Power Operated Relief Valve and Block Valve Isolation – Manual Initiation except for SG blowdown valve isolation
- TS 3.3.13, "ESFAS Main Control Room Isolation, Air Supply Initiation, and Electrical Load De-energization," Table 3.3.13-1
  - Function 1, Main Control Room Air Supply Iodine or Particulate Radiation – High 2
- TS 3.3.19, "DAS Manual Controls"
  - Function 4, Automatic Depressurization System (ADS) stage 1 valves
  - Function 5, ADS stage 2 valves
  - Function 6, ADS stage 3 valves
  - Function 7, ADS stage 4 valves,
  - Function 9, Containment recirculation valves
  - Function 10, Passive containment cooling drain valves
  - Function 11, Selected containment isolation valves
- TS 3.4.11, "Automatic Depressurization System (ADS) – Operating"
- TS 3.5.1, "Accumulators"
- TS 3.5.6, "IRWST – Operating" only for containment recirculation flow paths
- TS 3.6.1, "Containment"
- TS 3.6.2, "Containment Air Locks"
- TS 3.6.3, "Containment Isolation Valves"
- TS 3.6.6, "Passive Containment Cooling System (PCS)"
- TS 3.6.8, "pH Adjustment"
- TS 3.7.4, "Secondary Specific Activity"
- TS 3.7.6, "Main Control Room Habitability System (VES)" for the radiological protection functions
- TS 3.7.10, "Steam Generator (SG) Isolation Valves" only for PORV and PORV block valves (SG blowdown isolation valve is not excluded)
- TS 5.5, "Programs and Manuals"
  - TS 5.5.1, "Offsite Dose Calculation Manual (ODCM)"
  - TS 5.5.2, "Radioactive Effluent Control Program"
  - TS 5.5.12, "Main Control Room Envelope Habitability Program"
  - TS 5.5.13, "Ventilation Filter Testing Program (VFTP)"

- TS 5.6, “Reporting Requirements”
  - TS 5.6.1, “Annual Radiological Environmental Operating Report”
  - TS 5.6.2, “Radioactive Effluent Release Report”
- TS 5.7, “High Radiation Area”

* * *

TS LCO 3.0

* * *

LCO 3.0.7 Test Exception LCOs 3.1.8 and 3.1.10 allow specified Technical Specification (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Test Exception LCOs is optional. When a Test Exception LCO is desired to be met but is not met, the ACTIONS of the Test Exception LCO shall be met. When a Test Exception LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall be made in accordance with the other applicable Specifications.

Additionally, for Unit 4 only, Combined License Condition 2.D(9) provides the milestone effectiveness for specified TS requirements prior to becoming fully effective at initial criticality of the reactor core. Compliance with TS requirements that are excluded from becoming effective while operating in MODES 4, 5, and 6 in accordance with the COL Condition, is optional prior to operation in Mode 3.

* * *

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Attachment 2
to the Enclosure of NL-23-0135

COL Condition 2.D(9) and Technical Specification Retyped Pages

(This Attachment consists of seven pages, including this cover page)

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Attachment 3
to the Enclosure of NL-23-0135

Technical Specifications Bases Marked-up Pages
(For Information Only)

Insertions denoted by underlined Blue text.
Deletions denoted by ~~Red Strikeout text~~
Omitted text is identified by three asterisks (* * *)

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TS LCO 3.0 Bases

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LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.8 and 3.1.10 allow specified Technical Specification (TS) requirements to be changed to permit performance of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

Additionally, for Unit 4 only, Combined License (COL) Condition 2.D(9) provides the milestone effectiveness for specified TS requirements commencing with the 10 CFR 52.103(g) finding prior to initial criticality. The TS exclusions listed in the COL Condition are only portions of the TS requirements Applicable in MODES 4, 5, and 6. All TS requirements become fully effective at initial criticality of the reactor core. LCO 3.0.7 recognizing this COL provision, allows changing the TS requirements that are deferred to initial criticality, which would otherwise be required to be met.

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