## Vogtle PEmails

**From:** Gleaves, Bill
**Sent:** Tuesday, May 21, 2019 1:48 PM
**To:** Vogtle PEmails
**Subject:** FW: Emailing: LAR Slides for Emailing to the Public
**Attachments:** Draft LAR-19-003 for May 23rd Meeting (non-sensitive).pdf

Attached is the 3rd slide presentation for May 23rd public meeting after removal of the sensitive pages.

-----Original Message-----
From: Gleaves, Bill
Sent: Tuesday, May 21, 2019 11:06 AM
To: Patel, Chandu (Chandu.Patel@nrc.gov) <Chandu.Patel@nrc.gov>
Cc: Dixon-Herrity, Jennifer <Jennifer.Dixon-Herrity@nrc.gov>
Subject: Emailing: LAR Slides for Emailing to the Public

Chandu,
Attached are the 3 slide sets for sending to the public for this Thursday's meeting. I removed the sensitive information from the 19-003 file.
Billy

| | |
|---|---|
| **Hearing Identifier:** | Vogtle_COL_Docs_Public |
| **Email Number:** | 448 |

**Mail Envelope Properties** (BN7PR09MB29168FF0F0DA68E0922AAF0E9F070)

| | |
|---|---|
| **Subject:** | FW: Emailing: LAR Slides for Emailing to the Public |
| **Sent Date:** | 5/21/2019 1:47:43 PM |
| **Received Date:** | 5/21/2019 1:47:51 PM |
| **From:** | Gleaves, Bill |

**Created By:** Bill.Gleaves@nrc.gov

**Recipients:**
"Vogtle PEmails" <Vogtle.PEmails@nrc.gov>
Tracking Status: None

**Post Office:** BN7PR09MB2916.namprd09.prod.outlook.com

| **Files** | **Size** | **Date & Time** |
|---|---|---|
| MESSAGE | 557 | 5/21/2019 1:47:51 PM |
| Draft LAR-19-003 for May 23rd Meeting (non-sensitive).pdf | | 854950 |

| **Options** | |
|---|---|
| **Priority:** | Standard |
| **Return Notification:** | No |
| **Reply Requested:** | No |
| **Sensitivity:** | Normal |
| **Expiration Date:** | |
| **Recipients Received:** | |

**Southern Nuclear Operating Company**

**ND-19-####**

**Enclosure 1**

**Vogtle Electric Generating Plant (VEGP) Units 3 and 4**

**Request for License Amendment:**

**Reconciliation of Detailed AP1000 Radiation Analyses**

**(LAR-19-003)**



(This Enclosure consists of 19 pages, including this cover page.)

**Table of Contents**

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC) hereby requests an amendment to Combined License (COL) Nos. NPF-91 and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively.

## 1. SUMMARY DESCRIPTION

The proposed change would revise the COLs to incorporate the contribution of design basis passive residual heat removal (PRHR) heat exchanger (HX) leakage to the in-containment refueling water storage tank (IRWST) into normal operating doses. The change to normal operating doses involves crediting the north-east wall and west wall of the IRWST as radiation shielding walls in Plant-specific Tier 1 (and associated COL Appendix C) Table 3.3-1.

The requested amendment requires a departure from Updated Final Safety Analysis Report (UFSAR) Tier 2 information that involves a change to the COL Appendix C (and plant-specific Tier 1) information in Table 3.3-1 identifying definitions of wall thicknesses and applicable radiation shielding walls for nuclear island buildings. This enclosure requests approval of the license amendment necessary to implement these changes. All discussions of changes to COL Appendix C are also understood to impact the corresponding plant-specific Tier 1 Table 3.3-1.

## 2. DETAILED DESCRIPTION

COL Appendix C Section 3.3 contains ITAAC for the nuclear island structures, including the containment internal structures. COL Appendix C Table 3.3-1 defines the walls and floors of the nuclear island structures, including the containment internal structure, that provide shielding during normal operations; this table currently states that the north-east wall and west wall of the IRWST are not applicable radiation shielding walls during normal operations.

<u>Design Function Related to Activity</u>

Radioactive fission products are generated within the core, which have the potential of leaking to the reactor coolant system (RCS) by way of defects in the fuel cladding. The core neutron flux also results in activation of the coolant and corrosion products in the RCS. As discussed in UFSAR Section 11.1, "Source Terms," two source terms are presented for the primary and the secondary coolant. The first is a conservative, or design basis, source term that assumes the design basis fuel defect level (0.25 percent fuel defect). This source term serves as a basis for system design and shielding requirements. The second source term is an expected model. This source term represents the expected average concentrations of radionuclides in the primary and the secondary coolant. The expected results are based on the ANSI/ANS 18.1-1984 source term adjusted for the AP1000 plant.

UFSAR subsection 11.1.1, Design Basis Reactor Coolant Activity, discusses that for the design basis source term it is assumed that there is a significant fuel defect level, well above that anticipated during normal operation. It is assumed that small cladding defects are present in fuel rods producing 0.25 percent of the core power output (also stated as 0.25 percent fuel defects). The defects are assumed to be uniformly distributed throughout the core. The design basis source term based on 0.25 percent fuel defects is used to provide a consistent set of design values for interfaces among the radioactive waste processing systems. The RCS specific activity limit in COL Appendix A Technical Specification (TS) 3.4.10, RCS Specific

Activity, is based upon 0.25 percent fuel defects. In addition, the liquid and gaseous radioactive waste processing systems have the capability to process wastes based upon 1.0 percent fuel defects.

UFSAR subsections 11.2.3 and 11.3.3, Radioactive Releases, describe that releases of radioactivity from the plant may not exceed concentration limits in Title 10 of the Code of Federal Regulation (CFR) Part 20 (10 CFR Part 20), Appendix B nor may the releases result in the annual offsite dose limits specified in 10 CFR Part 50, Appendix I. UFSAR Subsections 11.2.3 and 11.3.3 also describe release estimation, calculation, pathways, and release management.

UFSAR Section 12.3, “Radiation Protection Design Features,” describes specific design features for maintaining personnel exposure as low as reasonably achievable (ALARA). As described in UFSAR subsection 12.3.1.2, “Radiation Zoning and Access Control,” access to areas inside the plant structures and plant yard area is regulated and controlled by posting of radiation signs, control of personnel, and use of alarms and locks. Plant areas are categorized into radiation zones according to design basis radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR Part 20.

During plant operation, access to radiologically restricted areas is through the access control area in the annex building. Rooms and corridors are evaluated for potential radiation sources during normal, shutdown, spent resin transfer, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. Each radiation zone defines the radiation level range expected in the zone. The radiation zone categories employed and zoning for each plant area under normal conditions are shown in UFSAR Figure 12.3-1, “Radiation Zones, Normal Operations/Shutdown,” sheets 1 through 16. Radiation zones shown in the figures are based upon conservative design data. Posting of radiation signs, control of personnel access, and use of alarms and locks are discussed in UFSAR Section 12.5, "Health Physics Facilities Design."

As discussed in UFSAR subsection 12.5.4, “Controlling Access and Stay Time,” high and very high radiation areas are segregated and identified in accordance with 10 CFR Part 20. The entrances to high and very high radiation areas are locked or barricaded and equipped with audible and/or visible alarms, as required. As defined in 10 CFR Part 20, “restricted area means an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials”. Additionally, “radiation area" means an area accessible to personnel in which radiation levels could result in an individual receiving a dose equivalent in excess of 0.005 rem in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.”

As indicated in UFSAR subsection 3.11.4, "Estimated Radiation and Chemical Environment," the plant-specific estimates of the radiation doses incurred by equipment during normal operation are shown in Table 3D.5-2, "60-Year Normal Operating Doses," and the estimated doses following a loss-of-coolant accident are defined in Table 3D.5-5, "Accident Environments."

As stated in UFSAR subsection 3.1.2, “Protection by Multiple Fission Product Barriers,” the reactor core and associated coolant, control, and protection systems are designed such that no fuel damage occurs during normal core operation and operational transients (Condition I) or during transient conditions arising from occurrences of moderate frequency (Condition II). For normal operation, the plant is designed to accommodate a fuel defect level of up to 0.25 percent. Fuel damage, as used here, is defined as penetration of the fission product barrier, that is, the fuel rod cladding. The small number of clad defects that may occur are within the capability of the plant cleanup system and are consistent with the plant design bases.

The Passive Residual Heat Removal heat exchanger (PRHR HX) is located inside the In-containment Refueling Water Storage Tank (IRWST). The PRHR HX receives input directly from the RCS hot leg. The PRHR HX is normally aligned to the RCS with an open inlet valve and closed discharge valves. The IRWST is the heat sink for the PRHR HX. COL Appendix A, Technical Specifications, Limiting Conditions for Operation (LCO) 3.4.7 limits RCS operational leakage from the PRHR HX into the IRWST to 500 gallons per day (gpd). The 500 gpd limit from the PRHR HX is based on the assumption that a single crack leaking this amount would not lead to a PRHR HX tube rupture under the stress condition of an RCS pressure increase event. If leakage is through many cracks, and the cracks are very small, then the above assumption is conservative. This is conservative because the thickness of the PRHR HX tubes is approximately 60 percent greater than the thickness of the steam generator (SG) tubes. Furthermore, a PRHR HX tube rupture would result in an isolable leak and would not lead to a direct release of radioactivity to the atmosphere, the PRHR HX tube area is approximately 4 percent of the area of one steam generator, and the PRHR HX environment is substantially more benign than is the dynamic environment experienced by steam generator tubes. Note that 500 gpd was the original leakage limit for steam generator tubes prior to the issuance of NEI 97-06, Steam Generator Program Guidelines, which reduced the limit for steam generator tubes to 150 gpd per steam generator.

<u>Background and Description of Activity</u>

Previous design analyses neglected leakage of RCS fluid through the PRHR HX to the IRWST, and neglected crud accumulation and buildup following refueling outages . Each of these pathways provides a means for introducing radioactivity into the IRWST. Most notably, leakage through the PRHR HX concurrent with 0.25% fuel defects is listed as acceptable for leakage rates of up to 500 gpd, which is the limit in Technical Specification 3.4.7, RCS Operational Leakage. Introducing 500 gallons/day of reactor coolant into the IRWST is a significant source term that cannot be neglected in design analyses.

Currently, in COL Appendix C, Table 3.3-1, several walls around the IRWST are designated as "No" for radiation shielding (not credited for shielding for normal operation). Some of these walls now need to be credited to prevent additional physical changes or equipment qualification impacts while accounting for the revised fuel defect and PRHR HX to IRWST leakage assumptions.

UFSAR Figure 12.3-1, Sheets 6 and 7 of 16 need to reflect that operation with design basis PRHR HX leakage and fuel defects can lead to radiation fields in excess of the fields shown on the figure.

UFSAR Figure 12.3-1, Sheet 8 of 16 needs to reflect that operation with design basis PRHR HX leakage and fuel defects can lead to radiation fields in excess of the fields shown on the figure in the vicinity of IRWST hatches and vents.

Considering design basis conditions for 60 years inside the IRSWT, a 60-year design basis dose listed in UFSAR Table 3D.5-2, 60-Year Normal Operating Doses, needs to reflect the 60-year gamma dose for the IRWST.

Normally the IRWST is not a significant source of dose to the operating deck above it. However, considering design basis conditions of PRHR HX leakage of 500 gpd (TS 3.4.7) and 0.25% fuel defects (TS 3.4.10) the contribution to the operating deck could be significant. Most of the space above the IRWST is shielded by approximately two feet of concrete and half an inch of steel. However, there are locations where the shielding is reduced, such as maintenance hatches and vents. For the aforementioned design basis conditions, the zoning very near these locations could exceed the current operating deck radiation zoning. This extends to shutdown conditions because the primary contributors to dose are radionuclides in the vapor space of the IRWST that will not decay appreciably within 24 hours.

<u>Proposed Licensing Basis Changes</u>

The change to incorporate the contribution of the design basis PRHR HX leakage into the IRWST necessitates the following proposed changes to the current licensing basis:

A. Credit the 5/8″-thick steel module forming the west wall of the IRWST, and the 2′-6″-thick concrete and steel north-east wall of the IRWST for normal dose shielding. The proposed change requires changing COL Appendix C Table 3.3-1, Definition of Wall Thickness for Nuclear Island Buildings, Turbine Building, and Annex Building, to indicate the north-east and west walls of the IRWST are applicable radiation shielding walls.

B. Calculate the gamma dose rate and 60-year gamma dose for environmental qualification inside the IRWST. The proposed change requires adding the IRWST gamma dose rate and 60-year gamma dose to UFSAR Table 3D.5-2, 60-Year Normal Operating Doses.

C. Changes to UFSAR Figure 12.3-1

  i. UFSAR Figure 12.3-1, Sheet 6, Radiation Zones, Normal Operations/Shutdown Nuclear Island, Elevation 100"-0" & 107'-2"

  The proposed change adds a note to room 11305 (the IRWST) on UFSAR Figure 12.3-1, sheet 6 specifying radiation fields in this area of the 100'-0" & 107'-2" elevations may exceed zone IV for conditions approaching design basis PRHR HX leakage of 500 gpd and 0.25% fuel defects. Continued operation with these design basis conditions can result in dose rates that are designated as Zone V for liquid portions of the IRWST and Zone VII for vapor space within the IRWST.

  ii. UFSAR Figure 12.3-1 Sheet 7, Radiation Zones, Normal Operations/Shutdown Nuclear Island, Elevation 117'-6"

The proposed change adds a note to room 11305 (the IRWST) on UFSAR Figure 12.3-1, sheet 7 specifying radiation fields in this area of the 117'-6" elevation may exceed zone IV for conditions approaching design basis PRHR HX leakage of 500 gpd and 0.25% fuel defects. Continued operation with these design basis conditions can result in dose rates that are designated as Zone V for liquid portions of the IRWST and Zone VII for vapor space within the IRWST.

iii. UFSAR Figure 12.3-1, Sheet 8, Radiation Zones, Normal Operations/Shutdown Nuclear Island, Elevation 135'-3"

The proposed change adds two notes to room 11500 in the area above the IRWST on UFSAR Figure 12.3-1, sheet 8.

- The first note clarifies that the area directly above hatches may reach Zone III levels during shutdown for conditions approaching design basis PRHR HX leakage of 500 gpd and 0.25% fuel defects.
- The second note clarifies that areas around the IRWST vents may reach Zone V levels for conditions approaching design basis PRHR HX leakage of 500 gpd and 0.25% fuel defects.
- The proposed change also adds an outline showing the Zone IV/II area in Room 11500 to Figure 12.3-1, Sheet 8.

D. Add a description of how the IRWST source term is derived, to UFSAR Subsection 11.1.1, Design Basis Reactor Coolant Activity, as new Subsection 11.1.1.5, IRWST.

## 3. TECHNICAL EVALUATION

A. COL Appendix C, Table 3.3-1 – Credit the IRWST Walls as Shielding

The proposed change credits the north-east wall and the west wall of the IRWST as applicable radiation shielding walls. Without shielding credit applied to these walls based upon their design, radiation levels in Rooms 11300 and 11400 would exhibit significant increases. The downstream impacts of this would likely cause physical changes to equipment or qualification impacts, resulting in significant impacts to the project and significant costs that are based on a non-physical assumption (i.e., that the walls do not exist). The proposed change to credit the north-east and west walls of the IRWST for shielding is not a physical change to the size, configuration, or materials of construction of the IRWST walls. The change uses the existing thicknesses, configurations, and materials of construction in calculating radiation levels in areas adjacent to the side of the walls opposite the sources of radiation within the IRWST.

B. UFSAR Table 3D.5-2, 60-Year Normal Operating Doses

The proposed change adds the IRWST to the locations listed for 60-year Normal Operating Doses.

Extended plant operations with design basis fuel defects (0.25 percent) and design basis PRHR HX leakage (500 gpd) may result in significant radiation levels within the IRWST.

There may also be significant accumulation of radioactive noble gases in the IRWST air space.

Expected sources are also analyzed to show that – under normal conditions – the environment in the IRWST is less severe than the conditions that would exist with design basis fuel defects and design basis PRHR HX leakage.

As stated in UFSAR Subsection 3D.5.1.2, Radiation Dose, the normal operating dose rates and consequent 60-year design expectation doses at various locations inside containment are specified in Table 3D.5-2. These values have been derived from theoretical calculations assuming an expected 60 years of continuous operation and steady-state operating conditions.

The plant-specific estimates of the radiation dose incurred by equipment during normal operation are shown in Table 3D.5-2. Because of the potential increased radioactive contamination of the IRWST water, the gamma dose rate and the 60-year total integrated dose (TID) rads air for safety-related and important-to-safety equipment are considered for equipment qualification (EQ). Due to this consideration, this change proposes to add the gamma dose rate and 60-year TID rads air for the IRWST to UFSAR Table 3D.5-2. The addition of the IRWST to Table 3D.5-2 does not change the design of the IRWST; it documents the consideration of the IRWST TID rads air for qualification of equipment in the IRWST.

EQ is not impacted for the following reasons:

- There is no safety-related equipment above the IRWST hatches.
- The safety-related equipment within the IRWST is the IRWST vents themselves, (PXS-MY-Y61/62/63/64). These vents are qualified to the design basis 60-year TID of 2.2x$10^7$ rads-air.
- During normal operation, the vent covers are exposed to only gamma radiation. The radiation dose for the IRWST is conservatively used since the vent covers are connected to the IRWST.
- Nonsafety-related hydrogen igniters located inside and at the IRWST vents provide hydrogen control. The hydrogen igniters are qualified to the design basis 60-year TID of 2.2x$10^7$ rads-air.
- The hydrogen igniters were tested to 57 mRad, bounding the required 50 mRad which considers both severe accident and 60-year operation.
- Most of the space above the IRWST is shielded by 23.5 inches of concrete and 0.5 inches of steel.

C. Changes to UFSAR Figure 12.3-1

The increase in radiation levels in areas above the IRWST hatches and areas around the IRWST vents are for design basis conditions and are localized to specific areas over the IRWST. They are not representative of the dose rate over most of the IRWST, even considering design basis conditions. The current zoning of the operating deck over the

IRWST is appropriate for both design basis and expected conditions within the IRWST, excluding the aforementioned vents and hatches.

Radiation fields inside the IRWST may exceed zone IV (≤100 mRem/hr) for conditions approaching design basis PRHR HX leakage of 500 gpd and 0.25% fuel defects. Continued operation with these design basis conditions can result in dose rates that are designated as Zone V (≤1 Rem/hr) for liquid portions of the IRWST and Zone VII (≤100 Rem/hr) for vapor space within the IRWST. The area directly above IRWST hatches may reach Zone III (≤15.0 mRem/hr) levels during shutdown for conditions approaching design basis PRHR HX leakage of 500 gpd and 0.25% fuel defects, and areas around the IRWST vents may reach Zone V levels (≤1 Rem/hr) for both shutdown and normal operation.

The change clarifies that radiation levels in areas above the IRWST hatches could increase from Zone II (≤2.5 mRem/hr) to Zone III (≤15.0 mRem/hr) levels during shutdown for conditions approaching design basis PRHR HX leakage concurrent with 0.25% fuel defects into the IRWST. The change also clarifies that radiation levels in areas around the IRWST vents may increase from Zone II (≤2.5 mRem/hr) to Zone V (≤1 Rem/hr) for conditions approaching design basis PRHR HX leakage concurrent with 0.25% fuel defects into the IRWST.

The fully-filled configuration at shortly following shutdown with design basis PRHR HX leakage concurrent with design basis fuel defects is most limiting. The corresponding dose rates are:

- 42 rem/hr (Zone VII) within the IRWST (a change from rad Zone IV to rad Zone VII),
- 210 mrem/hr (Zone V) between CA03 and the containment vessel (a change from rad Zone IV to rad Zone V),
- <2.5 mrem/hr (Zone II) in room 11400 outside the IRWST air space (no change in rad zoning), and
- 10 mrem/hr (Zone III) in the annulus (no change in rad zoning).

Using expected sources results in the following maximum dose rates:

- 77 mrem/hr (Zone IV) within the IRWST
- 0.77 mrem/hr (Zone II) between CA03 and the containment vessel, and
- 0.038 mrem/hr (Zone 0) in the annulus.

These areas are parts of a radiologically controlled area (RCA). Access to the RCA is controlled, radiation levels are monitored, and worker stay times are controlled by radiological work packages to maintain worker dose ALARA.

Regular maintenance of several components is explicitly considered in the annual occupational dose evaluation and the doses from those activities are included in the annual dose assessments. The total collective annual dose is 31.2 person-rem which includes a 10% uncertainty factor. Of 31.2 person-rem, 37.4% is from routine maintenance which includes IRWST-related activities. The IRWST maintenance activities constitute 0.11 person-rem of the total routine maintenance dose; or less than 1% of the total annual occupational dose. This is based on dose rates assuming design basis leakage of 500 gpd from the PRHR HX and an RCS source term based on ANSI/ANS 18.1-1984 (adjusted for AP1000 specific parameters).

As stated in UFSAR Subsection 12.3.1.2, Radiation Zoning and Access Control, access to areas inside the plant structures and plant yard area is regulated and controlled by posting of radiation signs, control of personnel, and use of alarms and locks (UFSAR Section 12.5). During plant operation, access to radiologically restricted areas is through the access control area in the annex building.

Plant areas are categorized into radiation zones according to design basis radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR Part 20. Rooms, corridors, and pipeways are evaluated for potential radiation sources during normal, shutdown, spent resin transfer, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. Each radiation zone defines the radiation level range expected in the zone. The radiation zone categories employed and zoning for each plant area under normal conditions is shown in Figure 12.3-1.

Radiation zones shown in the figure are based upon conservative design data. Actual in-plant zones and control of personnel access are based upon surveys conducted by the Licensee. Access control provisions implement the requirements of 10 CFR Part 20 and utilize the alternative access control methods outlined in Regulatory Guide 8.38.

Based on actual operating plant data, ingress and egress of plant operating personnel to radiologically restricted areas is controlled and monitored as discussed in UFSAR Subsection 12.3.1.2 such that radiation levels and exposures are within the limits prescribed in 10 CFR Part 20.

As stated in UFSAR Subsection 12.5.4, Controlling Access and Stay Time, areas in the plant are classified as non-radiation areas and restricted radiologically controlled areas for radiation protection purposes. Restricted areas are further categorized as radiation areas, high radiation areas, airborne radioactivity areas, contamination areas, and radioactive materials areas, to comply with 10 CFR Part 20 and plant procedures and instructions.

Entrance to the RCA is normally through the access control area at the health physics area entry/exit location in the annex building.

High and very high radiation areas are segregated and identified in accordance with 10 CFR Part 20. The entrances to high and very high radiation areas are locked or barricaded and equipped with audible and/or visible alarms, as required.

The aforementioned controls ensure that the change to the radiation zoning for the IRWST and areas adjacent to the IRWST does not have an adverse effect on maintaining worker dose ALARA.

D. Adding New Subsection 11.1.1.5, IRWST to UFSAR Subsection 11.1.1, Design Basis Reactor Coolant Activity

The proposed change to account for the design basis leakage of RCS from the PRHR HX, concurrent with design basis fuel defects, into the IRWST is not a change to design basis leakage from the RCS. Technical Specification 3.4.7, RCS Operational Leakage limits RCS operational leakage into the IRWST through the PRHR HX to 500 gpd. This limit is unchanged. As stated in UFSAR Subsection 3.1.6, Fuel and Reactivity Control, the radioactive waste management systems are designed to minimize the potential for an inadvertent release of radioactivity from the facility and to provide confidence that the discharge of radioactive wastes is maintained below regulatory limits of 10 CFR Part 50, Appendix I, during normal operation. The radioactivity build-up in the IRWST would be treated before entry to containment like any other potential build-up of radioactivity. No new release pathway is being introduced and radionuclides potentially introduced into the IRWST are subject to existing cleanup systems.

The spent fuel cooling system (SFS) is designed to maintain the water in the IRWST consistent with requirements to limit the radioactivity of the water in the refueling cavity during a refueling. The treatment of assumed radioactive particles in the water is that they are perfectly spherical in a laminar flow. This is conservative for the following reasons:

- Purification via the SFS will introduce turbulence and likely slow the particles descent. The maximum purification flow is 200 gpm. The calculation assumed only 10 gpm for the purposes of purification of the liquid source.
- Actual particles are not spherical and will have a lower terminal velocity.
- Particles are assumed to reach terminal velocity instantly and fall for half the depth of the tank.

The gaseous radwaste and liquid radwaste processing systems include continuous radiation monitoring of their discharge paths. High radiation automatically closes a discharge isolation valve. The liquid radwaste system also has provisions to prevent inadvertent siphoning of its monitor tank contents which could cause an uncontrolled discharge. As stated in UFSAR subsection 11.1.1, Design Basis Reactor Coolant Activity, the liquid and gaseous radioactive waste processing systems have the capability to process wastes based upon 1.0 percent fuel defects. The liquid radwaste system is designed to handle a 10 gpm primary coolant system leak for one hour without discharge or overflow.

Plate-out sources consider 7 cycles of operation (~10.5 years). This was determined to be sufficiently long for the accumulated radionuclides to come to quasi-equilibrium. This is functionally the point at which Co-60 decay during the cycle is equal to the Co-60 “washed” into the IRWST following refueling plus in-leakage and subsequent plate-out contribution.

The increase in the radioactive contamination levels in the IRWST liquid is contained by the IRWST and by the containment structure. The proposed changes to account for the increased leakage from the PRHR HX to the IRWST do not create a release pathway or increase the possibility of release of radioactivity to the environment.

# 4. REGULATORY EVALUATION

## 4.1 Applicable Regulatory Requirements/Criteria

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 proposed change involves a departure from COL Appendix C; therefore, this activity requires an amendment to the COL.

10 CFR 52, Appendix D, VIII.A.4 states that exemptions from Tier 1 information are governed by the requirements in 10 CFR 52.63(b)(1) and 52.98(f). 10 CFR 52.63(b)(1) allows a licensee who references a design certification rule to request an exemption from Tier 1 information. 10 CFR 52.98(f) requires NRC approval for any modification to, addition to, or deletion from the terms and conditions of a COL. This proposed change involves a change to a figure in COL Appendix C, with a corresponding change to Tier 1 information in the associated plant-specific DCD. Therefore, NRC approval is required prior to making the proposed plant-specific change in this license amendment request.

10 CFR 52, Appendix D, Section VIII.B.5.a allows a licensee who references Appendix D to depart from Tier 2 information, without prior NRC approval, unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2* information, or the Technical Specifications, or requires a license amendment under paragraphs B.5.b or B.5.c of Section VIII. As discussed above, the proposed changes to UFSAR Table 3D.5-2, Subsection 11.1.1.5, and Figure 12.3-1 (Sheets 6 through 8) involve changes to Tier 1 Table 3.3-1. Therefore, an exemption request is submitted with this license amendment request.

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1, “Quality standards and records,” requires that structures, systems, and components (SSCs) important to safety are designed, built and tested to quality standards commensurate with the importance of the safety functions performed. The change to account for the design basis leakage from the PRHR HX concurrent with design basis fuel defects into the IRWST and credit existing walls for radiation shielding is not a change to the design or construction of the affected walls of the IRWST. The change calculates the radiation shielding afforded by the existing characteristics of the affected walls. The quality assurance program provides adequate assurance that the quality of these walls is commensurate with their shielding function and compliance with GDC 1 is maintained.

10 CFR Part 50, Appendix A, GDC 4, “Environmental and dynamic effects design bases,” requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. The IRWST vents and the hydrogen igniters located at the vents are qualified to the design basis 60-year normal TID of $2.2 \times 10^7$ rads-air. Dynamic effects are not changed by crediting the design basis leakage of the PRHR HX into the IRWST. The SSCs important to safety are qualified for the conditions anticipated in the IRWST due to the

increased source term from design basis leakage from the PRHR HX concurrent with design basis fuel defects into the IRWST, thereby maintaining compliance with GDC 4.

10 CFR Part 50, Appendix A, GDC 60, "Control of releases of radioactive materials to the environment," requires that the nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for the retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment. The proposed change to account for the design basis leakage of RCS from the PRHR HX, concurrent with design basis fuel defects, into the IRWST is not a change to design basis leakage from the RCS. Technical Specification 3.4.7, RCS Operational Leakage, limits RCS operational leakage into the IRWST through the PRHR HX to 500 gpd; this limit is unchanged. As stated in UFSAR Subsection 3.1.6, Fuel and Reactivity Control, the radioactive waste management systems are designed to minimize the potential for an inadvertent release of radioactivity from the facility and to provide confidence that the discharge of radioactive wastes is maintained below regulatory limits of 10 CFR Part 50, Appendix I, during normal operation. No new release pathway is being introduced and radionuclides potentially introduced into the IRWST are subject to existing cleanup systems. The gaseous radwaste and liquid radwaste processing systems include continuous radiation monitoring of their discharge paths. High radiation automatically closes a discharge isolation valve. The liquid radwaste system also has provisions to prevent inadvertent siphoning of its monitor tank contents which could cause an uncontrolled discharge. As stated in UFSAR subsection 11.1.1, Design Basis Reactor Coolant Activity, the liquid and gaseous radioactive waste processing systems have the capability to process wastes based upon 1.0 percent fuel defects. No new release pathway is being introduced. Radionuclides potentially introduced into the IRWST are subject to existing cleanup systems which are sized to treat RCS leakage (with design basis fuel defects) as high as 10 gallons per minute. The design of radioactive waste management systems is unaffected by the proposed change, maintaining compliance with GDC 60.

10 CFR Part 50, Appendix A, GDC 61, "Fuel storage and handling and radioactivity control," requires that fuel storage and handling, radioactive waste, and other systems which may contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions. These systems are designed: (1) with a capability to permit appropriate periodic inspection and testing of components important to safety; (2) with suitable shielding for radiation protection; (3) with appropriate containment, confinement, and filtering systems; (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal; and, (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. The proposed change to account for the design basis leakage from the PRHR HX concurrent with design basis fuel defects into the IRWST has no effect on residual heat removal capability or inspection and testing of systems important to safety. Existing IRWST and containment structures provide suitable shielding for protection from exposure to radiation. The ability to prevent

reduction in fuel storage coolant inventory is unaffected by the change to account for design basis leakage from the PRHR HX concurrent with design basis fuel defects into the IRWST. And, as stated in UFSAR Subsection 3.1.1, “Overall Requirements,” the SFS maintains the water IRWST consistent with activity requirements of the water in the refueling cavity during refueling, maintaining compliance with GDC 61.

### 4.2 Precedent

No precedent is identified.

### 4.3 Significant Hazards Consideration

The requested amendment proposes a change to Combined License (COL) Appendix C (and the corresponding plant-specific Design Control Document (DCD) Tier 1) information related to crediting the north-east wall and west wall of the IRWST as radiation shielding walls.

The requested amendment proposes a change to Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Table 3.3-1 of COL Appendix C (and a corresponding change to the plant-specific DCD Tier 1 information).

The change to COL Appendix C (and corresponding plant-specific Tier 1) proposes crediting the north-east wall and west wall of the IRWST as radiation shielding walls. The northeast and west walls of the IRWST are not physically modified by this change.

An evaluation to determine whether or not a significant hazards consideration is involved with the proposed amendment was completed by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

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

<u>Response</u>: No.

The proposed changes incorporate the contribution of design basis fuel defects and passive residual heat removal (PRHR) heat exchanger (HX) leakage to the in-containment refueling water storage tank (IRWST) into normal operating doses.

To reduce the dose rates in the vicinity of the IRWST, this proposed change involves crediting the north-east wall and west wall of the IRWST as radiation shielding walls. There is no physical change to the size, configuration, or materials of construction of the IRWST walls. The change uses the existing thicknesses, configurations, and materials of construction in calculating radiation levels in areas adjacent to the side of the walls opposite the sources of radiation within the IRWST.

As part of this proposed change, the potential increase in radioactive contamination of the IRWST is accounted for in the plant-specific estimates of the radiation doses incurred by equipment during normal operation. These doses are considered in the equipment qualification (EQ) of safety-related and important-to-safety equipment. However, there is no impact to EQ because such equipment is either not located where it would incur the estimated dose or is qualified for more severe doses (e.g., severe accident doses). Therefore, there is no impact to the capability of safety-related and important-to-safety equipment to perform their functions credited in reducing the probability, or mitigating the consequences, of an accident.

The proposed changes to the radiation zones around the IRWST only involve normal operations/shutdown and are localized to specific areas within and above the IRWST. No post-accident radiation zones are changed by this activity.

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

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

Response: No.

The proposed changes incorporate the contribution of design basis fuel defects and PRHR HX leakage to the IRWST into normal operating doses.

To reduce the dose rates in the vicinity of the IRWST, this proposed change involves crediting the north-east wall and west wall of the IRWST as radiation shielding walls. There is no physical change to the size, configuration, or materials of construction of the IRWST walls. The change uses the existing thicknesses, configurations, and materials of construction in calculating radiation levels in areas adjacent to the side of the walls opposite the sources of radiation within the IRWST.

As part of this proposed change, the potential increase in radioactive contamination of the IRWST is accounted for in the plant-specific estimates of the radiation doses incurred by equipment during normal operation. These doses are considered in the equipment qualification (EQ) of safety-related and important-to-safety equipment. However, there is no impact to EQ because such equipment is either not located where it would incur the estimated dose or is qualified for more severe doses (e.g., severe accident doses). Therefore, there is no impact to the capability of safety-related and important-to-safety equipment to perform their design functions.

The proposed changes to the radiation zones around the IRWST only involve normal operations/shutdown and are localized to specific areas within and above the IRWST. No post-accident radiation zones are changed by this activity.

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

#### 4.3.3 Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed changes incorporate the contribution of design basis fuel defects and PRHR HX leakage to the IRWST into normal operating doses.

To reduce the dose rates in the vicinity of the IRWST, this proposed change involves crediting the north-east wall and west wall of the IRWST as radiation shielding walls. There is no physical change to the size, configuration, or materials of construction of the IRWST walls. The change uses the existing thicknesses, configurations, and materials of construction in calculating radiation levels in areas adjacent to the side of the walls opposite the sources of radiation within the IRWST.

As part of this proposed change, the potential increase in radioactive contamination of the IRWST is accounted for in the plant-specific estimates of the radiation doses incurred by equipment during normal operation. These doses are considered in the equipment qualification (EQ) of safety-related and important-to-safety equipment. However, there is no impact to EQ because such equipment is either not located where it would incur the estimated dose or is qualified for more severe doses (e.g., severe accident doses). Therefore, there is no impact to the capability of safety-related and important-to-safety equipment to perform their functions credited in reducing the probability, or mitigating the consequences, of an accident.

The proposed changes to the radiation zones around the IRWST only involve normal operations/shutdown and are localized to specific areas within and above the IRWST. No post-accident radiation zones are changed by this activity.

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

Based on the above, 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.

### 4.4 Conclusions

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. The above evaluations demonstrate that the requested change can be accommodated without an increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without a significant reduction in a margin of safety. Having arrived at negative declarations with regard to the criteria of 10 CFR 50.92, this assessment determined that the requested change does not involve a Significant Hazards Consideration.

## 5. ENVIRONMENTAL CONSIDERATIONS

Southern Nuclear Operating Company (SNC or "Licensee") is requesting an amendment to Combined License (COL) Nos. NPF-91 and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively. The requested amendment proposes a change to Updated Final Safety Analysis Report (UFSAR) Tier 2 information, which involves a change to COL Appendix C (and the corresponding plant-specific Design Control Document (DCD) Tier 1) information related to crediting the northeast and west walls of the IRWST as radiation shielding walls.

The requested amendment proposes changes to Tier 2 information in UFSAR Table 3D.5-2, Subsection 11.1.1.5, and Figure 12.3-1 (Sheets 6 through 8), which involve changes to Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Table 3.3-1 of COL Appendix C, and corresponding changes to the plant-specific DCD Tier 1 information.

The change to the UFSAR proposes to incorporate the contribution of design basis passive residual heat removal (PRHR) heat exchanger (HX) leakage to the in-containment refueling water storage tank (IRWST) into normal operating doses. Sections 2 and 3 of this license amendment request provide the details of the proposed change.

The Licensee has determined that the anticipated construction and operational effects of the proposed amendment meet 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, Significant Hazards Consideration, 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 determined that (1) the requested amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) the requested amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) the requested amendment does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the requested 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 change to account for the design basis leakage of reactor coolant from the PRHR HX, concurrent with design basis fuel defects, into the IRWST is not a change to design basis leakage from the RCS. The RCS leakage limit is unchanged. The radioactive waste management systems are designed to minimize the potential for an inadvertent release of radioactivity from the facility and to provide confidence that the discharge of radioactive wastes is maintained below regulatory limits of 10 CFR 50, Appendix I, during normal operation. The radioactivity build-up in the IRWST would be contained by the IRWST and by the containment structure and would be treated before entry to containment like any other potential build-up of radioactivity. No new release pathway is being introduced and radionuclides potentially introduced into the IRWST are subject to existing cleanup systems.

The proposed changes to account for the increased leakage from the PRHR HX to the IRWST do not create a release pathway or increase the possibility of release of radioactivity to the environment. There is no impact to the assumptions or analyses in the completed safety analysis for radiation doses as a result of the change.

Additionally, the proposed change does not affect any aspect of plant construction or operation that introduces a change to any effluent types (for example effluents containing chemicals or biocides, sanitary system effluents, and other effluents), and do not affect any non-radiological effluent release quantities. Accordingly, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.

Therefore, it is concluded that the requested 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 changes involve (1) accounting for the design basis leakage of reactor coolant from the PRHR HX, concurrent with design basis fuel defects, into the IRWST and (2) crediting the north-east wall and west wall of the IRWST as applicable radiation shielding walls. The areas affected are inside containment and are parts of a radiologically controlled area (RCA). Access to the RCA is controlled, radiation levels are monitored, and worker stay times are controlled by radiological work packages to maintain worker dose as low as reasonably achievable (ALARA).

Plant areas are categorized into radiation zones according to design basis radiation levels and anticipated personnel occupancy with consideration given toward maintaining personnel exposures ALARA and within the standards of 10 CFR 20. Rooms, corridors, and pipeways are evaluated for potential radiation sources during normal, shutdown, spent resin transfer, and emergency operations; for maintenance occupancy requirements; for general access requirements; and for material exposure limits to determine appropriate zoning. Each radiation zone defines the radiation level range expected in the zone. Radiation zones are based upon conservative design data. Actual in-plant zones and control of personnel access are based upon surveys conducted by the licensee. Based on actual operating plant data, ingress or egress of plant operating personnel to radiologically restricted areas is controlled and monitored such that radiation levels and exposures are within the limits prescribed in 10 CFR 20.

As stated in UFSAR Subsection 12.5.4, Controlling Access and Stay Time, areas in the plant are classified as non-radiation areas and restricted radiologically controlled areas for radiation protection purposes. Restricted areas are further categorized as radiation areas, high radiation areas, airborne radioactivity areas, contamination areas, and radioactive materials areas, to comply with 10 CFR 20 and plant procedures and instructions.

High and very high radiation areas are segregated and identified in accordance with 10 CFR 20. The entrances to high and very high radiation areas are locked or barricaded and equipped with audible and/or visible alarms, as required.

The aforementioned controls ensure that the change to the radiation zoning for the IRWST and areas adjacent to the IRWST does not have an adverse effect on maintaining worker dose ALARA.

The proposed change does not modify walls, floors, or other structures that provide radiation shielding. Company and station policies maintain radiation exposure of personnel within limits defined by 10 CFR Part 20, "Standards for Protection Against Radiation." Administrative procedures and practices are implemented to maintain radiation exposure of personnel ALARA. Therefore, it is concluded that the requested amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above review of the requested amendment, it has been determined that anticipated construction and operational effects of the requested amendment do 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 individual or cumulative occupational radiation exposure. Accordingly, the requested 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), an environmental impact statement or environmental assessment of the proposed amendment is not required.

## 6. REFERENCES

None.

**Southern Nuclear Operating Company**

**ND-19-**Error! Reference source not found.

**Enclosure 2**

**Vogtle Electric Generating Plant (VEGP) Units 3 and 4**

**Exemption Request:**

**Reconciliation of Detailed AP1000 Radiation Analyses**

**(LAR-19-003)**



(This Enclosure consists of 7 pages, including this cover page.)

## 1.0 Purpose

Southern Nuclear Operating Company (the Licensee) requests a permanent exemption from the provisions of 10 CFR 52, Appendix D, Section III.B, *Design Certification Rule for the AP1000 Design, Scope and Contents*, to allow a departure from elements of the certification information in Tier 1 of the generic AP1000 Design Control Document (DCD). The regulation, 10 CFR 52, Appendix D, Section III.B, requires an applicant or licensee referencing Appendix D to 10 CFR Part 52 to incorporate by reference and comply with the requirements of Appendix D, including certified information in DCD Tier 1. The Tier 1 information for which a plant-specific departure and exemption is being requested includes relocating the auxiliary steam header isolation valve from the same header as the turbine bypass valves to a new header.

This request for exemption provides the technical and regulatory basis to demonstrate that 10 CFR 52.63, §52.7, and §50.12 requirements are met and will apply the requirements of 10 CFR 52, Appendix D, Section VIII.A.4 to allow departures from generic Tier 1 information due to a proposed change to Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Table 3.3-1 to credit the north-east wall and west wall of the in-containment refueling water storage tank (IRWST) as radiation shielding walls

## 2.0 Background

The Licensee is the holder of Combined License Nos. NPF-91 and NPF-92, which authorize construction and operation of two Westinghouse Electric Company AP1000 nuclear plants, named Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively.

<u>Design Function Related to Activity</u>

As described in UFSAR Subsection 6.3.2.2.3, The in-containment refueling water storage tank (IRWST) is a large, stainless-steel lined tank located underneath the operating deck inside the containment. The IRWST is AP1000 Equipment Class C and is designed to meet seismic Category I requirements.

The IRWST is sized to provide the flooding of the refueling cavity for normal refueling, the post-loss of coolant accident flooding of the containment for reactor coolant system long-term cooling mode, and to support the passive residual heat removal (PRHR) heat exchanger (HX) operation.

<u>Reason for Activity</u>

The contribution of design basis passive residual heat removal (PRHR) heat exchanger (HX) leakage to the in-containment refueling water storage tank (IRWST) is being incorporated into normal operating doses. This change to normal operating doses involves crediting the north-east wall and west wall of the IRWST as radiation shielding walls in Tier 1 Table 3.3-1, which currently shows the north-east wall and west wall of the IRWST as not being an applicable radiation shielding wall during normal operations.

Description of Activity

The north-east wall and west wall of the IRWST are credited as radiation shielding walls in Tier 1 Table 3.3-1.

## 3.0 Technical Justification of Acceptability

This activity only includes a change to credit the north-east wall and west wall of the IRWST as radiation shielding walls. The proposed change to credit the north-east and west walls of the IRWST for shielding is not a physical change to the size, configuration, or materials of construction of the IRWST walls. The change uses the existing thicknesses, configurations, and materials of construction in calculating radiation levels in areas adjacent to the side of the walls opposite the sources of radiation within the IRWST.

The proposed change does not result in a modification to, addition to, or removal of a structure, system, or component (SSC) such that a design function is adversely affected, have no impact on plant operating procedures or a method of control that adversely affects a design function, do not result in an adverse change to a method of evaluation or use of an alternate method of evaluation, do not represent tests or experiments outside the reference bounds of the design basis, and do not alter the assumptions or results of the ex-vessel severe accident assessment.

## 4.0 Justification of Exemption

10 CFR Part 52, Appendix D, Section VIII.A.4 and 10 CFR 52.63(b)(1) govern the issuance of exemptions from elements of the certified design information for AP1000 nuclear power plants. Since SNC has identified a change to the Tier 1 information as discussed in Enclosure 1 of the accompanying License Amendment Request, an exemption from the certified design information in Tier 1 is needed.

10 CFR Part 52, Appendix D, and 10 CFR 50.12, §52.7, and §52.63 state that the NRC may grant exemptions from the requirements of the regulations provided six conditions are met: 1) the exemption is authorized by law [§50.12(a)(1)]; 2) the exemption will not present an undue risk to the health and safety of the public [§50.12(a)(1)]; 3) the exemption is consistent with the common defense and security [§50.12(a)(1)]; 4) special circumstances are present [§50.12(a)(2)]; 5) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption [§52.63(b)(1)]; and 6) the design change will not result in a significant decrease in the level of safety [Part 52, App. D, VIII.A.4].

The requested exemption satisfies the criteria for granting specific exemptions, as described below.

### 1. This exemption is authorized by law

The NRC has authority under 10 CFR 52.63, §52.7, and §50.12 to grant exemptions from the requirements of NRC regulations. Specifically, 10 CFR 50.12 and §52.7 state that the NRC may grant exemptions from the requirements of 10 CFR Part 52 upon a proper showing. No law exists that would preclude the change covered by this exemption request. Additionally, granting of the proposed exemption does not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission's regulations.

Accordingly, this requested exemption is "authorized by law," as required by 10 CFR 50.12(a)(1).

2. **This exemption will not present an undue risk to the health and safety of the public**

The proposed exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would allow a change to elements of the plant-specific Tier 1 DCD to depart from the AP1000 certified (Tier 1) design information. The plant-specific DCD Tier 1 will continue to reflect the approved licensing basis for VEGP Units 3 and 4 and will maintain a consistent level of detail with that which is currently provided elsewhere in Tier 1 of the DCD. Therefore, the affected plant-specific DCD Tier 1 ITAAC will continue to serve its required purpose.

The proposed change to credit the north-east wall and west wall of the IRWST as radiation shielding walls. The proposed change to credit the north-east and west walls of the IRWST for shielding is not a physical change to the size, configuration, or materials of construction of the IRWST walls. The change uses the existing thicknesses, configurations, and materials of construction in calculating radiation levels in areas adjacent to the side of the walls opposite the sources of radiation within the IRWST.

Therefore, the requested exemption from 10 CFR 52, Appendix D, Section III.B would not present an undue risk to the health and safety of the public.

3. **The exemption is consistent with the common defense and security**

The requested exemption from the requirements of 10 CFR 52, Appendix D, Section III.B would allow the licensee to depart from elements of the plant-specific DCD Tier 1 design information. The proposed exemption does not alter the design, function, or operation of any structures or plant equipment that are necessary to maintain a safe and secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures.

Therefore, the requested exemption is consistent with the common defense and security.

4. **Special circumstances are present**

10 CFR 50.12(a)(2) lists six "special circumstances" for which an exemption may be granted. Pursuant to the regulation, it is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request. The requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

The rule under consideration in this request for exemption is 10 CFR 52, Appendix D, Section III.B, which requires that a licensee referencing the AP1000 Design Certification Rule (10 CFR Part 52, Appendix D) shall incorporate by reference and comply with the requirements of Appendix D, including Tier 1 information. The VEGP Units 3 and 4 COLs reference the AP1000 Design Certification Rule and incorporate by reference the requirements of 10 CFR Part 52, Appendix D, including Tier 1 information. The underlying purpose of Appendix D, Section III.B is to describe and define the scope and contents of the AP1000 design certification, and to require compliance with the design certification information in Appendix D.

The proposed exemption would credit the north-east wall and the west wall of the IRWST as applicable radiation shielding walls.

The proposed exemption to credit the northeast and west walls of the IRWST for shielding is not a physical change to the size, configuration, or materials of construction of the IRWST walls. The change uses the existing thicknesses, configurations, and materials of construction in calculating radiation levels in areas adjacent to the side of the walls opposite the sources of radiation within the IRWST. Therefore, there is no impact to the structural integrity of the IRWST, and the IRWST design function of providing the heat sink for the passive residual heat removal (PRHR) heat exchanger (HX) is not affected.

Therefore, special circumstances are present, because application of the current generic certified design information in Tier 1 as required by 10 CFR Part 52, Appendix D, Section III.B, in the particular circumstances discussed in this request is not necessary to achieve the underlying purpose of the rule.

**5. The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption.**

Based on the nature of the change to the plant-specific Tier 1 information and the understanding that this change supports the design function of the supported equipment, it is expected that this exemption may be requested by other AP1000 licensees and applicants. However, a review of the reduction in standardization resulting from the departure from the standard DCD determined that even if other AP1000 licensees and applicants do not request this same departure, the special circumstances will continue to outweigh any decrease in safety from the reduction in standardization because the key design functions of the equipment associated with this request will continue to be maintained. Furthermore, the justification provided in the license amendment request and this exemption request and the associated mark-ups demonstrate that there is a limited change from the standard information provided in the generic AP1000 DCD, which is offset by the special circumstances identified above.

Therefore, the special circumstances associated with the requested exemption outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption.

6. **The design change will not result in a significant decrease in the level of safety.**

   The exemption revises the plant-specific DCD Tier 1 information to credit the northeast and west walls of the IRWST for shielding is not a physical change to the size, configuration, or materials of construction of the IRWST walls. The change uses the existing thicknesses, configurations, and materials of construction in calculating radiation levels in areas adjacent to the side of the walls opposite the sources of radiation within the IRWST. Therefore, there is no impact to the structural integrity of the IRWST, and the IRWST design function of providing the heat sink for the passive residual heat removal (PRHR) heat exchanger (HX) is not affected.

## 5.0 Risk Assessment

A risk assessment was not determined to be applicable to address the acceptability of this proposal.

## 6.0 Precedent Exemptions

None

## 7.0 Environmental Consideration

The Licensee requests a departure from elements of the certified information in Tier 1 of the generic AP1000 DCD. The Licensee has determined that the proposed departure would require a permanent exemption from the requirements of 10 CFR 52, Appendix D, Section III.B, *Design Certification Rule for the AP1000 Design, Scope and Contents,* with respect to installation or use of facility components located within the restricted area, as defined in 10 CFR Part 20, or which changes an inspection or a surveillance requirement; however, the Licensee evaluation of the proposed exemption has determined that the proposed exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Based on the above review of the proposed exemption, the Licensee has determined that the proposed activity 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 individual or cumulative occupational radiation exposure. Accordingly, the proposed exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment of the proposed exemption is not required.

Specific details of the environmental considerations supporting this request for exemption are provided in Section 5 of the associated License Amendment Request provided in Enclosure 1 of this letter.

## 8.0 Conclusion

The proposed change to Tier 1 are necessary to credit the northeast and west walls of the IRWST for shielding. The exemption request meets the requirements of 10 CFR 52.63, *Finality of design certifications*, 10 CFR 52.7, *Specific exemptions*, 10 CFR 50.12, *Specific exemptions*, and 10 CFR 52 Appendix D, *Design Certification Rule for the AP1000*. Specifically, the exemption request meets the criteria of 10 CFR 50.12(a)(1) in that the request is authorized by law, presents no undue risk to public health and safety, and is consistent with the common defense and security. Furthermore, approval of this request does not result in a significant decrease in the level of safety, satisfies the underlying purpose of the AP1000 Design Certification Rule, and does not present a significant decrease in safety as a result of a reduction in standardization.

## 9.0 References

None.

**Southern Nuclear Operating Company**

**ND-19-**Error! Reference source not found.

**Enclosure 3**

**Vogtle Electric Generating Plant (VEGP) Units 3 and 4**

**Proposed Changes to Licensing Basis Documents – Public Information**

**(LAR-19-003)**



(This Enclosure consists of 4 pages, including this cover page.)

**Revise COL Appendix C (and corresponding plant-specific Tier 1) Table 3.3-1 to indicate that the north-east wall and the west wall of the IRWST are applicable radiation shielding walls during normal operations as shown below:**

**Table 3.3-1 (cont.)**
**Definition of Wall Thicknesses for Nuclear Island Buildings, Turbine Building, and Annex Building[1]**

| Wall or Section Description | Column Lines[7] | Floor Elevation or Elevation Range[7][8] | Concrete Thickness[2][3][4][5][9] | Applicable Radiation Shielding Wall (Yes/No) |
|---|---|---|---|---|
| * * * | * * * | * * * | * * * | * * * |
| North-east wall of in-containment refueling water storage tank | Parallel to column line N | From 103'-0" to 135'-3" | 2'-6" | ~~No~~Yes |
| West wall of in-containment refueling water storage tank | Not applicable | From 103'-0" to 135'-3" | 5/8" steel plate with stiffeners | ~~No~~Yes |
| * * * | * * * | * * * | * * * | * * * |

**Revise UFSAR Table 3D.5-2 to include the 60-year normal operating doses for the IRWST as shown below:**

**Table 3D.5-2**
**60-Year Normal Operating Doses**

| Location | Gamma Dose Rate (Rad air hour) | 60-Year Gamma Dose (Rads air) |
|---|---|---|
| Inside Containment: | | |
| * * * | * * * | * * * |
| Adjacent to Reactor Vessel Wall | $\leq 3.6 \times 10^4$ | $1.9 \times 10^{10}$[(a)] |
| IRWST | $4.2 \times 10^1$ | $2.2 \times 10^7$ |
| Outside Containment: | | |
| * * * | * * * | * * * |

**Insert new UFSAR Subsection 11.1.1.5 titled "IRWST" after UFSAR Subsection 11.1.1.4 as shown below (Note: a corresponding update to the Chapter 11 Table of Contents is also made to reflect this new subsection):**

**11.1.1.4 Nitrogen-16**

* * *

**11.1.1.5 IRWST**

The IRWST liquid source term is calculated as a mass balance using the IRWST liquid as a control volume. Nuclides are produced via RCS in-leakage (via PRHR HX and refueling cavity drain-down) and decay of parent radionuclides. The reactor coolant leakage into the IRWST is assumed to have radionuclide concentrations as defined in Table 11.1-2. Nuclides are removed via SFS cleanup, radioactive decay, draining (overflow) to the sump, evolution of gaseous radionuclides, and settling within the IRWST. Nuclides that are removed from the tank liquid via evolution into the air space or settling to the tank floor are tracked independently using similar equations.

**11.1.2 Design Basis Secondary Activity**

* * *

**Southern Nuclear Operating Company**

**ND-19-####**

**Enclosure 4**

**Vogtle Electric Generating Plant (VEGP) Units 3 and 4**

**Proposed Changes to Licensing Basis Documents – Withheld Information**

**(LAR-19-003)**



(This Enclosure consists of 4 pages, including this cover page.)