UNITED STATES
NUCLEAR REGULATORY COMMISSION
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March 12, 2019

MEMORANDUM TO: Robert J. Pascarelli, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

FROM: Michael J. Wentzel, Acting Chief ***/RA/***
PRA Licensing Branch A
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SUBJECT: SAFETY EVALUATION INPUT FOR GRAND GULF NUCLEAR STATION UNIT 1, LICENSE AMENDMENT REQUEST TO IMPLEMENT TECHNICAL SPECIFICATION TASK FORCE-425, REVISION 3, RELOCATE SURVEILLANCE FREQUENCIES TO LICENSEE CONTROL

By application dated April 12, 2018 and supplemented by letters dated June 7, 2018 and November 30, 2018, Entergy Operations Inc, requested changes to the technical specifications, for Grand Gulf Nuclear Station Unit 1 in accordance with part 50.90 of Title 10 of the *Code of Federal Regulations*. The requested change is the adoption of U.S. Nuclear Regulatory Commission-approved Technical Specification Task Force (TSTF-425), Revision 3, "Relocate Surveillance Frequencies to Licensee Control—RITSTF Initiative 5b."

The Probabilistic Risk Assessment Licensing Branch A (APLA) reviewed the proposed changes using the generic requirements identified in TSTF-425. On the basis of our review, and as discussed in the attached safety evaluation, the APLA staff finds that the methodology and approach used by the licensee are consistent with TSTF-425 and therefore acceptable.

Docket No. 50-416

Enclosure:
Safety Evaluation

CONTACT: Jigar J. Patel, DRA/APLA
301-415-2832

SUBJECT: SAFETY EVALUATION INPUT FOR GRAND GULF NUCLEAR STATION UNIT 1, LICENSE AMENDMENT REQUEST TO IMPLEMENT TECHNICAL SPECIFICATION TASK FORCE-425, REVISION 3, RELOCATE SURVEILLANCE FREQUENCIES TO LICENSEE CONTROL DATED: 3/12/2019

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DATE	2/06/2019	3/12/2019	3/ 12/2019

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. [ ] TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS INC

GRAND GULF NUCLEAR STATION UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By application dated April 12, 2018 (Reference 1), as supplemented by three letters dated June 7, 2018, November 30, 2018 and March 6, 2019 (References 2, 3 and 4), Entergy Operations (or the licensee), requested changes to the Technical Specifications (TSs) for Grand Gulf Nuclear Station (GGNS).

The proposed changes would revise the GGNS TSs to adopt the U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) Change Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control-RITSTF [Risk-Informed TSTF] Initiative 5b" (Reference 5) for GGNS.

The three supplemental letters dated June 7, 2018, November 30, 2018 and March 6, 2019 provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on May 24, 2016 (81 FR 32807).

2.0 REGULATORY EVALUATION

2.1 Description of the Proposed Changes

The licensee proposed to modify the GGNS TSs by relocating specific surveillance frequencies to a licensee-controlled program (i.e., the Surveillance Frequency Control Program (SFCP) in accordance with Nuclear Energy Institute (NEI) 04-10, Revision 1 (Reference 6). The licensee stated that the proposed change is consistent with the adoption of NRC-approved TSTF-425, Revision 3. When implemented, TSTF-425, Revision 3, relocates most periodic frequencies of TS surveillances to the SFCP, and provides requirements for the new SFCP in the Administrative Controls section of the TSs. All surveillance frequencies can be relocated except the following:

- Frequencies that reference other approved programs for the specific interval, such as the Inservice Testing Program or the Primary Containment Leakage Rate Testing Program;
- Frequencies that are purely event-driven (e.g., "each time the control rod is withdrawn to the 'full out' position");

- Frequencies that are event-driven, but have a time component for performing the surveillance on a one-time basis once the event occurs (e.g., "within 24 hours after thermal power reaching 2: [greater than or equal to] 95% RTP [rated thermal power}"); and

- Frequencies that are related to specific conditions (e.g., battery degradation, age and capacity) or conditions for the performance of a surveillance requirement (SR).

In letter dated September 19, 2007 (Reference 7), the NRC staff approved Topical Report NEI 04-10, Revision 1, as an acceptable methodology for referencing in licensing actions to the extent specified and under the limitations delineated in NEI 04-10, Revision 1, and the safety evaluation (SE) providing the basis for NRC acceptance of NEI 04-10, Revision 1.

2.2 Applicable Commission Policy Statements

In the "Final Policy Statement: Technical Specifications Improvements for Nuclear Power Plants," dated July 22, 1993 (58 FR 39132), the NRC addressed the use of probabilistic safety analysis (PSA, currently referred to as probabilistic risk assessment or PRA) in STS. In this 1993 publication, the NRC states, in part:

> The Commission believes that it would be inappropriate at this time to allow requirements which meet one or more of the first three criteria [of Title 1O of the *Code of Federal Regulations* (1O CFR), Section 50.36] to be deleted from Technical Specifications based solely on PSA (Criterion 4). However, if the results of PSA indicate that Technical Specifications can be relaxed or removed, a deterministic review will be performed....
>
> The Commission Policy in this regard is consistent with its Policy Statement on "Safety Goals for the Operation of Nuclear Power Plants," 51 FR 30028, published on August 21, 1986. The Policy Statement on Safety Goals states in part, "* * * probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgments can be made*** about the degree of confidence to be given these [probabilistic] estimates and assumptions. This is a key part of the process for determining the degree of regulatory conservatism that may be warranted for particular decisions. This defense-in-depth approach is expected to continue to ensure the protection of public health and safety." ...
>
> The Commission will continue to use PSA, consistent with its policy on Safety Goals, as a tool in evaluating specific line-item improvements to Technical Specifications, new requirements, and industry proposals for risk-based Technical Specification changes....

Approximately 2 years later, the NRC provided additional detail concerning the use of PRA in the "Final Policy Statement: Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," dated August 16, 1995 (60 FR 42622). In this publication, the NRC states, in part:

> The Commission believes that an overall policy on the use of PRA methods in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner

that would promote regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach....

PRA addresses a broad spectrum of initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for multiple and common cause failures. The treatment therefore goes beyond the single failure requirements in the deterministic approach. The probabilistic approach to regulation is, therefore, considered an extension and enhancement of traditional regulation by considering risk in a more coherent and complete manner....

Therefore, the Commission believes that an overall policy on the use of PRA in nuclear regulatory activities should be established so that the many potential applications of PRA can be implemented in a consistent and predictable manner that promotes regulatory stability and efficiency. This policy statement sets forth the Commission's intention to encourage the use of PRA and to expand the scope of PRA applications in all nuclear regulatory matters to the extent supported by the state-of-the-art in terms of methods and data....

Therefore, the Commission adopts the following policy statement regarding the expanded NRC use of PRA:

(1) The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

(2) PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

(3) PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

(4) The Commission's safety goals for nuclear power plants and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on nuclear power plant licensees.

2.3 Applicable Regulations

In 10 CFR 50.36, "Technical specifications," the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) SRs; (4) design features; and (5) administrative controls. These categories will remain in the GGNS TSs.

Section 50.36(c)(3) of 10 CFR states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The FR notice published on July 6, 2009 (74 FR 31996), which announced the availability of TSTF-425, Revision 3, states that the addition of the SFCP to the TSs provides the necessary administrative controls to require that surveillance frequencies relocated to the SFCP are conducted at a frequency to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. The FR notice also states that changes to surveillance frequencies in the SFCP are made using the methodology contained in NEI 04-10, Revision 1, including qualitative considerations, results of risk analyses, sensitivity studies and any bounding analyses, and recommended monitoring of structures, systems, and components (SSCs), and are required to be documented.

Existing regulatory requirements, such as 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" (i.e., the Maintenance Rule), and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," require licensee monitoring of surveillance test failures and implementing corrective actions to address such failures. Such failures can result in the licensee increasing the frequency of a surveillance test. In addition, by having the TSs require that changes to the frequencies listed in the SFCP be made in accordance with NEI 04-10, Revision 1, the licensee will be required to monitor the performance of SSCs for which surveillance frequencies are decreased to assure reduced testing does not adversely impact the SSCs.

2.4 Applicable NRC Regulatory Guides and Review Plans

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2 (Reference 8), describes an acceptable risk-informed approach for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications," Revision 1 (Reference 9), describes an acceptable risk-informed approach specifically for assessing proposed TS changes.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (Reference 10), describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors (LWRs).

NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (Reference 11) provides general guidance for evaluating the technical basis for proposed risk-informed changes. Guidance on evaluating PRA technical adequacy is provided in SRP, Chapter 19, Section 19.1, Revision 3, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load" (Reference 12). More specific guidance related to risk-informed TS changes is provided in SRP, Chapter 16, Section 16.1, Revision 1, "Risk-Informed Decision-making: Technical Specifications" (Reference 13), which includes changes to surveillance test intervals (STIs) (i.e., surveillance frequencies) as part of risk-informed decision-making. Section 19.2 of the SRP references the same criteria as RG 1.174, Revision 2, and RG 1.177, Revision 1, and states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

- The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change;
- The proposed change is consistent with the defense-in-depth philosophy;
- The proposed change maintains sufficient safety margins;
- When proposed changes result in an increase in core damage frequency (CDF) or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement;
- The impact of the proposed change should be monitored using performance measurement strategies.

3.0 TECHNICAL EVALUATION

The licensee's adoption of TSTF-425, Revision 3, provides for administrative relocation of applicable surveillance frequencies, and provides for the addition of the SFCP to the Administrative Controls section of TSs. The changes to the Administrative Controls section of the TSs will also require the application of NEI 04-10, Revision 1, for any changes to surveillance frequencies within the SFCP. The licensee's application for the changes described in TSTF-425, Revision 3, included documentation regarding the PRA technical adequacy consistent with RG 1.200, Revision 2. NEI 04-10, Revision 1, states that PRA methods are used with plant performance data and other considerations to identify and justify modifications to the surveillance frequencies of equipment at nuclear power plants. This is consistent with guidance provided in RG 1.174, Revision 2, and RG 1.177, Revision 1, in support of changes to STIs.

3.1 Key Principles

RG 1.777, Revision 1, identified five key safety principles required for risk-informed changes to TSs. Each of these principles are addressed by NEI 04-10, Revision 1. Sections 3.1.1 through 3.1.5 of this section contain a discussion of the five principles, including the NRC staff's evaluation of how the licensee's license amendment request (LAR) satisfied each principle.

3.1.1 The Proposed Change Meets Current Regulations

Section 50.36(c)(3) of 1O CFR requires that TSs include surveillances, which are "requirements relating to test, calibration, or inspection to assure that necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." The licensee is required by its TSs to perform surveillance tests, calibration, or inspection on specific safety-related equipment (e.g., reactivity control, power distribution, electrical, and instrumentation) to verify system operability. Surveillance frequencies are based primarily upon deterministic methods, such as engineering judgment, operating experience, and manufacturer's recommendations. The licensee's use of NRC-approved methodologies identified in NEI 04-10, Revision 1, provides a way to establish risk-informed surveillance frequencies that complements the deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

The SRs remain in the TSs, as required by 10 CFR 50.36(c)(3). This change is analogous with other NRC-approved TS changes in which the SRs are retained in TSs, but the related surveillance frequencies are relocated to licensee-controlled documents, such as surveillances performed in accordance with the Inservice Testing Program and the Primary Containment Leakage Rate Testing Program. Thus, this proposed change complies with 10 CFR 50.36(c)(3) by retaining the requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The regulatory requirements in 10 CFR 50.65 and 10 CFR Part 50, Appendix B, and the monitoring required by NEI 04-10, Revision 1, ensure that surveillance frequencies are sufficient to assure that the requirements of 10 CFR 50.36 are satisfied and that any performance deficiencies will be identified and appropriate corrective actions taken. The licensee's SFCP ensures that SRs specified in the TSs are performed at intervals sufficient to assure that the above regulatory requirements are met. Based on the foregoing, the NRC staff concludes that the proposed change meets the first key safety principle of RG 1.177, Revision 1, by complying with current regulations.

3.1.2 *The Proposed Change Is Consistent with the Defense-in-Depth Philosophy*

The defense-in-depth philosophy (i.e., the second key safety principle of RG 1.177, Revision 1) is maintained if:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation;
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided;
- System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers). (Because the scope of the proposed methodology is limited to revision of surveillance frequencies, the redundancy, independence, and diversity of plant systems are not impacted.);
- Defenses against potential common cause failures (CCFs) are preserved, and the potential for the introduction of new CCF mechanisms is assessed;

- Independence of barriers is not degraded;

- Defenses against human errors are preserved;

- The intent of the General Design Criteria in 10 CFR Part 50, Appendix A, is maintained.

The changes to the Administrative Controls section of the TSs will require the application of NEI 04-10, Revision 1, for any changes to surveillance frequencies within the SFCP.

NEI 04-10, Revision 1, uses both the CDF and the large early release frequency (LERF) metrics to evaluate the impact of proposed changes to surveillance frequencies. In accordance with RG 1.174, Revision 2, and RG 1.177, Revision 1, changes to CDF and LERF are evaluated using a comprehensive risk analysis, which assesses the impact of proposed changes, including contributions from human errors and CCFs. Defense-in-depth is also included in the methodology explicitly as a qualitative consideration outside of the risk analysis, as is the potential impact on detection of component degradation that could lead to an increased likelihood of CCFs. The NRC staff concludes that both the quantitative risk analysis and the qualitative considerations provide reasonable assurance that defense-in-depth is maintained to ensure protection of public health and safety, satisfying the second key safety principle of RG 1.177, Revision 1.

3.1.3 *The Proposed Change Maintains Sufficient Safety Margins*

The engineering evaluation that will be conducted by the licensee under the SFCP when frequencies are revised will assess the impact of the proposed frequency change to assure that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring the proposed surveillance test frequency change is not in conflict with approved industry codes and standards or adversely affects any assumptions or inputs to the safety analysis; or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist.

The design, operation, testing methods, and acceptance criteria for SSCs specified in applicable codes and standards (or alternatives approved for use by the NRC) will continue to be met as described in the plants' licensing bases, including the Updated Final Safety Analysis Report and TS Bases, because these are not affected by changes to the surveillance frequencies. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. On this basis, the NRC staff concludes that safety margins are maintained by the proposed methodology and, therefore, the third key safety principle of RG 1.177, Revision 1, is satisfied.

3.1.4 *When Proposed Changes Result in an Increase in CDF or Risk, the Increases Should Be Small and Consistent with the Intent of the Commission's Safety Goal Policy Statement*

RG 1.177, Revision 1, provides a framework for evaluating the risk impact of proposed changes to surveillance frequencies, which requires identification of the risk contribution from impacted surveillances, determination of the risk impact from the change to the proposed surveillance frequency, and performance of sensitivity and uncertainty evaluations. The changes to the Administrative Controls section of the TSs will require application of NEI 04-10, Revision 1, in the SFCP. NEI 04-10, Revision 1, satisfies the intent of RG 1.177, Revision 1, guidance for evaluation of the change in risk, and for assuring that such changes are small by providing the technical methodology to support risk-informed TSs for control of surveillance frequencies.

3.1.4.1 *PRA Technical Adequacy*

The technical adequacy of the licensee's PRA must be commensurate with the safety significance of the proposed TS change and the role the PRA plays in justifying the change. That is, the greater the change in risk or the greater the uncertainty in that risk from the requested TS change, or both, the more rigor that must go into ensuring the technical adequacy of the PRA.
RG 1.200 (Reference 10) provides regulatory guidance for assessing the technical adequacy of a PRA. The current revision (i.e., Revision 2) of this RG endorses, with clarifications and qualifications, the use of the following:

(1) American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009, "Addenda to ASME RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (i.e., the PRA Standard) (Reference 15),

(2) NEI 00-02, "PRA Peer Review Process Guidance" (Reference 16), and

(3) NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2 (Reference 17).

The licensee performed an assessment of the PRA models used to support the SFCP using the guidance of RG 1.200, Revision 2, to ensure that the PRA models are capable of determining the change in risk due to changes to surveillance frequencies of SSCs, using plant-specific data and models. Capability Category (CC) II of the NRC-endorsed PRA standard is the target capability level for supporting requirements for the internal events PRA (IEPRA) for this application. Any identified deficiencies to those requirements are further assessed to determine any impacts to proposed decreases to surveillance frequencies, including the use of sensitivity studies where appropriate, in accordance with NEI 04-10, Revision 1.

The GGNS PRA model Revision 1 underwent a peer review in October 1997 by the Boiling Water Reactor Owners Group (BWROG). Subsequently, a full scope industry peer review of the GGNS PRA model Revision 4 was conducted by the BWROG in September 2015. This peer review documented 66 new Facts and Observations (F&Os) including 39 findings, 26 suggestions and one best practice. The full scope peer review findings from 2015 were closed by an independent assessment conducted August 2017.

NRC staff observed the August 2017 independent assessment for GGNS's review and closure of F&Os on August 2017 at Jackson, MS. The closure process followed the guidance outlined in Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13, "Close-out of Facts and Observations (F&Os)" (Reference 17), as accepted by the NRC in a letter dated May 3, 2017, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-out of Facts and Observations" (Reference 18). Appendix X provides guidance to licensees for closing F&Os that were opened during the peer review process. As detailed in NRC's observation report of GGNS's closeout process (Reference 19), the NRC observers could not conclude that the licensee fully adhered to the endorsed guidance in conducting the F&O closure audit. Therefore, in order for the NRC to review the technical adequacy of the GGNS PRA with regard to SFCP, the LAR was supplemented by the F&Os and associated resolutions and conclusions (Reference 2).

The supplemental information listed 39 finding-level facts and observations from the 2015 full-scope industry peer review that were all closed by an independent assessment (IA) team on August 2017. The IA team documented the basis for each F&O to validate whether the F&O constituted a PRA upgrade, maintenance update; as well as ensuring that capability category II of the ASME PRA standard was met for each F&O. NRC staff confirms IA's team's assessment that none of the changes made to the GGNS PRA were considered PRA upgrade or use of a new PRA method.

NRC staff requested additional information to clarify the impact of F&O 4-14 which described the inadequate justification for the dismissal of previous plant failures from inclusion in the PRA. The IA team closed the finding due to additional justification stating that all failures included in the PRA must have occurred during the time frame for the PRA update (Sept 2006 – Aug 2012) and must meet the definition of a PRA functional failure. In response to RAI, GGNS stated that the peer reviewer noted that the time frame selected was appropriate and correctly used in the model; the data is representative of multiple refueling cycles; and the use of data prior to Sept 2006 would result in a more substantial overlap with generic data reference used for the update. To ensure the six year plant specific data provided in the PRA model continues to adequately represent the uniformity in plant design, operational practice, and experience, a sensitivity study of plant failure data since August 2012 was performed to determine if a further PRA update is required (Reference 4). This sensitivity study updated the initiating events, unavailability and reliability data to include initiating events, unavailability time and function failures over the six year period of Sept 2012 – Aug 2018 for risk significant components. An update was not applied to those failure modes and initiating events that did not occur in the six year period (Sept 2012 – Aug 2018) but existed in the prior data sampling period. This results in conservative outcome since incorporating additional exposure time without any new events; or demands without any new functional failures, will lower the initiating events frequency and the failure frequency, respectively. One potential issue identified during the unavailability analysis was the previous PRA analysis used calendar hours and reactor critical hours for the calculation of different test and maintenance unavailabilities. To maintain consistency, the current calculation continued to use the same methodology; however, this is non conservative since the use of calendar times results in smaller unavailability. The sensitivity study recommends corrective action to investigate the inconsistency in the unavailability exposure periods. For the objective of the sensitivity study, the NRC staff finds the non-conservative treatment of the unavailability analysis acceptable since the impact of the non-conservatism is offset by the conservatism in the initiating event and reliability analysis. The results of the quantification, incorporating data from prior and updated data, shows an increase of 3.80E-07 CDF and 1.64E-07 LERF which is within the acceptance guidelines of RG 1.174. Therefore, the NRC staff finds GGNS's response to RAI and sensitivity analysis acceptable.

The NRC staff finds the dispositions to the remaining open F&Os acceptable for this application. Based on the review of the information provided in the April 12, 2018, LAR (Reference 1), and supplemental letters dated June 7, 2018, November 30, 2018 and March 6, 2019 (References 2, 3 and 4), the NRC staff concludes that the review of the PRA is consistent with Regulatory Position 2.3.1, "Technical Adequacy of the PRA," of RG 1.177, Revision 1 (Reference 9) for this application. As summarized in this SE, the NRC staff concludes that any deficiencies identified during the review of the PRA have been resolved to support the evaluation of changes proposed to surveillance frequencies within the SFCP.

3.1.4.2 *Scope of the PRA*

The proposed changes to the Administrative Controls section of the TSs would require the licensee to evaluate each proposed change to a relocated surveillance frequency using NEI 04-10, Revision 1, to determine its potential impact on CDF and LERF from internal events, fires, seismic, other external events, and shutdown conditions. In cases where a PRA of sufficient scope or quantitative risk models are unavailable, the licensee uses bounding analyses, or other conservative quantitative evaluations. A qualitative screening analysis may be used when the surveillance frequency impact on plant risk is shown to be insignificant.

The licensee has at-power internal events and internal flooding PRA models. In accordance with NEI 04-10, Revision 1, the licensee will use these PRA models to perform quantitative evaluations to support the development of changes to surveillance frequencies in the SFCP. This is acceptable because the NRC- approved methodology in NEI 04-10, Revision 1, allows for more refined analysis to be performed supporting changes to surveillance frequencies in the SFCP. Section 3.3 of the LAR states "GGNS does not currently have a fire PRA model. Therefore, a bounding fire risk evaluation, based on information from the Individual Plant Examination of External Events (IPEEE) and other available insights for fire risk, will be performed for STI changes in accordance with the guidance of NEI 04-10 Revision 2." In response to NRC's request for additional information, the licensee clarified that the GGNS fire PRA model was not developed in accordance with NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities" and that analysis performed for the IPEEE and the fire PRA do not provide quantitative fire risk information that can be directly compared to the internal events PRA model on a quantitative basis. GGNS further stated, when assessing fire risk, a bounding assessment is performed considering the qualitative aspects of the risk, including the impact of fire initiators in applicable fire zones when one or more SSC is unavailable. GGNS currently offsets fire risk configurations by implementing risk management actions (RMAs) in accordance with the plant risk assessment for maintenance activities procedure to minimize the likelihood of a fire. For the SFCP, GGNS will employ the bounding qualitative analysis for surveillance frequency changes that includes the appropriate RMAs. A more detailed qualitative analysis is performed if the affected SSC is not modeled or included in the equipment out-of-service monitor (EOOS).

For other hazard groups for which a PRA model does not exist, a qualitative or bounding analysis, consistent with NEI 04-10, Revision 1, is performed to provide justification for the acceptability of the proposed test interval change. GGNS does not have a seismic, high winds or external flooding PRA, therefore a qualitative or bounding approach will be used to assess external event hazard risk for STI changes. Similarly, GGNS does not maintain a shutdown PRA model; however, GGNS does operate under a shutdown risk management program outlined in NUMARC 91-06 that will be used for shutdown risk evaluation; or an application specific shutdown may be performed for STI changes. Since the licensee's proposed analysis of external hazards is consistent with NEI 04-10 methodology for STI change evaluations in the absence of quantifiable PRA models, the NRC finds the licensee's treatment of external hazards acceptable.

Based on the application of NRC-approved NEI 04-10, Revision 1, as required by proposed TS 5.5.17, the NRC staff concludes that the licensee's evaluation methodology is sufficient to ensure the risk contribution of each surveillance frequency change is properly identified for evaluation and is consistent with Regulatory Position 2.3.2, "Scope of the Probabilistic Risk Assessment for Technical Specification Change Evaluations," of RG 1.177, Revision 1.

3.1.4.3 *PRA Modeling*

The licensee's methodology includes the determination of whether the SSCs affected by a proposed change to a surveillance frequency are modeled in the PRA. Where the SSC is directly or implicitly modeled, a quantitative evaluation of the risk impact may be carried out. The methodology adjusts the failure probability of the impacted SSCs, including any impacted CCF modes, based on the proposed change to the surveillance frequency. Where the SSC is not modeled in the PRA, bounding analyses are performed to characterize the impact of the proposed change to the surveillance frequency. Potential impacts on the risk analyses due to screening criteria and truncation levels are addressed by the requirements for PRA technical adequacy, consistent with guidance contained in RG 1.200, Revision 2, and by sensitivity studies identified in NEI 04-10, Revision 1.

The NRC staff concludes that the GGNS PRA modeling is consistent with the guidance in NEI-04-10, Revision 1, and therefore the modeling is sufficient to ensure an acceptable evaluation of risk for the proposed changes in surveillance frequency, and is consistent with Regulatory Position 2.3.3, "Probabilistic Risk Assessment Modeling," of RG 1.177, Revision 1.

3.1.4.4 *Assumptions for Time Related Failure Contributions*

The failure probabilities of SSCs modeled in PRAs may include a standby time-related contribution and a cyclic demand-related contribution. In Section 3.4, "Identification of Key Assumptions," of the LAR dated April 12, 2018, the licensee states that the determination of standby failure rates are a key source of uncertainty and therefore, sensitivity studies will be performed on standby failure rates for STI evaluations. The NEI 04-10, Revision 1 criteria adjust the time-related failure contribution of SSCs affected by the proposed change to a surveillance frequency. This is consistent with RG 1.177, Revision 1, Section 2.3.3, which permits separation of the failure rate contributions into demand and standby for evaluation of SRs. If the available data do not support distinguishing between the time-related failures and demand failures, then the change to surveillance frequency is conservatively assumed to impact the total failure probability of the SSC, including both standby and demand contributions. The SSC failure rate per unit time is assumed to be unaffected by the change in test frequency, such that the failure probability is assumed to increase linearly with time. This assumption will be confirmed by the required monitoring and feedback implemented after the change in surveillance frequency is implemented. The NEI 04-10, Revision 1, process requires consideration of qualitative sources of information with regard to potential impacts of test frequency on SSC performance, including industry and plant-specific operating experience, vendor recommendations, industry standards, and code-specified test intervals. Thus, the NRC staff concludes that the licensee's process is not reliant upon risk analyses as the sole basis for the proposed changes because the licensee has, and will, apply the associated guidance in NRC-approved NEI 04-10, Revision 1.

The potential benefits of a reduced surveillance frequency, including reduced downtime and reduced potential for restoration errors, test-caused transients, and test-caused wear of equipment, are identified qualitatively, but are not quantitatively assessed. The NRC staff concludes that the licensee applied NRG-approved NEI 04-10, Revision 1, to employ reasonable assumptions with regard to extensions of STIs, and the requested changes are consistent with Regulatory Position 2.3.4, "Assumptions in Completion Time and Surveillance Frequency Evaluations," of RG 1.177, Revision 1.

3.1.4.5 *Sensitivity and Uncertainty Analyses*

The proposed amended TSs would require that changes to the frequencies listed in the SFCP be made in accordance with NEI 04-10, Revision 1. Therefore, the licensee will be required to have sensitivity studies that assess the impact of uncertainties from key assumptions of the PRA, uncertainty in the failure probabilities of the affected SSCs, impact on the frequency of initiating events, and any identified deviations from CC II of the PRA standard. Where the sensitivity analyses identify a potential impact on the proposed change, revised surveillance frequencies are considered, along with any qualitative considerations that may bear on the results of such sensitivity studies. In accordance with NEI 04-10, Revision 1, as required by proposed TS 5.5.14, the licensee will also perform monitoring and feedback of SSC performance, once the revised surveillance frequencies are implemented. Therefore, the NRC staff concludes that the licensee appropriately considered the possible impact of PRA model uncertainty and sensitivity to key assumptions and model limitations, and the LAR is consistent with Regulatory Position 2.3.5, "Sensitivity and Uncertainty Analyses Relating to Assumptions in Technical Specification Change Evaluations," of RG 1.177, Revision 1, because the licensee has, and will, apply the associated guidance in NRG-approved NEI 04-10, Revision 1.

3.1.4.6 *Acceptance Guidelines*

In accordance with NEI 04-10, Revision 1, as required by proposed TS 5.5.14, the licensee will quantitatively evaluate the change in total risk (including internal and external events contributions) in terms of CDF and LERF for both the individual risk impact of a proposed change in surveillance frequency and the cumulative impact from all individual changes to surveillance frequencies using NEI 04-10, Revision 1, in accordance with the TS SFCP. Each individual change to surveillance frequency must show a risk impact below 1E-6 per year for change to CDF, and below 1E-7 per year for change to LERF. These changes to CDF and LERF are consistent with the acceptance criteria of RG 1.174, Revision 2 (Reference 8), for very small changes in risk. Where the RG 1.174, Revision 2, acceptance criteria are not met, the process in NEI 04-10, Revision 1, either considers revised surveillance frequencies that are consistent with RG 1.174, Revision 2, or the process terminates without permitting the proposed changes. Where quantitative results are unavailable for comparison with the acceptance guidelines, appropriate qualitative analyses are required to demonstrate that the associated risk impact of a proposed change to surveillance frequency is negligible or insignificant. Otherwise, bounding quantitative analyses are required that demonstrate the risk impact is at least one order of magnitude lower than the RG 1.174, Revision 2, acceptance guidelines for very small changes in risk. In addition to assessing each individual SSC surveillance frequency change, the cumulative impact of all changes must result in a risk impact less than 1E-5 per year for change to CDF, and less than 1E-6 per year for change to LERF. The total CDF and total LERF must be reasonably shown to be less than 1E-4 per year and 1E-5 per year, respectively. These values are consistent with the acceptance criteria of RG 1.174, Revision 2, as referenced by RG 1.177, Revision 1 (Reference 9), for changes to surveillance frequencies.

Consistent with the NRC staff's SE dated September 19, 2007, for NEI 04-10, Revision 1 (Reference 7), the TS SFCP will require the licensee to calculate the total change in risk (i.e., the cumulative risk) by comparing a baseline model that uses failure probabilities based on surveillance frequencies prior to being changed per the SFCP, to a revised model that uses failure probabilities based on the changed surveillance frequencies. The NRC staff further notes that the licensee includes a provision to exclude the contribution to cumulative risk from individual changes to surveillance frequencies associated with insignificant risk increases (i.e., less than 5E-8 per year for CDF and 5E-9 per year for LERF) once the baseline PRA models are updated to include the effects of the revised surveillance frequencies.

The quantitative acceptance guidance of RG 1.174, Revision 3, is supplemented by qualitative information to evaluate the proposed changes to surveillance frequencies, including industry and plant-specific operating experience, vendor recommendations, industry standards, the results of sensitivity studies, and SSC performance data and test history. The final acceptability of the proposed change is based on all of these considerations and not solely on the PRA results. Post-implementation performance monitoring and feedback are also required to ensure continued reliability of the components. The licensee's application of NRC-approved NEI 04-10, Revision 1, provides acceptable methods for evaluating the risk increase associated with proposed changes to surveillance frequencies, consistent with Regulatory Position 2.4 of RG 1.177, Revision 1. Therefore, the NRC staff concludes that the proposed methodology satisfies the fourth key safety principle of RG 1.177, Revision 1, by assuring that any increase in risk is small, consistent with the intent of the Commission's Safety Goal Policy Statement.

3.1.5 *The Impact of the Proposed Change Should Be Monitored Using Performance Measurement Strategies*

The licensee's proposed TS 5.5.14 requires application of NEI 04-10, Revision 1 (Reference 6), in the SFCP. NEI 04-10, Revision 1, requires performance monitoring of SSCs whose surveillance frequencies have been revised as part of a feedback process to ensure that the change in test frequency has not resulted in degradation of equipment performance and operational safety. In response to RAI, GGNS stated performance monitoring strategies will be implemented to monitor changes to surveillance frequencies consistent with the requirements of NEI 04-10 Revision 1. GGNS's performance monitoring strategies include the following: (A) confirmation that no failure mechanisms that are related to the revised STI become important enough to alter the failure rates assumed in the justification of the program changes (B) performance monitoring ensures adequate component capability exists (C) component monitoring is expected for high safety significant SSCs as defined by the GGNS maintenance rule program (D) performance will be monitored per the monitoring requirements of the maintenance rule program; however monitoring unique to revised STI may be specified (E) output of the performance monitoring will be periodically re-assessed and appropriate adjustments made to the surveillance frequencies if needed. The monitoring and feedback includes consideration of Maintenance Rule (i.e., 10 CFR 50.65) monitoring of equipment performance. In the event of SSC performance degradation, the surveillance frequency will be reassessed in accordance with the methodology, in addition to any corrective actions that may be required by the Maintenance Rule. The performance monitoring and feedback specified in NEI 04-10, Revision 1 and GGNS's response to RAI, is sufficient to reasonably assure acceptable SSC performance and is consistent with Regulatory Position 3.2 of RG 1.177, Revision 1. Thus, the NRC staff concludes that the fifth key safety principle of RG 1.177, Revision 1, is satisfied.

4.0 REFERENCES

1. Larson, E., Entergy Operations, letter to U.S. Nuclear Regulatory Commission, Subject: "Application for Technical Specification Change Regarding Risk- Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," dated April 12, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18102B445).

2. Larson, E., Entergy Operations, letter to U.S. Nuclear Regulatory Commission, Subject: “Supplemental Information Needed for Acceptance of Requested Licensing Action RE: Adoption of Technical Specification Task Force (TSTF) Traveler TSTF-425, Revision 3,” dated May 23, 2018 (ADAMS Accession No. ML18158A514)

3. Larson, E., Entergy Operations, letter to U.S. Nuclear Regulatory Commission, Subject: “Response to Request for Additional Information for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program,” dated November 30, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18337A136)

4. Larson, E., Entergy Operations, letter to U.S. Nuclear Regulatory Commission, Subject: “Internal Events Probabilistic Risk Assessment (PRA) Sensitivity Study Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (TSTF-425),” dated March 06, 2019 (ADAMs Accession No. ML19066A088)

5. Technical Specifications Task Force, letter to U.S. Nuclear Regulatory Commission, Subject: Transmittal of TSTF, Revision 3, "Relocate Surveillance Frequencies to Licensee Control-RITSTF Initiative 5b," dated March 18, 2009 (ADAMS Package Accession No. ML090850642).

6. NEI 04-10, Revision 1, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies," April 2007 (ADAMS Accession No. ML071360456).

7. Nieh, H. K., U.S. Nuclear Regulatory Commission, letter to Mr. Biff Bradley, Nuclear Energy Institute, Subject: "Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) 04-10, Revision 1, 'Risk-Informed Technical Specifications Initiative 58, Risk-Informed Method for Control of Surveillance Frequencies (TAC No. MD6111),'" dated September 19, 2007 (ADAMS Accession No. ML072570267).

8. U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, Revision 2, May 2011 (ADAMS Accession No. ML100910006).

9. U.S. Nuclear Regulatory Commission, "An Approach for Plant-Specific, Risk-Informed Decision-making: Technical Specifications," Regulatory Guide 1.177, Revision 1, May 2011 (ADAMS Accession No. ML100910008).

10. U.S. Nuclear Regulatory Commission, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200, Revision 2, March 2009 (ADAMS Accession No. ML090410014).

11. U.S. Nuclear Regulatory Commission, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," NUREG-0800, Section 19.2, June 2007 (ADAMS Accession No. ML071700658).

12. U.S. Nuclear Regulatory Commission, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load," NUREG-0800, Section 19.1, Revision 3, September 2012 (ADAMS Accession No. ML12193A107).

13. U.S. Nuclear Regulatory Commission, "Risk-Informed Decision Making: Technical Specifications," NUREG-0800, Section 16.1, Revision 1, March 2007 (ADAMS Accession No. ML070380228).

14. American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009, "Addenda to ASME RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009, New York, NY.

15. NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," Revision 1, May 2006 and NEI 00-02 Appendix D, "Self Assessment Process for Addressing ASME PRA Standard RA-Sb-2005, as endorsed by NRC Regulatory Guide 1.200," October 2006 (ADAMS Accession Nos. ML061510619 and ML063390593, respectively).

16. NEI 05-04 "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," NEI 05-04, Revision 2, November 2008 (ADAMS Accession No. ML083430462).

17. Anderson, V., Nuclear Energy Institute, letter to U.S. Nuclear Regulatory Commission, Subject: “Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)” dated February 21, 2017 (ADAMS Accession No. ML17086A431)

18. Gitter, J. and Jane Ross-Lee, M., U.S. Nuclear Regulatory Commission, letter to Krueger, G., Nuclear Energy Institute, Subject: “U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13 Close-Out of Facts and Observations (F&Os)” dated May 3, 2017 (ADAMS Accession No. ML17079A427)

19. Rosenberg, S., U.S. Nuclear Regulatory Commission, memorandum to Giitter, J., U.S. Nuclear Regulatory Commission, Subject: “Nuclear Regulatory Commission Report on Observations of Implementation of the Industry Independent Assessment Team Close-Out of Facts and Observations After Appendix X Issued” dated December 1, 2017 (ADAMS Accession No. ML17265A812)