October 14, 2015

MEMORANDUM TO: Anthony J. Mendiola, Chief
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SUBJECT: SUMMARY OF AUGUST 27, 2015, CONFERENCE CALL WITH THE NUCLEAR ENERGY INSTITUTE

On August 27, 2015, the U.S. Nuclear Regulatory Commission (NRC) staff held a conference call with representatives from the Nuclear Energy Institute (NEI). The purpose of the meeting was to discuss NEI 06-09, Revision 0-A, "Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines," and its implementation in Technical Specifications Task Force Traveler (TSTF)-505, Revision 1, "Provide Risk-Informed Extended Completion Times RITSTF [Risk-Informed Technical Specifications Task Force] Initiative 4b." Information related to the conference call can be found in the Agencywide Documents Access and Management System package for the meeting at Accession No. ML15226A469.

The NRC staff began by reviewing some history on the development of NEI 06-09. NEI developed the document to describe a methodology for implementing risk-managed technical specifications. The implementation of this methodology to the standard technical specifications was developed subsequent to the review and approval of NEI 06-09. Continuing, the NRC staff reported that Diablo Canyon has filed a license-amendment request (LAR) for instrumentation and control (I&C) using TSTF-505 and is the first application to address I&C.

The NRC staff stated that as it conducted this LAR review, questions emerged about whether the implementation of TSTF-505 was consistent with the safety evaluation for NEI 06-09 Revision 0-A in accordance with Regulatory Guide (RG) 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications." There are five criteria in RG 1.177 but the NRC staff is unclear how all of these five criteria (e.g., redundancy and diversity) are maintained in the implementation of the DC TSTF-505 LAR.

In its opening remarks, the NEI representative stated that the white paper provided by the NRC staff in advance of the meeting was useful in helping prepare for the meeting. A copy of the white paper is enclosed with these minutes. The NEI representative further stated that it was not able to provide comments on the white paper.

The NRC staff recognized that NEI had limited time to review the white paper and recommended that a workshop be held to discuss it in more detail. It was then recommended that a public meeting be held to discuss the technical issues to gain a better understanding before a workshop was conducted. Topics suggested for the meeting included the following:

- the interpretation of NEI 06-09 and its safety evaluation
- the role of design requirements and the application of Title 10 of the *Code of Federal Regulations* 50.36, "Technical Specifications"
- the TS and probabilistic risk assessment (PRA) definition of functionality
- the definition of loss of function

It was suggested by the NEI representative that senior NRC management continue to be involved and NEI would include its senior sponsor.

After a thorough discussion of plans to hold a meeting and workshop, the NRC staff noted that there is currently an effort to schedule a meeting between the NRC staff and NEI. The meeting is focused on the definition of PRA functionality. It was agreed that this meeting would be held and then a decision would be made on the need for additional meetings and a workshop.

There were no action items from this meeting.

Enclosure:
TSTF-505 Implementation – Clarification
Questions

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Enclosure:
TSTF-505 Implementation – Clarification Questions

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ADAMS Accession Nos.: Pkg: ML15226A469; Summary: ML15240A003; Notice: ML15230A521 *via email

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Technical Specifications Task Force Traveler-505
Implementation – Clarification Questions

Context: A nuclear power plant (NPP) control system controls plant operations within parameters required by the safety analysis report (SAR) using both manual and automatic means. When NPP operation exceeds monitored parameters and enters into an unsafe condition as a result of control system failure or protection system failure or through operational error the NPP protection system is designed to restore the plant to a safe state. Failure modes of the NPP control system include control system failure. Control system failure is mitigated by the protection system, whereas the protection system cannot be tolerated to fail and therefore it is designed to meet the single failure criterion (among other criteria). In summary, improper operation or failure must be mitigated by the protection system even when the protection system is degraded by a single failure.

Separation of Protection and Control: NPPs, including Westinghouse plants, may share protection system parameter inputs between both reactor protection and control systems. Regulatory requirements for sharing equipment and parameter inputs include General Design Criteria No. 24, and Institute of Electrical and Electronics Engineers (IEEE) 279-1991 Clauses 4.7.1-3, which essentially requires that additional redundancy must be designed into the protection system when there is share equipment between protection and control systems. In addition, IEEE 279-1971 Clause 4.7.1 requires that shared equipment must be classified as part of the protection system, and Clause 4.7.3 requires that the any failure of shared protection and control equipment must be mitigated by the protection system even when the remaining portions of the protection system are degraded by a single failure. The Westinghouse Standard Technical Specification limit the time a plant is allowed to operate with one required instrument channel inoperable in either the reactor trip system (RTS) or engineered safety features actuation system (ESFAS) or both. This technical specification condition remedial action has been determined to be consistent with SAR design criteria and supporting analyses (see Updated Final SAR (UFSAR) Chapter 15 and the associated U.S. Nuclear Regulatory Commission (NRC or the Commission) safety evaluation thereof). Adoption of Technical Specifications Task Force Traveler (TSTF)-505 proposes to allow plant operation in a condition where two or more RTS & ESFAS instrumentation channels are inoperable, potentially coupling protections and control systems, and potentially creating the possibility of a new or different kind of accident. The regulations state (see item 50.92(c) of Title 10 of the *Code of Federal Regulations* (10 CFR 50.92(c)) that "The Commission may make a final determination, under the procedures in § 50.91, that a proposed amendment to an operating license…involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not…Create the possibility of a new or different kind of accident from any accident previously evaluated." Information is needed to understand how: (1) no new events requiring protection exist in the proposed new conditions, (2) all original events are protected against, and/or (3) Probabilistic Risk Assessment (PRA) modeling adequately addresses protection and control interactions.

Meeting Regulatory Requirements: The NRC used the criteria in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decision Making: Technical Specifications" to evaluate the acceptability of Nuclear Energy Institute (NEI) 06-09 Rev.0-A; RG 1.177 contains five criteria. The first criteria is that proposed changes meet current regulations; the NRC staff concluded the proposed technical specifications will meet the current regulations in part because the "LCOs [limiting conditions for operation] themselves would remain unchanged as would the remedial actions or shutdown requirements." It is the intent of the technical review branch for instrumentation and control to confirm this understanding by comparing the changes in the textual descriptions of the remedial actions against specific regulatory requirements as described below.

Safe Operation Criteria: The regulations state (see 10 CFR 50.92(a)) that "In determining whether an amendment to a license…will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses…to the extent applicable and appropriate." The regulations also state (see 10 CFR 50.57(a)(2)) that an operating license may be issued by the Commission upon finding that the facility will operate in conformity with the application as amended. In general, Chapter 15 of the UFSAR states that analysis of each fault condition is based on a conservative set of initial conditions corresponding to the most adverse set of conditions that can occur during operation with deviations as permitted by the plant technical specification (TS). In summary, the regulations do not allow operation in a condition that has not been demonstrated to be acceptable by the bounding analysis of Chapter 15. Therefore, an applicant may: (1) demonstrate that all changed conditions are bounded by the current UFSAR analyses, (2) perform additional analyses to be added to the UFSAR, and/or (3) demonstrate operation consistent with the PRA model and supporting analyses provide reasonable assurance of safety.

Operable vs. PRA Functional: After the approval of NEI 06-09 Rev. 0-A, instrument channels could be considered to be in one of four conditions: (1) operable, (2) operable but degraded or non-conforming (see RIS 2013-05), (3) PRA Functional, and (4) non-functional. As explained in the attachment to RIS 2013-05:

> "Section 3.8 of Part 9900 [now superseded by IMC 0326] includes Operable/Operability as a defined term and discusses its meaning in the context of the CLB design by the following statement:
>
> > In order to be considered operable, an SSC [structures, systems and components] must be capable of performing the safety functions specified by its design, <u>within the required range of design physical conditions</u>, initiation times, and mission times. [Emphasis added]"

Therefore, PRA Functional and non-functional refer to conditions which do not meet the definition of operability. Additional clarity is needed to address how this concept applies to instrumentation. What analysis or rationale ensures that operation consistent with PRA Functional (as opposed to TS Operable) criteria ensures that operation is safe?

TS Derivation from SAR: The regulations (see 10 CFR 50.36(b)) also state: "The technical specifications will be derived from the analyses and evaluation included in the safety analysis report…" In the current TS, LCO requirements are derived from the safety analysis report. If the requirements of the LCOs are not met for certain conditions the staff does provide remedial actions (Required Actions, and Completion Times) in lieu of the requirement to shut down the plant. The regulatory basis for risk-informed completion times (RICTs) is unclear since RICTs are only allowed where there is no "total loss of safety function" but how "the total loss of function" determination corresponds to the safety analysis report is not described. Individual

instrument channels can be considered to be “PRA Functional” but how the definition of “PRA Functional” corresponds to the safety analysis report is not described.
Spatial Dependency: Further, the regulations also state (see IEEE 279-1971 Clause 3(3)) that the design bases documentation must include the minimum number and location of sensors required to monitor adequately, for protection function purposes, the variables required in order to provide protective actions that have special dependence. TSTF-505 proposes new conditions with reduced numbers of spatially relevant sensors from those allowed by current Standard Technical Specification LCOs, without explicitly identified analysis or justification. Information is needed to: (1) demonstrate that allowed reductions in the number of spatially dependent sensors are bounded by the current UFSAR analyses, (2) demonstrate by performing additional analyses (to be added to the UFSAR), and/or (3) demonstrate that operation consistent with the PRA model and supporting analyses adequately ensures safety.

PRA Modeling: Section 3.2.3 of NEI 06-09 Rev.0-A states:

> “If a degraded or nonconforming condition existing on a component can be explicitly modeled by the station’s PRA, then a situation specific RICT can be calculated. In these cases the PRA analysis supporting the RICT calculation must be documented, retrievable, and able to be referenced using normal operator documentation mechanisms (e.g., Control Room Logs or other equivalent methods). In the RICT calculation, equipment PRA functionality may be considered. The evaluation for the applicability of crediting “PRA functionality” shall be conducted in accordance with the guidance provided in Item 11 of Section 2.3.1. This guidance is intended to address separate operability and PRA functionality assessments which would allow a component to be considered both inoperable and PRA functional based on an evaluation of the same degraded condition.
>
> If the condition causing a component to be inoperable is not modeled in the PRA, and the condition has been evaluated and documented in the risk-management technical specification program as having no risk impact, then the RICT may be calculated assuming availability of the inoperable component and its associated system, subsystem or train. If there is no documented basis for exclusion, or if the condition was screened as low probability, then the inoperable component must be considered not functional.”

It is not clear how specifically identified degraded or nonconforming instrumentation are modeled or evaluated in each plant specific PRA (or if they are calculated to be no risk impact), for example:

(A) Slightly non-conservative setpoint
(B) Slightly non-conservative response time
(C) Operation outside the design basis
(D) Spatially dependent variables

It is also not clear how any conservative or bounding modeling assumptions are shown to be conservative or bounding (e.g., when new or different accidents exist).

Past Staff Positions: Previously, the Pressurized Water Reactor Owners Group has requested changes to the TS allowing limited operation in a condition where two channels were inoperable are considered to not meet regulatory requirements (Agencywide Document and Access

Management System (ADAMS) Accession No. ML043280419). In response (ADAMS Accession No. ML050350312) the NRC stated:

> “permitting plant operation with two channels inoperable in a two out of four channel configuration would not establish defense-in-depth criteria required for a deterministic evaluation without any hardship justification.”

Finally, the staff subsequently stated (ADAMS Accession No. ML051030036) that is was not obvious how either a risk informed or a deterministic evaluation allowing limited operation in a condition where two channels were inoperable could be supported successfully. The NRC also expressed its concern that in this condition, the design may not have the ability to reliability perform its safety function for all analyzed events (ADAMS Accession No. ML061600308). Neither the NEI 06-09 Rev.0-A or TSTF-505 addresses the hardship consideration or the reliability concerns.