SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-43

DOMINION ENERGY KEWAUNEE, INC.

KEWAUNEE POWER STATION

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated August 24, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092440398), as supplemented by letters dated October 22, 2009 (ADAMS Accession No. ML093070092), April 13, 2010 (ADAMS Accession Nos. ML101060517 and ML101040090), May 12, 2010 (ADAMS Accession No. ML101380399), July 1, 2010 (ADAMS Accession No. ML101890404), July 16, 2010 (ADAMS Accession No. ML102370370), August 18, 2010 (ADAMS Accession No. ML102371064), September 7, 2010 (ADAMS Accession No. ML102730383), September 8, 2010 (ADAMS Accession No. ML102580700), October 15, 2010 (ADAMS Accession No. ML102920037), and December 2, 2010 (ADAMS Accession No. ML103400328). Dominion Energy Kewaunee (the licensee) requested changes to the technical specifications (TS) for the Kewaunee Power Station (KPS). The proposed changes would revise the current technical specifications (CTS) to the improved technical specifications (ITS).

The following Safety Evaluation (SE) of the proposed ITS conversion is based on the application, the information provided to the NRC through the Kewaunee ITS Conversion web page hosted by Excel Services Corporation (as docketed by the licensee), and supplements provided, as discussed above.

To expedite its review of the application, the Nuclear Regulatory Commission (NRC) staff issued its requests for additional information (RAIs) through the Kewaunee ITS Conversion web page, and the licensee addressed the RAIs by providing responses on the web page. Entry into the database was protected so that only the licensee and NRC reviewers could enter information into the database to add RAIs (NRC) or provide responses to the RAIs (licensee); however, a public, read-only version of the website was available to allow members of the public to view the questions asked and the responses provided (http://www.excelservices.com/).

To be in compliance with the regulations for written communications for license amendment requests, the licensee has submitted a copy of the database in a submittal to the NRC. The

RAIs and responses for the various TS Sections are available in ADAMS in the following documents:

TS Section	ADAMS Document
1.0	ML101890404
2.0	ML101890404
3.0	ML101890404
3.1	ML101890404
3.2	ML102370370
3.3	ML102371064
3.4	ML101890404
3.5	ML102370370
3.6	ML101890404
3.7	ML102370370
3.8	ML102371064
3.9	ML102371064
4.0	ML101890404
5.0	ML102370370

During the review process, the public was able to access the website by going to www.excelservices.com. In the final phase of the review, the questions, responses, and all attachments were submitted as supplemental to the original submittal under oath or affirmation on the licensee's docket.

The additional information provided in the supplemental letters, did not expand the scope of the application as noticed and did not change the NRC staff's initial proposed finding of no significant hazards consideration published in the *Federal Register* on December 15, 2009 (74 FR 66384).

2.0 BACKGROUND

KPS has been operating in accordance with the TS issued with the original Facility Operating License dated December 21, 1973, as amended. The proposed conversion to the ITS is based upon:

- NUREG-1431, Revision 3.0, "Standard Technical Specifications – Westinghouse Plants," (ADAMS Accession No. ML062510017) as modified by: Technical Specifications Task Force (TSTF)-475-A, Rev. 1 "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action," TSTF-479-A, Rev. 0 "Changes to Reflect Revision of 10 CFR 50.55a," TSTF-482-A, Rev. 0 "Correct LCO 3.0.6 Bases," TSTF-485-A, Rev. 0 "Correct Example 1.4-1," TSTF-490-A, Rev. 0 "Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec," TSTF-491-A, Rev. 2 "Removal of Main Steam and Main Feedwater Valve Isolation Times from Technical Specifications," TSTF 493, Rev. 4 "Clarify Application of Setpoint Methodology for LSSS Functions," TSTF-497-A, Rev. 0 "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years

or Less," TSTF-511-A, Rev. 0 "Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26."

- KPS CTS
- "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (Final Policy Statement), published on July 22, 1993 (58 FR 39132); and
- 10 CFR 50.36, "Technical specifications," as amended July 19, 1995 (60 FR 36953).

Hereinafter, the proposed TS for KPS are referred to as the ITS, the existing TS are referred to as the CTS, and the improved standard TS, as issued in NUREG-1431, are referred to as the Improved Standard Technical Specifications (ISTS). The corresponding Bases are ITS Bases, CTS Bases, and ISTS Bases, respectively. In addition to basing the ITS on the ISTS, the Final Policy Statement, and the requirements in 10 CFR 50.36, the licensee retained portions of the CTS as a basis for the ITS.

During the course of its review, the NRC staff utilized the KPS ITS conversion database to issue RAIs. The RAIs served to clarify the ITS with respect to the guidance in the Final Policy Statement and the ISTS. The NRC staff required that the licensee docket the KPS ITS conversion database in a sworn statement with regards to its accuracy, as well as docket all RAIs and responses under oath and affirmation, in a supplement to the license amendment. The licensee also proposed changes of a generic nature that were not in the ISTS. The NRC staff requested that the licensee submit such generic changes as proposed changes to the ISTS using the industry TSTF Travelers. These generic issues were considered for specific applications in the KPS ITS.

Consistent with the Commission's Final Policy Statement and 10 CFR 50.36, the licensee proposed transferring some CTS requirements to licensee-controlled documents (such as the KPS Technical Requirements Manual (TRM), for which changes to the documents by the licensee are controlled by a regulation (e.g., 10 CFR 50.59) and which may be made without prior NRC approval. NRC-controlled documents, such as the TS, may not be changed by the licensee without prior NRC approval. In addition, human factors principles were emphasized to add clarity to the CTS requirements being retained in the ITS, and to define more clearly the appropriate scope of the ITS. Further, significant changes were proposed to the CTS Bases to make each ITS requirement clearer and easier to understand.

The overall objective of the proposed amendment, consistent with the Final Policy Statement, is to rewrite, reformat, and streamline the KPS CTS to provide clearer, more readily understandable requirements to ensure safer operation of the plant, while still satisfying the requirements of 10 CFR 50.36. During its review, the NRC staff relied on the Final Policy Statement, 10 CFR 50.36, and the ISTS as guidance for acceptance of CTS changes. This SE provides a summary basis for the NRC staff's conclusion that use of the licensee's proposed ITS based on ISTS, as modified by plant-specific changes, is acceptable for continued operation of the Kewaunee Power Station. This SE also explains the NRC staff's conclusion that the ITS

are consistent with the KPS current licensing basis (CLB) and the requirements of 10 CFR 50.36.

This SE relies on the following license conditions to be included in the facility operating license: (1) the schedule for the first performance of new and revised surveillance requirements (SRs); and (2) the relocation of CTS requirements into licensee-controlled documents as part of the implementation of the ITS.

3.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act (the "Act") requires that applicants for nuclear power plant operating licenses will provide:

> (S)uch technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization . . . of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TS. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and the mitigation of accident consequences. As recorded in the Statements of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports" (33 FR 18610, December 17, 1968), the Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS.

For several years, the NRC and industry representatives sought to develop guidelines for improving the content and quality of nuclear power plant TS. On February 6, 1987, the Commission issued an interim policy statement on TS improvements, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, utility owners groups and the NRC staff developed ISTS (e.g., NUREG-1431) that would establish model TS based on the Commission's policy for each primary reactor type. In addition, the NRC staff, licensees, and owners groups developed generic administrative and editorial guidelines in the form of a "Writer's Guide" for preparing TS, which gives appropriate consideration to human factors engineering principles and was used throughout the development of plant-specific ITS.

In September 1992, the Commission issued NUREG-1431, Revision 0, which was developed using the guidance and criteria contained in the Commission's Interim Policy Statement. The ISTS in NUREG-1431 were established as a model for developing the ITS for Westinghouse-

type plants. The ISTS reflect the results of a detailed review of the application of the Interim Policy Statement criteria which have been incorporated in 10 CFR 50.36(c)(2)(ii) to generic system functions, which were published in a “Split Report” issued to the nuclear steam supply system vendor owners groups in May 1988. ISTS also reflect the results of extensive discussions concerning various drafts of ISTS so that the application of the TS criteria and the Writer's Guide would consistently reflect detailed system configurations and operating characteristics for all reactor designs. As such, the generic Bases presented in NUREG-1431 provide an abundance of information regarding the extent to which the ISTS present requirements that are necessary to protect public health and safety.

The ISTS in NUREG-1431, Revision 3.0, “Standard Technical Specifications – Westinghouse Plants” (ADAMS Accession No. ML062510073) as modified by the following TSTF Travelers, applies to Kewaunee: TSTF-475-A, Rev. 1 “Control Rod Notch Testing Frequency and SRM Insert Control Rod Action,” TSTF-479-A, Rev. 0 “Changes to Reflect Revision of 10 CFR 50.55a” TSTF-482-A, Rev. 0 “Correct LCO 3.0.6 Bases,” TSTF-485-A, Rev. 0 “Correct Example 1.4-1,” TSTF-490-A, Rev. 0 “Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec,” TSTF-491-A, Rev. 2 “Removal of Main Steam and Main Feedwater Valve Isolation Times from Technical Specifications,” TSTF-493, Rev 4 “Clarify Application of Setpoint Methodology for LSSS Functions,” TSTF-497-A, Rev. 0 “Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less,” TSTF-511-A, Rev. 0 “Eliminate Working Hour Restrictions from TS 5.2.2 to Support Compliance with 10 CFR Part 26.”

On July 22, 1993, the Commission issued its Final Policy Statement, expressing the view that satisfying the guidance in the policy statement also satisfies Section 182a of the “Act” and 10 CFR 50.36. The Final Policy Statement described the safety benefits of the ISTS and encouraged licensees to use the ISTS as the basis for plant-specific TS amendments and for complete conversions to ITS based on the ISTS. In addition, the Final Policy Statement gives guidance for evaluating the required scope of the TS and defines the guidance criteria to be used in determining which of the LCOs and associated SRs should remain in the TS. The Commission noted that in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TS, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in Portland General Electric Co. (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

> There is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

By this approach, existing LCO requirements that fall within or satisfy any of the criteria in the Final Policy Statement should be retained in the TS; those LCO requirements that do not fall

within or satisfy these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36 (60 FR 36953, July 19, 1995). The four criteria are stated as follows:

Criterion 1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3 A structure, system, or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4 An SSC which operating experience or probabilistic risk assessment (PRA) has shown to be significant to public health and safety.

Part 4.0 of this SE explains the NRC staff's determination that the conversion of the Kewaunee CTS to ITS based on ISTS, as modified by plant-specific changes, is consistent with the Kewaunee current licensing basis, the requirements and guidance of the Final Policy Statement, and 10 CFR 50.36.

4.0 TECHNICAL EVALUATION

In its review of the Kewaunee ITS application, the NRC staff evaluated five kinds of CTS changes as defined by the licensee. The NRC staff's review also included an evaluation of whether existing regulatory requirements are adequate for controlling future changes to requirements that are removed from the CTS and placed in licensee-controlled documents. The following are the five types of CTS changes:

A Administrative - Changes to the CTS that do not result in new requirements or change operational restrictions and flexibility.

M More Restrictive - Changes to the CTS that result in added restrictions or reduced flexibility.

L Less Restrictive - Changes to the CTS that result in reduced restrictions or added flexibility.

LA Removed Details - Changes to the CTSs that eliminate detail and relocate the detail to a licensee-controlled document. Typically, this involves details of system design and system description including design limits, description of system operation, procedural details for meeting TS requirements or reporting

requirements, and cycle-specific parameter limits and TS requirements redundantly located in other licensee-controlled documents.

R Relocated Specifications - Changes to the CTS that relocate requirements that do not meet the selection criteria of 10 CFR 50.36(c)(2)(ii).

The ITS application included a justification for each proposed change to the CTS in a numbered discussion of change (DOC), using the above letter designations as appropriate. In addition, the ITS application included an explanation of each difference between ITS and ISTS requirements in a numbered justification for deviation.

The changes to the CTS, as presented in the ITS application, are listed and described in the following five tables (for each ITS section) provided as Attachments 1 through 5 to this SE:

- Table A - Administrative Changes
- Table L - Less Restrictive Changes
- Table LA - Removed Detail Changes
- Table M - More Restrictive Changes
- Table R - Relocated Specifications

These tables provide a summary description of the proposed changes to the CTS. The tables are only meant to summarize the changes being made to the CTS. The details as to what the actual changes are and how they are being made to the CTS or ITS are provided in the licensee's application and supplemental letters.

In several sections of the DOC tables, an equivalency is drawn between CTS modes (i.e. HOT STANDBY) and ITS MODES. These comparisons are general equivalencies only, since except for ITS MODE 5 and CTS COLD SHUTDOWN, there is no exact equivalence between the two tables of definitions.

The NRC staff's evaluation and additional description of the kinds of changes to the CTS requirements listed in Tables A, M, L, and R attached to this SE are presented in Sections A-E as follows:

- Section A - Administrative Changes (A)
- Section B - More Restrictive Changes (M)
- Section C - Less Restrictive Changes (L)
- Section D - Removed Detail Changes (LA)
- Section E - Relocated Specifications (R)

The control of specifications, requirements, and information relocated from the CTS to licensee-controlled documents is described in Section F below.

A. Administrative Changes to the CTS (A)

Administrative changes are intended to incorporate human factors principles into the form and structure of the ITS so that plant operations personnel can use them more easily. These changes are editorial in nature or involve the reorganization or reformatting of CTS requirements without affecting technical content or operational restrictions. In order to ensure consistency, the NRC staff review of the licensee proposed TS used the ISTS as guidance to reformat and make other administrative changes that do not involve technical changes to CTS. Administrative changes are not intended to add, delete, or relocate any technical requirements of the CTS. Examples of changes that Staff has found acceptable include:

- Identifying plant-specific wording for system names
- Reformatting, renumbering, and rewording of Current Technical Specifications (CTS) with no change in intent
- Splitting up requirements currently grouped under a single current specification and moving them to more appropriate locations in two or more specifications of the ITSs
- Presentation changes that involve rewording or reformatting for clarity (including moving an existing requirement to another location within the TS) but that do not involve a change in requirements
- Wording changes and additions that are consistent with CTS interpretation and practice and that more clearly or explicitly state existing requirements
- Deletion of obsolete TS
- Deletion of redundant TS requirements that exist elsewhere in the TS.

Table A attached to this SE lists the administrative changes being made in the Kewaunee ITS conversion. Table A is organized in ITS order by each A-type DOC to the CTS, provides a summary description of the administrative change that was made, and provides CTS and ITS references. The NRC staff reviewed all of the administrative and editorial changes proposed by the licensee and finds them acceptable because they are compatible with the ISTS, do not result in any change in operating requirements, and are consistent with the Commission's regulations.

B. More Restrictive Changes to the CTS (M)

The licensee, in electing to implement the ISTS, proposed a number of requirements that are more restrictive than those in the CTS. The ITS requirements in this category include requirements that are either new to Kewaunee, more conservative than corresponding

requirements in the CTS, or have additional restrictions that are not in the CTS, but are in the ISTS.

These changes include additional requirements that decrease allowed outage times, increase the frequency of surveillances, impose additional surveillances, increase the scope of specifications to include additional plant equipment, increase the applicability of specifications, or provide additional actions. These changes are generally made to conform to NUREG-1431 and have been evaluated to not be detrimental to plant safety.

Table M attached to this SE lists the more restrictive changes being made in the Kewaunee ITS conversion. Table M is organized in ITS order by each M-type DOC to the CTS. It provides a summary description of each more restrictive change that was adopted, and references the affected CTS and ITS. The staff reviewed each M-type DOC and found the changes to be acceptable because these changes provide additional restrictions on plant operation that enhance safety.

Two DOC's currently in the Table L (Less Restrictive) describe changes that, upon further staff review, were determined to be, on balance, better described as "M", or More Restrictive changes. They are:

- DOC 3.8.1 L05

- This DOC is identified in the Attached DOC Table as being a Less Restrictive Change. The description of the change is accurate, however, on balance, the new TS is actually More Restrictive because, in part, this new TS adds a requirement that was not in the KPS CTS.

- TS 3.8.2 L01

- This DOC is identified in the Attached DOC Table as being a Less Restrictive Change. The description of the change is accurate, however, on balance, the new TS is actually More Restrictive because this new LCO requires an AC Source in MODES 5 and 6. This was not required in the KPS CTS.

C. <u>Less Restrictive Changes to the CTS (L)</u>

Less restrictive changes include deletions of and relaxations to portions of the CTS requirements that are being retained in the ITS. When requirements have been shown to give little or no safety benefit, their relaxation or removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new NRC staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups' comments on ISTS. In developing the ISTS, NRC staff reviewed generic relaxations contained in the ISTS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The Kewaunee plant design was also reviewed to determine if their specific design basis and licensing basis are consistent with the technical basis for the model requirements in the ISTS and thus provide a basis for ITS.

All changes to the CTS that involved deletions of or relaxations to portions of CTS requirements can be grouped in the following nine categories:

Change Categories:

Category 1 – Relaxation of LCO Requirements

Category 2 – Relaxation of Applicability

Category 3 – Relaxation of Completion Time

Category 4 – Relaxation of Required Action

Category 5 – Deletion of Surveillance Requirement

Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria

Category 7 – Relaxation of Surveillance Frequency

Category 8 – Deletion of Reporting Requirements

Category 9 – Allowed Outage Time, Surveillance Frequency, and Bypass Time Extensions Based on Generic Topical Reports

The following discussion addresses why these categories of less restrictive changes are acceptable:

Category 1 – Relaxation of LCO Requirement

Certain CTS LCOs specify limits on operational and system parameters beyond those necessary to ensure meeting safety analysis assumptions and, therefore, are considered overly restrictive. The CTS also contain operating limits that have been shown to give little or no safety benefit to the operation of the plant. The ITS, consistent with the guidance in the ISTS, would delete or revise such operating limits.

These changes reflect the Standard Technical Specifications (STS) approach to provide LCO requirements that specify the protective conditions that are required to meet safety analysis assumptions for required features. These conditions replace the lists of specific devices used in the CTS to describe the requirements needed to meet the safety analysis assumptions.

TS changes represented by this type allow operators to more clearly focus on issues important to safety. The resultant ITS LCOs maintain an adequate degree of protection consistent with the safety analysis. They also improve focus on issues important to safety and provide reasonable operational flexibility without adversely affecting the safe operation of the plant. Changes involving the relaxation of LCOs are consistent with the guidance established by the ISTS taking into consideration the Kewaunee CLB. Therefore, based on the above, Category 1 changes are acceptable.

Category 2 – Relaxation of Applicability

The CTS require compliance with an LCO during the applicable Mode(s) or other conditions specified in the Specification's Applicability statement. CTS Applicability includes specific defined terms of reactor conditions and more general terms such as "all MODES." Generalized applicability conditions are not used in ITS, therefore the ITS eliminates CTS requirements such as "all MODES" replacing them with ITS defined MODES or applicable conditions that are consistent with the application of the plant safety analyses assumptions for OPERABILITY of the required features.

Further, where CTS Applicability requirements are inconsistent with the applicable accident analyses assumptions for a system, subsystem, or component specified in the LCO, the licensee proposed to change the LCO to establish a consistent set of requirements in the ITS. These modifications or deletions are acceptable because, during the operational or other conditions specified in the ITS applicability requirements, the LCOs are consistent with the applicable safety analyses. Changes involving relaxation of applicability requirements are consistent with the guidance established by the ISTS, taking into consideration the Kewaunee CLB. Therefore, based on the above, Category 2 changes are acceptable.

Category 3 – Relaxation of Completion Time

Upon discovery of a failure to meet an LCO, the TS specify time limits for completing Required Actions of the associated TS Conditions. Required Actions establish remedial measures that must be taken within specified Completion Times. Completion Times specify limits on the duration of plant operation in a degraded condition. Incorporating longer Completion Times is acceptable because such Completion Times continue to be based on the operability status of redundant TS required features, the capacity and capability of remaining TS-required features, provision of a reasonable time for repairs or replacement of required features, vendor-developed standard repair times, and the low probability of a DBA occurring during the repair period. Changes involving relaxation of Completion Times are consistent with the guidance established by the Commission, taking into consideration the Kewaunee CLB. These changes are generally made to conform to NUREG-1431, and have been evaluated to not be detrimental to plant safety. Therefore, based on the above, Category 3 changes are acceptable.

Category 4 – Relaxation of Required Action

LCOs specify the lowest functional capability or performance level of equipment that is deemed adequate to ensure safe operation of the facility. When an LCO is not met, the TS specify actions to restore the equipment to its required capability or performance level, or to implement remedial measures providing an equivalent level of protection. These actions minimize the risk associated with continued operation while providing time to repair inoperable features. Some of the Required Actions are modified to place the plant in a MODE in which the LCO does not apply. Adopting Required Actions from NUREG-1431 is acceptable because the Required Actions take into account the OPERABILITY status of redundant systems of required features, the capacity and

capability of the remaining features, and the compensatory attributes of the Required Actions as compared to the LCO requirements.

Compared to CTS required actions, certain proposed ITS actions would result in extending the time period during which the licensee may continue to operate the plant with specified equipment inoperable. Upon expiration of this time period, further action, which may include shutting down the plant, is required. These ITS actions provide measures that adequately compensate for the inoperable equipment, and are commensurate with the safety importance of the inoperable equipment, plant design, and industry practice. Therefore, these action requirements will continue to ensure safe operation of the plant. Changes involving relaxations of action requirements are consistent with the guidance established by the ISTS, taking into consideration the Kewaunee CLB. Therefore, based on the above, Category 4 changes are acceptable.

Category 5 – Deletion of Surveillance Requirement

The CTS require maintaining LCO specified structures, systems, and components (SSCs) operable by meeting SRs in accordance with specified SR frequencies. This includes conducting tests to demonstrate that such SSCs are operable and that LCO specified parameters are within specified limits. When the test acceptance criteria and any specified conditions for the conduct of the test are met, the equipment is deemed operable. The changes of this category relate to deletion of CTS SRs, deletion of acceptance criteria, and deletion of the conditions required for performing the SR.

The ITS eliminates unnecessary CTS Surveillance Requirements that do not contribute to verification that the equipment used to meet the Limiting Condition for Operation (LCO) can perform its required functions. Deleting the SRs, including acceptance criteria and/or conditions for performing the SRs, for these items is consistent with the objective of the ISTS, without reducing confidence that the equipment is operable. Appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its safety functions. For example, the CTS contain SRs that are not included in the ISTS for a variety of reasons. These include deletion of SRs for measuring values and parameters that are not necessary to meet ISTS LCO requirements. In addition, the ISTS may not include reference to specific acceptance criteria contained in the CTS, because these acceptance criteria are not necessary to meet ISTS LCO requirements, or are defined in other licensee controlled documents.

This deletion of SRs is acceptable because appropriate testing standards are retained for determining that the LCO required features are operable as defined by the ISTS taking into consideration the Kewaunee CLB. Therefore, based on the above, Category 5 changes are acceptable.

Category 6 – Relaxation of Surveillance Requirement Acceptance Criteria

Prior to placing the plant in a specified operational Mode or other condition stated in the applicability of an LCO, and in accordance with the specified SR time interval thereafter, the CTS require establishing the operability of each LCO-required component by

meeting the SRs associated with the LCO. This usually entails performance of tests to demonstrate the operability of the LCO-required components, or the verification that specified parameters are within LCO limits. A successful demonstration of operability requires meeting the specified acceptance criteria, as well as any specified conditions, for the conduct of the test. Relaxations of CTS SRs can include relaxing both the acceptance criteria and the conditions of performance.

For example, the ITS allows some Surveillance Requirements to verify OPERABILITY under actual or test conditions. Adopting the ITS allowance for these conditions is acceptable because required features cannot distinguish between an “actual” signal or a “test” signal. Also included are changes to CTS SRs that are replaced in the ITS with separate and distinct testing requirements that, when combined, provide OPERABILITY verification of all components required in the LCO for the features specified in the CTS. Changes that provide exceptions to Surveillance Requirements to allow for variations that do not affect the results of the test are also included in this category.

These relaxations of CTS SRs optimize test requirements for the affected safety systems and increase operational flexibility. These CTS SR relaxations are consistent with the guidance established by the ISTS in consideration of the Kewaunee CLB. Therefore, based on the above, Category 6 changes are acceptable.

Category 7 – Relaxation of Surveillance Frequency

Prior to placing the plant in a specified operational Mode or other condition stated in the applicability of an LCO, and in accordance with the specified SR time interval (frequency) thereafter, the CTS require establishing the operability of each LCO required component by meeting the SRs associated with the LCO. This usually entails performance of tests to demonstrate the operability of the LCO-required components, or the verification that specified parameters are within LCO limits. A successful demonstration of operability requires meeting the specified acceptance criteria, as well as any specified conditions, for the conduct of the test, at a specified frequency based on the reliability and availability of the LCO-required components.

Category 7 relaxations of CTS SRs include extending the interval between the SRs. Increasing the time interval between Surveillance tests in the ITS results in decreased equipment unavailability due to testing. Relaxation of Surveillance Frequency can also include the addition of Surveillance Notes which allow testing to be delayed until appropriate unit conditions for the test are established, or exempt testing in certain MODES or specified conditions in which the testing cannot be performed.

Reduced testing is also acceptable where operating experience or other deterministic criteria have demonstrated that these components usually pass the Surveillance when performed at the specified interval, thus the Surveillance Frequency is acceptable from a reliability standpoint. Surveillance Frequency changes to incorporate alternate train testing have also been shown to be acceptable where other qualitative or quantitative test requirements are required that are established predictors of system performance.

These CTS SR frequency relaxations are consistent with the guidance established by the ISTS taking into consideration the Kewaunee CLB. Therefore, based on the above, Category 7 changes are acceptable.

Category 8 – Deletion of Reporting Requirements

The CTS contain requirements that are redundant to reporting regulations in 10 CFR. Consistent with the ISTS, the ITS would omit many of the CTS reporting requirements and, in many cases, relies on the reporting requirements of 10 CFR 50.73 or other regulatory requirements. The ITS changes to reporting requirements are acceptable because the regulations provide adequate reporting requirements, or the reports do not affect continued plant operation. Therefore, this change has no effect on the safe operation of the plant. Deletion of these requirements reduces the administrative burden on the licensee and in turn allows increased attention to plant operations important to safety. Therefore, Category 8 changes have no adverse impact on the safe operation of the plant and are acceptable.

Category 9 – Allowed Outage Time, Surveillance Frequency, and Bypass Time Extensions Based on Generic Topical Reports

Kewaunee is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1431, Rev. 3.0, “Standard Technical Specifications, Westinghouse Plants.” Part of this conversion includes adoption of Technical Specification task Force (TSTF) travelers TSTF-411, TSTF-418 and WCAP 10271.

These TSTFs and WCAP are associated with changes to certain reactor protection system channel completion times, bypass times and surveillance test intervals; engineered safety system actuation system surveillance test intervals, logic cabinet completion times, bypass times, and surveillance test intervals; and reactor trip breakers surveillance test interval, completion times, and bypass times.

The proposed changes have been generically evaluated and approved by the NRC in WCAP-15376-P-A, Revision 1, “Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times,” March 2003; WCAP-10271-P-A, Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System, May 1986; WCAP-10271 Supplement 1-P-A, Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System- Supplement 1, May 1986; WCAP-l0271-P-A Supplement 2, Revision 1, Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System Supplement 2, Revision 1; and WCAP-14333-P-A, Rev. 1, Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times.

In the nine categories presented above, the proposed less restrictive changes to the CTS are acceptable because they will not adversely impact safe operation of the facility. The ITS requirements are consistent with the CLB, operating experience, and plant accident and transient analyses, and provide reasonable assurance that public health and safety will be protected.

Table L attached to this SE lists the less restrictive changes being made in the Kewaunee ITS conversion. Table L, which is organized in ITS order by each L-type DOC to the CTS, provides a summary description of the less restrictive change that was made, the CTS and ITS references, and a reference to the specific change type discussed above.

Two DOC's in the Table L describe changes that, upon further staff review, were determined to be, on balance, better described as "M", or More Restrictive changes. They are:

DOC 3.8.1 L05

This DOC is identified in the Attached DOC Table as being a Less Restrictive Change. The description of the change is accurate, however, on balance, the new TS is actually More Restrictive because, in part, this new TS adds a requirement that was not in the KPS CTS.

TS 3.8.2 L01

This DOC is identified in the Attached DOC Table as being a Less Restrictive Change. The description of the change is accurate, however, on balance, the new TS is actually More Restrictive because this new LCO requires an AC Source in MODES 5 and 6. This was not required in the KPS CTS.

D. <u>Removed Details (LA)</u>

When requirements or detailed information has been shown to give little or no safety benefit, their removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new NRC staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the owners groups' comments on the ISTS. The NRC staff reviewed generic relaxations contained in the ISTS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The Kewaunee design was also reviewed to determine if the specific design basis and licensing basis are consistent with the technical basis for the model requirements in the ISTS and thus provide a basis for ITS.

All of the changes to the CTS involving the removal of specific, detailed information from individual specifications evaluated to be Types 1 through 4 as described below:

Type 1 - Removing Details of System Design and System Description, Including Design Limits

The design of the facility is required to be described in the Updated Safety Analysis Report (USAR) by 10 CFR 50.34. In addition, the quality assurance requirements of Appendix B to 10 CFR Part 50 require that plant design be documented in controlled procedures and drawings and maintained in accordance with an NRC-approved Quality Assurance Topical Report (QATR). The regulation at 10 CFR 50.59 specifies controls for changing the facility as described in the USAR. The regulation at 10 CFR 50.54(a) specifies criteria for changing the QATR. The TRM is a general reference in the USAR

and changes to it are accordingly also subject to 10 CFR 50.59. The ITS Bases also contain descriptions of system design. ITS 5.5.12 specifies controls for changing the Bases. Removing details of system design is acceptable because the associated CTS requirements being retained without these details are adequate to ensure safe operation of the facility.

In addition, retaining such details in TS is unnecessary to ensure proper control of changes. Cycle-specific design limits are contained in the Core Operating Limits Report (COLR) in accordance with GL 88-16, “Removal of Cycle-Specific Parameter Limits From Technical Specifications,” dated October 3, 1988. ITS Section 5.6, “Reporting Requirements,” includes the programmatic requirements for the COLR. Therefore, it is acceptable to remove Type 1 details from the CTS and place them in licensee-controlled documents.

Type 2 - Removing Descriptions of System Operation

The plans for normal and emergency operation of the facility are required to be described in the USAR by 10 CFR 50.34. ITS 5.4.1.a and 5.4.1.e will require written procedures to be established, implemented, and maintained for plant operating procedures recommended in Appendix A of Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operation),” Revision 2, dated February 1978, and in all programs specified in ITS Section 5.5, respectively. The ITS Bases also contain descriptions of system operation. Controls specified in 10 CFR 50.59 apply to changes in procedures as described in the USAR and TRM. ITS 5.5.12 specifies controls for changing the Bases. Removing details of system operation is acceptable because the associated CTS requirements being retained without these details are adequate to ensure safe operation of the facility. In addition, retaining such details in TS is unnecessary to ensure proper control of changes. Therefore, it is acceptable to remove Type 2 details from the CTS and place them in licensee-controlled documents.

Type 3 - Removing Procedural Details for Meeting TS Requirements or Reporting Requirements

Details for performing TS SRs or for regulatory reporting are more appropriately specified in the plant procedures. Changes to procedural details include those associated with limits retained in the ITS. For example, ITS 5.4.1 requires that written procedures covering activities that include all programs specified in Specification 5.5 be established, implemented, and maintained. ITS 5.5.6, “Inservice Testing Program,” requires a program to provide controls for inservice testing (IST) of American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves. The program includes testing frequencies specified in the ASME Operation and Maintenance Standards and Codes, and applicable addenda.

Prescriptive procedural information in a TS requirement is unlikely to contain all procedural considerations necessary for the plant operators to comply with TS and all regulatory reporting requirements, and referral to plant procedures is therefore required in any event. Therefore, it is acceptable to remove Type 3 details from the CTS and place them in licensee-controlled documents.

Type 4 - Removal of LCO, SR, or other TS requirement to the TRM, USAR, Offsite Dose Calculation Manuel (ODCM), NFQAPD, Containment Leak rate Testing (CLRT) Program, IST Program, ISI Program, or Setpoint Control Program

Certain CTS administrative requirements are redundant with respect to current regulations and thus are relocated to the USAR or other appropriate licensee-controlled documents. The Final Policy Statement allows licensees to relocate to licensee-controlled documents, and CTS requirements that do not meet any of the criteria for mandatory inclusion in the TS.

Examples of the proposed changes include moving details out of the Current Technical Specifications (CTS) and into the Technical Specifications Bases, the Updated Safety Analysis Report (USAR), the Containment Leak Rate Testing (CLRT) Program, the Technical Requirements Manual (TRM), and other documents under regulatory control such as the ODCM, the Nuclear Facility Quality Assurance Program Description (NFQAPD), the Inservice Testing (IST) Program, the Inservice Inspection (ISI) Program, and the Setpoint Control Program. The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. Typically, the information moved is descriptive in nature and its removal conforms to NUREG-1431. Changes made in accordance with the provisions of licensee-controlled documents are subject to the specific requirements of those documents. For example, 10 CFR 50.54(a) governs changes to the NFQAPD, and ITS 5.5.12 governs changes to the ITS Bases. Therefore, it is acceptable to remove these details from CTS and place them in licensee-controlled documents.

To the extent that information has been relocated to licensee-controlled documents, such information is not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to public health and safety. Further, where such information is contained in LCOs and associated requirements in the TS, the NRC staff has concluded that they do not fall within any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii) and discussed in the Final Policy Statement (see Section 2.0 of this SE). Accordingly, existing detailed information, such as generally described above, may be removed from the CTS and not included in the ITS.

Table LA attached to this SE lists the less restrictive removal of detail changes being made in the Kewaunee ITS conversion. Table LA is organized in ITS order by each LA-type DOC and includes the following:

1. The ITS/CTS number, followed by the DOC number, (e.g. LA01);

2. The reference numbers of the associated CTS requirements;

3. A summary description of the relocated details and requirements;

4. The name of the licensee-controlled document to contain the relocated details and requirements (location);

5. The regulation (or ITS Specification) for controlling future changes to relocated requirements (change control process); and

6. A characterization of the type of change.

E. Relocated Specifications (R)

The Final Policy Statement (58 FR 39132 dated July 22, 1993) states that LCOs and associated requirements that do not satisfy or fall within any of the four specified criteria (now contained in 10 CFR 50.36(c)(2)(ii)) may be relocated from existing TS to appropriate licensee-controlled documents as noted in Section D above.

This section discusses the relocation of entire specifications from the CTS to licensee controlled documents. These specifications generally would include LCOs, Action Statements (i.e., Actions), and associated SRs. In its application and supplements, the licensee proposed relocating such specifications from the CTS to licensee-controlled documents. The NRC staff has reviewed the licensee's submittals and finds that relocation of these requirements is acceptable in that the LCOs and associated requirements were found not to fall within the scope of 10 CFR 50.36(c)(2)(ii) and changes to licensee controlled documents will be adequately controlled by 10 CFR 50.59, as applicable. These provisions will continue to be implemented by appropriate station procedures (i.e., operating procedures, maintenance procedures, surveillance and testing procedures, and work control procedures).

Table R attached to this SE lists the relocated changes requested as part of the Kewaunee ITS conversion and lists all specifications that are being relocated from the CTS to licensee controlled documents. Table R includes the following in columns:

1. References to the ITS/CTS section and DOC number;

2. References to the relocated CTS requirement;

3. Summary descriptions of the relocated CTS requirement;

4. Names of the document that will contain the relocated specifications (i.e., the new location); and

5. The method for controlling future changes to the relocated specifications (i.e., the regulatory change control process)

The specifications relocated from the CTS are not required to be in the TS because they do not fall within the criteria for mandatory inclusion in the TS as stated in 10 CFR 50.36(c)(2)(ii). These specifications are not needed to obviate the possibility that an abnormal situation or event will give rise to an immediate threat to the public health and safety. The NRC staff concludes that appropriate controls have been established for all of the current specifications and information being moved to the TRM. These relocations are the subject of a new license condition discussed in Section 7.0 of this SE. Until incorporated in licensee controlled documents, changes to these specifications and information will be controlled in accordance with the current applicable procedures and regulations.

F. Control of Specifications, Requirements, and Information Relocated from the CTS

In the ITS conversion, the licensee proposes to relocate specifications, requirements, and detailed information from the CTS to licensee-controlled documents. This is discussed in Sections 4.D and 4.E of this SE. The facility and procedures described in the USAR and TRM can be revised in accordance with the provisions of 10 CFR 50.59, to ensure that records are maintained and appropriate controls are established over those requirements removed from the CTS and future changes to the requirements. Other licensee-controlled documents contain provisions for making changes consistent with applicable regulatory requirements. The documentation of these changes will be maintained by the licensee in accordance with the record retention requirements specified in the Nuclear Facility Quality Assurance Program Description (NFQAPD) and such applicable regulations as 10 CFR 50.59.

The license condition for the relocation of requirements from the CTS, which is discussed in Section 7.0 of this SE, will address the implementation of the ITS conversion and the schedule for the relocation of the CTS requirements into licensee-controlled documents.

G. Evaluation of Other TS Changes (Beyond Scope Issue Reviews, Accident Dose Branch (AADB) Review, and Component Performance and Testing Branch (CPTB) Review), as Part of the Conversion to ITS

Before and during the review of the subject License Amendment Request, several Beyond Scope Items (BSIs) were identified. The term BSI refers to a review effort of sufficient magnitude or specialization to justify considering the use of a separate Task Assignment Control (TAC) number to monitor its labor effort. For example, TAC No. ME2419 captured the costs of the Diesel Generator Air Start Evaluation. In the case of the review of the setpoint methodology, a separate review directly supported this Safety Evaluation and its evaluation document could have been imbedded in it. However, specialized needs and use of specialized expertise justified the creation of TAC No. ME3460 to separately review aspects of the setpoint methodology. The end result was a distinct pair of Safety Evaluations that were attached to this Safety Evaluation (designated as Safety Evaluation Attachments) rather than imbedded in it.

All Safety Evaluation Attachments are described below and share in common the content of SE Sections 8.0 (STATE CONSULTATION), 9.0 (ENVIRONMENTAL CONSIDERATION), and 10.0 (CONCLUSION), which are omitted from the separate attachments. The Safety Evaluation Attachments immediately follow this SE.

None of these BSIs changed the scope of the original application. Most of these issues were able to be resolved as part of the normal review process. Any changes to the original submittal that were required to address these issues are reflected in the supplements to the LAR, in the issued TS resulting from the ITS Conversion, and in the Tables A, L, LA, M, and R provided with this SE.

BSIs resolved as part of and within this Safety Evaluation were:

1. ME2419 – Diesel Generator Air Start System
   The licensee requested deviations from the standard TS for the above system. During the course of discussions, the licensee chose to adopt the standard TS for the system and the BSI required no further action.

2. ME2421 – AFW Pump Undervoltage Protection
   In their submittal, the licensee requested to remove requirements related to these undervoltage relays. During the course of the staff's review, it was determined that the function performed by these relays is redundant to other plant instrumentation currently covered by technical specifications. Therefore, this change is acceptable.

3. ME3122 – Snubbers
   During the course of the review, a BSI was established (TAC ME3122) related to LCO 3.0.8 regarding snubbers not available to perform their associated support function(s). The licensee requested a deviation from the ISTS allowed time that a snubber may be unavailable prior to declaring affected supported system(s) LCO(s) not met. The request was to extend that time from 12 to 24 hours. Kewaunee's current licensing basis provides 72 hours in most cases due to plant specific equipment and configuration issues that require an extended time to replace failed snubbers. An analysis of the application of the new 24 hour period to all snubbers covered by LCO 3.0.8 showed that in nearly all cases this will result in a more conservative specification than the CTS, and is appropriate based on a plant specific analysis and plant specific configurations and conditions. This deviation may not be broadly applicable as precedent for future License Amendment Requests from other Licensees.

Three of the BSIs required separate Safety Evaluation Attachments (designated as Attachments 1 through 4 (Attachments 2 and 3 address one BSI), as explained below).

Safety Evaluation Attachment 1, performed under TAC No. ME2420, addresses the review of TS SR 3.7.5.1 related to Auxiliary Feed Pump operability.

Safety Evaluation Attachments 2 and 3 address a BSI that was established to support the review of setpoint methodology submitted by the licensee as part of their request to adopt TSTF-493 Option B. For the review of this BSI, Attachment 2 (performed under TAC No. ME2139) directly supports this Safety Evaluation, and Attachment 3 (performed under TAC No. ME3460) directly supports Attachment 2.

Safety Evaluation Attachment 4 (performed under TAC No. ME3544) addresses a BSI that was established to support the review by the Electrical Engineering Branch (EEEB) of the licensee's rewrite of TS 3.8 regarding DC Sources. The review of this BSI also addressed the effect on the Battery Monitoring and Maintenance Program described in TS Section 5.0.

Two Safety Evaluation Attachments were prepared as separate efforts by their respective reviewing NRC Technical Branches. They, when added to this Safety Evaluation, complete the

comprehensive review of the ITS Conversion. To simplify referral to this information, they are included as attachments to this Safety Evaluation.

Safety Evaluation Attachment 5 has been prepared by the Accident Dose Branch (AADB) to review the adoption of TS 3.4.16, RCS specific activity, and TSTF-490, Revision 0, regarding deletion of E-Bar (E) definition and revision to RCS specific activity.

Safety Evaluation Attachment 6 has been prepared by the Component Performance and Testing Branch (CPTB) to review TS changes with respect to inservice testing (IST) requirements

H. Implementation of TSTF-493 Rev 4 "Clarify Application of Setpoint Methodology for LSSS Functions"

Changes in accordance with several of the above categories are a result of the licensee's adoption of Technical Specification Task Force (TSTF) Traveler TSTF-493 "Clarify Application of Setpoint Methodology for LSSS Functions". The Notice of Availability (NOA) for this Traveler was published in the *Federal Register* 75 FR 26294 on May 11, 2010. This TSTF was nearing approval when this TS Conversion was submitted. As a result, the Licensee opted to serve as the pilot plant for TSTF-493. Because of that decision and the timing of the NOA, the original submittal for technical specifications affected by TSTF-493 was subject to several revisions submitted to the NRC as LAR supplements and responses to Requests for Additional Information during the review of the overall conversion. The final version of the Kewaunee ITS and the Discussions of Changes from CTS attached to this SE reflect the final, noticed version of TSTF-493, modified as required based on the evaluation of the submitted setpoint methodology.

5.0 DELETED LICENSE CONDITIONS

Two License Conditions are being removed as a part of the overall License Amendment to convert Kewaunee to ITS. The first is not specifically related to technical specifications. License Condition 2.C(8) expired with the completion of Kewaunee refueling outage R-29 in May 2008, and is no longer applicable.

The second, License Condition 2.C(9), does relate to technical specifications, and has also expired (with the completion of the Fall 2006 refueling outage). This condition allowed a temporary relaxation of Surveillance Test Intervals and referenced a table of TS Surveillance Requirements (Table 2.C.(9) on page 5a of the facility operating license). The table has also been removed. Since this condition and the associated table are no longer applicable, their removal is acceptable.

6.0 LICENSEE COMMITMENTS

In reviewing the proposed ITS conversion for Kewaunee, the NRC staff has relied upon the licensee's commitment to relocate certain requirements from the CTS to licensee-controlled documents as described in Table LA, "Removed Details" (Attachment 3 to this SE) and Table R, "Relocated Specifications" (Attachment 5 to this SE). These tables, and Sections 4.D and 4.E of this SE, reflect the relocations described in the licensee's submittals on the conversion. The

NRC staff requested and the licensee submitted a set of license conditions to make these commitments enforceable (see Section 7.0 of this SE). Such commitments from the licensee are important to the ITS conversion because the acceptability of removing certain requirements from the TS is based on those requirements being relocated to licensee-controlled documents where further changes to the requirements will be controlled by applicable regulations or other requirements (e.g., 10 CFR 50.59).

7.0 LICENSE CONDITIONS

In the series of supplements to the original submittal, the licensee agreed to license conditions which describe (1) the relocation of certain CTS requirements and license conditions, as applicable, to other license controlled documents prior to ITS implementation, and (2) a schedule to begin performing new and revised SRs after ITS implementation. The following license conditions are included in the Facility Operating License:

1. License Amendment No. 207 authorizes the relocation of certain Technical Specifications previously included in Appendix A to other licensee-controlled documents. At the time of implementation of this amendment, the subject requirements shall have been relocated to the specified documents, as described in Table LA, Removed Detail Changes, and Table R, Relocated Specifications, attached to the NRC staff's Safety Evaluation, which is enclosed in this amendment.

2. The schedule for performing SRs that are new or revised in Amendment No. 207 shall be as follows:

    a) For SRs that are new in this amendment, the first performance is due at the end of the first Surveillance interval, which begins on the date of implementation of this amendment.

    b) For SRs that existed prior to this amendment, whose intervals of performance are being reduced, the first reduced Surveillance interval begins upon completion of the first Surveillance performed after implementation of this amendment.

    c) For SRs that existed prior to this amendment, whose intervals of performance are being extended, the first extended Surveillance interval begins upon completion of the last Surveillance performed prior to implementation of this amendment.

    d) For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance subject to the modified acceptance criteria is due at the end of the first Surveillance interval that began on the date the Surveillance was last performed prior to the implementation of this amendment.

The NRC staff has reviewed the above schedule for the licensee to begin performing the new and revised SRs and concludes that it is acceptable. The licensee states that its

implementation date for the new ITS documents is to be concurrent with the date of this amendment. This implementation date is acceptable.

Because the commitments discussed in Section 6.0 of this SE are being relied upon for the amendment, a license condition is included in the amendment that will enforce the relocation of requirements from the CTS to licensee-controlled documents. The relocations are described in Table LA and Table R, which are Attachments 3 and 5 to this SE. The license condition states that implementation of this amendment shall include relocation of these requirements to the specified documents. The relocation of these requirements to the specified documents is to be concurrent with the date of this amendment. This implementation date is acceptable.

8.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified on September 23, 2010, of the proposed issuance of the amendment. The State official had no comments.

9.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and Finding of No Significant Impact was published in the *Federal Register* on December 23, 2010 (75 FR 80855), for the proposed conversion of the CTS to ITS for KPS. Accordingly, the Commission has determined that issuance of these amendments will not result in any significant environmental impacts other than those evaluated in the Generic Environmental Impact Statement for License Renewal of Nuclear Plants [NUREG-1437], Supplement 40, Regarding Kewaunee Power Station, Final Report, dated August, 2010 (ADAMS Accession No. ML102150106).

The Commission also issued a Notice of Consideration of Issuance of Amendment and Opportunity for a Hearing on December 15, 2009 (74 FR 66384). There have been no comments or requests for hearing.

10.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that: (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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

11.0 REFERENCES

1. NUREG-1431, Revision 3.0, “Standard Technical Specifications – Westinghouse Plants.” (ADAMS Accession No. ML062510017)

2. Letter from Leslie N. Hartz Dominion Energy Kewaunee (DEK) “License Amendment Request 249: Kewaunee Power Station Conversion to Improved Technical Specifications,” dated August 24, 2009. (ADAMS Accession No's ML092440398, ML092440416 through ML092440435, and ML092440441)

3. Letter from William R. Matthews (DEK), “License Amendment Request 249: Conversion to Improved Technical Specifications - Setpoint Methodology Supplement,” dated October 22, 2009. (ADAMS Accession No. ML093070096)

4. Letter from J. Alan Price (DEK), “License Amendment Request 249: Conversion to Improved Technical Specifications Response to Request for Additional Information RE: TSTF-490,” dated April 13, 2010. (ADAMS Accession No. ML101060517)

5. Letter from J. Alan Price (DEK), “License Amendment Request 249: Conversion to Improved Technical Specifications - Setpoint Methodology,” dated April 13, 2010. (ADAMS Accession No. ML101040090)

6. Letter from J. Alan Price (DEK), “License Amendment Request 249: Kewaunee Power Station Conversion to Improved Technical Specifications - Request to Change Proposed Service Water and Main Steam Isolation Valve Specifications,” dated May 12, 2010. (ADAMS Accession No. ML101380399)

7. Letter from J. Alan Price (DEK), “License Amendment Request 249: Kewaunee Power Station Conversion to Improved Technical Specifications - Supplement to Volumes 3, 4, 5, 6, 9, 11, and 15” dated July 1, 2010. (ADAMS Accession No. ML101890404)

8. Letter from J. Alan Price (DEK), “License Amendment Request 249: Kewaunee Power Station Conversion to Improved Technical Specifications - Supplement to Volumes 7, 10, 12, and 16, and Proposed License Conditions,” dated July 16, 2010. (ADAMS Accession No. ML102370369)

9. Letter from Leslie N. Hartz (DEK), “License Amendment Request 249: Kewaunee Power Station Conversion to Improved Technical Specifications - Supplement to Volumes 1, 2, 8, 13 and 14,” dated August 18, 2010. (ADAMS Accession No. ML102371023)

10. Letter from Leslie N. Hartz (DEK), “License Amendment Request 249: Kewaunee Power Station Conversion to Improved Technical Specifications – Submittal of Information Requested by NRC Staff,” dated September 7, 2010. (ADAMS Accession No. ML102730383)

11. Letter from Leslie N. Hartz (DEK), “License Amendment Request 249: Kewaunee Power Station Conversion to Improved Technical Specifications - Supplement to Volumes 8 and 16,” dated September 8, 2010. (ADAMS Accession No. ML102580700)

12. Letter from J. Alan Price (DEK), “License Amendment Request 249: Kewaunee Power Station Conversion to Improved Technical Specifications - Supplement to Volume 8,” dated October 15, 2010. (ADAMS Accession No. ML102920037)

Safety Evaluation Attachments:

1. Safety Evaluation Attachment 1 (from Balance of Plant Branch – SBPB)
2. Safety Evaluation Attachment 2 (from Technical Specifications Branch – ITSB)
3. Safety Evaluation Attachment 3 (from Instrumentation and Control Branch – EICB)
4. Safety Evaluation Attachment 4 (from Electrical Engineering Branch – EEEB)
5. Safety Evaluation Attachment 5 (from Accident Dose Branch – AADB)
6. Safety Evaluation Attachment 6 (from Component Performance and Testing Branch – CPTB)

List of Regulatory Commitments

Tables of Technical Specification Changes:

1. Table A - Administrative Changes
2. Table L - Less Restrictive Changes
3. Table LA - Removed Details
4. Table M - More Restrictive Changes
5. Table R - Relocated Specifications

List of Standard Acronyms and Abbreviations

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Date: February 2, 2011

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FROM THE BALANCE OF PLANT BRANCH

PERTAINING TO AUXILIARY FEEDWATER PUMP OPERABILITY

FOR KEWAUNEE POWER STATION IMPROVED TECHNICAL SPECIFICATION

CONVERSION LICENSE AMENDMENT REQUEST (TAC NO. ME2420)

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated August 24, 2009, Dominion Energy Kewaunee, Inc., the licensee, requested a revision to Operating license DPR-43 for the Kewaunee Power Station. The licensee proposed to revise their current Technical Specifications (TS) to become consistent with Improved Technical Specifications (ITS) as described in NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," Revision 3.0.

As part of the proposed amendment, the licensee provided a list of "Beyond Scope Changes." One of these changes proposes to add a special allowed outage time to ITS surveillance requirement (SR) 3.7.5.1, to allow 4 hours for the cross-tie valves on the steam-driven auxiliary feedwater (TDAFW) pump system not to be in their normal post accident position. The licensee requests this exception in order to test the motor-driven auxiliary feedwater (MDAFW) pumps while the plant is in mode requiring auxiliary feedwater (AFW) pumps to be operable.

This exception to the current ITS allows the AFW system to be temporarily out of its normal standby alignment, without declaring the TDAFW pump train inoperable. Specifically, during this alignment, if the turbine driven AFW pump receives a signal to autostart, then the pump will not automatically deliver flow simultaneously to both of the two steam generators (SGs) with one of the cross-tie valves closed.

2.0 REGULATORY EVALUATION

The applicable regulatory requirements are:

- Title 10 of the *Code of Federal Regulations*, Part 50, Appendix A, General Design Criterion (GDC) 34 specifies, in part, that the system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

- GDC 44 for cooling water specifies, in part, that a system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided.

A description of the licensee's proposed change follows. Currently, ITS for TS SR 3.7.5.1 reads as follows:

	Surveillance	Frequency
SR 3.7.5.1	----------NOTE---------- AFW train(s) may be considered operable during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ----------	
	Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

The licensee proposes to add an additional note to SR 3.7.5.1 to read as follows:

	Surveillance	Frequency
SR 3.7.5.1	----------NOTE---------- 1. AFW train(s) may be considered operable during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. 2. One AFW header cross-tie valve is allowed to be closed for up to 4 hours during testing of the motor driven AFW pump and the turbine driven AFW train may be considered operable, provided the cross-tie valve is capable of being remotely realigned. ----------	
	Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

The licensee is also proposing to amend the TS Bases Section SR 3.7.5.1, to add an explanation that the above provision is reasonable based upon the cross-tie valves retaining their capability to be manually remotely aligned to their functional position from the control room.

3.0 TECHNICAL EVALUATION

The AFW system provides a redundant, independent, and diverse means of supplying feedwater to the SGs for cooling the reactor coolant system (RCS) under emergency conditions. The AFW system consists of two MDAFW pumps and one TDAFW pump, each pump having a design capacity of 240 gallons per minute. The MDAFW pumps' discharges are interconnected by a cross-over pipe, which may be isolated by two normally-open motor valves (AFW cross-tie valves), which enable the supply of feedwater to one or both SGs. The TDAFW pump discharges into this cross-over pipe, enabling the TDAFW pump to supply either or both SGs depending on the position of the cross-tie valves. The AFW system provides a means of pumping sufficient feedwater to the SGs in order to remove sufficient RCS heat in order to prevent the RCS from going water solid, thus preventing overpressurization and subsequent opening of a pressurizer relief valve. The capacity of any two AFW pumps is sufficient to provide the required feedwater flow to mitigate the loss of main feedwater transient.

The licensee's safety analyses require the AFW cross-tie valves to be in different positions, depending on the type of accident. The licensee normally maintains the AFW cross-tie valves (AFW-10A and AFW-10B) open, allowing the TDAFW pump to flow to both SGs through cross-over piping into the MDAFW pumps' discharge piping. The licensee's proposed TS SR provision will allow operators to close one cross-tie valve for up to 4 hours for surveillance testing.

The cross-tie valves are DC powered, motor operated valves. They do not receive an automatic signal to open. The cross-tie valves can be remotely operated from the control room, remotely operated from the dedicated shutdown panel, or locally operated using manual handwheels. Operators are required to determine the correct position of the valves in order to provide their safety function depending upon the plant's equipment conditions during an accident. For main steam line break (MSLB) and steam generator tube rupture, the operator is required to close the cross-tie valve to the ruptured or faulted generator. For loss of normal feedwater (LONF), the desired position for the cross-tie valves is open, so that flow is delivered equally to the two SGs. Therefore, as long as the cross-tie valves maintain their ability to be remotely operated to the correct position, the valves can perform their design function.

The limiting design basis accident and transient for the AFW system is the LONF accident. The licensee's design basis for LONF assumes, "AFW flow is available 1 minute after reactor trip and is distributed equally to two steam generators." In the ITS submittal Attachment 1, Volume 12, the licensee states that during testing the TDAFW pump will only be able to deliver flow to one SG, because it is isolated from the SG to which the MDAFW pump is being tested. The staff noted that depending on which crosstie valve is closed during the MDAFW pump test, the TDAFW pump may only inject into the same SG fed by the non-test MDAFW pump, leaving one SG not being supplied by an operable AFW pump. The NRC staff asked the licensee to evaluate whether the design bases accidents assumptions can be met with this cross-tie valve closed.

The licensee responded to the staff's inquiry,

> In all configurations, two AFW pumps (TDAFW pump and the non-test MDAFW pump) are available to automatically deliver AFW flow to either one SG or two SGs, at any given time. Two AFW pumps delivering flow to one SG meets the LONF analysis requirements since the AFW flow from two pumps is significantly greater than the AFW flow assumed in the LONF analysis. The LONF analysis was performed assuming one AFW pump delivering flow to two SGs. The configuration of AFW delivery to two SGs is consistent with the normal AFW system configuration at power. Fluid System Analysis (Calculation C11783) Auxiliary Feedwater System supports two AFW pumps are capable of supplying enough flow to one SG to provide decay heat removal to meet the LONF accident analysis.

Therefore, based upon the licensee's analysis of AFW system flow to one or both SGs, the AFW system can meet the design requirements credited to mitigate a LONF accident.

AFW is also credited in the MSLB analysis. Maximum AFW flow from three AFW pumps is assumed for the first 10 minutes of the accident until operators can isolate the flow to the affected SG by tripping the AFW pumps or closing the isolation valves. Afterwards, flow from only one AFW train is required to meet the AFW requirement to remove sufficient decay heat. The MSLB event requires the operators to manually re-position the cross-tie valves based upon the event, and the assumptions allow sufficient time for the operators to perform this task. Therefore, without consideration of an additional single failure of the cross-tie valve, the licensee's proposed exception having the cross-tie valve closed would not preclude the AFW system from performing its intended function during a MSLB. The 4 hour allowed completion time is acceptable considering the very low probability of a MSLB accident coincident with failure of the cross-tie valve to re-open,

The staff finds the licensee's justification for the "Beyond Scope Change" for AFW SR 3.7.5.1 is reasonable based upon:

- the TDAFW pump train remains capable of performing its design safety function with operator actions, because the cross-tie valve maintains its ability to be manually remotely controlled from the control room,

- the licensee has calculations that show a LONF accident, the limiting design-basis accident for AFW, can be mitigated with the cross-tie valve closed.

Therefore, based on the information provided and considerations discussed above, the staff finds the licensee's proposal acceptable to add a provision in TS SR 3.7.5.1 for a 4 hour LCO for the TDAFW pump train cross-tie valve to be closed during testing of the MDAFW pumps.

4.0 CONCLUSION TO SAFETY EVALUATION

Based on a review of the information that was provided and as discussed in the Technical Evaluation Section, the staff has determined that the proposed "Beyond Scope Change" change to TS SR 3.7.5.1 is appropriate. The proposed change is consistent with NRC practices and

policies as generally reflected in the STS and as reflected by applicable precedents that have been approved. Therefore, the staff has determined that the proposed change to TS SR 3.7.5.1 should be approved.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FROM TECHNICAL SPECIFICATIONS BRANCH (ITSB)

PERTAINING TO THE SETPOINT CONTROL PROGRAM

FOR KEWAUNEE POWER STATION IMPROVED TECHNICAL SPECIFICATION

CONVERSION LICENSE AMENDMENT REQUEST (TAC NO. ME2139)

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated August 24, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092440398), Dominion Energy Kewaunee, Inc. (the licensee) proposed an amendment to the Technical Specifications (TSs) for Kewaunee Power Station. By letter dated October 22, 2009, (ADAMS Accession No. ML093070092) the licensee supplemented their application. One of the changes in this amendment request would revise Kewaunee's current TS to Improved TS consistent with the Improved Standard Technical Specifications described in NUREG-1431, Revision 3.0, "Standard Technical Specifications - Westinghouse Plants." In this amendment the licensee proposed revising the TSs by adding an Administrative Control TS 5.5.16, "Setpoint Control Program (SCP)" and relocating the reactor trip settings, engineered safety features initiation instrument setting limits, and instrument operation conditions for reactor trip setting limits from TS Section 2.3, "Limiting Safety System Settings – Protective Instrumentation" and TS Section 3.5, "Instrumentation System" to the Technical Requirements Manual. The licensee stated that the application is consistent with Option B of NRC-approved Revision 4 to TSTF-493. The availability of this TSs improvement was announced in the *Federal Register* on May 11, 2010 (75 FR 26294).

TSTF-493 Option B provides a way for licensees to be able to make any needed changes to a specific set of instrument channel Technical Specification values through the use of a 10 CFR 50.59 safety evaluation (SE) process, using a licensee-controlled setpoint control program that utilizes a NRC approved setpoint calculation methodology. The setpoint methodology should be able to accomplish the following types of activities:

- Evaluate and process changes to existing instrument channel Allowable Value, Limiting Trip Setpoint, Nominal Trip Setpoint, As-Found Tolerance, As-Left Tolerance, to accommodate operational or maintenance issues, assuming no changes in equipment are planned.

- Evaluate and process changes to these values to existing instrument channel Allowable Value, Limiting Trip Setpoint, Nominal Trip Setpoint, As-Found Tolerance, As-Left Tolerance to address any required instrument channel design change due to obsolescence or planned modifications to improve facility performance.

- Establish new instrument channel settings or revision to existing settings based on changes in any of the following circumstances: changes in analytical limits due to changes in accident or transient analysis modeling; receipt of new information regarding instrument channel error effects e.g., 10 CFR Part 21 reports of vendor product specification non-conformance notices, 10 CFR Part 21 reports regarding instrument performance issues, NRC generic communications or INPO notices regarding instrument operating experience reports, changes in calibration or functional test procedure methods or calibration test equipment, and other circumstances likely to occur during the course of the remaining licensed reactor operating lifespan.

In its letter of October 22, 2009, the licensee submitted Kewaunee's setpoint methodology for NRC review. The setpoint methodology for Kewaunee did not include sufficient information to demonstrate how they would accomplish the above activities; therefore the NRC staff requested additional information from the licensee to complete its review of the proposed application. In order to complete the application in a timely manner, to allow the licensee to implement the proposed amendment prior to their 2011 refueling outage, the licensee modified its license amendment request by removing any reference to the NRC-approved methodology in the setpoint control program. The effect of this change is that the new setpoint control program will deviate from the TSTF-493 setpoint control program by not permitting the licensee to make changes to setpoint values relocated to a licensee controlled document. Setpoint changes will still require a license amendment.

The SCP will ensure that instrumentation will function as required to initiate protective systems or actuate mitigating systems at values equal to or more conservative than the point assumed in applicable safety analyses. The SCP provides requirements to control instrumentation setpoints so that they will function as assumed in applicable safety analyses.

The proposed change will resolve operability determination issues associated with potentially non-conservative TS Allowable Values (AVs)[1] calculated using some methods in the industry standard ISA-S67.04-1994 Part 2, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." The concern is that when these values are used to assess instrument channel performance during testing, non-conservative decisions about the equipment operability may result. In addition, the proposed change will resolve operability determination issues related to relying on AVs associated with TSs Limiting Safety System

[1] The instrument setting "Allowable Value" is a limiting value of an instrument's as-found trip setting used during surveillances. The AV is more conservative than the Analytical Limit (AL) to account for applicable instrument measurement errors consistent with the plant-specific setpoint methodology. If during testing, the actual instrumentation setting is less conservative than the AV, the channel is declared inoperable and actions must be taken consistent with the TS requirements.

Settings (LSSSs)[2] to ensure that TSs requirements, not plant procedures, will be used for assessing instrument channel operability.

The regulatory basis for the proposed TSs changes is described in Section 2.0 of this SE. The technical evaluation is discussed in Section 3.0 of this SE.

2.0 REGULATORY EVALUATION

Plant protective systems are designed to initiate reactor trips (scrams) or other protective actions before selected unit parameters exceed ALs assumed in the safety analysis in order to prevent violation of the reactor core safety limits (SLs) and reactor coolant system (RCS) pressure SL from postulated anticipated operational occurrences (AOOs) and to assist the engineered safety features (ESF) systems in mitigating accidents. The reactor core SLs and RCS pressure SL ensure that the integrity of the reactor core and RCS is maintained.

Kewaunee Power Station was designed and constructed to meet the intent of the Atomic Energy Commission's (AEC) General Design Criteria (GDC), as originally proposed on July 11, 1967 (32 *Federal Register* 10213). Construction of the plant was about 50 percent complete and the Final Safety Analysis Report (FSAR, Amendment No. 7) had been filed with the AEC before publication of the GDC on February 20, 1971. As a result, the licensee was not required to reanalyze the plant design or resubmit the FSAR. However, AEC staff did evaluate the plant design against the GDC in effect in 1972 and found that the plant design generally conforms to the intent of those criteria.

The proposed GDC in 1967 for instrumentation were:

> The regulation at 10 CFR Part 50, Appendix A, GDC 12, Instrumentation and control systems, states:
>
> > Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.
>
> The regulation at 10 CFR Part 50, Appendix A, GDC 14, Core protection systems, states:
>
> > Core protections systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

The GDC in 1972 for instrumentation were:

> The regulation at 10 CFR Part 50, Appendix A, GDC 13, Instrumentation and control, states:

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[2] 10 CFR 50.36(c)(1(II)(a) states: "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions."

> Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

The regulation at 10 CFR Part 50, Appendix A, GDC 20, Protection system functions, states:

> The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

The Commission's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36. The regulation at 10 CFR 50.36 requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The regulation requires, in part, that the TSs include items in the following categories: (1) Safety limits, limiting safety systems settings, and limiting control settings; (2) Limiting conditions for operation; (3) Surveillance requirements; (4) Design features; and (5) Administrative controls. However, the regulation does not specify the particular requirements to be included in TSs.

Instrumentation required by the TSs are designed to assure that the applicable safety analysis limits will not be exceeded during accidents and AOOs. This is achieved by specifying the Limiting Trip Setpoints (LTSP), including testing requirements to assure the necessary quality of systems, in terms of parameters directly monitored by the applicable instrumentation systems for LSSSs, as well as specifying limiting conditions for operation (LCOs) on other plant parameters and equipment in accordance with 10 CFR 50.36(c)(2).

- Section 50.36(c)(1)(i)(A) states in part:

  > Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity.

- Section 50.36(c)(1)(ii)(A) states in part:

  > Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system

does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor.

- Section 50.36(c)(2) states in part:

  Limiting conditions for operation. Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

- Section 50.36(c)(3) states in part:

  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.

- Section 50.36(c)(5), states in part:

  Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner.

In addition to the regulatory requirements stated above, the NRC staff also considered the previously approved guidance in NUREG–1431, Revision 3, ‘‘Standard Technical Specifications, Westinghouse Plants,’’ dated September 2006, and Regulatory Guide (RG) 1.105, Revision 3, “Setpoints for Safety-Related Instrumentation,” for determining the acceptability of revising instrumentation TSs requirements. RG 1.105, Revision 3, describes a method acceptable to the NRC staff for complying with the NRC’s regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TSs limits. The RG endorses Part 1 of ISA-S67.04-1994, “Setpoints for Nuclear Safety-Related Instrumentation,” subject to NRC staff clarifications. The ISA standard provides a basis for establishing setpoints for nuclear instrumentation for safety systems and addresses known contributing errors in the channel. Part 1 establishes a framework for ensuring that setpoints for nuclear safety-related instrumentation are established and maintained within specified limits.

3.0 TECHNICAL EVALUATION

3.1 Background

The licensee added the term "Limiting Trip Setpoint" as generic terminology for the setpoint value. Kewaunee uses the following terms: Allowable Value (AV), As Found Tolerance (AFT), As Left Tolerance (ALT), Limiting Trip Setpoint (LTSP) and Nominal Trip Setpoint (NTSP). Kewaunee defines the AV, AFT, ALT, LTSP and NTSP as:

- Allowable Value - is the threshold value used to determine channel operability during the performance of channel functional tests and channel calibrations. The

AV is the limiting as found setting for the channel trip setpoint that accounts for all of the Non-Channel Operational Test error components from the Channel Statistical Allowance (CSA) Calculation in accordance with Methods 1 or 2 from ISA-RP67.04.02-2000 and ISA-RP67.04-Part II-1994.

- As Found Tolerance - The As Found Tolerance is equal to the statistical combination of the rack error components and rack drift.

- As Left Tolerance - The As Left Tolerance is equal to the statistical combination of the rack error components minus the rack drift.

- Limiting Trip Setpoint - The LTSP is the limiting setting for the channel trip setpoint considering all credible instrument errors associated with the instrument channel.

- Nominal Trip Setpoint - The desired setpoint for the variable. Initial calibration and subsequent re-calibrations should be made at the Nominal Trip Setpoint value specified in approved plant documentation. The NTSP is the Limiting Trip Setpoint with margin added. The NTSP is always equal to or more conservative than the LTSP.

3.2 TS 5.5.16, Setpoint Control Program

3.2.1 Setpoint Control Program Scope and Requirements for Processing Setpoint Changes

The licensee proposed adding a SCP to the Administrative Controls section of the TSs which deviates from TSTF-493 Option B SCP requirements. The modified Option B SCP establishes the requirements necessary for ensuring that setpoints for automatic protective devices are initially within and remain within the TSs requirements through the addition of SCP, but changes to NTSP and AV instrument values will require prior review and approval by the NRC for the reasons described below.

In proposing new program requirements, the licensee is relocating the reactor trip settings, engineered safety features initiation instrument setting limits, and instrument operation conditions for reactor trip setting limits from TS Section 2.3, “Limiting Safety System Settings – Protective Instrumentation” and TS Section 3.5, “Instrumentation System” to a licensee controlled SCP. Under 10 CFR 50.36 the instrument LCOs are still retained in TS by requiring the adherence to values referenced in the SCP, and by requiring surveillance requirements (SRs) to perform testing to verify the operability of the instruments in accordance with the SCP. The program is documented in the Technical Requirements Manual. The licensee identified the following TSs as the specifications to which the SCP applies:

- TS 3.3.1, “Reactor Protection System (RPS) Instrumentation

- TS 3.3.2, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions”

- TS 3.3.5, “Loss of Offsite Power (LOOP) Diesel Generator (DG) Start Instrumentation”
- TS 3.3.6, “Containment Purge and Vent Isolation Instrumentation”
- TS 3.3.7, “Control Room Post Accident Recirculation (CRPAR) Actuation Instrumentation”

TSs SRs which verify AVs or LTSPs are revised to state that the SRs must be performed in accordance with the SCP.

The current method of controlling instrument setpoints to assure conformance to 10 CFR 50.36 is to specify the value in the TSs. Relocating the TSs values to licensee controlled documents and requiring the values to be approved by the NRC assures conformance to 10 CFR 50.36. The NRC safety evaluation that approved Kewaunee’s values can be found in this Safety Evaluation as Safety Evaluation Attachment 3. The controls on the relocated setpoints continue to ensure that the lowest functional capability or performance levels of instrumentation required for safe operation is met. This permits operation at any specific value determined by the licensee, once approved by the NRC, to be within the acceptance criteria.

It is essential to plant safety that a plant is operated within the bounds of the parameter limits and that a requirement to maintain the plant within the appropriate bounds must be retained in the TSs. However, the specific values of these limits may be modified by the licensee, without affecting nuclear safety, provided that these changes are approved by the NRC and are within all applicable limits of the plant safety analysis that are addressed in the FSAR.

The NRC staff has not approved processing changes to Kewaunee Power Station instrumentation setpoints under 10 CFR 50.59 using an approved setpoint methodology as described in Option B of TSTF-493. NRC approval using 10 CFR 50.90 is required to change the listed value of the NTSP, AV, AFT, and ALT (as applicable) for each function described in the SCP paragraph a.

The NRC staff therefore finds that the scope of the SCP described in TS 5.5.16 is sufficient under 10 CFR 50.36(c)(1)(ii)(A) to ensure instrument Functions necessary to assure safety functions will actuate at the point assumed in the applicable safety analysis and will be periodically assessed. The NRC staff also finds that the process on which to base future changes to AVs and LTSP limits required by TSs will ensure changes are made in accordance with 10 CFR 50.90.

3.2.2 Content of the SCP

TS 5.5.16 Paragraph a states:

> The program shall list the Functions in the following specifications to which it applies:
>
> 1. LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation";

2. LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions";

3. LCO 3.3.5, "Loss of Offsite Power (LOOP) Diesel Generator (DG) Start Instrumentation";

4. LCO 3.3.6, "Containment Purge and Vent Isolation Instrumentation"; and

5. LCO 3.3.7, "Control Room Post Accident Recirculation (CRPAR) Actuation Instrumentation."

The licensee described the plant-specific evaluation for the list of instrument Functions that are described in TS 5.5.16 Paragraph a. The licensee's proposed SCP Paragraph lists the TSs for Functions with setpoints controlled by the program. The NRC staff reviewed the licensee's list, and finds that the TSs listed in SCP Paragraph are consistent with the Functions that are required to be controlled by the SCP as identified in the approved TSTF-493, Revision 4.

TS 5.5.16 Paragraph b states:

> The program shall list the value of the Nominal Trip Setpoint (NTSP), Allowable Value (AV), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) (as applicable) for each Function described in Paragraph a. The NRC staff has not approved processing changes to Kewaunee Power Station instrumentation setpoints under 10 CFR 50.59 using an approved setpoint methodology as described in Option B of TSTF-493. NRC approval using 10 CFR 50.90 is required to change the value of the NTSP, AV, AFT, and ALT (as applicable) for each Function described in Paragraph a.

The licensee SCP Paragraph b establishes program requirements to ensure LTSP, NTSP, AV, AFT, and ALT (as applicable) of the Functions described in SCP Paragraph a. In addition, Paragraph b of the program contains a list of LTSP, NTSP, AV, AFT, and ALT (as applicable) values for the Functions of the TSs described in Paragraph a.

TS 5.5.16 Paragraph c states:

> The program shall establish methods to ensure that Functions described in Paragraph a. will function as required by verifying the as-left and as-found settings are consistent with the list of values established by Paragraph b. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.

SCP Paragraph c establishes program methods to ensure the instrument Functions with relocated setpoints will function as required by verifying the AFT and ALT. Evaluation of channel performance is described and will verify that the channel will continue to perform in accordance with safety analysis assumptions. The assessment will establish an acceptable level of confidence in the channel performance prior to returning the channel to service. For

channels determined to be operable but degraded, the licensee stated that after returning the channel to service channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition to establish a reasonable expectation for continued operability.

TS 5.5.16 Paragraph d states:

> The program shall identify the Functions described in Paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). The NTSP of these Functions are Limiting Safety System Settings. These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS, CHANNEL OPEATIONAL TESTS, and TRIP ACTUATING DEVICE OPERATIONAL TESTS that verify the NTSP.
>
> 1. The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified NTSP.
>
> 2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified NTSP by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.
>
> 3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.
>
> 4. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the NTSP at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the NTSP provided that the as-found and as-left tolerances apply to the actual setpoint used to confirm channel performance).

The licensee has demonstrated the calculations for the LTSP, NTSP, AV, AFT and ALT are consistent with the setting limits for the instrument Functions identified in SCP Paragraph d. The licensee has revised the affected TSs Surveillances and revised TSs Tables (see Table 1

below) where these Functions are listed. Since the settings of these Functions are calculated and submitted to the NRC for approval, they are acceptable to the NRC staff.

Table 1

Surveillance Requirement	Table
3.3.1.7 Perform Channel Operational Test	Table 3.3.1-1 Reactor Protection System Instrumentation
3.3.1.8 Perform Channel Operational Test	Table 3.3.2-1 Engineered Safety Feature Actuation System Instrumentation
3.3.1.10 Perform Channel Calibration	Table 3.3.6-1 Containment Purge and Vent Isolation Instrumentation
3.3.1.11 Perform Channel Calibration	Table 3.3.7-1 Control Room Post Accident Recirculation Actuation Instrumentation
3.3.1.12 Perform Channel Calibration	
3.3.2.4 Perform Channel Operational Test	
3.3.2.6 Perform Channel Calibration	
3.3.5.1 Perform Trip Actuating Device Operational Test	
3.3.5.2 Perform Channel Calibration	
3.3.6.3 Perform Channel Operational Test	
3.3.6.4 Perform Channel Calibration	
3.3.7.2 Perform Channel Operational Test	
3.3.7.4 Perform Channel Calibration	

In accordance with the SCP, the general requirement in Paragraph c for all the affected Functions is augmented with additional requirements in Paragraph d. In accordance with SCP Paragraph d the licensee identified the Functions of Specifications described in SCP Paragraph a that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A).

The surveillance test requirements for Paragraph d includes the exclusion criteria that are used to determine which Functions must receive the additional requirements in Paragraph d. The licensee identified instruments that would be excluded (i.e., meets TSTF-493, Revision 4 Attachment A Exclusion Criteria) because their functional purpose can be described as (1) a manual actuation circuit, (2) an automatic actuation logic circuit, or (3) an instrument function that derives input from contacts which have no associated sensor or adjustable device (i.e., limit switches, breaker position switches, etc.). Instrument Functions not meeting one or more of the exclusion criteria are identified in the SCP.

Since automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) are consistent with the plant-specific license basis and since performance based testing is applied to all current TSs Surveillances that evaluate such settings, including the AFT and ALT (as applicable) the requirement of SCP, Paragraph d are met.

TS 5.5.16 Paragraph e states, “The program shall be specified in the Technical Requirements Manual.”

SCP Paragraph e specifies that the program requirements of TS 5.5.16 are implemented by the SCP which is incorporated in the Technical Requirements Manual. Changes to the values listed in the SCP in the Technical Requirements Manual are required to have NRC approval using 10 CFR 50.90.

3.2.3 Evaluation

Based on the review of the licensee’s application, the NRC staff concludes that the licensee setpoints are consistent with Regulatory Guide (RG) 1.105, Revision 3, “Setpoints for Safety-Related Instrumentation” and are therefore acceptable. Additionally, the NRC staff concludes that by implementing plant procedures to meet the SCP program requirements, the determination of instrument function operability will be controlled by the requirements of TS 5.5.16 during SR testing specified in the TSs. By meeting the requirements of Paragraph c, the licensee has also demonstrated that these instruments will perform their safety function. The NRC staff further concludes that by meeting the requirements of Paragraph d the proposed TSs changes meet the requirements of 10 CFR 50.36(c)(1)(ii)(A) and therefore, are acceptable.

Therefore, Paragraphs a, b, c, and d meet the requirements of 10 CFR 50.36 and consequently are acceptable.

4.0 CONCLUSIONS TO SAFETY EVALUATION

The Commission has concluded, based on the considerations discussed above, that: (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.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FROM THE INSTRUMENT AND CONTROLS BRANCH (EICB)

PERTAINING TO THE REVISED SETTING LIMITS AND OPERABILITY DETERMINATION

TOLERANCES

FOR KEWAUNEE POWER STATION IMPROVED TECHNICAL SPECIFICATION

CONVERSION LICENSE AMENDMENT REQUEST (TAC NO. ME3460)

DOCKET NO. 50-305

1.0 INTRODUCTION

1.1 Background

By letter dated August 24, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092440398), and supplemented by letters dated October 22, 2009 (ADAMS Accession No. ML093070092), April 13, 2010 (ADAMS Accession No. ML101040090), September 7, 2010 (ADAMS Accession No. ML102730381), and November 10, 2010 (ADAMS Accession No. ML103210492), Dominion Energy Kewaunee, Inc. (Dominion, the licensee) proposed a license amendment to revise the Kewaunee Power Station (Kewaunee) Unit 1 current Technical Specifications (TSs) to Improved Technical Specifications (ITS) consistent with the Improved Standard Technical Specifications (ISTS) described in NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," Revision 3. Concurrent with this revision, Dominion proposed adoption of technical specification task force (TSTF) traveler TSTF-493 (Revision 4), Option B. Provided certain conditions are met (as elaborated below), Option B of TSTF-493 would allow the relocation of the settings and Allowable Values for certain safety related instrument channel settings from the plant Technical Specifications to a controlled document referenced within the plant UFSAR that is under licensee control. The set of instrument channel Allowable Values applicable to TSTF-493 are those safety related functions delineated within Attachment 2 of the TSTF-493 Revision 4 Model Application included within the U.S. Nuclear Regulatory Commission (NRC) *Federal Register* Notice of Availability (75 FR 26294), "Clarify Application of Setpoint Methodology for LSSS Functions." When adopting Option B, the licensee must:

- Provide a plant-specific evaluation that includes the calculation basis for the Limiting Trip Setpoint (LTSP), Nominal Trip Setpoint (NTSP), Allowable Value

(AV), As-Found Tolerance band (AFT), and As-Left Tolerance band (ALT) for the list of instrument functions that are described in Attachment 2 of the Model Application.

- Provide a description of the program methods for ensuring the set of appropriate safety related instrument channels will function as required by verifying that the limits of the As-Found and As-Left tolerances are consistent with the established setpoint methodology. The description must show how the proposed plant licensing basis meets the guidance provided in Regulatory Information Summary (RIS) 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels;" and Regulatory Guide 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation." The description must also identify the measures to be taken to ensure that the associated instrument channel is capable of performing its safety functions in accordance with applicable design requirements and associated analyses, including information regarding the controls employed to ensure that the As-Left trip setting after completion of the required periodic surveillance is consistent with the setpoints determined using the proposed setpoint methodology, and the plant corrective action process for restoring channels to operable status.

- Provide documentation, including summary calculations, of the methodology used for establishing the LTSP, NTSP, AV, AFT, and ALT, indicating the related Analytical Limits (ALs) and the source of these ALs.

The Licensee's proposed revised Technical Specifications are summarized below.

1.2 <u>Instrument Channel Settings to be included in the ITS:</u>

The licensee determined that for the Kewaunee Units 1 & 2 Improved Technical Specifications to meet the requirements of TSTF-493, Revision 4 and address the ISTS for Westinghouse plants, it is necessary to recognize that some safety-related instrument channels are considered "Primary" Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) trips and permissives credited in the plant safety analyses, and some are considered "Back-up" RTS and ESFAS trips and permissives that are not credited in the plant safety analyses. The Primary trips have an associated Analytical Limit, while Back-up trip and permissive functions do not. Since the original Kewaunee custom Technical Specifications did not contain some of the instrument functions considered as "Back-up" functions, to ensure that the Kewaunee Improved Technical Specifications address all of the ISTS requirements for Westinghouse plants and the requirements for the set of safety-related reactor trip and ESFAS functions as outlined in TSTF-493 Revision 4, the licensee has ensured that the following safety-related Primary <u>and</u> Back-up RTS and ESFAS trip and permissive functions and their nominal and limiting settings, allowable values, and as-found and as-left tolerances are included in its submittal:

1.2.1 Primary and Back-up Reactor Trips

Power Range Neutron Flux High Setpoint Reactor Trip:
AL: 118.00% RTP
AV: 111.185% RTP
LTSP: 110.96% RTP
AFT Limit: 106.50% RTP
NTSP: 105.00% RTP
AFT: ±1.50% RTP
ALT: ±1.00%RTP

Power Range Neutron Flux Low Setpoint Reactor Trip:
AL: 35.00%RTP
AV: 28.19% RTP
LTSP: 27.96% RTP
AFT Limit: 26.00%RTP
NTSP: 24.50% RTP
AFT: ±1.5% RTP
ALT: ±1.00% RTP

Power Range Neutron Flux High Positive Rate Reactor Trip: (Not credited in Safety Analyses)
NTSP: 5.00% RTP
Static As-Found Tolerance (AFT): 5.00% RTP ± 1.30% RTP
Static As-Left Tolerance (ALT): 5.0% RTP ± 0.50% RTP
Dynamic As-Found Tolerance: Time constant of 2.3 seconds ± 0.20 seconds
Dynamic As-Left Tolerance: Time constant of 2.3 seconds ± 0.20 seconds

Power Range Neutron Flux High Negative Rate Reactor Trip: (Not credited in Safety Analyses)
NTSP: 5.00% RTP
Static As-Found Tolerance (AFT): 5.00% RTP ± 1.30% RTP
Static As-Left Tolerance (ALT): 5.0% RTP ± 0.50% RTP
Dynamic As-Found Tolerance: Time constant of 2.3 seconds ± 0.20 seconds
Dynamic As-Left Tolerance: Time constant of 2.3 seconds ± 0.20 seconds

Intermediate Range Neutron Flux High Reactor Trip: (Not credited in Safety Analyses)
NTSP: 20.00% RTP
As-Found Tolerance (AFT) band: 20.00% RTP ± 5.00% RTP
As-Left Tolerance (ALT) band: 20.00% RTP ± 4.9% RTP

Source Range Neutron Flux High Reactor Trip: (Not credited in Safety Analyses)
NTSP: 1.0 E+5 Counts/second
AFT: +0.466 E+5 Counts/second, -0.318 E+5 Counts/second (logarithmic scale)
ALT: +0.358 E+5 Counts/second, -0.264 E+5 Counts/second (logarithmic scale)

Overtemperature ΔT Reactor Trip:
AL: 130.0% Delta T Power
AV: 124.117% Delta T Power
LTSP: 121.597% Delta T Power
AFT Limit: 121.25% Delta T Power
NTSP: 118.25% Delta T Power
AFT: ± 3.0% Delta T Power
ALT: ± 2.4% Delta T Power

Overpower ΔT Reactor Trip: (Not credited in Safety Analyses)
NTSP: (Continuously computed value)
Static AFT: Computed nominal setpoint ± 2.288% ΔT Power
Static ALT: Computed nominal setpoint ±1.724% ΔT Power

Pressurizer Low Pressure Reactor Trip:
AL: 1835.30 PSIG
AV: 1855.94 PSIG
LTSP: 1858.82 PSIG
AFT Limit: 1894.00 PSIG
NTSP: 1904 PSIG
AFT: ± 10.00 PSIG
ALT: ± 5.7 PSIG

Pressurizer High Pressure Reactor Trip:
AL: 2410.30 PSIG
AV: 2389.78 PSIG
LTSP: 2387.64 PSIG
AFT Limit: 2386.00 PSIG
NTSP: 2377.00 PSIG
AFT: ± 9.0 PSIG
ALT: ± 4.0 PSIG

Reactor Coolant Flow Low Reactor Trip (Normalized):
AL: 87.0% Flow
AV: 90.271% Flow
LTSP: 90.515% Flow
AFT Limit: 91.9% Flow
NTSP: 93.0% Flow
AFT: ± 1.1% Flow
ALT: ± 0.55% Flow

Reactor Coolant Pump Undervoltage Trip: (Not credited in Safety Analyses)
NTSP: 92.00 VAC (76.667% of normal voltage)
AFT: ± 1.06 VAC (0.885% of normal voltage)
ALT: ± 1.00 VAC (0.833% of normal voltage)

Reactor Coolant Pump Underfrequency Trip: (Not credited in Safety Analyses)
NTSP: 57 Hz
AFT: ± 0.3 Hz
ALT: ± 0.1 Hz

Pressurizer High Level Reactor Trip: (Not credited in Safety Analyses)
NTSP: 85% Level
AFT: ± 1.12% Level
ALT: ± 0.5% Level

Steam Generator Water Level Low Low Reactor Trip/Auxiliary Feedwater Initiation:
AL: 0.0% Narrow Range (NR) Level
AV: 4.087% NR Level
LTSP: 4.496% NR Level
AFT Limit: 15.88% NR Level
NTSP: 17.0% NR Level
AFT: ± 1.12% Level
ALT: ± 0.5% Level

Steam Generator Water Level Low Coincident Reactor Trip: (Not credited in Safety Analyses)
NTSP: 25.5% NR Level
AFT: ± 1.12% Level
ALT: ± 0.5% Level

Steam Flow Feed Flow Mismatch Coincident Reactor Trip: (Not credited in Safety Analyses)
NTSP: 0.87 E+6 Pounds mass per hour (PPH)
AFT: ± 0.063 E+6 PPH
ALT: ± 0.045 E+6 PPH

Turbine Trip Low Fluid Oil Pressure: (Not credited in Safety Analyses)
NTSP: 45.0 PSIG
AFT: ± 0.5 PSIG
ALT: ± 0.5 PSIG

1.2.2 Reactor Trip Permissives

Permissive P-6, Intermediate Range Neutron Flux (Unblock): (Not credited in Safety Analyses)
NTSP: 1.0 E-5% Rated Power
AFT: Between 1.0 E-5% Rated Power and 1.27% Rated Power (no rack drift error component)
ALT: Between 1.0 E-5% Rated Power and 1.27% Rated Power (no rack drift error component)

Permissive P-7, Block Low Power Reactor Trips and Enable High Power Reactor Trips: (Not credited in Safety Analyses)
P-10 NTSP: 11.0% RTP
P-10 AFT: ± 1.2% RTP
P-10 ALT: ± 0.5% RTP
P-13 NTSP: 8.8% Turbine Load
P-13 AFT: ± 1.25% Turbine Load
P-13 ALT: ± 0.56% Turbine Load

Permissive P-8, (Unblock) Power Range Neutron Flux: (Not credited in Safety Analyses)
NTSP: 9.5% RTP
AFT: ± 1.3% RTP
ALT: ± 0.5% RTP

Permissive P-10, Power Range Neutron Flux Unblock Low Power Reactor Trips and Block High Power Reactor Trips: (Not credited in Safety Analyses)
NTSP: 9.0% RTP
AFT: ± 1.3% RTP
ALT: ± 0.5% RTP

1.2.3 Primary and Back-up Engineered Safety Features Actuation System Initiations

High Containment Pressure—Safety Injection:
AL: 5.0 PSIG
AV: 4.328 PSIG
LTSP: 4.237 PSIG
AFT Limit: 3.935 PSIG
NTSP: 3.6 PSIG
AFT: ± 0.335 PSIG
ALT: ± 0.15 PSIG

High-High Containment Pressure (Containment Spray):
AL: 23.0 PSIG
AV: 21.827 PSIG
AFT Limit: 21.671 PSIG
LTSP: 21.622 PSIG
NTSP: 21.00 PSIG
AFT: ± 0.671 PSIG
ALT: ± 0.300 PSIG

High-High Containment Pressure (Steam Line Isolation):
AL: 17.0 PSIG
AV: 15.827 PSIG
AFT Limit: 15.671 PSIG
LTSP: 15.622 PSIG
NTSP: 15.00PSIG
AFT: ± 0.671 PSIG
ALT: ± 0.300 PSIG

Pressurizer Low Pressure (Safety Injection):
AL: 1685 PSIG
AV: 1754.94 PSIG
LTSP: 1755.62 PSIG
AFT Limit: 1820 PSIG
NTSP: 1830 PSIG
AFT: ± 10.0 PSIG
ALT: ± 4.0 PSIG

High Steam Flow Coincident with Safety Injection and Coincident with T_{avg}—Low Low:
AL: 1.75 E+6 lbs/hr
AV: 0.981 E+6 lbs/hr
LTSP: 0.944 E+6 lbs/hr
AFT Limit: 0.899 E+6 lbs/hr
NTSP: 0.75 E+6 lbs/hr
AFT: ± 0.149 E+6 lbs/hr
ALT: ± 0.067 E+6 lbs/hr

High High Steam Flow Coincident with Safety Injection:
AL: 7.76 E+6 lbs/hr
AV: 7.673 E+6 lbs/hr
LTSP: 7.668 E+6 lbs/hr
AFT Limit: 4.3699 E+6 lbs/hr
NTSP: 4.3439 E+6 lbs/hr
AFT: ± 0.026 E+6 lbs/hr
ALT: ± 0.011 E+6 lbs/hr

Low Low T_{avg} Coincidence Input into Steam Line Isolation: (Not credited in Safety Analyses)
NTSP: 541.0 Deg. F
AFT: ± 1.38 Deg. F
ALT: ± 0.95 Deg. F

Steam Line Pressure Low:
AL: 465.3 PSIG
AV: 489.31 PSIG
LTSP: 496.366 PSIG
AFT Limit: 496.85 PSIG
NTSP: 514.0 PSIG
AFT: ± 17.15 PSIG
ALT: ± 10.0 PSIG

Steam Generator Water Level Low Low Trip/Auxiliary Feedwater Initiation:
AL: 0.0% Narrow Range (NR) Level
AV: 4.087% NR Level
LTSP: 4.496% NR Level
AFT Limit: 15.88% NR Level
NTSP: 17.0% NR Level
AFT: ± 1.12% Level
ALT: ± 0.5% Level

Steam Generator Water Level High High:
AL: 100.00% NR Level
AV: 92.486% NR Level
LTSP: 92.077% NR Level
AFT Limit: 67.62% NR Level
NTSP: 66.50% NR Level
AFT: ± 1.12% NR Level
ALT: ± 0.50% NR Level

Safeguards Bus Undervoltage (Loss of Voltage): (Not credited in Safety Analysis)
NTSP: 101.31 VAC (84.15% of bus voltage)
AFT: ± 0.241 VAC (± 0.200% of bus voltage)
ALT: ± 0.200 VAC (± 0.166% of bus voltage)
Time Delay NTSP: 1.75 seconds
Time Delay AFT: ± 0.25 seconds
Time Delay ALT: ± 0.10 seconds

Safeguards Bus Second Level Undervoltage (Degraded Voltage): (Not credited in Safety Analysis)
NTSP: 112.57 VAC (93.50% of bus voltage)
AFT: ± 0.215 VAC (± 0.215% of bus voltage)
ALT: ± 0.200 VAC (± 0.200% of bus voltage)
Time Delay NTSP: 6.72 seconds
Time Delay AFT: ± 0.68 seconds
Time Delay ALT: ± 0.10 seconds

Forebay Level: (Not credited in Safety Analysis)
NTSP: 162” Water Level (equivalent to 566 feet 0” above sea level)
AFT: ± 9” Water Level
ALT: ±4.5” Water Level

Containment Purge and Vent System Radiation Particulate Detector and Radioactive Gas Detector Containment Ventilation Isolation: (Not credited in Safety Analyses) (Not within scope of Setpoint Control Program)
NTSP: High Alarm setpoint is set conservative with respect to 2.2 E+5 Counts/Min + Background, based on limits established in the Offsite Dose Calculation Manual
AFT Limit: (2.2 E+5 Counts per minute + Background)
ALT Limit: 8.00E+4 Counts per minute

Containment Particulate Radiation Monitor (R11): (Not credited in Safety Analyses)
AFT Limit = ALT Limit = 8.00 E+4 Counts per Minute

Control Room Ventilation Radiation Monitor (R23): (Not credited in Safety Analyses)
AFT Limit = ALT Limit = 1.00 E+4 Counts/minute

Turbine Building Service Water Header Isolation: (Not credited in Safety Analyses) (Added per ISTS for Westinghouse plants)
NTSP: 82.5 PSIG
AFT = ALT = ± 1.0 PSIG

2.0 REGULATORY EVALUATION

The following regulatory bases and guidance documents pertain to the proposed TS change:

- Title 10, Section 50.36(c)(1)(ii)(A), of the *Code of Federal Regulations* states, "Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety function does not function as required, the licensee shall take corrective action, which may include shutting down the reactor."

- Regulatory Guide 1.105, Revision 3, *"Setpoints for Safety-Related Instrumentation,"* issued December 1999, describes a method acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

- Regulatory Issue Summary 2006-17, *"NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels,"* dated August 24, 2006.

In addition, the technical requirements from the following Technical Specification Task Force Traveler also pertain to this proposed TS change:

- Technical Specification Task Force (TSTF) Traveler TSTF-493 *"Clarify Application of Setpoint Methodology for LSSS Functions".* The Notice of Availability (NOA) for this Traveler was published in the *Federal Register* 75 FR 26294 on May 11, 2010. This TSTF was nearing approval when this TS Conversion was submitted. As a result, the Licensee opted to serve as the pilot plant for TSTF-493. Because of that decision and the timing of the NOA, the original submittal for technical specifications affected by TSTF-493 was subject to several revisions submitted to the NRC as LAR supplements and responses to Requests for Additional Information during the review of the overall conversion.

The final version of the Kewaunee ITS and the Discussions of Changes from CTS attached to this SE reflect the final, May 11, 2010 noticed version of TSTF-493.

3.0 TECHNICAL EVALUATION

3.1 Background

There are two key regulatory concerns pertinent to the establishment and maintenance of technical specification values for the Reactor Trip and Engineered Safety Features Actuation System (ESFAS) Initiations, which need to be evaluated as the licensee adopts the Improved Technical Specifications (ITS) and adopts TSTF-493, Revision 4. These are as follows:

1. The primary reactor trip and ESFAS initiation settings are considered Limiting Safety System Settings in the context of 10 CFR 50.36(c)(1)(ii)(A), such that the settings must be so chosen that the automatic protective action will correct the abnormal situation before a safety level is exceeded. Such an occurrence could challenge the integrity of the reactor core, reactor coolant pressure boundary, containment, and associated systems. The NRC staff has issued Regulatory Guide 1.105, Revision 3, to provide guidance for ensuring that setpoints for safety related instrumentation are initially within and remain within established technical specification limits. Limiting Safety System Settings are used to protect the plant safety limits, and must be established in a manner that provides a high confidence level (the NRC staff typically accepts a 95/95 tolerance interval) that the instrument channel will function accordingly so that safety limits are not exceeded.

2. In the context of 10 CFR 50.36, Limiting Safety System Settings form a part of automatic protection systems that must function properly when the reactor is in operation for the reactor system to be considered safe to operate. If, during operation, it is determined that the automatic safety system does not function as required, licensees must take appropriate action, which may include shutting down the reactor. Surveillance requirements relating to test, calibration, or inspection to ensure the necessary quality of systems and components is maintained are performed to ensure that the facility is operating within required limits, and therefore, Limiting Safety System Settings are specified as technical specification-defined limits, and must be maintained within technical specifications rather than within plant procedures. To do so, technical specification surveillance requirements must be developed to clearly spell out the required actions that must be taken if it is determined that a Limiting Safety System Setting is not being met, and the required actions that must be taken to restore a safety related instrument channel that accomplishes a Limiting Safety System Setting to operable status if it is found to be inoperable.

The NRC staff has performed a technical evaluation of the licensee's submittals associated with this license amendment request. This evaluation accomplishes the following objectives:

- Verify that the licensee's setpoint calculation methods are adequate to assure that control and monitoring setpoints are established and maintained in a manner consistent with plant safety function requirements.

- Verify that the licensee's setpoint calculation methods are adequate to assure with a high confidence level that required protective actions are initiated before the associated plant process parameters exceed their analytical limits.

- Confirm that the proposed actions to be taken at established calibration intervals and operability determination methods are consistent with safety analysis assumptions and NRC guidance.

Below is a summary of the findings of this evaluation.

3.2 Changes to the Setting Limits, and Development/Documentation of As-Found Tolerances and As-Left Tolerances for Kewaunee

In response to NRC staff concerns raised in the 2003-2006 time frame regarding instrument setpoint methodology, (particularly with regard to what has been referred to as "Method 3," as defined in the Instrumentation, Systems, and Automation (now referred to as International Society of Automation) (ISA) Standard ISA-RP67.04.02, "Methodology for the Determination of Setpoints for Nuclear Safety-Related Instrumentation"), and in response to NRC Regulatory Issue Summary RIS 2006-17, the licensee has developed and identified a methodology for establishing the settings for each channel within the scope of this license amendment request, as described further below.

In particular, within RIS 2006-17, the NRC staff provided guidance to licensees regarding the establishment of initial safety related setpoints and the continual maintenance of those setpoints by re-defining the following terms:

> "Limiting Trip Setpoint": The Limiting Trip Setpoint is the limiting setting for the channel trip setpoint considering all credible instrument errors associated with the instrument channel. The limiting trip setpoint is the limiting value to which the channel must be reset at the conclusion of periodic testing to ensure the Safety Limit will not be exceeded if a design basis event occurs before the next periodic surveillance or calibration.
>
> "Allowable Value": An Allowable Value is a limiting value of an instrument's as-found trip setting used during surveillances.

In RIS 2006-17, the NRC staff identified that "if the instrument channel trip setting is not left at a value that is conservative with respect to the Limiting Trip Setpoint, then there may not be assurance that the safety limit will be protected until the next periodic surveillance because instrument drift and other changes to the trip setting can occur. These uncertainties are to be

accounted for in the calculation of the Limiting Trip Setpoint. It is the NRC staff's position that the Limiting Trip Setpoint protects the Safety Limit."

During workshops and interactions with licensees during the 2003-2006 time frame, the NRC staff noted that the measurement of the Limiting Trip Setpoint during periodic surveillances has uncertainties associated with it. Instrument uncertainties associated with the performance of instrument Channel Operational Tests (known as instrument channel "COT uncertainties") are present when instrument channel surveillances are performed. The establishment of Limiting Trip Setpoints must ensure that an instrument channel must remain functional at the conclusion of surveillance testing while allowing for such COT uncertainties. Therefore, the establishment of Limiting Trip Setpoints must account for all COT uncertainties and all non-COT uncertainties. Also, Allowable Values (AVs) may be used to identify limits of operation beyond which, an instrument channel performance during a periodic surveillance test is found to be unacceptable and the channel is to be declared "inoperable." Such AV limits must provide a high assurance that instrument channels will accomplish required trip actions with sufficient margin between the Analytical Limit and the Allowable Value to account for all non-COT uncertainties. Finally, the NRC staff noted in RIS 2006-17 that excessive changes in the trip setpoint could go undetected if operability decisions are simply based on finding an instrument channel trip point conservative with respect to the Allowable Value during a periodic surveillance test. This condition could occur in the event that the instrument channel had been set at the beginning of the surveillance interval at a Nominal Trip Setpoint established conservative with respect to the Limiting Trip Setpoint. The NRC staff provided criteria for establishing "As-Left" and "As-Found" Tolerance limits in a manner that will not result in an excessive amount of instrument channel drift which could potentially go undetected if these tolerance limits were inappropriately determined.

In the determination of Reactor Protection System (RPS) and ESFAS setpoints, the licensee has recognized and utilized the above NRC staff concerns, revised terminology, and guidance incorporated into TSTF-493, Revision 4 and NRC Regulatory Issue Summary (RIS) 2006-17. The new terminology and requirements detailed in TSTF-493, Revision 4 and RIS 2006-17 have been incorporated into Kewaunee's Setpoint Control Program as outlined in its submittal documents. In addition to the new terminology and requirements, the licensee has agreed with the position of the NRC staff as outlined in RIS 2006-17, and has taken the position that the Limiting Trip Setpoint (LTSP) protects the Safety Limit. This revised position represents a change from the historical definition of the Allowable Value as delineated in Standardized Technical Specifications (STS), i.e., "a setting chosen to prevent exceeding a Safety Analysis Limit". Since the Limiting Trip Setpoint (LTSP) accounts for all credible instrument errors associated with the instrument channel, it is a more conservative setting than the associated Allowable Value as previously defined in pre-2006 versions of ISA Standard S67.04. With respect to Kewaunee's conversion to Improved Technical Specifications, the licensee has stated that it agrees with this revised position based on explanations and guidance provided in TSTF-493, Revision 4 and RIS 2006-17, and has reflected this in its project technical reports. In addition, the licensee has established new Nominal Technical Specification Setpoints (NTSPs) that are always more conservative than the LTSPs, and that are the values established as the Kewaunee Improved Technical Specification "Limiting Safety System Settings" that are maintained under its "Setpoint Control Program."

Further, the licensee has documented the requirements of its new "Setpoint Control Program" in Section 5.5.16 of the Kewaunee Improved Technical Specifications, specifically identifying the

Nominal Trip Setpoints (NTSPs) as the Technical Specification "Limiting Safety System Settings." The NTSPs of the safety functions for Technical Specification LCOs 3.3.1 (Reactor Protection System RPS Instrumentation), 3.3.2 (ESFAS Instrumentation Functions), 3.3.5 (Loss of Offsite Power (LOOP), Diesel Generator Start Instrumentation, 3.3.6 (Containment Purge and Vent Isolation Instrumentation), and 3.3.7 (Control Room Post Accident Recirculation Actuation Instrumentation) shall be demonstrated to be functioning during Channel Calibrations, Channel Operational Tests, and Trip Actuating Device Operational Tests that verify the NTSP.

Dominion has documented its methodology for establishing the AVs, LTSPs, NTSPs, AFTs, and ALTs in Dominion Technical Report EE-0116, *"Allowable Values for North Anna Improved Technical Specifications (ITS) Tables 3.3.1-1 and 3.3.2-1, Setting Limits for Surry Custom Technical Specifications (CTS), Sections 2.3 and 3.7, and Allowable Values for Kewaunee Power Station Improved Technical Specifications (ITS) Functions Listed in Specification 5.5.16,"* Revision 8, and in Dominion Technical Report EE-0132, "*Setpoint Methodology for Kewaunee Power Station--Kewaunee Unit 1,*" as revised. In these technical reports, Dominion has provided a detailed description of its methodology for determining LTSPs, AVs, NTSPs, and AFTs based on the analytical limits (ALs) (where applicable) that are documented in Licensee's controlled document, Dominion Technical Report NE-0994, "*Safety Analysis Limits for Technical Specification Instrumentation Companion to EE-0101,*" Revision 17.

The NRC staff notes that to address the determination of the term "Total Loop Uncertainty" (TLU) as identified in some NRC staff guidance documents, the licensee's Technical Report EE-0116 uses the same nomenclature. However the NRC staff also notes that within the individual licensee instrument channel setpoint calculation documents the licensee has equated the term "Channel Statistical Allowance" (CSA) with the term Total Loop Uncertainty in its topical report EE-0116. The staff notes that within Westinghouse Electric Company technical reports of the late 1970s and early 1980s the term CSA is used to estimate the likely maximum uncertainty for anticipated channel performance deviation which is then to be compared with the margin between the previously established Nominal Trip Setpoints and their associated analysis limits, (identified as the "total allowance") to ensure that the estimate of such deviation will not exceed these margins. However, for the purpose of establishing a high confidence margin between the new Nominal Trip Setpoints and their associated Analytical Limits, the NRC staff notes that the CSA term is equivalent with the TLU term because it is applied to the required margin between the Limiting Trip Setpoint and the Analytical Limit, while the new desired trip Nominal Trip setpoint to be established through this setpoint methodology program is conservative with respect to the Limiting Trip Setpoint. Technical Report EE-0116 provides a demonstration for each instrument channel within the scope of the ITS as to how the LTSPs and AV's for primary reactor trip and ESFAS initiations are established in a manner that is consistent with "Methods 1 and 2" as described in ISA Standard ISA-RP67.04.02, *"Methodology for the Determination of Setpoints for Nuclear Safety-Related Instrumentation."* As described in greater detail below, this methodology ensures that the tolerance interval established between the AL and the AV includes a documented allowance for the appropriate algebraic and statistical combination of all instrument channel normal and accident performance errors that are not observable during a channel operability test (i.e., non-COT errors,) and it ensures that the interval between the AL and the Limiting Trip Setpoint (LTSP) contains a documented allowance for <u>all</u> instrument channel performance errors (i.e., the interval between the AL and the LTSP contains an allowance for the total of all COT and non-COT errors).

3.2.1 Method for Establishing AVs and LTSPs for Primary Reactor Trip, and ESFAS Initiation Settings

In particular, for the technical specification settings that were identified as “Primary” reactor trip or engineered safety feature actuation system settings having an Analytical Limit (AL), and evaluated in the plant safety analysis, the licensee implemented a methodology for ensuring that the instrument channel would perform in a manner that would not allow the AL to be exceeded during a plant transient or analyzed accident. Specifically, the licensee has used the following criteria to demonstrate that the analytical limit (AL) will be protected if all three of the following conditions are satisfied:

1. The interval between the Analytical Limit (AL) and the Limiting Trip Setpoint (LTSP) is equal to or greater than the total loop uncertainty (TLU) (referred to as the “Channel Statistical Allowance” (CSA) within various Dominion technical documents) for that channel. This TLU (or CSA) consists of an appropriate algebraic and statistical combination of the all the identified error terms for the channel, inclusive of all channel operability test (COT) terms and all identified error terms that are not observable during a channel operability test (i.e., the non-COT error terms). In other words, the Limiting Trip Setpoint (LTSP) for Kewaunee has been selected in a manner that accounts for (with a high confidence level) all credible instrument channel performance uncertainties associated with the operation of the instrument channel. An evaluation of the licensee’s methodology for determining the TLU is described in Section 3.4, below.

2. The interval between the Analytical Limit (AL) and the Allowable Value (AV) is equal to or greater than the appropriate algebraic and statistical combination of the non-channel operational test (non-COT) error components of the TLU. In other words, the Allowable Value (AV) for Kewaunee has been selected in a manner that accounts for (with a high confidence level) all credible non-COT instrument channel uncertainties associated with the operation of the instrument channel.

3. The interval between the LTSP and the AV is equivalent to the magnitude of the COT error components of the TLU without the inclusion of any excessive margin.

The net effect of the implementation of these three criteria is to ensure that during the time interval between successive instrument channel calibrations of primary reactor trip or ESFAS initiation instrument channel bistable devices, the expected maximum estimated deviation of the instrument channel trip setting due to operation of the instrument channel under normal operating conditions has been accounted for at a high confidence level in the settings. Hence, instrument channels that had been calibrated to (or set at) values conservative to or equal to the “Limiting Trip Setpoint” (LTSP), have sufficient margin such that channel trip performance between successive surveillance intervals would not be expected to deviate to the extent that the trip setpoint encroaches upon the margin established for the non-COT instrument channel performance under normal, transient, or abnormal operating conditions. That is, there would be sufficient margin remaining between that trip point and the AL to ensure that during the next successive surveillance interval the instrument channel would not have deviated from its setting

more than an amount that accounts for the high confidence estimate of the allowance for such normal operating performance drift deviation, inclusive of all COT uncertainties.

3.2.2 Method for Establishing NTSPs

Further, to ensure that all primary reactor trip and ESFAS initiation instrument channel settings are established in a conservative manner, the licensee has identified instrument channel Nominal Trip Setpoints (NTSPs) that are all more conservative than the LTSPs. Settings that were originally contained in the Kewaunee Custom Technical Specifications (CTS) were compared against the revised LTSPs and AVs established using the criteria described above. In the cases where it was found by the licensee that the original CTS settings were non-conservative with respect to the new AVs and/or LTSPs established using the newer criteria described above, taking into account the new AFT tolerance band around the NTSP, for the Kewaunee improved technical specifications (ITS) the licensee revised these CTS settings and established new NTSPs for the Improved Technical Specifications. That is, the licensee's staff has selected NTSPs such that when the instrument channel is adjusted to the values identified as NTSPs and then established appropriate corresponding As-Found Tolerance bands (see next paragraph) that account for the allowed deviation of the settings during the next successive surveillance interval, the worst-case deviation of the channel trip point from the value at which the channel had been adjusted at the conclusion of the surveillance interval calibration process would not allow the channel trip point to exceed the channel's "Allowable Value."

3.2.3 Method for Establishing AFTs and ALTs

Further, the licensee established limits for "As-Found" setting tolerances (AFTs). Generally, the AFTs were established for primary reactor trip and ESFAS initiation channels by adding and subtracting the appropriate statistical combination of the COT loop error components to the NTSP in order to establish the upper and lower limits of the AFT tolerance band, and then ensuring that the worst case limit of the As-Found Tolerance band remains conservative with respect to the Allowable Value. The COT loop error components are generally inclusive of the manufacturer's reference accuracy of the components in the loop, the calibration equipment accuracy, and rack drift, with no additional margin added.

The licensee has established "As-Left" tolerance bands (ALTs) for primary reactor trip and ESFAS initiation channels based on an amount equivalent to 1 times the component manufacturer's "reference accuracy" value for the device performing the trip action (i.e., the comparator/bistable device on the rack,) without including any additional measurement and test equipment uncertainty term or drift term. ALTs for back-up reactor trip and ESFAS initiation functions without formal CSA calculations were largely documented based on an evaluation of COT error minus the magnitude allowed for rack drift. The NRC staff position regarding limiting safety system settings (LSSSs) is that such As-Left Tolerances should be representative of a small, but achievable calibration acceptance tolerance around the NTSP implemented during the channel calibration process—i.e., small in comparison to the AFT computed as described above.

In the Dominion Technical Report EE-0116, the licensee states that the limits of the As-Found Tolerance bands for "primary" instrument channel functions analyzed in the plant safety analysis

"must be set equal to or, preferably, conservative with respect to the Allowable Value" determined using the criteria as described above.

The NRC staff has determined that this methodology is consistent with the NRC staff's guidance in RIS 2006-17.

3.2.4 Back-up Reactor Trip, Permissive, and ESFAS Initiation Settings

For instrument channels serving as "back-up" reactor trip and ESFAS functions, for which no ALs have been established, there are several safety system settings for which no formal Channel Statistical Allowance calculations have been determined to date. However, to comply with the scope of the Westinghouse Improved Technical Specifications, the licensee has proposed to treat these channels as within the scope of the licensee's Setpoint Control Program. For such cases the licensee utilized the existing CTS settings as Nominal Trip Setpoints (NTSPs) and then determined appropriate "As-Found" tolerance bands (AFTs) around these values that have been determined consistent with the criteria described above, or in some cases through a more subjective (less quantitative) process. For example, some settings are selected to be close enough to normal operating conditions to be able to easily detect an adverse change in process variable, but statistically high enough to not cause a spurious trip or isolation to occur. Example: Radiation monitoring functions that are set to a "statistically significant level above background" that are designed to detect increases in particulate or noble gas levels in relatively high volume ventilation flow streams.

For such non-safety analysis related settings, the ALTs are based on existing calibration procedure allowances, and AFTs were established based on an evaluation of historical performance that demonstrates consistent performance during successive calibrations within the manufacturer's published instrument performance specifications. In cases where it was found that historical performance demonstrates that significant instrument channel deviation between calibration surveillance intervals was small in comparison with the setting allowances appropriate to the calibrated span for the channel function, and the type of instruments involved, (e.g., high performance electronic equipment, such as radiation detection instrument channels), the As-Found Tolerance was set to be the same as (equal to) the As-Left Tolerance band for that channel. This method does not allow much room for drift/deviation between successive surveillance intervals; however, in its Setpoint Control Program (see below), for all channels, the licensee has committed to ensuring that at the completion of every surveillance test, the instrument channel setpoint shall be reset to a value that is within the As-Left Tolerance around the NTSP—otherwise the channel is found to be inoperable. If the As-Found value of the trip setting differs from the previous As-Left value or the specified NTSP by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel functionality shall be evaluated before declaring the Surveillance Requirements to be met and returning the instrument channel to service. The identification of such a condition during the conduct of a Technical Specification Surveillance Requirement shall be noted and the identification of that instrument channel shall immediately be entered into the plant corrective action program. The NRC staff notes that although this proposed practice may not allow for much deviation from one surveillance to the next successive surveillance without potentially requiring a corrective action report to be prepared, it is a conservative practice from a safety perspective.

3.3 Setpoint Control Program

Evaluation of Technical Specification Operability Determination and Required Actions

The licensee has proposed to implement a Setpoint Control Program, as described in its Improved Technical Specifications Section 5.5.16. This program establishes the requirements for ensuring that setpoints for automatic protective devices are initially within and remain within the assumptions of the applicable safety analysis; and provides a means for identifying whether changes to established instrumentation setpoints are needed, based on operating performance. It also identifies instrument channel surveillance requirement methods to assess channel operability and ensure safety instrumentation will function as required. The program ensures that surveillance testing of automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A) verifies that instrumentation will function as required. The NRC staff evaluation of the conformance of the licensee's Setpoint Control Program to the requirements of TSTF-493 may be found in this Safety Evaluation as Safety Evaluation Attachment 2; however, the NRC staff has issued specific criteria pertaining to the methods for assessing LSSSs during periodic testing and calibration of instrument channels accomplishing LSSSs. These criteria were contained in a Regulatory Issue Summary, RIS 2006-17. The NRC staff has evaluated the licensee's proposed Setpoint Control Program for compliance with the intent of RIS 2006-17, and has identified several important features of the licensee's proposed program, as summarized below.

The licensee's program includes the following scope of safety related instrument functions, which are discussed in Safety Evaluation Attachment 2, also:

- LCO 3.3.1 (Reactor Protection System (RPS) Instrumentation),
- LCO 3.3.2 (ESFAS Instrumentation Functions),
- LCO 3.3.5 (Loss of Offsite Power (LOOP) Diesel Generator Start Instrumentation,
- LCO 3.3.6 (Containment Purge and Vent Isolation Instrumentation), and
- LCO 3.3.7 (Control Room Post Accident Recirculation Actuation Instrumentation).

The program ensures that these functions are periodically demonstrated to be functioning during Channel Calibrations, Channel Operational Tests, and Trip Actuating Device Operational Tests that verify the NTSP.

The program requires that the Nominal Trip Setpoint (NTSP), Allowable Value (AV), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) (as applicable) of the safety functions described within the scope identified above are calculated using an NRC-approved setpoint methodology.

> Note
>
> Based on an initial evaluation of the materials submitted by the licensee with this license amendment request, the NRC staff has not evaluated the licensee's instrument setpoint methodology program for Kewaunee, the approval of which is a prerequisite for enabling the licensee to make full implementation of Option B of TSTF-493, Revision 4. The staff is working with the licensee and the BWR and PWR Owners Groups to clarify its acceptance criteria for developing and submitting a programmatic licensee setpoint methodology with appropriate controls for approval and adoption of new setpoints as they are needed. Therefore, this SER is limited to providing the NRC staff evaluation of the adequacy of the portion of the licensee's submittal concerning the technical approach used by the licensee in establishing and maintaining the instrument channel settings as documented in the Improved Technical Specifications. At a later date, after the NRC staff issues a clarification to its May 11, 2010 *Federal Register* Notice of Availability (75 FR 26294) for the TSTF-493, regarding the programmatic requirements and acceptance criteria for establishing a program enabling the revision or development of new Technical Specification instrument channel settings under a licensee-controlled setpoint methodology program, the NRC staff will review the licensee's submittal in response to those clarifications and perform a separate safety evaluation of that submittal under a 10 CFR 50.90 amendment process.

In addition, the program lists the value of the NTSP, AV, AFT, or ALT (as applicable) for each function described above and the licensee has identified, in general terms, the setpoint methodology used to calculate these values, as described in the evaluation above.

The program also establishes the methods to ensure that the safety functions described above will function as required by periodically verifying that the As-Left and As-Found settings are consistent with those established by the setpoint methodology. If the As-Found value of the instrument channel trip setting is found to be less conservative than the specified AV, then the Surveillance Requirement is not met and the instrument channel shall be immediately declared inoperable.

The program identifies the safety functions described above that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). The NTSP of these functions are considered to be the Limiting Safety System Settings. These safety functions shall be demonstrated to be functioning as required by applying the following test requirements and acceptance criteria during CHANNEL CALIBRATIONS, CHANNEL OPERATIONAL TESTS, and TRIP ACTUATING DEVICE OPERATIONAL TESTS that verify the NTSPs:

1. The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified NTSP.

2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified NTSP by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the Surveillance Requirement met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.

3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the Surveillance Requirement is not met and the instrument channel shall be immediately declared inoperable.

4. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the NTSP at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the NTSP provided that the as-found and as-left tolerances apply to the actual setpoint used to confirm channel performance).

The NRC staff finds the above approach for verifying the operability of safety related setpoints to be acceptable, and consistent with the intent of the criteria established within Regulatory Issue Summary RIS 2006-17. The licensee's Setpoint Control Program will be documented in the Kewaunee Technical Requirements Manual. The staff further finds that the surveillance requirements described above are consistent with the concepts identified in TSTF-493.

3.4 Evaluation of the Adequacy of the Licensee's Methods for Establishment of Limiting Safety System Settings through the Analysis and Combination of Uncertainty Allowances

In Regulatory Guide 1.105 Revision 3, the NRC has endorsed, subject to limitations and additional guidance, the use of industry standard ANSI/ISA S67.04-1994, Part 1, "Setpoints for Nuclear Safety-Related Instrumentation used in Nuclear Power Plants." Of primary concern to the NRC staff is that the settings of safety systems be selected in a manner that provides a high confidence that the instrument channel will perform its required safety action while ensuring that the analytical limit for the setting is not exceeded, commensurate with the instrument channel's expected performance under all normal and anticipated accident and operational occurrences. Specifically, for technical specification-related instrument channels, paragraph 50.36(c)(1)(ii)(a) of Title 10 of the *Code of Federal Regulations*, Part 50 requires that "where a limiting safety system is specified for a variable on which a safety limit has been placed, the setting be so

chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

The NRC staff's position regarding high confidence in establishing safety system settings that ensure that ALs will not be exceeded can be met by a) ensuring that a mathematical expression representative of the total set of instrument channel uncertainties reasonably expected to be present during channel operations and calibration conditions has been identified as an appropriate combination of error terms, and b) that all the error terms are combined in a manner representative of their likely contribution to total error, and c) that the data representing each error term has been appropriately estimated and quantified in a manner consistent with the error expression model. In the review of uncertainties in determining a trip setpoint and its allowable values, the NRC staff typically uses 95/95 tolerance limits as an acceptance criterion for this data. That is, there is a 95 percent likelihood that the constructed tolerance limits contain 95 percent of the population of interest for the error term being evaluated.

The nuclear industry has developed several methodologies for establishing Limiting Safety System Settings (LSSS) which combine uncertainties in a part-algebraic and part-statistical manner to model or represent the estimate of total instrument channel uncertainty. The validity of such methodologies requires that input uncertainties used in the estimate of total loop uncertainty be statistically valid to support the model used.

3.4.1 Evaluation of the Methodology for Establishing AV, LTSP, NTSP Values, and AFT/ALT tolerance Bands, and Adequacy of the Expression for Total Loop Uncertainty

To perform an evaluation of a licensee's setpoint program, the NRC staff evaluates the analysis methods to determine whether the analysis parameters and assumptions are consistent with the safety analysis results (in the form of final established analytical limits), system design bases, technical specifications, plant design, and expected maintenance practices. The staff examines the relationships between the analytical limit, limiting trip setpoint, allowable value, nominal trip setpoint, as-left tolerance band and as-found tolerance bands to ensure that the licensee's determination of these terms is consistent with NRC staff guidance provided in Regulatory Guide 1.105 and RIS 2006-17. The staff also reviews the expressions for total loop uncertainty to identify whether there is reasonable assurance that the licensee has accounted for all likely instrument channel uncertainty terms expected to be present for various types of measurement instrument channels, and evaluates whether the licensee's methods for combining these terms appears to realistically represent the expected performance behavior for the channel. In addition, the staff evaluates how the licensee's program maintains an understanding of the relationship between the instrument and process measurement units so that, for example, the settings for any non-linear instrument channel spans are appropriately translated into process units if they were originally based on error estimates determined in instrument units. (Example, flow setpoints may be based on differential pressure measurement devices and the process units for flow are based on a function of the square root of the instrument units). In addition, the staff evaluates the appropriateness of the application of environmental conditions applicable to the location within the plant of sensors versus the location of the rack devices that perform any signal conditioning, comparisons, and trip outputs.

To perform the evaluations for this license amendment request, the NRC staff evaluated in detail the licensee's technical reports, engineering standards, and key reference documents

within those standards and reports to gain an understanding of the set of methods utilized by the licensee to establish the AV, LTSP, NTSP, AFT, and ALT values for the primary RPS trip and permissive channels, and the ESFAS initiation instrument channels. In addition, the staff performed a sampling of several instrument channel calculations of channel statistical allowance, and a set of corresponding instrument channel calibration procedures that contain much of the data used to establish these values, and which in turn, implement some of the values that were determined within the calculations. In addition, the staff reviewed several manufacturers' published bulletins depicting instrument performance parameters to understand how the licensee used the information from the vendor to establish allowance for various instrument channel performance uncertainties.

In the documents submitted with the LAR, the licensee has stated that the setpoint methodology for Kewaunee is based on ANSI/ISA S67.04-Part 1-1982, "Setpoints for Nuclear Safety Related Instrumentation," and ISA-RP67.04-Part-II-1994, "Methodologies for the Determination of Setpoints for Nuclear Safety Related Instrumentation." The NRC staff notes that the primary use of these ISA Standards has been with regard to the definition of terms and the use of criteria for establishing the relationships among the terms AL, AV, LTSP, and NTSP. However, the NRC staff noted that in addition to these documents, expressions for "Total Loop Uncertainty" as referred to within the ISA documents are based in part on a Dominion Engineering Nuclear Standard, STD-EEN-0304, "*Calculating Instrumentation Uncertainties by the Square Root of the Sum of the Squares Method*," Revision 6, effective June 19, 2009. This engineering standard, in turn, consolidates guidance from several Westinghouse Electric Corporation documents, from previous NRC evaluations of North Anna Power Station RPS/ESFAS Setpoint Methods, and from a 1985 NRC NUREG/CR document regarding the appropriated application of statistical analysis to the evaluation of instrument channel uncertainty, among others.

Of particular pertinence to this evaluation are the following two documents referenced within Dominion Engineering Standard STD-EEN-0304, listed within the standard as "Reference 3.11 and Reference 3.10, respectively. These references provide criteria pertinent to the establishment of sufficient channel statistical allowances to provide a high confidence that the established limiting trip setpoints will ensure that the ALs will not be exceeded during a design basis transient or accident:

- "Westinghouse Setpoint Methodology for Control and Protection Systems," as published in IEEE Transactions on Nuclear Science, Volume 33, No. 1, February 1986 by C. R. Tuley and R. B. Miller, of Westinghouse Electric Corporation, and
- NUREG/CR-3659, "Mathematical Model for Assessing the Uncertainties of Instrumentation Measurements for Power and Flow of PWR Reactors," USNRC, February 1985 (prepared by G. M. Hesson, W. C. Cliff, and D. L. Stevens, all from the Pacific Northwest National Laboratory) (ADAMS Accession No. ML0815503351)

The licensee has provided model mathematical expressions within STD-EEN-0304 representing "Channel Statistical Allowance" (CSA) for instrument channels. These expressions further represent instrument channels that are calibrated as a "string" (i.e., where the instrument channel modules are first calibrated individually, and then the collection of modules is calibrated together as a unit from transmitter input through bistable output,) as well as instrument channels

calibrated as a sensor (or multiple sensors) feeding individual instrument modules. These expressions are each further represented in a form that includes a term representing systematic errors.

In Dominion Technical Reports EE-0132, Revision 0, "*Setpoint Methodology for Kewaunee Power Station*" and EE-0116, Revision 8, "*Allowable Values for North Anna Improved Technical Specifications (ITS) Tables 3.3.1-1 and 3.3.2-1, Setting Limits for Surry Custom Technical Specifications (CTS), Sections 2.3 and 3.7, and Allowable Values for Kewaunee Power Station Improved Technical Specifications (ITS) Functions Listed in Specification 5.5.16*," the licensee provided a description of how these mathematical expressions described in STD-EEN-0304 are utilized to represent typical instrument channels that are used to accomplish Kewaunee Power Station RPS reactor trips and ESFAS initiations—specifically, Kewaunee Single-Parameter Protection Functions, Dual Parameter Protection Functions, and Multiple-Parameter Protection Functions. In these examples, the licensee indicates how sensor and rack module error terms are first grouped and then combined overall to represent an expression that contains the errors for all sensor through rack converter and power supply modules, and finally the bi-stable trip unit (comparator) that accomplishes the trip initiation function.

In its discussions regarding how "Allowable Value" has been determined for Kewaunee Reactor Trip and ESFAS initiation channels, the licensee provided an explanation that of the various methods described in ANSI/ISA RP67.04.02 for determining the Allowable Value, "Method 3" as described within this standard was not used. Instead, the licensee provided a clear description regarding how it has utilized either "Method 1" or "Method 2" as presented within the ANSI/ISA standard. Using Methods 1 or 2 will ensure that the Allowable Value (equivalent to the Minimum or Maximum Allowable Value for Surry and North Anna) will account for all credible instrument and process errors that are not tested or quantified during the performance of the Channel Operational Test (COT), and thus ensure that an allowance for these instrument and process errors will always remain between the Allowable Value and the Analytical Limit, such that in the event that a channel is calibrated at the Limiting Trip Setpoint, an allowance will exist between the Limiting Trip Setpoint and the Allowable Value to accommodate the uncertainties in measurement of the setting during periodic surveillances that does not encroach upon the allowance estimated to account for all non-COT errors. (However, the licensee for Kewaunee plans to calibrate all primary RPS and ESFAS trips, initiations, and permissives and other LCOs at the established Nominal Trip Setting, which has been selected to be at a value that is always more conservative than the Limiting Trip Setpoint.)

Dominion Technical Reports EE-0132, Revision 0 and EE-0116, Revision 8 describe how the licensee has applied a four-step process for establishing the LTSP, AV, NTSP, and AFT for primary RPS and ESFAS trips, permissives, and other limiting conditions for operation that are credited in the Kewaunee Safety Analyses. This four-step process ensures that the Allowable Values for Kewaunee are determined based on the ANSI/ISA Recommended Practice "Method 1" or "Method 2." This process is briefly summarized as follows:

1. The licensee first determined the Minimum (decreasing trip) or Maximum (increasing trip) Limiting Trip Setpoint (LTSP). The Maximum Limiting Trip Setpoint is arrived at by subtracting the Total Loop Uncertainty (TLU) from the Analytical Limit (AL) (also known as the Safety Analysis Limit). (The Minimum Limiting Trip Setpoint is arrived at by adding the Total Loop Uncertainty (TLU) to

the Analytical Limit (AL).) The Analytical Limits were documented in a controlled licensee document, Dominion Technical Report No. NE-0994, "Safety Analysis Limits for Technical Specification Instrumentation Companion to EE-0101," Revision 17.

2. Next, the licensee determined the Minimum (decreasing trip) or Maximum (increasing trip) Allowable Value (AV). This Maximum Allowable Value is arrived at by subtracting the statistical combination (i.e., Square Root of the Sum of the Squares "SRSS") of the NON COT Loop Error Components (i.e., the loop error terms that are not tested or quantified during the Channel Operational Test "COT") from the Analytical Limit (AL). (The Minimum Allowable Value is arrived at by adding the statistical combination of the NON COT Loop Error Components to the Analytical Limit (AL).)

3. Once these values were determined, the licensee proceeded to establish the appropriate Nominal Trip Setpoint (NTSP). After the LTSP was determined in step 1, the current Nominal Trip Setpoint for the function was evaluated by the licensee for acceptability. In several instances, the current Nominal Trip Setpoint in the Technical Specifications had to be moved in a more conservative direction to obtain additional margin to the Analytical Limit and/or to allow for the full COT error allowance between the Nominal Trip Setpoint and the non-conservative limit of the As Found Tolerance (AFT). (That is, the Nominal Trip Setpoint (NTSP) was established to be conservative with respect to the Limiting Trip Setpoint (LTSP); however, it was additionally constrained by the establishment of the As-Found Tolerance limit.) The As-Found Tolerance limit was established in a manner that was always conservative with respect to the Allowable Value, so in certain cases the NTSP had to be set further away from the Allowable Value to accommodate the AFT.

4. Finally, the licensee determined the magnitude of the upper and lower limits of the As Found Tolerance (AFT). After the AV was determined in step 2, and the NTSP was established in Step 3, the upper and lower limits of the As Found Tolerance band were determined based on the SRSS expression representing COT error performance and drift, which is then applied to the value determined for the NTSP. The AFT band limit for an increasing trip function is arrived at by adding the statistical combination (i.e., Square Root of the Sum of the Squares "SRSS") of the COT Loop Error Components (i.e., the loop error terms that are tested or quantified during the Channel Operational Test "COT") to the Nominal Trip Setpoint (NTSP). The AFT for a decreasing trip function is arrived at by subtracting the statistical combination of the COT Loop Error Components from the Nominal Trip Setpoint. In all cases, the worst-case (non-conservative direction) limit of the As Found Tolerance band was set conservative with respect to the Allowable Value. The NRC staff also noted that the licensee's determination of As-Left Tolerance bands was consistent in the application of a band equivalent to plus and minus a value corresponding to the manufacturer's published reference accuracy for the module performing the actual trip output.

The NRC staff notes that the Westinghouse Electric Corporation documents referenced within the licensee's Engineering Standard STD-EEN-0304 provide the bases for identifying the various types of instrument channel uncertainties which may be present during the operation and maintenance of the Kewaunee primary RPS trip and permissive, and ESFAS initiation instrument channels. Additionally, these documents provide mathematical expressions of instrument channel uncertainty allowances that address the algebraic and statistical combination of these terms deemed appropriate by the reactor supplier to be representative of instrument channels for which there are field mounted instruments located in environments that are subject to harsh conditions during design basis events, and rack-mounted modules located in milder environmental areas of the plant that are still subject to power supply variations, EMI/RFI effects, seismic effects, and other effects. The Westinghouse documents were prepared during the late 1970s through the mid-1980s, and are reflective of the configuration of field sensor and remote-mounted rack components typical of the hardware used during that time frame. (i.e., the Westinghouse technical documents were based on the use of Foxboro, Barton, and Veritrack/Tobar and Rosemount sensors, and Foxboro, Hagan 7100 series, and Westinghouse 7100 and 7300 series rack modules.) Although some of the rack hardware has since been replaced at Kewaunee by newer versions of rack modules (e.g., some original Foxboro rack modules have been replaced with newer NUS equivalent modules) the newer modules were sufficiently similar to enable the same mathematical expressions to be representative of the anticipated channel performance. Also, the mathematical error analysis model expressions representing a field sensor (or sensors) located in potentially harsh environments that supply signals to a set of rack modules in a more controlled environment is still valid, and in its calculations of Total Loop Uncertainty, the licensee has incorporated module performance error terms representative of the newer module performance, where applicable.

The Westinghouse reports referenced within the licensee's Engineering Standard STD-EEN-0304 provide the bases for establishing expressions of Total Loop Uncertainty (referred to as Channel Statistical Allowances in some of the licensee's referenced documents) identify a particular set of safety instrument channel performance uncertainty terms as being representative of all channel error terms. These include, but are not limited to:

> Process Measurement Accuracy (PMA) – an allowance for non-instrument related effects which have a direct bearing on the accuracy of an instrument channel's reading, e.g., temperature stratification in a large diameter pipe, or the effects of changing fluid density on level measurements.
>
> Primary Element Accuracy (PEA) – (example:) the error due to the use of a metering device, e.g., elbow, orifice, or venturi. The licensee uses this term as an allowance for the inaccuracies of the system element that converts the measured variable energy into a form suitable for measurement.
>
> Sensor Calibration Accuracy (SCA) - the "reference (calibration) accuracy" for a sensor or transmitter as defined by SAMA Standard PMC 20.1-1973. The NRC staff notes that the SAMA definition has identified the fact that the term "reference accuracy" is often associated with the effects of hysteresis, linearity, repeatability, reproducibility, *and drift*. However, the NRC staff notes that within the US nuclear industry, the common usage of the term "reference accuracy" in the nuclear industry usually includes the effects of hysteresis, linearity,

deadband, and repeatability. The term “reproducibility” is often equated with the term “repeatability” for bistable devices and for analog components it represents the degree of conformity to repeated tests of applied inputs over the calibrated span of the device. Also, in the nuclear industry, the term “drift” is treated separately from “reference accuracy.” The NRC staff notes that the licensee has identified field sensor drift as the term “Sensor Drift” (SD) and the drift associated with rack-mounted modules is identifies as the term “Rack Drift” (RD). The licensee has defined this term in the form of a “limit” that errors will not exceed when a sensor is used under specified operating conditions.

Sensor Pressure Effects (SPE) - the change in input-output relationship due to a change in the static head pressure from the calibration conditions (if calibration is performed at line pressure) or the accuracy to which a correction factor is introduced for the difference between calibration and operating conditions for a differential pressure transmitter. The licensee has applied this term as an “allowance” for the effects of steady state pressure on a device. For differential pressure instruments, it represents the change in input-output relationship due to a change in applied static pressure (usually to account for the change that occurs when a sensor is exposed to operating conditions after having been calibrated at calibration conditions).

Sensor Temperature Effects (STE) - the change in input-output relationship due to a change in the ambient temperature (for expected normal operating conditions) from the reference calibration conditions about a transmitter. The licensee applies this term as an allowance for the effect of changes in input-output relationship due to changes in ambient temperature surrounding the instrument, and notes that changes in process medium are not included in this term, but rather in the process measurement accuracy (PMA) term.

Sensor Drift (SD) - the change in input-output relationship over a period of time at reference conditions. The licensee applies this term as an allowance for such effects.

Environmental Allowance (EA) - the change in sensor/transmitter output due to adverse radiation and temperature effects from a limiting accident condition. Typically this value is determined from a conservative (bounding) set of enveloping conditions. The licensee applies this term as an allowance for such effects, usually applied to the sensor. The NRC staff noted when evaluating the licensee’s documentation of calculations of Channel Statistical Allowance that the licensee’s standard expressions for channel statistical allowances included the EA term within the SRSS combination of error terms. The staff noted that typically, such terms representing the effects of harsh environment on field devices are not based on a sufficient number of samples of test data to warrant treatment of this term as a pure random variable. Further, one of the Westinghouse technical reports referenced within the licensee’s engineering standard (Reference 3.11) specifically noted that the EA term should be algebraically added rather than combined using SRSS means. However, the staff also noted that the licensee Engineering Standard EEN-0304, Revision 6

provides guidance for performing an evaluation of the data associated with the publication of environmental allowance values that are not clearly identified as representative of sufficient data to warrant treatment as 2-sigma bounds of random variables, representing a 95 percent confidence interval estimate. The engineering standard states: “When anything other than the manufacturer's stated allowance is chosen, or the EA term is not random, the EA term shall be taken out of the square root function and directly added to the SE and radical terms.” Further, the staff noted that for the calculations of channel statistical allowance associated with the primary Reactor Trip and Permissive and ESFAS Initiations, (e.g., Calculation C10818 for Pressurizer Low Pressure Safety Injection Initiation under Design Basis Conditions) the applicable error effect associated with harsh environment was Insulation Resistance error effect, and this error term was removed from inside the SRSS combination and accounted for by algebraically summing this error effect with the other terms.

Rack Calibration Accuracy (RCA) - the reference (calibration) accuracy, as defined by SAMA Standard PMC 20.1-1973, for the process rack modules (cards) in an instrument loop. When the licensee has made use of this term within calculations of loop uncertainty, they are representing that each module or card is individually calibrated to a specific tolerance, and then all rack modules are string-calibrated to within an overall tolerance value.

Rack Comparator Setting Accuracy (RCSA) - the reference (calibration) accuracy, as defined by SAMA Standard PMC 20.1-1973, of the instrument loop comparator (bistable). The licensee applies this term in the form of a “limit” that errors will not exceed when the device is calibrated (verified) during a string calibration.

Rack Temperature Effects (RTE) - change in input-output relationship for the process rack module string due to a change in the ambient temperature from the reference calibration conditions. The licensee applies this term as an allowance for the effects of such changes.

Rack Drift (RD) - the change in input-output relationship over a period of time at reference conditions.” The licensee has applied this term when describing the change in input-output relationship for the modules mounted on the electronic rack that processes the input signal from the field device. The licensee applies this term as an allowance for the effects of such changes.

The licensee’s error methodology standard STD-EEN-0304, Revision 6 further identified other terms that are needed to appropriately characterize the instrument channel uncertainty, given the uniqueness of the installation, hardware, event conditions, calibration methods, and other unique features. These include:

Systematic Error (SE) - an error which, in the course of a number of measurements made under the same conditions of the same value of a given quantity, either remains constant in absolute value and sign or varies according

to a definite law when the condition changes. Examples are cable and terminal blocks in high temperature and/or moisture environments, and reference leg heat-up (differential pressure devices used for level measurements). The licensee has combined this term with other error terms using algebraic addition.

Sensor Measuring and Test Equipment Error (SMTE) – an allowance for the accuracy rating of the measuring and test equipment used in performing sensor calibrations.

Sensor Power Supply Effect (SPSE) – an allowance that accounts for the loop power supply voltage fluctuation effects on the transmitter output

Rack Module Measuring and Test Equipment (RMTE/MnMTE) - an allowance for the accuracy rating of measuring and test equipment used in performing rack module calibrations, and

Rack Readability Allowance (RRA) – an allowance for the inability to read analog indicators because of parallax distortion. Due to the devices installed and the method for calibration used, this term is not used in the calculation of Channel Statistical Allowance for the primary Reactor Trip and Permissive functions or ESFAS Initiation functions at Kewaunee Power Station.

The Westinghouse and Dominion technical reports provide expressions of Channel Statistical Allowance (CSA) that represent the combined uncertainty of the sensing and rack components that are representative of the Kewaunee primary RPS Trip and Permissive function and ESFAS Initiation instrument channels. These technical reports also provide several analyses and justification of the bases for identifying which terms may be considered random and independent, random and dependent, and non-random in nature, and which terms may be partially dependent on which other terms. These analyses also provide the bases for identifying Channel Statistical Allowance expressions that include a combination of uncertainty terms in a manner such that non-random uncertainties are combined algebraically, random terms demonstrated and justified as purely independent terms are combined using the Square-Root-of-the-Sum-of-the-Squares (SRSS) method, and random terms demonstrated and justified to be partly dependent on other random terms are first combined algebraically with the other dependent random terms before their resultant sum is combined with other random terms using the SRSS method.

The Westinghouse publication listed as Reference 3.11 in the Dominion Engineering Standard provided a basis for the identification and use of the combination of uncertainty terms representative of field sensors feeding signals to rack mounted modules. The expressions found in the licensee's engineering standard were not identical to those in the Westinghouse reference; however, the expressions used by the licensee were based on the same principles indicated within the Westinghouse publication and conservatively included additional error terms appropriate to the number of modules within the rack, the uncertainties associated with the measurement and test equipment uncertainties and methods used for calibrating the field sensors and the rack-mounted equipment. In addition, as described above, the licensee expression for channel statistical allowance included an uncertainty term representing the systematic error that could be present due to specific channel configuration and field conditions.

Systematic error is error that is deterministic (non-random) in nature and the NRC staff noted that the licensee has combined this term within expressions of total loop uncertainty as algebraic additions to the SRSS combination of the random error terms.

The NRC staff has found that this means of identifying expressions of total loop uncertainty is consistent with industry practice, Westinghouse publications regarding appropriate instrument setpoint methodology for Westinghouse reactor facilities, and provides one of the major components necessary to achieve a high confidence that estimates of instrument channel performance uncertainty using this expression will contribute toward an overall high assurance that the instrument channel will perform its required safety trip action during a transient or design-basis accident prior to the analytical limit being reached. Another necessary component needed to establish such high confidence is the establishment of estimates of individual error terms within these expressions that are based on an appropriate quantity and quality of data needed to support the model of error expression implied by the use of the SRSS methodology.

3.4.2 Evaluation of Data Quality Representative of the Instrument Channel Performance Uncertainty Terms

Regulatory Position C.1 of Revision 3 (1999) of Regulatory Guide 1.105 states that “Section 4 of ISA-S67.04-1994 specifies the methods, but not the criterion, for combining uncertainties in determining a trip setpoint and its allowable values. The 95/95 tolerance limit is an acceptable criterion for uncertainties. That is, there is a 95 percent probability that the constructed limits contain 95 percent of the population of interest for the surveillance interval selected.” Revision 2 (1986) of Regulatory Guide 1.105 stated “Paragraph 4.3 of the standard specifies the methods for combining uncertainties in determining a trip setpoint and its allowable values. Typically, the NRC staff has accepted 95 percent as a probability limit for errors. That is, of the observed distribution of values for a particular error component in the empirical data base, 95 percent of the data points will be bounded by the value selected. If the data base follows a normal distribution, this corresponds to an error distribution approximately equal to a "two sigma" value.”

The SRSS model used for representing the appropriate combination of random error terms is only valid if sufficient data exists to support statistical analyses, the data representing random error terms are normally distributed, and the data represents uncertainty terms that are random and independent. If so, the central limit theorem/SRSS methods can be used to combine like units of standard deviations of random error. If the endpoints of each of the uncertainty term intervals are being estimated based on a sufficient sample of random variables, then the level of confidence associated with a particular tolerance interval estimate is a measure of the <u>likelihood</u> that the calculated interval does, in fact, cover at least the specified portion of the population. That is, if the tolerance limits have been based on a statistically sufficient quantity of sample data, then the confidence that the interval contains 95 percent of the population of interest increases.

The NRC staff position is that an acceptable tolerance interval to use for such estimates of instrument performance characteristics is the number of standard deviations of normally distributed random uncertainty variables corresponding to a 95 percent confidence that the interval contains 95 percent or more of the population of interest. Assuming that the sample size on which this data has been based is large (typically hundreds of data points or more,) this

would correspond to approximately ±2-standard deviations of the sample data distributed about the mean of the normal distribution. However, if the raw data sample size is not sufficiently large, (e.g., there has not been an appropriate number of tests or measurements of the characteristic of interest to support statistical test methods) the tolerance interval representing the uncertainty term should be determined through best-estimate means, such as bounding assumptions supported by adequate technical justification, (e.g., documented historical operating experience) or through use of appropriate statistical means (e.g., applying Student's t-factors to increase the estimated size of the interval to account for unknowns).

The NRC staff evaluated the licensee's submitted engineering standards, technical reports, referenced Westinghouse technical reports, and has performed an evaluation of a sample of actual calculations of Channel Statistical Allowance that utilized the criteria contained in the licensee's technical reports and referenced documents. The staff notes that the licensee's calculation documents do not contain statements that enforce the application of the 95/95 acceptance criterion for estimating the magnitude of random error term uncertainty intervals when estimating Total Loop Uncertainty for establishing the required margin between the Analytical Limits and Limiting Trip Setpoints. However, the NRC staff noted that in several areas, the licensee's engineering standard EEN-0304, Revision 6 provides guidance requiring users to evaluate sources of data extracted from purchasing specifications and manufacturer's published specifications in a manner that acknowledges that where such data is used, it should be representative of a 95 percent confidence level to achieve conservative results. Attachment A to EEN-0304, for example, provides guidance for establishing high confidence interval estimates by conservatively bounding sufficiently large sample sizes of historical calibration data for modeling the performance of "as-found" values associated with instrument drift. Elsewhere in EEN-0304, Revision 6, cautionary statements are provided regarding the evaluation of data to be used in the estimate of Environmental Allowance (EA) so as to ensure that 95 percent confidence limits are used or else the data should be treated as a non-random error term and algebraically added within the calculation of Channel Statistical Allowance.

In its actual calculations of channel statistical allowance (which were used to identify Total Loop Uncertainty) the NRC staff noted that licensee utilized bounding assumptions for several of the error terms associated with calibration uncertainties and channel operational test (COT) errors, and justified the use of these assumptions. For example, in its estimate of Channel Statistical Allowance, the term for Rack Drift (RD) is included in the margin between the Analytical Limit and the Limiting Trip Setpoint. Although the Westinghouse technical reports referenced within the licensee's engineering standard identified that a conservative estimate for rack drift based on observation of actual historical performance of the Westinghouse 7100 and 7300 series modules to be ±0.5 percent of span, for conservatism the licensee utilized a value of ±1.0 percent to conservatively bound the estimate uncertainty tolerance interval for this term. Also, in the licensee's estimates of Channel Statistical Allowance, is an uncertainty term representing measuring and test equipment error used to account for estimates of Sensor (SMTE) and Rack (RMTE and/or MnMTE). The NRC staff noted that the licensee evaluated the use of all possible measurement ranges of the Fluke 45 precision voltmeter that is used for calibrating rack mounted equipment and then bounded the worst-case value of the manufacturer's published calibration equipment performance in the total CSA calculation. However, within the actual calibration procedures there were instructions for the instrument calibration technicians to utilize the best performing (most accurate) calibration range, which results in additional conservatism in the estimate of total loop uncertainty.

To verify whether the sensor performance data used by the licensee represents 95/95 percent tolerance intervals, an independent check of several of the values used by the licensee in its estimates of total loop uncertainty was performed by the NRC staff. For example, when evaluating licensee calculation C10818, the staff obtained a copy of the Rosemount Product Manual referenced within the calculation (Product Manual 00809-0100-4631, Revision BA, dated April 2007) and the Product Data Sheet associated with this device (Data Sheet 00813-0100-4631, Revision BA, dated April 2007) and was able to verify that the licensee utilized appropriate values representative of 2-sigma-or-better performance of sensor reference accuracy, sensor temperature effect, and sensor drift over a 30-month interval. The staff noted, however, that within the licensee's calculation documents, there were no statements confirming that an analysis of the vendor performance data was performed to demonstrate that it has been confirmed with the vendor that the data used is truly representative of a 95/95 tolerance as acceptance criteria. Although the NRC staff is cognizant of the data quality represented by the manufacturer in this instance, the licensee should document in its calculations that it has verified that each manufacturer published data term used is truly representative of the 95/95 acceptance criterion, to accomplish the intent of the guidance provided in Appendix A of its own engineering standard EEN-0304, Revision 6. Alternatively, if it cannot be demonstrated that such data is representative of the 95/95 tolerance interval acceptance criterion, appropriate statistical corrections (e.g., student's t-factors) or other suitably documented bounding analyses should be performed to ensure that the tolerance intervals for uncertainty terms used in the calculations of total loop uncertainties for limiting safety system settings support the modeling of random error as a normal distribution.

Other data used by the licensee to depict systematic error terms were supported by sketches of the instrument channel installation circumstances, and accompanying analyses to justify the magnitude of the error terms being estimated. For example, to estimate the magnitude of the systematic error for the static head correction associated with the pressurizer pressure instruments, the licensee provided a sketch of the actual elevations of the pressure taps, condensing chambers, and transmitter mounting heights and converted the resulting head correction factors into percent of span. Elevation data for this sketch was determined from actual field verification. The correction head was calculated based on a reference temperature representative of normal ambient temperature in the vicinity of the instrument sensing lines during the times when these devices would be expected to perform their required safety actions. The NRC staff finds this to be an appropriate supporting tool to document the conservatism or bounding analysis that provides a high confidence that the channel total loop uncertainty has been conservatively established.

In other areas evaluated by the NRC staff, it was noted that within the licensee instrument channel calibration procedures evaluated, it was identified that the setting tolerances depicted in tables of performance acceptance criteria ("Acceptance Range," equivalent to As-Left Tolerance) during the calibration process were, in fact, based on values evaluated within the calculation of channel statistical allowance associated with that channel.

Although the materials submitted by the licensee in support of this license amendment request do not directly document that the licensee has estimated the uncertainty terms used in the calculation of channel statistical allowances or total loop uncertainties in a manner that achieves the NRC staff acceptance criterion of 95/95 for every error term, the NRC staff finds that the materials submitted depict that sufficient conservatisms have been built into the estimates of the

terms used in the calculation of total loop uncertainty to provide a reasonable assurance that the LTSP values have been determined in a conservative manner. In instances where it was not clear whether the instrument performance or instrument error allowance values depicted had been estimated to be representative of meeting the 95/95 criterion, the licensee provided and documented suitable bounding assumptions to ensure that the error term used is reasonably conservative. The staff finds that this is an acceptable alternative approach to meeting the staff's guidance in Regulatory Guide 1.105 Revision 3.

4.0 CONCLUSIONS TO SAFETY EVALUATION

On the basis of its review of the licensee's submittal, the NRC staff concludes that the licensee's documented methodology for determining the LTSPs, AVs, NTSPs, AFTs, and ALTs associated with the proposed Improved Technical Specification changes as identified in its license amendment request submittals appropriately addresses the regulatory concerns and requirements identified within 10 CFR 50.36(c)(1)(ii)(A) and, therefore, is found to be acceptable. The staff also concludes that the instrument channel uncertainty allowances established between the Analytical Limits and the LTSPs for the primary reactor trip and ESFAS initiation instrument channels will provide a high confidence that the instrument channel will function appropriately to ensure that safety limits will not be exceeded during anticipated transients, operational occurrences, or design-basis accidents. The NRC staff further concludes that by performing the testing and corrective actions identified within the surveillance requirements identified in the proposed Setpoint Control Program outlined in revised Technical Specification Section 5.5.16, the plant Technical Specifications, rather than plant procedures, will provide the required administrative control for the assessment of instrument channel operability, and that the descriptions of the Surveillance Requirement actions meet the intent of RIS 2006-17.

The NRC staff has not approved the processing of changes to Kewaunee Power Station instrumentation setpoints under 10 CFR 50.59 using an NRC-approved setpoint methodology as provided for and described within Option B of TSTF-493. Therefore, NRC approval using the 10 CFR 50.90 amendment process is still required to change any listed values of the LTSP, AV, NTSP, AFT, and ALT (as applicable) for each function described as being within the Setpoint Control Program scope identified above. However, when evaluating such future changes for Kewaunee, the NRC staff will review and evaluate the adequacy of any future proposed changes to the settings in light of the licensee's implementation of its methodologies as described within this Safety Evaluation Report.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FROM THE ELECTRICAL ENGINEERING BRANCH (EEEB)

PERTAINING TO EMERGENCY POWER FOR PROTECTION SYSTEMS

FOR KEWAUNEE POWER STATION IMPROVED TECHNICAL SPECIFICATION

CONVERSION LICENSE AMENDMENT REQUEST (TAC NO. ME2139)

DOCKET NO. 50-305

1.0 INTRODUCTION

By application dated August 24, 2009, as supplemented by letters dated October 22, 2009 (ADAMS Accession No. ML093070092), April 13, 2010 (ADAMS Accession Nos. ML101060517 and ML101040090), May 12, 2010 (ADAMS Accession No. ML101380399), July 1, 2010 (ADAMS Accession No. ML101890404), July 16, 2010 (ADAMS Accession No. ML102370370), August 18, 2010 (ADAMS Accession No. ML102371064), Dominion Energy Kewaunee, Inc. requested an amendment to Operating License Number DPR-43 in the form of changes to the Custom Technical Specifications (CTS) for Kewaunee Power Station (KPS) in accordance with Part 50 Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR). The proposed changes would convert the existing KPS TSs to the Improved Standard TS (ISTS). The Electrical Engineering Branch (EEEB) reviewed the conversion of the direct current (DC) power sources portion of the ISTS conversion request.

Existing KPS CTS Limiting Condition of Operation (LCO) 3.7, “Auxiliary Electrical Systems,” and Surveillance Requirements (SRs) (CTS 4.6, “Periodic Testing of Emergency Power System”) address operability of the electrical power system. In converting to the ISTS, existing CTSs 3.7 and 4.6 will be replaced with new TSs 3.8.4, 3.8.5, 3.8.6, and 5.5.15.

In arriving at its conclusion concerning the emergency power for protection systems, the evaluation took into consideration the use by KPS of batteries of higher capacity than required.

2.0 REGULATORY EVALUATION

The following U.S. Nuclear Regulatory Commission (NRC) requirements and guidance document are applicable to the NRC staff’s review of the licensee amendment request:

- KPS Criterion 24, “Emergency Power for Protection Systems,” requires that in the event of loss of all off-site power, sufficient alternate sources of power shall

be provided to permit the required functioning of the protection systems. KPS is supplied with normal, reserve and emergency power to provide for the required functioning of the protection systems. In the event of a reactor and turbine trip, two diesel generators supply emergency power. Any one diesel generator is capable of supplying the emergency power requirements of the plant. The Instrumentation and Controls portions of the protection systems are supplied from the 125 Volt station batteries during the diesel startup period.

- KPS Criterion 39, "Emergency Power for Engineered Safety Features," requires that alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the Engineered Safety Features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

The Commission's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, "Technical specifications." The regulation of 10 CFR 50.36 requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The regulation requires, in part, that the TSs include items in the following categories: (1) Safety limits, limiting safety systems settings, and limiting control settings; (2) Limiting conditions for operation; (3) Surveillance requirements; (4) Design features; and (5) Administrative controls.

- Section 50.36(c)(2)(ii), specifies four criteria to be used in determining whether a TS LCO needs to be established for a particular item. These criteria are summarized as follows:

  (A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

  (B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

  (C) Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

  (D) Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

- Section 50.36(c)(3), states that "[s]urveillance 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."

- Section 50.63(a)(1), Loss of all alternating current [AC] power, states that "Each light-water-cooled nuclear power plant licensed to operate under subpart C of 10 CFR part 52 after the Commission makes a finding under §52.103(g) of this chapter, and each design for a light-water-cooled nuclear power plant approved under a standard design approval, standard design certification, and manufacturing license under part 52 of this chapter must be able to withstand for a specified duration and recover from a station blackout [SBO] as defined in [Section] 50.2." An SBO consists of the loss of all AC power to the essential and nonessential switchgear buses at a nuclear power plant. This involves the simultaneous loss of offsite power (LOOP), turbine trip, and the loss of the onsite emergency power supplies (typically emergency diesel generators). Nuclear power plants are designed to cope with a LOOP event through the use of onsite power supplies (e.g., an alternate AC source or the nuclear power plant's safety-related batteries).

- Section 50.65(a)(3), Requirements for monitoring the effectiveness of maintenance at nuclear power plants, states in part that:

  Performance and condition monitoring activities and associated goals and preventive maintenance activities shall be evaluated at least every refueling cycle provided the interval between evaluations does not exceed 24 months. …Adjustments shall be made where necessary to ensure that the objective of preventing failures of structures, systems, and components through maintenance is appropriately balanced against the objective of minimizing unavailability of structures, systems, and components due to monitoring or preventive maintenance.

- Regulatory Guide (RG) 1.129, Revision 2, "Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants," provides guidance for meeting the intent of General Design Criteria 1, 17, and 18 with respect to the maintenance, testing, and replacement of vented lead-acid storage batteries in nuclear power plants. This RG endorses, in part, the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 450-2002, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications."

3.0 TECHNICAL EVALUATION

3.1 Design Features of the Class 1E DC Power System

The station Class 1E DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety-related equipment and preferred AC vital bus power (via DC to AC power converters (i.e., inverters)).

Each battery charger is normally in the float-charge mode. Float-charge is the condition in which the charger is supplying the connected loads and the battery cells are receiving adequate

current to charge the battery optimally. This assures the internal losses of a battery are overcome and the battery is maintained in a fully charged state.

3.2 Evaluation of Proposed Changes

Currently, CTSs 3.7 and 4.6 contains the following requirements for the DC systems:

CTS 3.7 - Auxiliary Electrical Systems

a. The reactor shall not be made critical unless all of the following requirements are satisfied:

6. Both station batteries and both DC systems are OPERABLE, except during testing and surveillance as described in TS 4.6.b.

b. During power operation or recovery from inadvertent trip, any of the following conditions of inoperability may exist during the time intervals specified. If OPERABILITY is not restored within the time specified, then within 1 hour action shall be initiated to achieve HOT STANDBY within the next 6 hours.

3. One battery may be inoperable for a period not exceeding 24 hours provided the other battery and two battery chargers remain OPERABLE with one charger carrying the DC supply system.

Basis

The plant safeguards 125-V [Volt] DC power is normally supplied by two batteries each of which will have a battery charger in service to maintain full charge and to assure adequate power for starting the diesel generators and supplying other emergency loads. A third charger is available to supply either battery.

CTS 4.6 - Periodic Testing of Emergency Power System

The following tests and surveillance shall be performed:

b. Station Batteries

1. The voltage of each cell shall be measured to the nearest hundredth volt each month. An equalizing charge shall be applied if the lowest cell in the battery falls < 2.13 volts. The temperature and specific gravity of a pilot cell in each battery shall be measured.

2. The following additional measurements shall be made quarterly: the specific gravity and height of electrolyte in every cell and the temperature of every fifth cell.

3. All measurements shall be recorded and compared with previous data to detect signs of deterioration.

4. The batteries shall be subjected to a load test during the first REFUELING and once every 5 years thereafter. Battery voltage shall be monitored as a function of time to establish that the battery performs as expected during heavy discharge and that all electrical connections are tight.

Basis

Station Batteries, TS 4.6.b Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide indication of a cell becoming unserviceable long before it fails.

If a battery cell has deteriorated, or if a connection is loose, the voltage under load will drop excessively, indicating need for replacement or maintenance.

The licensee has proposed revising CTSs 3.7 and 4.6. CTS 3.7.a.6, in part, requires the batteries to be OPERABLE. CTS 3.7.b.3 allows one battery to be inoperable for up to 24 hours before a reactor shutdown is required. CTS 4.6.b provides SRs for various battery parameters. Thus, the battery parameter requirements are covered by CTS 3.7.a.6. ISTS 3.8.6 requires the battery parameters to be within limits. The requirements for the batteries are included in ISTS 3.8.4 and ISTS 3.8.5. This changes the KPS CTS by dividing the requirements for the batteries and the requirements for battery parameters into separate specifications.

3.2.1.1 TS LCO 3.8.4 Change (1) CONDITIONS A.1, A.2, A.3

The proposed change would add new Condition A to address the condition in which one required battery charger is inoperable. This change would establish a 72-hour Completion Time (CT) for an inoperable battery charger provided that battery terminal voltage is restored to greater than or equal to the minimum established float voltage within 2 hours, and battery float current is verified to be less than or equal to 2 amps once per 12 hours.

Evaluation of TS 3.8.4 Change (1)

New Condition A would apply when one required battery charger is inoperable (e.g., the voltage limit of SR 3.8.4.1 is not maintained). There are three associated Required Actions for new Condition A. The Required Actions provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to operable status in a specific time period. Required Action A.1 requires the battery terminal voltage to be restored to greater than or equal to the minimum established float voltage within 2 hours. Required Action A.2 requires the battery float current to be verified to be ≤ 2 amps once per 12 hours. Required Action A.3 requires the battery charger to be restored to operable status within 72 hours.

New Required Action A.1 provides assurance that a battery discharge is terminated by requiring that the battery terminal voltage be restored to greater than or equal to the minimum established

float voltage within 2 hours. The battery charger, in addition to maintaining the battery operable, provides DC control power to AC circuit breakers and thus supports the recovery of AC power following events such as loss of offsite power or SBO. The 2-hour CT provides an allowance for returning an inoperable charger to operable status or for re-establishing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. This provides assurance that the battery will be restored to its fully charged condition from any discharge that might have occurred due to the battery charger being inoperable. At the end of the 2 hours, a terminal voltage of at least the minimum established float voltage provides indication that the battery is on the exponential charging current portion of its recharging cycle.

In response to a staff request for additional information (RAI), the licensee provided a license condition to incorporate the minimum established float voltage limit in the KPS Updated Safety Analysis Report (USAR). This provides reasonable assurance that the minimum established float voltage value will be appropriately maintained by the licensee to accurately reflect the design of the plant.

New Required Action A.2 would require that once per 12 hours, the battery float current be verified to be less than or equal to 2 amps. This would confirm that if the battery has been discharged as the result of an inoperable battery charger, it had been fully recharged. If at the expiration of the 12-hour period, the battery float current is greater than 2 amps, then the battery is considered inoperable (see Section 3.2.3.1 of this SE for a more detailed discussion on the 2-amp float current value). This verification provides assurance that the battery has sufficient capacity to perform its safety function.

New Required Action A.3 requires restoring the inoperable battery charger to operable status within 72 hours. Based on the presumption that (1) the DC bus remains energized; (2) the battery discharge is terminated based on restoration of the battery terminal voltage (New Required Action A.1); and (3) the battery is fully recharged based upon battery float current (New Required Action A.2), the staff finds that the licensee has established reasonable assurance to support a 72-hour restoration time for an inoperable battery charger (New Required Action A.3).

The NRC staff approval of the 72-hour CT for an inoperable battery charger is principally based on the availability of a spare battery charger that is appropriately sized to perform the design function of the battery charger being replaced (i.e., alternate method).

In response to a staff RAI, the licensee submitted a license condition to incorporate the minimum requirements for the alternate means (i.e., spare battery charger) that will be used to obtain the extended battery charger CT in the KPS USAR. This provides reasonable assurance that the requirements will be appropriately maintained by the licensee to accurately reflect the design of the plant.

Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements that when an LCO is not met, the licensee shall follow any remedial action permitted by the TS until the condition can be met.

3.2.1.2 TS LCO 3.8.4 Change (2) CONDITION B

The proposed change would add new Condition B to address the condition where one DC electrical power subsystem is inoperable for reasons other than Condition A. New Required Action B.1 would require restoration of the DC electrical power subsystem to OPERABLE status within 2 hours.

Evaluation of TS 3.8.4 Change (2)

New condition B is similar to existing CTS 3.7.b.3 with the exception that it extends the CT for restoring the DC electrical power subsystem to OPERABLE status to 2 hours versus the 1-hour CT in existing CTS 3.7.b.3. The NRC staff finds that the proposed 2-hour CT provides reasonable assurance that safety will continue to be maintained given the Condition, and therefore, is acceptable.

Furthermore, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements when an LCO is not met, the licensee shall follow any remedial action permitted by the TS until the condition can be met.

3.2.1.3 TS LCO 3.8.4 Change (3) CONDITION C

The proposed change would add new Condition C to address the condition where the Required Action and associated CT is not met. New Required Action C.1 would require KPS to be in MODE 3 within 6 hours and MODE 5 in 36 hours.

Evaluation of TS 3.8.4 Change (3)

New Condition C is similar to existing CTS 3.7.b with the exception that it adds a requirement to be in MODE 3 within 6 hours and in MODE 5 within 36 hours. The NRC staff considers this change to be more conservative than the existing KPS CTS, and therefore, is acceptable.

3.2.1.4 TS SR 3.8.4 Change (4) SR 3.8.4.1

The proposed change would add new SR 3.8.4.1 that would require the licensee to verify battery terminal voltage is greater than or equal to the minimum established float voltage. The Frequency would be 7 days. The value for the minimum established float voltage would be removed from the TSs and controlled by new TS 5.5.15, “Battery Monitoring and Maintenance Program.”

Evaluation of TS 3.8.4 Change (4)

The purpose of SR 3.8.4.1 is to verify battery terminal voltage while the system is on a float charge to ensure the effectiveness of the battery chargers is not degraded. The battery float voltage selected by the battery manufacturer is the minimum voltage which ensures an optimum charging voltage is applied to the battery. The minimum established float voltage will maintain the battery plates in a condition that supports optimizing battery grid life. Maintaining this voltage limit ensures that the battery will be capable of providing its designed safety function. Furthermore, new TS 5.5.15 will require battery pilot cells to be selected from those that

represent the lowest voltage cells in the battery. This ensures that the other cells will be above the pilot cell voltage and above the TS limit. With all battery cells above 2.07 V (required by new TS 3.8.6.2), there is adequate assurance that that the minimum battery terminal voltage is at an acceptable threshold for establishing battery operability. The surveillance frequency is consistent with the recommendations provided in IEEE Std. 450-2002.

In response to a staff RAI, the licensee submitted a license condition to incorporate the minimum established float voltage limit in the KPS USAR. Incorporating the minimum established float voltage limit into the plant's USAR provides reasonable assurance that the value will be appropriately maintained by the licensee to accurately reflect the design of the plant.

The NRC staff finds this change to be conservative because it is more stringent than the requirements contained in the KPS CTS. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements for surveillances by ensuring that the necessary quality of systems and components is maintained and that the LCOs will be met.

3.2.1.5 TS SR 3.8.4 Change (5) SR 3.8.4.2

The proposed changes would add new SR 3.8.4.2 that will require the licensee to verify that each battery charger supplies greater than or equal to 150 amps at greater than or equal to the minimum established float voltage for greater than or equal to 8 hours. The proposed change also adds an alternate criterion to new SR 3.8.4.2.

Evaluation of TS 3.8.4 Change (5)

SR 3.8.4.2 specifies battery charger current requirements for each DC source, and its purpose is to verify the design capacity of each battery charger. The proposed change would require the licensee to verify that that battery charger supplies 150 amps at greater than or equal to the minimum established float voltage for greater than or equal to 8 hours. The voltage requirements are based on the battery charger voltage level after a response to a loss of AC power. Battery manufacturers establish this voltage limit to provide the optimum charge on the battery and to maintain the battery plates in a condition that supports maintaining the battery grid life. Maintaining this voltage limit should ensure that the battery will be capable of providing its designed safety function. Furthermore, new TS 5.5.17 will require battery pilot cells to be selected to represent the lowest voltage cells in the battery. This ensures that the other cells are above the pilot cell voltage, which must remain above the TS limit. With all battery cells above 2.07 V (required by new TS 3.8.6.2), there is adequate assurance that that the terminal voltage is satisfactory.

The licensee also proposed adding an alternative criterion to SR 3.8.4.2, which states, "Verify each required battery charger can recharge the battery to the fully charged state within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state." This is an alternate method for verifying the design capacity of each battery charger because normal battery loads may not be available following the battery service test and may need to be supplemented with additional loads. The duration of this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage, temperature, and the

exponential decay in charging current. If each battery charger is capable of recharging its respective battery within 24 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the design minimum discharge state, the proposed alternate testing criteria would satisfy the purpose of SR 3.8.4.2.

The NRC staff finds this change to be conservative because it is more stringent than the requirements contained in the KPS CTS. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements for surveillances by ensuring that the necessary quality of systems and components is maintained and that the LCOs will be met.

3.2.1.6 TS SR 3.8.4 Change (6) SR 3.8.4.3

The proposed changes would add new SR 3.8.4.3 which would require the licensee to verify, every 18 months, that the battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.

The SR would also include the following two notes:

1. The modified performance discharge test in SR 3.8.6.6 may be performed in lieu of SR 3.8.4.3.

2. This surveillance shall not normally be performed in MODE 1, 2, 3, or 4. Credit may be taken for unplanned events that satisfy this SR.

Evaluation of TS 3.8.4 Change (6)

According to KPS USAR Section 8.2.3.4, although the batteries are sized for an 8-hour period, they are only required to carry the loads for the 4-hour SBO duration. The NRC staff clarifies that the KPS batteries are also required for other design-basis accidents (DBAs) described in the KPS USAR. In response to a staff RAI, the licensee stated that the manufacturer's 8-hour rating for each of the KPS safeguards batteries is 1770 ampere-hours (AH); a continuous rating of 221 amps for 8 hours, and a maximum momentary rating of 1557 amps for 1 minute. The prescribed 8-hour surveillance test load profile bounds the actual (i.e., calculated) load profile for the 4-hour SBO duty cycle (this test is then continued at a constant load of 221 amps for a total of approximately 8 hours, or a minimum established battery terminal voltage, as a demonstration of as-designed battery capacity). The total 4-hour surveillance test load draws approximately 890 AH; with a maximum momentary load of approximately 360 amps for the 1st minute, followed by a continuous load of 221 amps for the remaining 3 hrs 59 min.

Thus, when these batteries are fully charged, based on the manufacturer's design rating (1770 AH, the batteries are nominally 199 percent of the capacity necessary to satisfy the 4-hour surveillance test load (~890 AH) which bounds the 4-hour SBO duty cycle. The licensee also provided graphical charts to show that the modified performance discharge test completely encompassed the load profile of the battery service discharge test. During its review of the graphical charts, the staff identified a minor discrepancy in the load profile during the first minute loading/discharge. Specifically, the NRC staff identified a difference of 0.66 A (i.e., 274.01 – 273.35). In response to a staff RAI, the licensee stated that this error was determined to be a

procedure discrepancy. A change to the controlling calculation revised the battery's bounding load profile for the 1st minute from 273.35 A to 274.01 A, an increase of 0.66 A. However, a follow on revision to the surveillance procedure, SP-38-102B, Station Battery BRB101 Load Test Electrical Maintenance was inadequate.

As a result of this finding, the licensee generated a condition report (CR) to identify that the surveillance procedure, SP-038-102B, did not bound the calculation-based load profile for the 1st minute of the test. The licensee's engineering department evaluated the surveillance data from October 2009 and determined that battery voltage at the end of the 1st minute, the time period in question, was above the minimum required voltage, and thus acceptable. This CR specifically noted the need for a revision to the surveillance procedure, SP-38-102B to correct the required actual battery test load profile. The Corrective Action Program, via the Condition Report, will drive the necessary procedure revisions. The 18-month frequency is consistent with the guidance contained in RG 1.32.

Based on the above, the NRC staff finds that the modified performance discharge test completely encompasses the load profile of the battery service discharge test. The staff finds this change to be conservative because it is more stringent than the requirements contained in the KPS CTS. The staff also finds that successful performance of this SR would adequately verify the capability of the KPS batteries to perform their intended design and safety function. Therefore, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements for surveillances by ensuring that the necessary quality of systems and components is maintained and that the LCOs will be met.

3.2.2 TS 3.8.5 (DC Sources - Shutdown) Changes

New TS 3.8.5 would require one DC electrical power subsystem to be OPERABLE to support one subsystem of the DC Electrical Power Distribution System required by LCO 3.8.10, "Distribution System – Shutdown," in MODE 5 and 6 and during movement of irradiated fuel assemblies.

3.2.2.1 <u>TS LCO 3.8.5 Change (1) CONDITION A</u>

The proposed change would add new Condition A to state "One required DC electrical power subsystem inoperable," with the following Required Actions that must be implemented immediately:

> New Required Action A.1 would require suspension of movement of irradiated fuel assemblies.
>
> New Required Action A.2 would require suspension of operations involving positive reactivity additions that could result in loss of required shutdown margin (SDM) or boron concentration.
>
> New Required Action A.3 would require the licensee to initiate action to restore the required DC electrical power subsystem to OPERABLE status.

Evaluation of TS 3.8.5 Change (1)

The new LCO will require one DC electrical power subsystem consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling within the subsystem to be OPERABLE by LCO 3.8.10. The allowance for sufficiently conservative actions is made (i.e., to suspend movement of irradiated fuel assemblies and operations involving positive reactivity additions) that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the reactor coolant system for minimum SDM or refueling boron concentration. This may result in an overall reduction in reactor coolant system boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive moderator temperature coefficient must also be evaluated to ensure they do not result in a loss of required SDM. Suspension of these activities must not preclude completion of actions to establish a safe conservative condition.

The NRC staff finds this change to be conservative because the proposed LCO will be applicable under more conditions than in the KPS CTS. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements that when an LCO is not met, the licensee shall follow any remedial action permitted by the TS until the condition can be met.

3.2.2.2 TS SR 3.8.5 Change (2) SR 3.8.5.1

The proposed change would add SR 3.8.5.1 to reflect changes previously described in Section 3.2.1 of this safety evaluation report.

Evaluation of TS 3.8.5 Change (2)

The licensee proposed adding SR 3.8.5.1 to be consistent with the proposed addition of TS 3.8.4. New SR 3.8.5.1 would require the licensee to perform all surveillances required by SR 3.8.4.1 through SR 3.8.4.3. The NRC staff reviewed the proposed changes and has determined that the changes are consistent with the proposed changes to TS 3.8.4 and meet the intent of the SR.

The NRC staff finds this change to be conservative because it is more stringent than the requirements contained in the KPS CTS. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements for surveillances by ensuring that the necessary quality of systems and components is maintained and that the LCOs will be met.

3.2.3 TS 3.8.6 (Battery Parameters) Changes

New TS 3.8.6 would require battery parameters for Train A and B batteries to be within limits when associated DC electrical power subsystems are required to be OPERABLE.

3.2.3.1 TS LCO 3.8.6 Change (1)

CTS 3.7.a.6 requires the station batteries and the associated battery parameters tested as specified in CTS 4.6.b, to be within limits prior to making the reactor critical. The ISTS LCO 3.8.6 requires the Train A and Train B battery parameters to be within limits "when associated DC electrical power subsystems are required to be OPERABLE." ISTS 3.8.4 requires both Train A and Train B batteries to be OPERABLE in MODES 1, 2, 3, and 4 and ITS 3.8.5 requires one of the Train A or B batteries to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies.

Evaluation of TS 3.8.6 Change (1)

The purpose of CTS 3.7.a, in part, is to ensure the station batteries are OPERABLE to mitigate the consequences of a transient or DBA. The station batteries are required to be OPERABLE in MODES 1, 2, 3, and 4 when a DBA (e.g., loss of coolant accident) may occur. In MODE 1 and MODE 2 the reactor is either critical or there is a potential for the reactor to become critical. In MODE 3 and MODE 4 the reactor is not critical, however the reactor coolant temperature is always above 200 degrees Fahrenheit (°F) and there is considerable energy in the reactor core and the station batteries must be available to support equipment necessary to mitigate the consequences of a pipe break. The station batteries are required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies so that the unit can be maintained in the shutdown or refueling condition for extended periods, to ensure sufficient instrumentation and control capability is available for monitoring and maintaining unit status, and to mitigate events postulated during shutdown, such as a fuel handling accident. Therefore, it is necessary to require the battery parameters to be within limits to ensure the batteries are OPERABLE in all MODES of Operation.

The NRC staff finds this change to be conservative because the LCO will be applicable under more conditions than in the KPS CTS. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements that when an LCO is not met, the licensee shall follow any remedial action permitted by the TS until the condition can be met.

3.2.3.2 TS LCO 3.8.6 Change (2) CONDITION A

The proposed change would add new TS LCO 3.8.6, Condition A to address the condition when one battery with one or more battery cells float voltage is less than 2.07 V.

Evaluation of TS 3.8.6 Change (2)

The licensee proposed adding new TS 3.8.6 Condition A to address the condition when one battery with one or more battery cells float voltage is less than 2.07 V. This new Condition would be applicable when a battery is found with one or more battery cells with a float voltage less than 2.07 V. Condition A contains remedial measures (Required Actions) acceptable to the NRC staff for the condition of a degraded battery cell. The Required Actions require the licensee to verify: (a) the battery terminal voltage to be greater than or equal to the minimum established float voltage (SR 3.8.4.1) within 2 hours, and (b) each battery's float current is less than or equal to 2 amps (SR 3.8.6.1) within 2 hours, and (c) restore affected cell voltage to greater than or equal to 2.07 V within 24 hours. The above actions ensure that there is still

sufficient capacity for the battery to perform its intended function. Continued operations for up to 24 hours will allow the restoration of the affected cell(s) voltage to greater than or equal to 2.07 V.

In response to a staff RAI, the licensee submitted a license condition to incorporate the minimum established float voltage limit into the plant's USAR. Incorporating the minimum established float voltage limit into the plant's USAR provides reasonable assurance that the value will be appropriately maintained by the licensee to accurately reflect the design of the plant.

The NRC staff finds this change to be conservative because it is more stringent than the requirements contained in the KPS CTS. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements that when an LCO is not met, the licensee shall follow any remedial action permitted by the TS until the condition can be met.

3.2.3.3 TS LCO 3.8.6 Change (3) CONDITION B

The proposed change would add new TS 3.8.6, Condition B to address battery state of charge.

Evaluation of TS 3.8.6 Change (3)

The licensee proposed adding new TS 3.8.6, Condition B to address battery state of charge. This new Condition would be applicable when a battery is found with a float current greater than 2 amps. A float current of greater than 2 amps provides an indication that a partial discharge has occurred. The Required Action is to verify within 2 hours that the battery terminal voltage is greater than or equal to the minimum established float voltage (SR 3.8.4.1), thus confirming battery charger operability. If the terminal voltage is satisfactory and there are no battery cells with a float voltage less than 2.07 V, Required Action B.2 of Condition B ensures that within 12 hours the battery will be restored to its fully-charged condition from any discharge that might have occurred due to a temporary loss of the battery charger.

If the terminal voltage is found to be less than the minimum established float voltage, it indicates that the battery charger is either inoperable or is operating in the current limit mode. If the battery charger is operating in the current limit mode for 2 hours, it indicates that the battery has been substantially discharged and likely cannot perform its required design functions. In this case, new Condition F would be entered.

If the terminal voltage is found to be satisfactory, but there are one or more battery cells with a float voltage less than 2.07 V and a float current > 2 amps, the associated "OR" statement in the revised Condition F of TS LCO 3.8.6 would be applicable, and the battery must immediately be declared inoperable. If the terminal voltage is satisfactory and there are no cells less than 2.07 V, and the out-of-limit float current condition is due to one or more battery cells with low voltage, the battery is not substantially discharged and the 12-hour CT to restore battery float current to within limits is reasonable.

Currently, there are no specific requirements for monitoring a battery's state of charge at KPS. The licensee proposed adding a requirement to monitor float current to determine battery state of charge. Float current monitoring is recognized by the industry as being a more direct and

expeditious method for determining battery state of charge. The licensee provided a letter from the manufacturer of the batteries used at the KPS verifying the acceptability of using float current monitoring as a reliable and accurate indication of a battery state of charge for the life of the battery. In response to a staff RAI, the licensee noted that for the KPS batteries a 2-amp float current limit provides an indication that the battery is at least 95 percent charged based. Based on this, the licensee provided a license condition to maintain a 5 percent design margin for the batteries. The licensee also committed to add a description in the USAR to describe how the design margin for the batteries corresponds to the float current value indicating that the battery is the required percent charged. This provides assurance that the battery will be fully charged when the 2-amp float current limit is reached.

The NRC staff notes that no changes to the float current limit should be required for replacement batteries of the same size and model number. For replacement batteries of a different model/size and/or manufacturer, float current limit changes would need to be verified as part of design change documentation package for installing replacement batteries.

In response to a staff RAI, the licensee stated that it will use a Fluke meter that has a +/-0.2 amp uncertainty to measure charging current. The NRC staff finds that this uncertainty is acceptable, particularly given the large battery capacity margin available while in the exponential decay charging portion of the recharge cycle. Based on this information, the NRC staff finds that the licensee has provided adequate information to show that the equipment that will be used to monitor float current under SR 3.8.6.1 will have the necessary accuracy and capability to measure electrical currents in the expected range.

The NRC staff finds that the licensee's verifications of the battery manufacturer specifications and the KPS USAR Section 8 description to maintain a 5-percent design margin for the batteries corresponds to a 2-amp float current value which indicates that the battery is 95 percent charged provides adequate assurance that float current monitoring is an accurate method for determining the operability of the batteries. Based on these requirements, the NRC staff finds that float current monitoring is suitable for determining a battery's state of charge. The proposed changes will also ensure the battery parameters (maintenance, testing, and monitoring) are appropriately monitored and maintained in accordance with the new Battery Monitoring and Maintenance Program specified in TS Administrative Controls Section 5.5.

This change is acceptable because the Required Actions establish reasonable remedial measures that must be taken in response to the degraded conditions to allow continued operation while providing time to repair inoperable features. The NRC staff finds that the Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, time for repairs or replacement, and the low probability of a DBA occurring during the repair period. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements that when an LCO is not met, the licensee shall follow any remedial action permitted by the TS until the condition can be met.

3.2.3.4 TS LCO 3.8.6 Change (4) CONDITION C

The proposed change would add new TS 3.8.6, Condition C to address the level of the electrolyte in a cell.

Evaluation of TS 3.8.6 Change (4)

The licensee proposed adding new TS 3.8.6 Condition C to address the electrolyte level in a cell. This new Condition C would be applicable when a battery is found with one or more cells with an electrolyte level less than the minimum established design limits. If the electrolyte level is above the top of the battery plates, but below the minimum limit (i.e., minimum level indication mark on the battery cell jar), the battery should still have sufficient capacity to perform its intended safety function and could be considered operable. With the electrolyte level below the top of the plates, there is a potential for dry-out and plate degradation. New Required Actions C.1 (8-hour CT), C.2 (12-hour CT), and C.3 (31-day CT) (as well as provisions in the new Battery Monitoring and Maintenance Program) restore the electrolyte level, ensure that the cause of the loss of the electrolyte level is not due to a leak in the battery cell jar, and equalize and test the battery cells that have been discovered with an electrolyte level below the top of the plates.

In response to a staff RAI, the licensee provided a license condition to incorporate the minimum established design limits for electrolyte level in the KPS USAR. This provides adequate assurance that electrolyte level will be maintained at the required level to ensure battery operability. This also provides reasonable assurance that the value will be appropriately maintained by the licensee to accurately reflect the design of the plant.

This change is acceptable because the Required Actions establish reasonable remedial measures that must be taken in response to the degraded conditions to allow continued operation while providing time to repair inoperable features. The allowed restoration times provided in Condition C is permitted provided that only one of the two batteries is affected. The NRC staff finds that the Required Actions and associated CTs are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, time for repairs or replacement, and the low probability of a DBA occurring during the repair period. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements that when an LCO is not met, the licensee shall follow any remedial action permitted by the TS until the condition can be met.

3.2.3.5 TS LCO 3.8.6 Change (5) CONDITION D

The proposed change would add new TS 3.8.6, Condition D which would apply to a battery found with a pilot cell electrolyte temperature less than the minimum established design limit.

Evaluation of TS 3.8.6 Change (5)

The licensee proposes adding new TS 3.8.6 Condition D which would apply to a battery found with a pilot cell electrolyte temperature less than the minimum established design limit. A low electrolyte temperature limits the current and power available from the battery. This new

Condition D would be applicable when a battery is found with a pilot cell electrolyte temperature less than the minimum established design limit. The licensee proposed requiring restoration of the battery pilot cell temperature to greater than or equal to the minimum established design limits within 12 hours.

The KPS batteries are sized with correction margins to account for factors affecting performance that include temperature and aging and, as previously discussed, the licensee will be maintaining adequate battery design margins. In response to a staff RAI, the licensee stated that quarterly surveillance testing of the “A” and “B” safeguards batteries includes the measurement and recording of battery cell temperature for about every fifth cell, typically including from 11 to 14 cells for the 59-cell battery. Since the “A” and “B” safeguards batteries were replaced during the 2008 spring refueling outage, battery cell temperature data was reviewed for the first 7 instances of quarterly surveillance testing performed immediately thereafter.

For BRA101 (“A” train safeguards battery), the data indicates a maximum delta temperature, lowest measured cell temperature to highest measured cell temperature during any single quarterly surveillance, of 3.1 °F (71.9 – 75.0 °F). The average of battery cell temperatures for that same surveillance was 74.1 °F.

For BRB101 (“B” train safeguards battery), the data indicates a maximum delta temperature, lowest measured cell temperature to highest measured cell temperature during any single quarterly surveillance, of 4.7 °F (73.3 – 78.0 °F). The average of battery cell temperatures for that same surveillance was 75.1 °F.

Due to the use of 2.07 V as the minimum limit for cell float voltage and the use of pilot cell temperature in lieu of average cell temperature, changes are necessary in the method pilot cells are selected. In the past, pilot cells were selected to represent average cells in the battery. The change to 2.07 V now requires pilot cells to be selected to represent the lowest voltage cells in the battery. This ensures that the other cells are above the pilot cell voltage which must remain above the TS limit. The new Battery Monitoring and Maintenance Program specified in TS Administrative Controls Section 5.5 requires the licensee to select the lowest voltage cell(s) to be the pilot cell(s) in the battery bank. Pilot cell selection will be evaluated after each performance of SR 3.8.6.5, which is currently performed every 92 days, to ensure they continue to meet the selection criteria. Another difference in this approach is rotating pilot cells. The reason for rotation and not reusing cells was to prevent loss of electrolyte by repeated sampling (e.g., specific gravity monitoring). With the transition to float current monitoring, this concern is no longer valid and pilot cells should be selected based on the preceding discussion without regard to whether or not they have been used previously.

Since the maximum temperature deviation across the battery has not historically exceeded the IEEE Std. 450-2002 recommended maximum of 5 °F, cell temperature, the license does not need to take temperature into account when selecting battery pilot cells. Therefore, the licensee is proposing that they have performed the necessary analysis to demonstrate that sufficient margins exist in sizing to compensate for using the warmest cell as the pilot cell.

In response to a staff RAI, the licensee provided a license condition to incorporate the minimum established design limit for electrolyte temperature into the KPS USAR. Incorporating the

minimum established design temperature limit into the plant's USAR provides reasonable assurance that the value will be appropriately maintained by the licensee to accurately reflect the design of the plant.

Based on this information, the NRC staff finds that the pilot cell temperature is a sufficiently accurate representation of the temperature of the battery bank because: (1) the KPS batteries are designed with sufficient margins (i.e., temperature, aging, and design); and (2) the maximum deviation across the KPS batteries has not historically exceeded the IEEE Std. 450-2002 recommended maximum of 5 °F. The proposed 12-hour CT provides adequate time to restore the electrolyte temperature within established limits.

This change is acceptable because the Required Actions establish reasonable remedial measures that must be taken in response to the degraded conditions to allow continued operation while providing time to repair inoperable features. The NRC staff finds that the Required Action and associated CT are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, time for repairs or replacement, and the low probability of a DBA occurring during the repair period. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements that when an LCO is not met, the licensee shall follow any remedial action permitted by the TS until the condition can be met.

3.2.3.6 TS LCO 3.8.6 Change (6) CONDITION E

New TS 3.8.6, Condition E addresses the condition in which two batteries with parameters not within established limits.

Evaluation of TS 3.8.6 Change (6)

The licensee proposed adding new TS 3.8.6 Condition E to address the condition where two batteries with battery parameters not within limits. If this condition exists, there is not sufficient assurance that the batteries will be capable of performing their intended safety function. With two batteries involved, loss of function is possible for multiple systems that depend upon the batteries. The licensee proposed that battery parameters for one affected battery be restored to within limits within 2 hours (Required Action E.1). The NRC staff considers the 2-hour time period to be consistent with similar limiting conditions established by the TSs and reasonable considering the loss of function of components that depend on the batteries while also providing a relatively short duration to resolve the condition.

This change is acceptable because the Required Actions establish reasonable remedial measures that must be taken in response to the degraded conditions to allow continued operation while providing time to repair inoperable features. The NRC staff finds that the Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, time for repairs or replacement, and the low probability of a DBA occurring during the repair period. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements that when an LCO is

not met, the licensee shall follow any remedial action permitted by the TS until the condition can be met.

3.2.3.7 TS LCO 3.8.6 Change (7)

The proposed change would add new Condition F.

Evaluation of TS 3.8.6 Change (7)

New Condition F provides a default condition for battery parameters that fall outside the allowance of the Required Actions for Condition A, B, C, D, or E. Under this condition, it is assumed that there is insufficient capacity to supply the maximum expected load requirements. New Condition F also addresses the case where a battery is found with one or more battery cells having a float voltage less than 2.07 V and a float current greater than 2 amps. The licensee proposed adding a third condition to address SR 3.8.6.6 not being met. Since battery capacity may be insufficient to perform the intended design function, the Required Action is to declare the associated battery inoperable with an immediate Completion Time.

This change is acceptable because the Required Actions establish reasonable remedial measures that must be taken in response to the degraded conditions to allow continued operation while providing time to repair inoperable features. The NRC staff finds that the Required Actions are consistent with safe operation under the specified Condition, considering the OPERABLE status of the redundant systems or features. This includes the capacity and capability of remaining systems or features, time for repairs or replacement, and the low probability of a DBA occurring during the repair period. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements that when an LCO is not met, the licensee shall follow any remedial action permitted by the TS until the condition can be met.

3.2.3.8 TS SR 3.8.6 Change (8)

The proposed change would add new SR 3.8.6.1.

Evaluation of TS 3.8.6 Change (8)

The applicant proposed adding new SR 3.8.6.1, which will require verification that the float current for each battery is less than or equal to 2 amps every 7 days. The purpose of this SR is to determine the state of charge of the battery. Float charge is the condition in which the battery charger is supplying the continuous small amount of current (i.e., less than or equal to 2 amps) required to overcome the internal losses of a battery to maintain the battery in a fully charged state. The float current requirements are based on the float current indicative of a charged battery. As stated above in Section 3.2.3.3 (TS 3.8.6; Change (3)), the use of float current to determine the state of charge of the battery is consistent with the battery manufacturer recommendations.

The NRC staff finds this change to be conservative because it is more stringent than the requirements contained in the KPS CTS. Based on the above, the NRC staff concludes that the

proposed changes meet 10 CFR 50.36 requirements for surveillances by ensuring that the necessary quality of systems and components is maintained and that the LCOs will be met.

3.2.3.9 TS SR 3.8.6 Change (9)

The proposed change would add new SR 3.8.6.2.

Evaluation of TS 3.8.6 Change (9)

The licensee proposed adding new SR 3.8.6.2, which will require verification that the float voltage of pilot cells are greater than or equal to 2.07 V every 31 days. This voltage level represents the point where battery operability is in question. The Battery Monitoring and Maintenance Program in new TS Section 5.5.15 includes actions to restore battery cells with float voltage less than 2.13 V and actions to verify that the remaining cells are greater than or equal to 2.07 V when a cell or cells have been found to be less than 2.13 V.

The NRC staff finds this change is consistent with the KPS CTS. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements for surveillances by ensuring that the necessary quality of systems and components is maintained and that the LCOs will be met.

3.2.3.10 TS SR 3.8.6 Change (10)

The proposed change would add new SR 3.8.6.3, which will require verification that the connected cell electrolyte level of each battery is greater than or equal to the minimum established design limits every 31 days.

Evaluation of TS 3.8.6 Change (10)

The licensee proposed adding SR 3.8.6.3. Operation of the batteries at electrolyte levels greater than the minimum established design limit ensures that the battery plates do not suffer physical damage and continue to maintain adequate electron transfer capability.

The licensee also proposed adding specific limiting values for the battery electrolyte level to the Battery Monitoring and Maintenance Program. SR 3.8.6.3 would require the electrolyte level to be greater than or equal to the “minimum established design limits.” Relocation to the licensee controlled Battery Monitoring and Maintenance Program will allow flexibility to monitor and control this limit at values directly related to the battery ability to perform its required safety function. Incorporating the minimum established design level limit into the plant’s USAR provides reasonable assurance that the value will be appropriately maintained by the licensee to accurately reflect the design of the plant.

The NRC staff finds this change is consistent with the KPS CTS. The NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements for surveillances by ensuring that the necessary quality of systems and components is maintained and that the LCOs will be met.

3.2.3.11 TS SR 3.8.6 Change (11)

The proposed change would add new SR 3.8.6.4, which will require verification that the temperature of each battery pilot cell is greater than or equal to the minimum established design limits every 31 days.

Evaluation of TS 3.8.6 Change (11)

The licensee proposed adding SR 3.8.6.4 which will require verification that the temperature of each battery pilot cell is greater than or equal to the minimum established design limits every 31 days. Batteries have very large thermal inertia; the batteries are designed with margins to account for factors affecting performance (i.e., temperature, aging) (See Section 3.2.3.5 of this SE for additional details). As a result, the pilot cell temperature is an accurate representation of the temperature of the battery bank and is adequate to ensure that the minimum electrolyte temperature is maintained. The surveillance frequency is consistent with the recommendations provided in IEEE Std. 450-2002.

The licensee also proposed adding specific limiting values for the battery electrolyte temperature to the Battery Monitoring and Maintenance Program. SR 3.8.6.4 would require the electrolyte temperature to be greater than or equal to the "minimum established design limits." Depending on the available excess capacity of the associated incorporating the minimum established design temperature limit into the plant's USAR provides reasonable assurance that the value will be appropriately maintained by the licensee to accurately reflect the design of the plant safety related batteries the minimum temperature necessary to support operability of the battery can vary. Relocation to the Battery Monitoring and Maintenance Program will allow flexibility to monitor and control this limit at values directly related to the battery ability to perform its intended function.

Based on the above, the NRC staff finds that the pilot cell temperature is an accurate representation of the temperature of the battery bank and is consistent with the KPS CTS. The NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements for surveillances by ensuring that the necessary quality of systems and components is maintained and that the LCOs will be met.

3.2.3.12 TS SR 3.8.6 Change (12)

The proposed change would add new SR 3.8.6.5 which will require verification that float voltage of each connected cell is greater than or equal to 2.07 V every 92 days.

Evaluation of TS 3.8.6 Change (12)

The licensee proposed adding new SR 3.8.6.5. This voltage level represents the point at which battery operability cannot be assured. Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limit provided by the battery manufacturer. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less but greater than or equal to 2.07 V per cell, are addressed in the Battery Monitoring and Maintenance Program (TS Section 5.5.15).

Furthermore, the Battery Monitoring and Maintenance Program includes actions to restore battery cells with float voltage less than 2.13 V and actions to verify that the remaining cells are greater than or equal to 2.07 V when a cell or cells have been found to be less than 2.13 V. The 2.07 V individual cell limit reflects the Operability limit for the batteries. With each battery cell greater than or equal to 2.07 V, there is adequate assurance that that the battery terminal voltage is at an acceptable threshold for establishing battery operability. The surveillance frequency is consistent with the recommendations provided in IEEE Std. 450-2002.

The NRC staff finds this change is consistent with the KPS CTS. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements for surveillances by ensuring that the necessary quality of systems and components is maintained and that the LCOs will be met.

3.2.3.13 TS SR 3.8.6 Change (13)

The proposed change would add new SR 3.8.6.6, which will require verification that battery capacity is greater than or equal to 80 percent of the manufacturer’s rating when subjected to a performance discharge test or a modified performance discharge test.

Evaluation of TS 3.8.6 Change (13)

The licensee proposed adding new SR 3.8.6.6, which will require verification that battery capacity is greater than or equal to 80 percent of the manufacturer’s rating when subjected to a performance discharge test or a modified performance discharge test. This SR will contain a Note that would prohibit this surveillance from being performed in MODE 1, 2, 3, or 4 and allow credit to be taken for unplanned events that satisfy the SR. The frequency for performing this SR is normally 60 months. The frequency would increase to 24 months when a battery reaches 85 percent of the expected life with capacity greater than or equal to 100 percent of the manufacturer's rating. The frequency would further increase to 12 months when a battery shows degradation, or reaches 85 percent of the expected life with capacity less than 100 percent of the manufacturer's rating. This test frequency is consistent with IEEE Std. 450-2002.

In response to a staff RAI, the licensee has committed to incorporate the manufacturer’s expected/qualified life of the safety related batteries in the KPS USAR and the TS Bases. Incorporating the expected/qualified life of the safety related batteries in the KPS USAR and the TS Bases provides assurance that the surveillance will be performed at the expected frequency given the condition of the battery.

The NRC staff finds this change to be conservative because it is more stringent than the requirements contained in the KPS CTS. Based on the above, the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements for surveillances by ensuring that the necessary quality of systems and components is maintained and that the limiting conditions for operation will be met.

3.2.4 TS 5.5.15, Battery Monitoring and Maintenance Program

The proposed change would add a new Battery Monitoring and Maintenance Program to TS Section 5.5.

Evaluation of TS 5.5.15 Change

The licensee proposed adding a new administrative program titled "Battery Monitoring and Maintenance Program" to Administrative TS Section 5.5, to read as follows:

> Battery Monitoring and Maintenance Program
>
> This program provides controls for battery restoration and maintenance. The program shall be in accordance with the IEEE Std 450-2002, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," as endorsed by RG 1.129, Revision 2, with RG exceptions and program provisions as identified below:
>
> a. The program allows the following RG 1.129, Revision 2 exceptions:
>
> 1. Battery temperature correction may be performed before or after conducting discharge tests.
>
> 2. RG 1.129, Regulatory Position 1, Subsection 2, "References," is not applicable to this program.
>
> 3. In lieu of RG 1.129, Regulatory Position 2, Subsection 5.2, "Inspections," the following shall be used: "Where reference is made to the pilot cell, pilot cell selection shall be based on the lowest voltage cell in the battery after each performance of SR 3.8.6.5."
>
> 4. In RG 1.129, Regulatory Position 3, Subsection 5.4.1, "State of Charge Indicator," the following statements in Paragraph (d) may be omitted: "When it has been recorded that the charging current has stabilized at the charging voltage for three consecutive hourly measurements, the battery is near full charge. These measurements shall be made after the initially high charging current decreases sharply and the battery voltage rises to approach the charger output voltage."
>
> 5. In lieu of RG 1.129, Regulatory Position 7, Subsection 7.6, "Restoration," the following may be used: "Following the test, record the float voltage of each cell of the string."

b. The program shall include the following provisions:

1. Actions to restore battery cells with float voltage < 2.13 V;

2. Actions to determine whether the float voltage of the remaining battery cells is ≥ 2.13 V when the float voltage of a battery cell has been found to be < 2.13 V;

3. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates;

4. Limits on average electrolyte temperature, battery connection resistance, and battery terminal voltage; and

5. A requirement to obtain specific gravity readings of all cells at each discharge test, consistent with manufacturer recommendations.

The TS Battery Maintenance and Monitoring Program will ensure that battery parameters will be maintained and that actions will be implemented should the battery parameter(s) not be met.

The TS 5.5.15 Battery Monitoring and Maintenance Program provides assurance that the battery parameters will be monitored and controlled in accordance with the program, and that actions to restore deficient parameters will be implemented in accordance with the licensee's corrective action program. Furthermore, the battery and its preventive maintenance and monitoring program continue to be subject to the regulatory requirements of 10 CFR 50.65.

The NRC staff finds that this change provides assurance that the battery is maintained at required levels of performance, battery parameters are monitored, and that the limiting conditions for operation will continue to be met. Based on the above the NRC staff concludes that the proposed changes meet 10 CFR 50.36 requirements for surveillances by ensuring that the necessary quality of systems and components is maintained and that the limiting conditions for operation will be met.

4.0 REGULATORY COMMITMENTS

The following eight changes to the KPS USAR and its affected change pages are to be active and available to its users as part of the implementation of this ITS conversion and incorporated into the next scheduled update of the USAR. The NRC accepts and relies upon the dentition of "active" that applies to these changes to the KPS USAR. Namely, when used in the context of USAR revisions, an "active" USAR change/revision is one that has been through all of the requisite reviews and approvals, and is, from the time designated as "active" forward, considered as part of the then current USAR in effect. Further, within the electronic document management system, "active" is used to indicate that the changed or revised document is the current document of record.

a. DEK commits to incorporating the expected/qualified life of the safety related batteries in the battery monitoring program in the USAR.

b. The licensee will maintain a 5 percent design margin for the batteries to use float current monitoring as a state of charge indicator. The licensee will also add a description in the KPS USAR to describe how the design margin for the batteries corresponds to allow use of a 2-amp float current value as an indication that the battery is at least 95 percent charged.

c. The licensee will incorporate the minimum established float voltage limit in the KPS USAR.

d. The licensee will incorporate the minimum established design limits for electrolyte level in the KPS USAR.

e. The licensee will incorporate the minimum established design limit for electrolyte temperature into the KPS USAR.

f. The licensee will incorporate the minimum requirements for the alternate means (i.e., spare battery charger) that will be used to obtain the extended battery charger CT in the KPS USAR.

g. DEK commits to change (or verify) procedure(s) to include the measuring and test equipment manufacturer's recommended practice for instrument stabilization when measuring float current to confirm battery state of charge.

h. DEK commits to including vital switchgear and battery room temperature limits in the USAR.

5.0 CONCLUSION TO SAFETY EVALUATION

Based on the above evaluation, the NRC staff finds the proposed changes to the KPS TS provide assurance of the continued availability of the required DC power to shut down the reactor and to maintain the reactor in a safe condition after an anticipated operational occurrence or a postulated design-basis accident. The NRC staff also concludes that the proposed TS changes are in accordance with 10 CFR 50.36, 50.63, and 50.65, and meet the intent of KPS Criteria 24 and 39. Therefore, the NRC staff finds the proposed changes acceptable.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FROM THE ACCIDENT DOSE BRANCH (AADB)

PERTAINING TO ADOPTION OF TS 3.4.16, RCS SPECIFIC ACTIVITY, AND TSTF-490,

REVISION 0, REGARDING DELETION OF E BAR DEFINITION AND REVISION TO RCS

SPECIFIC ACTIVITY

FOR KEWAUNEE POWER STATION IMPROVED TECHNICAL SPECIFICATION

CONVERSION LICENSE AMENDMENT REQUEST (TAC NO. ME2139)

DOCKET NO. 50-305

1.0 INTRODUCTION

By application dated August 24, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092440398), as supplemented by letter dated April 13, 2010 (ADAMS Accession No. ML101060517), Dominion Energy Kewaunee, Inc. proposed to revise the technical specifications (TS) for Kewaunee Power Station (KPS, Kewaunee, or the licensee), Facility Operating License DPR-43. The proposed revision is the implementation of the Improved Technical Specifications (ITS) consistent with the Improved Standard Technical Specifications (ISTS) described in NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," Revision 3.0. In addition, the revision includes the adoption of Technical Specifications Task Force (TSTF)-490, Revision 0, "Deletion of E Bar Definition and Revision to RCS Specific Activity Technical Specification" for pressurized water reactor Standard Technical Specifications (STS).

Regulatory Issue Summary 2000-06, "Consolidated Line Item Improvement Process for Adopting Standard Technical Specifications Changes for Power Reactors," was issued on March 20, 2000 (ADAMS Accession No. ML003693442) to inform the addressees of the opportunity to participate as applicants in the consolidated line item improvement process (CLIIP) for TS amendments. The CLIIP facilitates licensees' adopting of U. S. Nuclear Regulatory Commission (NRC)-accepted changes to the STS for their specific plant TS. Each amendment application made as part of the CLIIP is to be processed and noticed in accordance with applicable rules and NRC procedures. A licensee's participation in the CLIIP is purely voluntary.

The STS for the five vendor designs include Babcock & Wilcox (NUREG-1430), Westinghouse (NUREG-1431), Combustion Engineering (NUREG-1432), General Electric Boiling Water Reactor/4 (BWR/4) (NUREG-1433), and General Electric BWR/6 (NUREG-1434). Changes to the STS NUREGs, which are potentially applicable to multiple plants, were proposed to the NRC by the Nuclear Energy Institute (NEI) sponsored TSTF through publicly available submittals. By letter dated September 13, 2005 (ADAMS Accession No. ML052630462), the TSTF submitted an application for NRC staff approval proposing the replacement of the current PWR TS 3.4.16 limit on reactor coolant system (RCS) gross specific activity with a new limit on RCS noble gas specific activity. The noble gas specific activity limit would be based on a new dose equivalent Xe-133 (DEX) definition that replaces the current E Bar ($\bar{E}$) average disintegration energy definition. In addition, the current dose equivalent I-131 (DEI) definition would be revised to allow the use of additional thyroid dose conversion factors (DCFs). The TSTF proposed these changes for incorporation into the STS as TSTF-490, Revision 0, which was referenced in the *Federal Register* Notice (FRN) 71 FR 67170, of November 20, 2006.

2.0 REGULATORY EVALUATION

The NRC staff issued a FRN (71 FR 67170, November 20, 2006) that requested public comment on the NRC's pending action to delete the E Bar definition and revise the RCS specific activity technical specification. In particular, following an assessment and draft safety evaluation by the NRC staff, the staff sought public comment on proposed changes to the STS, designated TSTF–490 Revision 0. *Federal Register* Notice 72 FR 12217 (March 15, 2007) gave notice that the NRC staff had revised the previous model license amendment request (LAR), model safety evaluation (SE), and model proposed no significant hazards consideration determination related to deletion of the $\bar{E}$ definition and revision to RCS specific activity technical specification. The request would revise the RCS specific activity specification for pressurized water reactors to utilize a new indicator, Dose Equivalent Xenon-133 instead of the current indicator known as $\bar{E}$.

As part of the regulatory standardization effort, the NRC staff prepared STS for each of the light–water reactor nuclear steam supply systems and associated balance–of–plant equipment systems. The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 FR 39132), which was subsequently codified by changes to Section 36 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) (60 FR 36953). NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," Revision 3.0, was developed under the NRC TS Improvement Program and issued in September 1992, completely replacing the previous STS. Licensees were encouraged to upgrade their technical specifications consistent with the criteria and conforming, to the extent practical, to Revision 3.0 to the improved STS.

The proposed TS conversion and applicable TSTF-490 implementation was reviewed to determine whether content and format were consistent with the applicable reference ITS 3.4.16, "RCS Specific Activity." Special attention was given to proposed TS provisions that departed from the reference TS to determine whether proposed differences are justified by uniqueness in plant design or other considerations. The NRC staff compared the proposed TS to the reference TS for whether or not the application states adequate technical justification for each departure from the reference TS. In addition, the NRC staff evaluated the impact of the

proposed changes as they relate to the radiological consequences of affected design-basis accidents (DBAs) that use the RCS inventory as the source term. The source term assumed in radiological analyses should be based on the activity associated with the projected fuel damage or the maximum RCS TS values, whichever maximizes the radiological consequences. The limits on RCS specific activity ensure that the offsite doses are appropriately limited for accidents that are based on releases from the RCS with no significant amount of fuel damage.

The Steam Generator Tube Rupture (SGTR) accident and the Main Steam Line Break (MSLB) accident typically do not result in fuel damage and therefore the radiological consequence analyses are based on the release of primary coolant activity at maximum TS limits. For accidents that result in fuel damage, the additional dose contribution from the initial activity in the RCS is not normally evaluated and it is considered to be insignificant in relation to the dose resulting from the release of fission products from the damaged fuel.

By letter dated March 17, 2003 (ML030210062), the Commission issued Amendment No. 166 to DPR-43 for Kewaunee Power Station, to revise the Technical Specifications to incorporate a full-scope application of an alternative source term (AST) methodology in accordance with 10 CFR, Section 50.67(b)(2). As a result of the licensee using the AST in their dose consequence analyses, the NRC staff used the regulatory guidance provided in NUREG-0800, Standard Review Plan Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” Revision 0, July 2000, and Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors”, July 2000, which provides the methodology and assumptions acceptable to the NRC staff for the evaluation of design basis radiological analyses using an AST. Specifically, the off-site dose criteria are 25 rem total effective dose equivalent (TEDE) at the exclusion area boundary for any 2-hour period following the onset of the postulated fission product release and 25 rem TEDE at the outer boundary of the low population zone for the duration of the postulated fission product release. In addition, 10 CFR Part 50.67(b)(2)(iii) requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident.

3.0 TECHNICAL EVALUATION

3.1 Technical Evaluation of Custom TS Conversion Changes

3.1.1 Administrative Changes

3.1.1.1 Wording and Cosmetic Changes

In the conversion of the KPS Current Technical Specifications (CTS) to the plant specific ITS, the licensee proposed certain administrative changes (wording preferences, editorial changes, reformatting, revised numbering, etc.) to ensure consistency with ISTS. The NRC staff considers these types of proposed changes to be administrative changes, in that they do not result in technical changes to the KPS CTS. Therefore, the NRC staff finds these changes acceptable.

3.1.1.2 Isotopic Analysis Requirement for Iodine

The licensee asserts that KPS CTS 3.1.c.2.C requires the Table TS 4.1-2 Sampling Test 1.f, RCS isotopic analysis for iodine, to be performed every 4 hours until the specific activity of the primary coolant system is restored to within limits. ITS 3.4.16 Required Action A.1 essentially requires this same analysis, however the explicit statement to perform the isotopic analysis for iodine "until restored to within its limits" has been deleted. The licensee indicated that this would change the KPS CTS by deleting the explicit statement to perform the isotopic analysis for iodine until the limits are met. The indicated purpose of the CTS 3.1.c.2.C and Table TS 4.1-2 Sampling Test 1.f is to ensure the Surveillance is performed to determine whether the specific activity is met. This current statement in KPS CTS is not necessary in the ITS, because ITS limiting condition for operation (LCO) 3.0.2 requires the Required Actions of the associated Conditions to be met upon discovery of failure to meet an LCO. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated. With this proposed change in place, the applicable Surveillance will still be required to be performed since ITS LCO 3.0.4 will require the Required Action to be performed until the LCO is met. This proposed change is considered as administrative because it does not result in technical changes to the KPS CTS. Therefore the NRC staff finds these changes acceptable.

3.1.1.3 Annual Reporting Requirements

The licensee asserts that KPS CTS 3.1.c.3 provides a cross-reference to CTS 6.9.a.2.D, the Annual Reporting Requirements. The current ITS 3.4.16 does not contain this cross-reference. This proposed change would change the current KPS CTS by deleting a cross-reference to another KPS CTS requirement. The licensee asserted that the purpose of the reference is to alert the user that a report may need to be generated due to the specific activity being outside the limit. However, KPS CTS 6.9.a.2.D has not been included in the KPS ITS. Therefore, the cross-reference is not needed. This proposed change is considered administrative because it does not result in technical changes to the KPS CTS. Therefore the NRC staff finds these changes acceptable.

3.1.1.4 CTS Required ACTION A

The licensee asserts that KPS CTS 3.1.c does not preclude the unit from becoming critical or increasing average temperature to greater than (>) 500°F when the DOSE EQUIVALENT I-131 is > 1.0 µCi/gm for less than (<) 48 hours. Thus, during this 48 hour time, the reactor can be made critical and the average temperature can be increased to > 500°F. The proposed KPS ITS 3.4.16 REQUIRED ACTION A includes a Note that specifies that LCO 3.0.4.c is applicable. This proposed change would revise the KPS CTS by specifying the applicable ITS LCO that is consistent with the current KPS CTS allowance. With the proposed changes in place, the proposed KPS ITS LCO 3.0.4.c will continue to ensure a similar allowance in that the reactor can be made critical and average temperature can be increased to > 500°F with the DOSE EQUIVALENT I-131 > 1.0 µCi/gm for < 48 hours. This proposed change is considered as administrative because it does not result in technical changes to the KPS CTS. Therefore the NRC staff finds these changes acceptable.

3.1.2 More Restrictive Changes

3.1.2.1 CTS RCS Specific Activity

The licensee asserts that the current KPS CTS 3.1.c.2 essentially requires that the specific activity of the reactor coolant shall be limited whenever the reactor is critical or the average coolant temperature is > 500°F. The proposed KPS ITS 3.4.16 Applicability, with TSTF-490-A incorporated, requires the RCS DOSE EQUIVALENT I-131 and RCS DOSE EQUIVALENT XE-133 specific activity to be within limits during MODES 1, 2, 3 and 4. In addition, when a unit shutdown is required by current KPS CTS 3.1.c.2.A and CTS 3.1.c.2.B, the current KPS CTS requires the unit to be in INTERMEDIATE SHUTDOWN with an average coolant temperature of < 500°F within 6 hours. Proposed KPS ITS 3.4.16 Required Action C.1 requires the unit to be in MODE 3 within 6 hours and Required Action C.2 requires the unit to be in MODE 5 within 36 hours. This proposed change would revise the current KPS CTS by applying the LCO in more MODES in the ITS than in CTS and by adding commensurate Required Actions to exit the new Applicability. The proposed change that deletes the E-bar requirement and replaces it with a DOSE EQUIVALENT XE-133 requirement is discussed in Sections 3.1.3.1 and 3.2.2 below.

The indicated purpose of KPS CTS 3.1.c is to ensure that the specific activity of the RCS is consistent with the assumptions of the MSLB accident and SGTR accident analyses. The licensee has asserted that with this proposed change in place, the applicable requirements will continue to ensure that the process variables are maintained in the MODES and other specified conditions assumed in the safety analyses and licensing basis. The NRC staff concludes that with the above changes implemented, the ITS 3.4.16 Applicability of MODES 1, 2, 3, and 4 is more restrictive than the CTS Applicability. Therefore, the NRC staff finds this proposed change acceptable because the implementation of the ITS 3.4.16 MODES of Applicability will be more restrictive than the CTS Applicability. The NRC staffs acceptance of the ITS 3.4.16 Applicability is discussed more in detail in Section 3.2.3 and Section 3.2.4 below.

3.1.2.2 CTS 3.1.C.2 Applicability

The current KPS CTS Applicability is "whenever the reactor is critical or the average coolant temperature is > 500°F". The reactor is considered critical in the OPERATING (ITS equivalent MODE 1) and HOT STANDBY (ITS equivalent MODE 2) MODES. The reactor coolant temperature ranges of HOT SHUTDOWN (ITS equivalent MODE 3) is greater than or equal to (≥) 540°F and INTERMEDIATE SHUTDOWN (ITS equivalent MODE 4) is > 200°F and < 540°F. However, the MODE associated with reactor coolant temperature ranges specified in the current KPS CTS are not equivalent to those of the proposed ITS. The reactor coolant temperature range associated with ITS MODE 3 is ≥ 350°F; for ITS MODE 4 the range is > 200°F and < 350°F. Kewaunee has indicated that as a result of the differences in the temperature ranges between the MODE definitions of KPS CTS and ITS, the current KPS CTS Applicability of "the average reactor coolant temperature of > 500°F" is attainable in equivalent ITS MODES 1, 2, and 3 but not in ITS MODE 4. Therefore, the NRC staff finds that proposed ITS 3.4.16 Applicability of MODES 1, 2, 3, and 4 is more restrictive than the current KPS CTS Applicability.

In addition, the licensee also indicated that during operation with RCS T_{avg} < 500°F, the release of activity would be minimal should a steam generator tube rupture occur since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety valves. This

condition would be satisfied once the unit enters ITS MODE 3. Furthermore, the proposed Required Actions for when a unit shutdown is required ensures that the proposed ITS LCO Applicability is exited. This proposed change was designated as more restrictive by the licensee because the Applicability is applicable in more MODES than in the current KPS CTS and commensurate actions to exit the proposed Applicability have been added. The NRC staff concludes that with the above changes implemented, the ITS 3.4.16 Applicability of MODES 1, 2, 3, and 4 is more restrictive than the CTS Applicability. The NRC staff's acceptance of the ITS 3.4.16 Applicability is discussed more in detail in Section 3.2.3 and Section 3.2.4 below.

3.1.2.3 CTS Reactor Coolant Sampling Requirement

The licensee asserts that the current KPS CTS Table TS 4.1-2 Item 1.b requires that DOSE EQUIVALENT I-131 reactor coolant concentration be sampled every 14 days. The proposed ITS Surveillance Requirement (SR) 3.4.16.2 requires that DOSE EQUIVALENT I-131 reactor coolant concentration is verified to be less than or equal to (≤) 1.0 µCi/gm every 14 days and between 2 and 6 hours after a THERMAL POWER change of ≥ 15 percent rated thermal power (RTP) within a 1 hour period. This proposed change will revise the current KPS CTS by adding a new Surveillance Requirement Frequency.

The indicated purpose of proposed ITS SR 3.4.16.2 is to verify that the reactor coolant DOSE EQUIVALENT I-131 specific activity is within the assumptions of the accident (MSLB and SGTR) analyses. This particular Surveillance is performed in order to ensure that iodine specific activity remains within the LCO limit during normal operation, and following fast power changes when iodine spiking is more apt to occur. The licensee stated that the proposed Frequency, between 2 and 6 hours after a power change ≥ 15 percent RTP within a 1 hour period, was established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results. This proposed change was designated as more restrictive by the licensee because a new SR Frequency has been added. Therefore, the NRC staff finds this proposed change acceptable because the implementation of the ITS SR 3.4.16.2 will be more restrictive than the current KPS CTS sampling requirement by imposing additional sampling requirements. Specifically, ITS SR 3.4.16.2 would require isotopic sampling for the DOSE EQUIVALENT I-131 specific activity to be applicable in more MODES than CTS, and the sampling frequency is increased to account for iodine spiking after a power change ≥ 15 percent RTP, which would provide more accurate sampling results. The NRC staff's acceptance of the proposed ITS 3.4.16 Applicability and ITS SR 3.4.16.2 is also discussed more in detail in Sections 3.2.3, 3.2.4, and 3.2.9 below.

3.1.3 Less Restrictive Changes

3.1.3.1 CTS Gross Radioactivity Determination, $\bar{E}$ Deletion, and ITS 3.4.16 REQUIRED ACTION B

The licensee asserts that the current KPS CTS 3.1.c.1.B requires that the gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium to be < 91/$\bar{E}$ µCi/cc. In addition, current KPS CTS 3.1.c.2.B states that if the limit is not met, then the unit must be shut down to INTERMEDIATE SHUTDOWN with an average coolant temperature < 500°F within 6 hours, with no restoration time prior to the shutdown provided. Furthermore, if the limit is not met, current KPS CTS 3.1.c.2.C requires that the sample and analysis requirements of

Table TS 4.12, item 1.f (an isotopic analysis for iodine), be performed every 4 hours. According to the submittal, Table TS 4.1-2, Item 1.a, requires a gross radioactivity determination (excluding tritium) 5 times per week, with a maximum time between tests of 3 days and item 1.e requires an $\bar{E}$ determination every 6 months with the sample being required after a minimum of 2 effective full-power days and 20 days of OPERATING MODE operation have elapsed since the reactor was last subcritical for > 48 hours. Proposed ITS 3.4.16 does not include any requirements related to $\bar{E}$.

Proposed ITS LCO 3.4.16 requires the DOSE EQUIVALENT XE-133 limit to be met. Proposed ITS SR 3.4.16.1 states that the DOSE EQUIVALENT XE-133 must be < 595 µCi/gm and requires verification of this limit every 7 days when in MODE 1. If DOSE EQUIVALENT XE-133 is not within the limit, proposed ITS 3.4.16 REQUIRED ACTION B provides 48 hours to restore the DOSE EQUIVALENT XE-133 to be within its limit prior to requiring a unit shutdown. Proposed ITS 3.4.16 REQUIRED ACTION B would also allow LCO 3.0.4.c to be applicable when in CONDITION B. Furthermore, when DOSE EQUIVALENT XE-133 is not within its limit, the ITS does not require the isotopic analysis for iodine to be performed every 4 hours. This proposed change would revise the current KPS CTS by deleting the $\bar{E}$ requirements on primary coolant gross specific activity and replacing it with the DOSE EQUIVALENT XE-133 requirements on primary coolant noble gas activity, consistent with Technical Specification Task Force (TSTF) change traveler TSTF-490-A.

From a radiological dose perspective, the licensee indicates that the proposed change incorporating the newly defined quantity DOSE EQUIVALENT XE-133 would result in an LCO that more closely relates to non-iodine RCS activity limits to the dose consequence analyses which form the bases. The licensee also indicates that DCFs used in the determination of DOSE EQUIVALENT I-131 and XE-133 is consistent with the DCFs used in the applicable dose consequence analysis. The licensee has considered this proposed change as less restrictive because the LCO is now being based on noble gas activity versus gross specific activity, and a limited amount of time (48 hours) is provided to restore the limit prior to requiring a unit shutdown. The NRC staff finds that these proposed changes are consistent with the NRC staffs approved position in NUREG-1431 and TSTF-490, Revision 0, and are therefore acceptable. The NRC staff's acceptance of the above stated changes is discussed more in detail in Sections 3.2.3, 3.2.6, 3.2.8, and 3.2.9 below.

3.1.3.2 Deletion of Surveillance for Reactor Coolant Sampling for Tritium

The licensee asserts that the current KPS CTS Table TS 4.1-2 Item 1.c requires a monthly reactor coolant sample for tritium activity. Proposed ITS 3.4.16, including the incorporation of TSTF-490-A, does not include this requirement. This proposed change would revise the current KPS CTS by deleting this Surveillance Requirement. The licensee indicates that the purpose of the current KPS CTS Table TS 4.1-2 Item 1.c and 1.e is to ensure plant operation is within the specified gross activity LCO limit (i.e., $\bar{E}$). TSTF-490-A, incorporated into the proposed ITS, changes the measurement of the gross specific activity of the reactor coolant to the primary coolant noble gas activity. The licensee states that the bases for this change lies in the fact that when $\bar{E}$ is determined using a design basis approach in which it is assumed that 1 percent of the power is being generated by fuel rods having cladding defects, and it is also assumed that there is no removal of fission gases from the letdown flow, the value of $\bar{E}$ is dominated by XE-133.

Kewaunee asserts that during normal plant operation there are typically only a small amount of fuel defects and the radioactive nuclide inventory can become dominated by tritium and corrosion and/or activation products, resulting in the determination of a value of Ē that is very different than would be calculated using the design basis approach. In addition, the accident dose analyses become disconnected from plant operation and the LCO becomes essentially meaningless. The licensee further states that if the indicated purpose of the LCO on gross specific activity is to support the dose analyses for design basis accidents (DBAs), then it would be more appropriate to have the LCO apply to the noble gas concentration with in the primary coolant. Thus, the current LCO on gross coolant activity, which is based on Ē, will be replaced by an LCO on reactor coolant noble gas activity, which would be based on DOSE EQUIVALENT XE-133. This is the proposed change described above in Section 3.1.3.1. The licensee designated this proposed change as less restrictive because a Surveillance which is required in the current KPS CTS will not be required in the proposed ITS. The NRC staff finds that the proposed change is consistent with the NRC staffs approved position in NUREG-1431 and TSTF-490, Revision 0, and is therefore acceptable. The NRC staff's acceptance of the above stated change is also discussed more in detail in Section 3.2.2 below.

3.2 Technical Evaluation of TSTF-490 TS Changes

3.2.1 Revision to the definition of DEI

The list of currently acceptable DCFs for use in the determination of DEI include the following:

- Table III of Technical Information Document (TID)-14844, Atomic Energy Commission (AEC), 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."
- Table E-7 of RG 1.109, Revision 1, NRC, 1977.
- International Commission on Radiological Protection (ICRP) 30, 1979, page 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."
- Committed Dose Equivalent (CDE) or Committed Effective Dose Equivalent (CEDE) dose conversion factors from Table 2.1 of Environmental Protection Agency (EPA) Federal Guidance Report No. 11.
- Table 2.1 of Environmental Protection Agency (EPA) Federal Guidance Report No. 11, 1988, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

Kewaunee is licensed to 10 CFR 50.67, "Accident Source Term", and has proposed that the CDE DCFs from ICRP 30 will be used in the definition of DEI. Therefore the new definition of DEI will read as follows:

> DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries per gram) that alone would produce the same dose when inhaled as the quantity and isotopic mixture of combined activities of iodine isotopes of I-131, I-132, I-133,

> I-134, and I-135 actually present. The determination of DOSE EQUIVALENT I-131 shall be performed using ICRP-30, 1979, page 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

The NRC staff conducted a confirmatory analysis, using various sources of DCFs and the RCS concentration values from the licensee's AST submittal. The NRC staff's analysis confirmed that for a given RCS isotopic concentration of iodine, the use of the DCFs from ICRP 30 will yield a DEI value of 1 µCi/gm which is consistent with the assumptions used for the current KPS SGTR and MSLB accident analyses. In addition, maintaining the DEI value to the TS limit of 1 µCi/gm will be conservative relative to the coolant concentrations used in the design basis dose consequence analyses for Kewaunee, as outlined and verified in the licensees response to request for additional information (RAI) response dated April 13, 2010 (ML101060517). Therefore, the NRC staff finds that the licensee's proposed revision of the DEI definition and the proposed use of the DCFs from ICRP 30 is consistent with the NRC staffs approved position in TSTF-490, Revision 0, and is therefore acceptable.

3.2.2 Deletion of the Definition of E Bar ($\bar{E}$) and the Addition of a New Definition for DE Xe-133

The new definition for DEX is similar to the definition for DEI. The determination of DEX will be performed in a similar manner to that currently used in determining DEI, except that the calculation of DEX is based on the acute dose to the whole body and considers the noble gases Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 which are significant in terms of contribution to whole body dose. Some noble gas isotopes are not included due to low concentration, short half-life, or small DCF. The calculation of DEX will use the effective DCFs from Table III.1 of EPA Federal Guidance Report No. 12 (FGR-12). Using this approach, the limit on the amount of noble gas activity in the primary coolant would not fluctuate with variations in the calculated values of $\bar{E}$. If a specified noble gas nuclide is not detected, the new definition states that it should be assumed the nuclide is present at the minimum detectable activity. This will result in a conservative calculation of DEX.

When $\bar{E}$ is determined using a design basis approach in which it is typically assumed that 1.0 percent of the power is being generated by fuel rods having cladding defects and it is also assumed that there is no removal of fission gases from the letdown flow, the value of $\bar{E}$ is dominated by Xe-133. The other noble gas nuclides have relatively small contributions. However, during normal plant operation there are typically only a small amount of fuel clad defects and the radioactive nuclide inventory can become dominated by tritium and corrosion and/or activation products, resulting in the determination of a value of $\bar{E}$ that is very different than would be calculated using the design basis approach. Because of this difference, the accident dose analyses become disconnected from plant operation and the LCO becomes essentially meaningless. It also results in a TS limit that can vary during operation as different values for $\bar{E}$ are determined.

This change will implement a LCO that is consistent with the whole body radiological consequence analyses which are sensitive to the noble gas activity in the primary coolant but not to other non-gaseous activity currently captured in the $\bar{E}$ definition. The licensee's current CTS specify the limit for primary coolant gross specific activity as 91/$\bar{E}$ µCi/cc due to nuclides with half-lives > 30 minutes, whenever the reactor is critical or the average coolant temperature

is > 500° F. Table TS 4.1-2 of the CTS currently requires the licensee to determine the gross radioactivity at a minimum of five (5) times per week, and to perform reactor coolant system isotopic analyses for Iodine once every four (4) hours until the specific activity of the primary coolant system is restored to within limits. The current STS LCO 3.4.16 specifies the limit for primary coolant gross specific activity as 100/ $\bar{E}$ µCi/gm. Kewaunee's current site-specific primary coolant gross specific activity of 91/$\bar{E}$ µCi/cc is equivalent to the current STS value of 100/$\bar{E}$ µCi/gm, where as the latter represents a standardized value set to be used by various licensees. Thus, both Kewaunee's current CTS definition and the current STS $\bar{E}$ definition include radioisotopes that decay by the emission of both gamma and beta radiation. In any case, the primary coolant gross specific activity for both would rarely, if ever, exceed 91/$\bar{E}$ µCi/cc since the calculated value is very high (the denominator is very low) because if beta emitters such as tritium (H-3) are included in the determination, as required by the $\bar{E}$ definition.

TS Section 1.1 definition for E - AVERAGE DISINTEGRATION ENERGY ($\bar{E}$) is deleted and replaced with a new definition for DEX which states:

> DOSE EQUIVALENT XE-133 shall be that concentration of Xe-133 (microcuries per gram) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it should be assumed to be present at the minimum detectable activity. The determination of DOSE EQUIVALENT XE-133 shall be performed using effective dose conversion factors for air submersion listed in Table III.1 of EPA Federal Guidance Report No. 12, 1993, "External Exposure to Radionuclides in Air, Water, and Soil."

The change incorporating the newly defined quantity DEX is acceptable from a radiological consequence dose perspective since it will result in an LCO that more closely relates the non-iodine RCS activity limits to the dose consequence analyses which form their bases. The licensee has proposed to use the DCFs from FGR-12 used in the evaluation of TSTF-490 for the calculation of DEX values. The NRC staff also confirmed that the licensee's proposed DEX value of 595 µCi/gm is a conservative representation of the mix of nuclides in the RCS and is used in the dose consequence analysis. Furthermore, the NRC staff finds that the licensee's proposed definition of DEX and the proposed use of the DCFs from FGR-12 is consistent with the NRC staffs approved position in TSTF-490, Revision 0, and is therefore acceptable.

3.2.3 LCO 3.4.16, "RCS Specific Activity"

The NRC staff evaluated the proposed change from the current KPS CTS to the ITS for LCO 3.4.16 in Section 3.1.3.1 above. For the proposed change to implement TSTF-490, ITS LCO 3.4.16 is modified to specify that iodine specific activity in terms of DEI and noble gas specific activity in terms of DEX shall be within limits. Currently, the limiting indicators are not explicitly identified in the ITS LCO, but are instead defined in current Condition B and SR 3.4.16.1 for gross non-iodine specific activity and in current Condition A and SR 3.4.16.2 for iodine specific activity. The change states, "RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits."

TSTF-490 proposes to revise TS 3.4.16 Required Action A.1 to remove the reference to Figure 3.4.16-1 "Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit versus Percent of RATED THERMAL POWER" and insert a limit of less than or equal to the site specific DEI spiking limit. The curve contained in Figure 3.4.16-1 was initiated by the AEC in a June 12, 1974 letter from the AEC on the subject, "Proposed Standard Technical Specifications for Primary Coolant Activity." Radiological dose consequence analyses for SGTR and MSLB accidents that take into account the pre-accident iodine spike do not consider the elevated RCS iodine specific activities permitted by Figure 3.4.16-1 for operation at power levels below 80 percent RTP. Instead, the pre-accident iodine spike analyses assume a DEI concentration 60 times higher than the corresponding long term equilibrium value, which corresponds to the specific activity limit associated with 100 percent RTP operation. TSTF-490 asserts and the NRC staff agrees that TS 3.4.16 Required Action A.1 should be based on the short term site specific DEI spiking limit to be consistent with the assumptions contained in the radiological consequence analyses.

3.2.4 LCO 3.4.16 APPLICABILITY

The NRC staff evaluated the proposed change from the current KPS CTS to the ITS for TS 3.4.16 Applicability in Section 3.1.2.2 of this SE. For the proposed change to implement TSTF-490, the APPLICABILITY requirement of LCO 3.4.16 is modified to be applicable in operation MODES 1, 2, 3, and 4. Currently, STS SR 3.4.16.1 and SR 3.4.16.2 do not require the SR to be performed in MODES 2, 3, and 4. These current conditions for sampling exclude sampling during the plant conditions where LCO 3.4.16 may be exceeded (e.g. MODES 2, 3, and 4). Additionally, after transient conditions (e.g. reactor trip, plant depressurization, shutdown or startup) that end in MODES 2, 3, and 4, the SR is currently not required to be performed. LCO 3.4.16 could potentially be exceeded after plant transient or power changes, therefore, sampling in MODES 2, 3, and 4 is applicable. In addition, isotopic spiking and fuel failures are more likely during transient conditions than during steady state plant operations.

However, it should also be noted that proposed ITS SR 3.4.16.1 and SR 3.4.16.2 are required to be met during all MODES of Applicability (MODES 1, 2, 3, and 4) in accordance with SR 3.0.1, which states:

> *SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3.*

Therefore, as stated in Kewaunee's response to Request for Additional Information dated April 13, 2010, at any time during MODES 1 through 4, if there is information or plant indication that proposed ITS SR 3.4.16.1 or ITS SR 3.4.16.2 may not be met, surveillance is required to be performed to ensure there is not a failure to meet the LCO. Nonetheless, the surveillance will be required to be performed during all MODES of Applicability (MODES 1, 2, 3, and 4). This ensures that the potential consequences of a steam line break or steam generator tube rupture are bounded by the approved accident analysis, from which the LCO limits are derived.

The NRC staff finds the proposed change acceptable because the applicability of ITS LCO 3.4.16 continue to ensure that the monitored process variables (e.g., Pressurizer pressure, RCS average temperature, RCS total flow rate, and RCS specific activity) are maintained in MODES 1, 2, 3, and 4 to ensure that the core operates within the limits assumed in the safety analyses of record. The NRC staff further finds that the licensee's proposed ITS LCO 3.4.16 APPLICABILITY is conservative because it applies to more operational modes and is therefore acceptable.

3.2.5 ITS 3.4.16 Condition A

The NRC staff evaluated the proposed change from the current KPS CTS to the ITS for the revision of Condition A of ITS 3.4.16 in Section 3.1.1.4 of this SE. ITS 3.4.16 Condition A is revised by replacing the DEI site-specific limit of "> 1.0 µCi/gm" with the words "not within limit" to be consistent with the revised ITS 3.4.16 LCO format. The site-specific DEI limit of ≤ 1.0 µCi/gm is now contained in proposed ITS SR 3.4.16.2. This proposed format change will not alter current STS requirements and is acceptable from a radiological dose perspective. ITS 3.4.16 Condition A, Required Action A.1 is revised to remove the reference to Figure 3.4.16-1 "Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit versus Percent of RATED THERMAL POWER" and insert a limit of less than or equal to the site-specific DEI spiking limit of 20 µCi/gm. The curve contained in Figure 3.4.16-1 was provided by the AEC in a June 12, 1974 letter from the AEC on the subject, "Proposed Standard Technical Specifications for Primary Coolant Activity." Radiological dose consequence analyses for SGTR and MSLB accidents that take into account the pre-accident iodine spike do not consider the elevated RCS iodine specific activities permitted by Figure 3.4.16-1 for operation at power levels below 80 percent RATED THERMAL POWER (RTP). Instead, the pre-accident iodine spike analyses assume a DEI concentration 20 times higher than the corresponding long-term equilibrium value, which corresponds to the specific activity limit associated with 100 percent RTP operation. It is acceptable that proposed ITS 3.4.16 Required Action A.1 should be based on the short-term site-specific DEI spiking limit to be consistent with the assumptions contained in the radiological consequence analyses. Therefore, the NRC staff finds the proposed change acceptable because the revised ITS 3.4.16 Condition A will still remain consistent with the NRC staffs approved position in TSTF-490, Revision 0. In addition, the site-specific DEI limit of ≤ 1.0 µCi/gm is now contained in proposed ITS SR 3.4.16.2, as described in Section 3.2.9 of this SE.

3.2.6 TS 3.4.16 Condition B Revision to include Action for DEX Limit

The NRC staff evaluated the proposed change from the current KPS CTS to the ITS for the revision of Condition B of ITS 3.4.16 in Section 3.1.3.1 of this SE. ITS 3.4.16 Condition B which states, "Gross specific activity of the reactor coolant not within limit," is replaced with a new condition which states, "Dose Equivalent Xe-133 not within limits." This change is made to be consistent with the proposed change to the ITS 3.4.16 LCO which requires the DEX specific activity to be within limits as discussed above in Section 3.2.3. The DEX limit is site-specific and the numerical value in units of µCi/gm is contained in revised ITS SR 3.4.16.1, as described below in Section 3.2.8. The site-specific limit of DEX in µCi/gm is typically based on the maximum accident analysis RCS activity corresponding to 1 percent fuel clad defects with sufficient margin to accommodate the exclusion of those isotopes based on low concentration, short half life, or small dose conversion factors. The primary purpose of the proposed

ITS 3.4.16 LCO on RCS specific activity and its associated Conditions is to support the dose analyses for DBAs. The whole body dose is primarily dependent on the noble gas activity, not the non-gaseous activity currently captured in the Ē definition.

The proposed Completion Time for proposed ITS 3.4.16 Required Action B.1 will require restoration of DEX to within limit in 48 hours. This is consistent with the Completion Time for proposed Required Action A.2 for DEI. The radiological consequences for the SGTR and the MSLB accidents demonstrate that the calculated thyroid doses are generally a greater percentage of the applicable acceptance criteria than the calculated whole body doses. It then follows that the proposed Completion Time for noble gas activity being out of specification in the proposed Required Action B.1 should be at least as great as the proposed Completion Time for iodine specific activity being out of specification in current Required Action A.2. Therefore, the proposed Completion Time of 48 hours for proposed Required Action B.1 is acceptable from a radiological dose perspective. A Note is also proposed to be added to the revised Required Action B.1 that states, "LCO 3.0.4.c is applicable." This Note would allow entry into a MODE or other specified condition in the LCO Applicability when proposed ITS LCO 3.4.16 is not being met, and is the same Note that is currently stated for proposed Required Actions A.1 and A.2. The proposed Note would allow entry into the applicable MODES from MODE 4 to MODE 1 (power operation) while the DEX limit is exceeded and the DEX is being restored to within its limit. This MODE change is acceptable due to the significant conservatism incorporated into the DEX specific activity limit, the low probability of an event occurring which is limiting due to exceeding the DEX specific activity limit, and the ability to restore transient specific excursions while the plant remains at, or proceeds to power operation. Therefore, the NRC staff finds the proposed change acceptable because the revised ITS 3.4.16 Condition B will still remain consistent with the NRC staffs approved position in TSTF-490, Revision 0.

3.2.7 TS 3.4.16 Condition C

The NRC staff evaluated the proposed change from the current KPS CTS to the ITS for the revision of Condition C of ITS 3.4.16 in Section 3.1.3.1 of this SE. The current STS 3.4.16 Condition C is revised to include Condition B (DEX not within limit) if the proposed Required Action and associated Completion Time of Condition B is not met. This is consistent with the changes made to proposed Condition B which now provide the same completion time for both components of RCS specific activity as discussed in the revision to Condition B. The proposed Condition C also replaces the limit on DEI from the deleted Figure 3.4.16-1, with a site-specific value of > 20 μCi/gm. This change makes proposed Condition C consistent with the changes made in proposed ITS 3.4.16 Required Action A.1.

The change to proposed ITS 3.4.16 Required Action C.1 requires the plant to be in MODE 3 within 6 hours and adds a new Required Action C.2 which requires the plant to be in MODE 5 within 36 hours. These changes are consistent with the proposed changes made to the ITS 3.4.16 Applicability. The proposed ITS LCO 3.4.16 is applicable throughout all of MODES 1 through 4 to limit the potential radiological consequences of an SGTR or MSLB that may occur during these MODES. In MODE 5 with the RCS loops filled, the steam generators are not normally used for decay heat removal. In this mode, however, due to the reduced temperature of the RCS, the probability of a DBA involving the release of significant quantities of RCS inventory is greatly reduced. Therefore, monitoring of RCS specific activity is not required. In MODE 5 with the RCS loops not filled and MODE 6, the steam generators are not used for

decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

A new ITS 3.4.16 Required Action C.2 Completion Time of 36 hours is also proposed for the plant to reach MODE 5. This Completion Time is reasonable, based on operating experience, to reach MODE 5 from full power conditions in an orderly manner and without challenging plant systems and the value of 36 hours is consistent with other proposed ITS which have a Completion Time to reach MODE 5. Therefore, the NRC staff finds the proposed change acceptable because the revised ITS 3.4.16 Condition C will still remain consistent with the NRC staffs approved position in TSTF-490, Revision 0.

3.2.8 SR 3.4.16.1 DEX Surveillance

The NRC staff evaluated the proposed change from the current KPS CTS to the ITS for the revision of Condition C of ITS 3.4.16 in Section 3.1.3.1 of this SE. The change replaces the current STS SR 3.4.16.1 surveillance for RCS gross specific activity with a surveillance to verify that the site-specific reactor coolant DEX specific activity is ≤ 595 μCi/gm. This change provides surveillance for the new LCO limit added to proposed ITS 3.4.16 for DEX. The revised ITS SR 3.4.16.1 surveillance requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days, which is the same frequency required under the current STS SR 3.4.16.1 surveillance for RCS gross non-iodine specific activity. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. The surveillance provides an indication of any increase in the noble gas specific activity.

The current STS surveillance requires the licensee to verify reactor primary coolant gross specific activity to ≤ 91/$\bar{E}$ μCi/gm. In the proposed LAR, the licensee is deleting the definition and reference to $\bar{E}$, the average disintegration energy, and adding a limit for primary coolant noble gas activity based on DOSE EQUIVALENT XE-133, and would take into account only the noble gas activity in the primary coolant. The change states, “Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 595 μCi/gm.” The results of the surveillance on DEX allow proper remedial action to be taken before reaching the LCO limit under normal operating conditions.

As described in Sections 3.2.4 and 3.2.6 above, and in accordance with proposed ITS SR 3.0.1, the APPLICABILITY requirement of proposed ITS LCO 3.4.16 is applicable in operation MODES 1, 2, 3, and 4 for Kewaunee. ITS SR 3.4.16.1 is also modified by inclusion of a NOTE which permits the use of the provisions of proposed ITS LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation. This allows entry into MODE 4, MODE 3, and MODE 2 prior to performing the surveillance. This also allows the surveillance to be performed in any of those MODES, prior to entering MODE 1.

The NRC staff finds the proposed change acceptable because the applicability of proposed ITS LCO 3.4.16, as described in Section 3.2.4 above, will continue to ensure that the monitored process variables (e.g., Pressurizer pressure, RCS average temperature, RCS total flow rate,

and RCS specific activity) are maintained in MODES 1, 2, 3, and 4 to ensure that the core operates within the limits assumed in the safety analyses of record. The NRC staff further finds that the licensee's proposed ITS SR 3.4.16.1 DEX Surveillance is conservative because it applies to more operational modes and is therefore acceptable.

3.2.9 SR 3.4.16.2 DEI Surveillance

As described in Section 3.2.4 above, and in accordance with proposed ITS SR 3.0.1, the APPLICABILITY requirement of proposed ITS LCO 3.4.16 is for MODES 1, 2, 3, and 4 for Kewaunee. Currently, a NOTE exists in STS SR 3.4.16.2 which reads, "Only required to be performed in MODE 1." In response to NRC staff RAIs, and to be consistent with the applicability and requirements of the proposed ITS SR 3.4.16.1, this NOTE is subsequently being deleted from proposed ITS SR 3.4.16.2. Hence, the DEI Surveillance is required to be met during all MODES of Applicability.

The NRC staff finds the proposed change acceptable because the applicability of proposed ITS LCO 3.4.16, as described in Section 3.2.4 above, will continue to ensure that the monitored process variables (e.g., Pressurizer pressure, RCS average temperature, RCS total flow rate, and RCS specific activity) are maintained in MODES 1, 2, 3, and 4 to ensure that the core operates within the limits assumed in the safety analyses of record. In addition, the current STS is also revised by adding commensurate surveillance requirements for sampling to ensure that the perspective DEI site specific activity of ≤ 1.0 μCi/gm is maintained. The NRC staff further finds that the licensee's proposed ITS SR 3.4.16.2 DEI Surveillance is conservative because it applies to more operational modes and is therefore acceptable.

3.2.10 Consistency of Site-Specific Limits and DCFs For DEX and DEI Surveillances

Consistent with the proposed, current, and/or revised definitions and limits for both DEX and DEI, and the DCFs used for the determination of DEI and DEX surveillances, the NRC staff verified, that the site-specific limits for both DEI and DEX, the DCFs, and the RCS radioisotopic concentrations, are consistent with the current applicable design-basis accident dose analyses for Kewaunee. The current KPS CTS definition of dose equivalent I-131 allows DEI to be calculated using DCFs from ICRP-30, and is based upon information presented in their LAR application to incorporate full-scope implementation of AST methodology dated March 19, 2002 (ADAMS Accession No. ML020870565), as supplemented by letters dated September 13, 2002 (ADAMS Accession No. ML022680167), October 21, 2002 (ADAMS Accession No. ML023040302), and approved in amendment 166 dated March 17, 2003 (ADAMS Accession No. ML030210062).

The acceptability for the pre-accident and concurrent iodine spike source terms are based on ICRP-30 DCFs, and the doses to be calculated using FGR-11, was also submitted and approved in the precedent documents stated above. In addition, RG 1.183 requires that the pre-accident and concurrent iodine spikes used in design basis analyses be based on the maximum value permitted by Technical Specifications, which is 20 μCi/gm for Kewaunee. The Kewaunee MSLB and SGTR accidents are analyzed using the maximum RCS activity. DCFs from FGR-11 are used to calculate the Total Effective Dose Equivalent consequences described using the guidance from RG 1.183, and the 1 μCi/gm DEI inventory.

The DCFs used by Kewaunee to determine dose from noble gases and the calculation of DEX are from FGR-12. DEX is that concentration of Xe-133 that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, and Xe-138 actually present. If a specific noble gas nuclide is not detected, it will be assumed to be present at the minimum detectable activity. Because, Xe-131m is not present in the current design basis accident (DBA) analysis source term, its exclusion from the calculation of DEX is conservative since the limit will be lower and the actual surveillance will include either the actual concentration detected or the minimum detectable activity.

The NRC staff evaluated the licensee's response to RAIs regarding the RCS concentrations for noble gas isotopes, the calculation of the site-specific limits on DEI and DEX, and how Kewaunee is maintaining consistency in regard to its current AST analyses. In its analyses, the licensee used primary coolant concentration values equivalent to the more limiting LCO of 1 µCi/gm DEI. Concentration values that are conservatively below the technical specification limits of 1 µCi/gm for DEI and 595 µCi/gm for the DEX, were verified by NRC staff. Therefore, the site-specific limits for both DEI and DEX, as described above in Sections 3.2.5, 3.2.6, and 3.2.7, and the DCFs used for the determination of DEI and DEX surveillances, as described above in Sections 3.2.1, 3.2.2, 3.2.8, and 3.2.9, are consistent with the current design-basis SGTR and MSLB accident dose analyses. The NRC staff further finds that the licensee's proposed site-specific limits and DCFs for DEX and DEI Surveillances is consistent with the NRC staffs approved position in TSTF-490, Revision 0, and is therefore acceptable.

3.2.11 SR 3.4.16.3 Deletion

The current STS SR 3.4.16.3 which required the determination of $\bar{E}$ is deleted. Proposed ITS 3.4.16 LCO on RCS specific activity supports the dose analyses for DBAs, in which the whole body dose is primarily dependent on the noble gas concentration, not the non-gaseous activity currently captured in the $\bar{E}$ definition. With the elimination of the limit for RCS gross specific activity and the addition of the new LCO limit for noble gas specific activity, this SR to determine $\bar{E}$ is no longer required.

3.3 Precedent

The technical specifications developed for the Westinghouse AP600 and AP1000 advanced reactor designs incorporate an LCO for RCS DEX activity in place of the LCO on non-iodine gross specific activity based on $\bar{E}$. This approach was approved by the NRC staff for the AP600 in NUREG-1512, "Final Safety Evaluation Report Related to the Certification of the AP600 Standard Design, Docket No. 52-003," dated August 1998 and for the AP1000 in the NRC letter to Westinghouse Electric Company dated September 13, 2004. In addition, the curve describing the maximum allowable iodine concentration during the 48-hour period of elevated activity as a function of power level was not included in the TS approved for the AP600 and API000 advanced reactor designs.

4.0 CONCLUSION

The proposed TS conversion and applicable TSTF-490 implementation was reviewed to determine whether the content and format were consistent with the applicable reference ITS 3.4.16, "RCS Specific Activity." The NRC staff has evaluated the licensee's request for applicable changes related to wording and cosmetic changes, isotopic analysis requirements for Iodine, annual reporting requirements, reactor coolant sampling requirements, and gross radioactivity determination for the KPS conversion from CTS to ITS versions of TS. Special attention was given to proposed conversions and/or TS provisions that departed from the reference TS to determine whether proposed differences are justified by uniqueness in plant design or other considerations. The NRC staff evaluated the proposed TS to the reference TS to determine whether the application stated adequate technical justification for each departure from the reference TS.

The NRC staff has also evaluated the licensee's request in accordance with TSTF-490 to revise the definition of DEI, delete the definition of E-bar, add a new definition for DEX, revise STS 3.4.16 to specify an LCO limit on both DEI and DEX, and revise the STS 3.4.16 Conditions and Required Actions. The NRC staff also evaluated the change in the Applicability of proposed ITS LCO 3.4.16 to reflect the MODES during which the SGTR and MSLB accidents could be postulated to occur, the revision of STS SR 3.4.16.1 to verify DEX is within the prescribed limit, the revision of STS SR 3.4.16.2 to delete the existing NOTE, the consistency of site-specific limits and DCFs for DEI and DEX surveillances, and the deletion of STS SR 3.4.16.3.

In addition, the NRC staff has evaluated the consistency of site-specific limits and DCFs for DEI and DEX surveillances. As described in the sections above, the NRC staff reviewed the licensee's assumptions, inputs, and methods to assess the impact of the proposed changes to the Kewaunee TS. Based on its review, the staff determined there is reasonable assurance that the proposed changes will not impact the dose consequences of the applicable DBAs because the proposed changes limit the RCS noble gas and iodine specific activity to ensure consistency with the values assumed in the site-specific DBA radiological consequence analyses.

Therefore, the proposed license amendment is acceptable with respect to the radiological dose consequences of the DBAs.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FROM THE COMPONENT PERFORMANCE AND TESTING BRANCH (CPTB)

PERTAINING TO INSERVICE TESTING REQUIREMENTS

FOR KEWAUNEE POWER STATION IMPROVED TECHNICAL SPECIFICATION

CONVERSION LICENSE AMENDMENT REQUEST (TAC NO. ME2139)

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated, August 24, 2009, Dominion Energy Kewaunee, Inc. (DEK) requested an amendment to current Technical Specifications (CTS) for Kewaunee Power Station (KPS). This amendment request would convert the CTS to Improved TS (ITS) consistent with the Improved Standard Technical Specifications (ISTS) described in NUREG-1431, Revision 3.0, "Standard Technical Specifications-Westinghouse Plants." The proposed changes would revise CTS 4.2 and relocate the inservice testing (IST) program portion of the CTS 4.2 to ITS 5.5.6. DEK states that the proposed changes are consistent with NUREG-1431, Revision 3.0.

The proposed changes also incorporate ISTS modifications approved by Technical Specification Task Force (TSTF) Travelers TSTF-479, "Changes to Reflect Revision to 10 CFR 50.55a," and TSTF-497, "Limit Inservice Testing Program Surveillance Requirement (SR) 3.0.2 Application to Frequencies of 2 Years or Less."

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(f)(5)(ii), requires that, if a revised IST program for a facility conflicts with the TS for that facility, the licensee shall apply to the Commission for amendment of the TS to conform the TS to the revised program. The licensee shall submit this application, as specified in 10 CFR 50.4, at least six months before the start of the period during which the provisions become applicable, as determined by 10 CFR 50.55a(f)(4).

In 1990, the American Society of Mechanical Engineers (ASME) published the initial edition of the ASME Operation and Maintenance of Nuclear Power Plants (OM) Code, which provides rules for IST of pumps and valves. The OM Code was developed and is maintained by the

ASME Committee on Operation and Maintenance of Nuclear Power Plants. The OM Code was developed in response to the ASME Board on Nuclear Codes and Standards directive that transferred responsibility for development and maintenance of rules for the IST of pumps and valves from the ASME, Section XI, Subcommittee on Nuclear Inservice Inspection to the ASME OM Committee. The ASME intended the OM Code to replace Section XI rules for IST of pumps and valves, and the rules for IST of pumps and valves have been deleted from Section XI. The DEK fourth 10-year interval IST program was updated to comply with the 1998 Edition through the 2000 Addenda of the OM Code as required by 10 CFR 50.55a(f)(4)(ii).

NUREG-1431, Revision 3.0, published in March 2004, contains the ISTS for Westinghouse plants. NUREG-1431, Revision 3.0 was modified via TSTF-479 in December 2005 and TSTF-497 in October 2006. TSTF-479 addressed changes to reflect revision of 10 CFR 50.55a, and TSTF-497 addressed changes to ISTS Section 5.5.8, "IST Program," to reflect the NRC position that the provisions of SR 3.0.2 are applicable to other normal and accelerated frequencies specified as two years or less for IST activities.

3.0 TECHNICAL EVALUATION

3.1 Specific Changes Requested

The amendment request would convert the CTS to ITS consistent with the ISTS described in NUREG-1431, Revision 3.0. With respect to IST requirements, the proposed changes would remove CTS Section 4.2, and relocate the IST program portion to ITS Section 5.5.6. The proposed changes would also incorporate the approved modifications identified in TSTF-479 and TSTF-497. Specifically, the proposed ITS Section 5.5.6 (as shown in bold) is as follows:

5.5.6 INSERVICE TESTING PROGRAM

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves.

a. Testing frequencies applicable to the ASME Code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing testing activities
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once 731 days

b. The provisions of SR 3.0.2 are applicable to the above required frequencies and to other normal and accelerated frequencies

specified as two years or less in the Inservice Testing Program for performing inservice testing activities

c. The provisions of SR 3.0.3 are applicable to inservice testing activities

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

3.2 Evaluation

In 1990, the ASME published the initial edition of the OM Code, which provides rules for IST of pumps and valves. The OM Code was developed and is maintained by the ASME Committee on Operation and Maintenance of Nuclear Power Plants. The OM Code was developed in response to the ASME Board on Nuclear Codes and Standards directive that transferred responsibility for development and maintenance of rules for the IST of pumps and valves from the ASME Code, Section XI, Subcommittee on Nuclear Inservice Inspection to the ASME OM Committee. The ASME intended the OM Code to replace Section XI rules for IST of pumps and valves, and the rules for IST of pumps and valves have been deleted from Section XI.

Section 50.55a(f) of 10 CFR, "Inservice Testing Requirements," requires, in part, that ASME Code Class 1, 2, and 3 pumps and valves meet the testing requirements of the OM Code. The DEK fourth 10-year interval IST program was updated to comply with the 1998 Edition through the 2000 Addenda of the OM Code as required by 10 CFR 50.55a(f)(4)(ii).

The proposed ITS Section 5.5.6 replaces the IST portion of the CTS Section 4.2. The new ITS Section 5.5.6 references the applicable OM Code and is consistent with 10 CFR 50.55a(f)(4)(ii). The changes do not eliminate any inservice tests and do not relinquish the licensee of its responsibility to seek relief from Code test requirements. The proposed changes will maintain consistency with the OM Code requirements and implement an IST Program that is consistent with TSTF-479, TSTF-497 and NUREG-1434, Revision 3.0. Therefore, the NRC staff finds the proposed changes to be administrative in nature and acceptable. In addition, certain test frequencies have been deleted from the ITS 5.5.6 for IST program. The NRC staff finds the deletion acceptable since the deleted frequencies are not used in the OM Code.

The proposed changes will allow the application of 25 percent extension provided for in SR 3.0.2 to specified normal and accelerated SR frequencies in the TS, but will limit the applicability to test intervals of two years or less. This is consistent with the intent that the extension would provide operational flexibility, but would not significantly degrade the reliability that results from performing the surveillance at a specified frequency.

Further, the proposal to limit the applicability to frequencies of 2 years or less limits the maximum incremental time period, between surveillances, which could be added by the 25 percent extension, is consistent with the approved provisions in TSTF-497 and guidance provided in Section 6.0 of NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants." Without this limitation, some components, such as safety and relief valves, which may be tested at surveillance intervals greater than 2 years, could have extensions applied which would be much greater than needed for operational flexibility.

4.0 CONCLUSION TO SAFETY EVALUATION

The proposed TS changes comply with the requirements of 10 CFR 50.92, “No Significant Hazards Consideration,” since the TS amendment does not: (i) involve a significant increase in the probability or consequences of an accident previously evaluated, or (ii) create the possibility of a new or different kind of accident from any accident previously evaluated, or (iii) involve a significant reduction in margin of safety. DEK is authorized to implement the proposed changes.