SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 158 TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 149 TO FACILITY OPERATION LICENSE NO. DPR-60

NUCLEAR MANAGEMENT COMPANY, LLC

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

1.0 INTRODUCTION

By application dated December 11, 2000, as supplemented by letters dated March 6, June 5, July 3, August 13, August 29, October 15, November 12, and December 12, 2001, and January 25, January 31, February 14, February 15, February 16, March 6, April 11, May 10, May 30, June 7, June 25, and June 28, 2002, the Nuclear Management Company, LLC (the licensee), requested changes to the Technical Specifications (TSs) for the Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2. The proposed amendments would convert the current TSs (CTS) to improved TSs (ITS). The Commission published notice of the proposed action (the ITS conversion) in the *Federal Register* on June 25, 2002 (67 FR 42808).

PINGP has been operating with the TSs issued with the original Facility Operating Licenses dated April 5, 1974 (for Unit 1), and October 29, 1974 (for Unit 2), as amended. The proposed conversion to the ITS is based upon:

- NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," (STS) Revision 1, dated April 1995;
- The current PINGP 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 TSs for PINGP are referred to as the ITS, the existing TSs are referred to as the CTS, and the improved standard TSs, such as in NUREG-1431, are referred to as the STS. The corresponding Bases are ITS Bases, CTS Bases, and STS Bases, respectively. For convenience, a list of acronyms used in this safety evaluation (SE) is provided in Attachment 1 to this SE.

In addition to basing the ITS on the STS, 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 Nuclear Regulatory Commission (NRC) staff issued several requests for additional information (RAIs) and conducted a series of telephone conference calls and meetings with the licensee. These RAIs, meetings, and conference calls served to clarify the ITS with respect to the guidance in the Final Policy Statement and the STS. In addition, based on these discussions, the licensee also proposed changes of a generic nature that were not in the STS. The NRC staff requested that the licensee submit such generic changes as proposed changes to the STS through the NRC/Nuclear Energy Institute's Technical Specifications Task Force (TSTF). These generic issues were considered for specific applications in the PINGP ITS. Consistent with the Final Policy Statement, the licensee proposed transferring some CTS requirements to licensee-controlled documents (such as the PINGP Updated Safety Analysis Report (USAR)), for which changes to the documents by the licensee are controlled by a regulation (e.g., 10 CFR 50.59) and which may be changed without prior NRC approval. NRC-controlled documents, such as the TSs, 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 amendments, consistent with the Final Policy Statement, is to rewrite, reformat, and streamline the TSs for PINGP, while still satisfying the requirements of 10 CFR 50.36. During its review, the NRC staff relied on the Final Policy Statement and the STS as guidance for acceptance of CTS changes. This SE provides a summary basis for the NRC staff's conclusion that the licensee can develop ITS based on STS, as modified by plant-specific changes, and that the use of the ITS is acceptable for continued operation of PINGP. This SE also explains the NRC staff's conclusion that the ITS, which are based on the STS as modified by plant-specific changes, are consistent with the PINGP current licensing basis and the requirements of 10 CFR 50.36.

The license conditions included in the proposed amendments will make enforceable the following aspects of the conversion: (1) the schedule for the first performance of new and revised surveillance requirements (SRs) (four conditions); (2) the relocation of CTS requirements into licensee-controlled documents as part of the implementation of the ITS; and (3) the schedule for completion of actions associated with verifying the maximum test face velocity for the ventilation systems included in ITS Section 5.5.9.

The NRC staff also acknowledges that, as indicated in the Final Policy Statement, the conversion to ITS is a voluntary process. Therefore, it is acceptable that the ITS differ from the STS to reflect the current licensing basis for PINGP. The NRC staff approves the licensee's changes to the CTS with the modifications documented in the licensee's supplemental submittals.

For the reasons stated *infra* in this SE, the NRC staff finds that the ITS issued with these license amendments comply with Section 182a of the Atomic Energy Act, 10 CFR 50.36, and the guidance in the Final Policy Statement, and that they are in accordance with the common defense and security and provide adequate protection of the health and safety of the public.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act requires that applicants for nuclear power plant operating licenses will state:

> [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 TSs. 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 TSs "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TSs.

For several years, NRC and industry representatives have sought to develop guidelines for improving the content and quality of nuclear power plant TSs. 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 improved STS (e.g., NUREG-1431) that would establish models of 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 TSs, which gives greater consideration to human factors principles and was used throughout the development of licensee-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 STS in NUREG-1431 were established as a model for developing the ITS for Westinghouse plants, in general. The STS reflect the results of a detailed review of the application of the Interim Policy Statement criteria to generic system functions, which were published in a "Split Report" issued to the nuclear steam supply system vendor owners groups in May 1988. STS also reflect the results of extensive discussions concerning various drafts of STS 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 STS present requirements that are necessary to protect public health and safety. The STS in NUREG-1431, Revision 1, as modified, apply to PINGP.

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 STS and encouraged licensees to use the STS as the basis for plant-specific TS amendments and for complete conversions to ITS based on the STS. In addition, the Final Policy Statement gave guidance for evaluating the required scope of the TSs and defined the guidance criteria to be used in determining which of the LCOs and associated SRs should remain in the TSs. The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TSs, 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:

> [T]here 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 TSs; 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 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 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 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.*

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

Part 3.0 of this SE explains the NRC staff's conclusion that the conversion of the PINGP CTS to ITS based on STS, as modified by plant-specific changes, is consistent with the PINGP current licensing basis and the requirements and guidance of the Final Policy Statement and 10 CFR 50.36.

3.0 EVALUATION

In its review of the PINGP ITS application, the NRC staff evaluated five kinds of changes to the CTS 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.

LR Less Restrictive-Relocated Details - Changes to the CTS that eliminate detail and relocate the detail to a licensee-controlled document. Typically, this involves details of system design, system description including design limits, description of system or plant operation, procedural detail for meeting TS requirements and relocated reporting requirements, and redundant requirement references.

R Relocated Specifications - Changes to the CTS that relocate the 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 STS requirements in a numbered justification for deviation (JFD).

In its review, the NRC staff identified the need for clarifications and additions to the December 11, 2000, ITS application in order to establish an appropriate regulatory basis for translation of CTS requirements into ITS. The NRC staff's comments were documented as RAIs and forwarded to the licensee by letters dated June 1, June 15, August 17, December 6, December 19, December 21, 2001, and January 8, 2002. The licensee provided responses to the RAIs in supplemental letters dated March 6, July 3, August 13, November 12, and December 12, 2001, and January 25, January 31, February 14, February 15, February 16, March 6, April 11, May 10, May 30, June 7, June 25, and June 28, 2002. The letters clarified the licensee's basis for translating the CTS requirements into ITS. For items that have been reviewed by the NRC staff as stated in this SE, the NRC staff finds that the licensee's submittals, including the responses to the RAIs, provide sufficient detail to allow the NRC staff to reach a conclusion regarding the adequacy of the licensee's proposed changes to the CTS.

The changes to the CTS, as presented in the ITS application, are listed and described in the following six tables (for each ITS section) attached to this SE:

- Table A - Administrative Changes
- Table M - More Restrictive Changes
- Table L - Less Restrictive Changes
- Table LR- Less Restrictive-Relocated Details
- Table R - Relocated Specifications
- Table U - Unused Numbers

These tables, except Table U, provide a summary description of the proposed changes to the CTS, references to the specific CTS requirements that are being changed, and the specific ITS requirements that incorporate the changes. 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. As noted in the table headers, CTS DOCs are not sequentially numbered. Table U, therefore, provides a convenient list for unused numbers (of CTS changes) for each ITS section.

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

- Section A Administrative Changes
- Section B More Restrictive Changes
- Section C Less Restrictive Changes
- Section D Less Restrictive-Relocated Details
- Section E Relocated Specifications

The control of specifications, requirements, and information relocated from the CTS is described in Section F below, and other CTS changes (i.e., beyond-scope changes) are described in Section G below.

A. Administrative Changes to the CTS

Administrative (nontechnical) 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. Every section of the ITS reflects this type of change. In order to ensure consistency, the NRC staff and the licensee have used the STS as guidance to reformat and make other administrative changes. Among the changes proposed by the licensee and found acceptable by the NRC staff are:

- Identifying plant-specific wording for system names, etc.;
- Splitting up requirements currently grouped under a single current specification and moving them to more appropriate locations in two or more specifications of the ITS;
- Combining related requirements currently presented in separate specifications of the CTS into a single specification of ITS;
- Presentation changes that involve rewording or reformatting for clarity (including moving an existing requirement to another location within the TSs) 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 TSs that no longer apply;
- Deletion of details that are strictly informational and have no regulatory basis; and
- Deletion of redundant TS requirements that exist elsewhere in the TSs.

Table A lists the administrative changes being made in the PINGP ITS conversion. Table A is organized in STS 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 Writer's Guide and the STS, do not result in any change in operating requirements, and are consistent with the Commission's regulations.

B. <u>More Restrictive Changes to the CTS</u>

The licensee, in electing to implement the specifications of the STS, proposed a number of requirements more restrictive than those in the CTS. The ITS requirements in this category include requirements that are either new, more conservative than corresponding requirements in the CTS, or have additional restrictions that are not in the CTS but are in the STS. Examples of more restrictive requirements are placing an LCO on plant equipment that is not required by the CTS, more restrictive requirements to restore inoperable equipment, and more restrictive SRs. Table M lists the more restrictive changes being made in the PINGP ITS conversion. Table M is organized in STS order by each M-type DOC to the CTS and provides a summary description of the more restrictive change that was adopted, and the CTS and ITS references. These changes are additional restrictions on plant operation that enhance safety and are acceptable.

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

Less restrictive requirements include deletions 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 TSs 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 staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups' comments on

STS. The NRC staff reviewed generic relaxations contained in the STS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The PINGP 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 STS and thus provide a basis for ITS.

All of the less restrictive changes to the CTS have been evaluated and found to involve deletions and relaxations to portions of CTS requirements that can be grouped in the following nine types:

Type 1 — Relaxation of LCO Requirement
Type 2 — Relaxation of Applicability
Type 3 — Relaxation of SR
Type 4 — Relaxation of Required Action
Type 5 — Relaxation of Completion Time
Type 6 — Deletion of Requirements Redundant to Regulations or Design Information
Type 7 — Relaxation of Surveillance Frequency from 18 months to 24 months
Type 8 — Relaxation of CTS 3.0.C Requirements
Type 9 — Relaxation of SR Frequency

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

Type 1 — Relaxation of LCO Requirement

Certain CTS LCOs specify operational and system parameters beyond those necessary to meet safety analysis assumptions and therefore are considered overly restrictive. The CTS also contain 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 STS, would delete or revise operating limits of this type. CTS LCO changes of this type include: (1) redefining operating modes, including mode title changes; (2) deleting or revising operational limits to establish requirements consistent with applicable safety analyses; (3) deleting requirements for equipment or systems which establish system capability beyond that assumed to function by the applicable safety analyses or which are implicit to the ITS requirement for systems, components, and devices to be operable; and (4) adding allowances to use administrative controls on plant devices and equipment during times when automatic control is required or to establish temporary administrative limits, as appropriate, to allow time for systems to establish equilibrium operation. 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. These changes are consistent with the guidance established by the STS in consideration of the PINGP current licensing basis and, in view of the above, are acceptable.

Type 2 — Relaxation of Applicability

The CTS require compliance with the LCO during the applicable Mode(s) or other conditions specified in the Specification statement. When CTS Applicability requirements are inconsistent with the applicable accident analyses assumptions for a system, subsystem, or component specified in the LCO, the LCO would be changed in the ITS to establish a consistent set of requirements. These modifications or deletions are acceptable because, during the conditions referenced in the ITS, the operability requirements are consistent with the applicable safety analyses. These changes are consistent with the guidance established by the STS in consideration of the PINGP current licensing basis and, in view of the above, are acceptable.

Type 3 — Relaxation of Surveillance Requirement

The CTS require maintaining LCO equipment operable by meeting SRs in accordance with specified SR frequencies. This requires conducting tests to demonstrate (1) equipment is operable or (2) LCO 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 type relate to relaxation of CTS SR acceptance criteria and/or the conditions for performing the SR.

Relaxing the SR acceptance criteria for these items provides operational flexibility, consistent with the objective of the STS, without reducing confidence that the equipment is operable. For example, the ITS would permit the use of an actual, as well as a simulated, actuation signal to satisfy SRs for automatically actuated systems. The CTS do not allow for the use of an actual actuation signal to satisfy SRs. TS-required features cannot distinguish between an “actual” signal and a “test” signal. The changes to TS acceptance criteria are acceptable because appropriate testing standards are retained for determining that the LCO-required features are operable.

Relaxing conditions for performing SRs include not requiring testing of deenergized equipment (e.g., instrumentation channel checks) or equipment that is already performing its intended safety function (e.g., position verification of valves locked in their safety actuation position). The changes also allow verification of the position of valves in high radiation areas by administrative means. ITS administrative controls (ITS 5.7) regarding access to high radiation areas make the likelihood of mispositioning valves small. These changes are acceptable because the changes do not affect the ability to determine whether equipment is capable of performing its intended safety function. These relaxations of CTS SRs optimize test requirements for the affected safety systems and increase operational flexibility. These changes are consistent with the guidance established by the STS in consideration of the PINGP current licensing basis and, in view of the above, are acceptable.

Type 4 — Relaxation of Required Action

LCOs are the lowest functional capability or performance level of equipment required for safe operation of the facility. When an LCO is not met, the CTS specify actions to be taken until the equipment is restored to its required capability or performance level, or remedial measures are established. 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.) For example, changes of this type include providing an option to (1) isolate a system, (2) place equipment in the state assumed by the safety analysis, (3) satisfy alternate criteria, (4) take manual actions in place of automatic actions, (5) “restore to operable status” within a specified timeframe, (6) place alternate equipment into service, or (7) use more conservative TS setpoints. The resulting ITS actions continue to provide measures that conservatively compensate for the inoperable equipment. The ITS actions are commensurate with safety importance of the inoperable equipment, plant design, and industry practice and do not compromise safe operation of the plant. These changes are consistent with the guidance established by the STS in consideration of the PINGP current licensing basis and, in view of the above, are acceptable.

Type 5 — Relaxation of Completion Time

Upon discovery of a failure to meet an LCO, the TS specify times for completing Required Actions of the associated TS conditions. Required Actions establish remedial measures that must be taken within specified completion times (allowed outage times). These times define limits during which operation in a degraded condition is permitted. Incorporating completion time extensions is acceptable because completion times take into account the operability status of the redundant systems of TS required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, vendor-developed standard repair times, and the low probability of a design-basis accident (DBA) occurring during the repair period. These changes are consistent with the guidance established by the STS in consideration of the PINGP current licensing basis and, in view of the above, are acceptable.

Type 6 — Deletion of Requirements Redundant to Regulations or Design Information

The CTS contain requirements that are redundant to regulations in 10 CFR. The CTS include requirements that a “Reportable Event” is any of those conditions specified in 10 CFR 50.73. However, consistent with the STS, the ITS would omit many of the CTS reporting requirements because the reporting requirements in the regulations cited do not need repeating in the TSs to ensure timely submission to the NRC. Therefore, this type of change has no impact on the safe operation of the plant. Deletion of these requirements is beneficial because it reduces the administrative burden on the licensee and in turn allows increased attention to plant operations important to safety. These changes are consistent with the guidance established by the STS in consideration of the PINGP current licensing basis and, in view of the above, are acceptable.

The CTS contain design information that is deleted in ITS. In conformance with the guidance of the STS, ITS fuel assembly design features are simplified. The ITS would include allowances for changes to core designs if certain conditions are met. The ITS would allow limited substitutions of filler rods for fuel rods if fuel assemblies comply with fuel safety design bases, and the installation of a limited number of untested lead test assemblies in nonlimiting core regions. All fuel assemblies with substitutions must be evaluated in accordance with NRC-approved codes and methods or tested to show they comply with all fuel safety design bases. Lead test assemblies will not challenge any reactor operating limits since, by the requirements of this specification, the substitutions are restricted to nonlimiting core regions. Additionally, core performance is monitored throughout the operating cycle to assure that the plant performs safely. Thus, public health and safety will be adequately protected if the plant is operated in accordance with the provisions of ITS 4.2.1, which implements guidance provided in Generic Letter (GL) 90-02, "Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications," Supplement 1. These changes are consistent with the guidance established by the STS in consideration of the PINGP current licensing basis and, in view of the above, are acceptable.

Type 7 — Relaxation of Surveillance Frequency from 18 months to 24 months

The CTS require maintaining LCO equipment operable by conducting SRs in accordance with specified SR intervals. The changes of this type relate to extending SR test intervals. Improved reactor fuels allow the licensee to consider an increase in the duration of the fuel cycle for the facility. The TSs that specify an 18-month surveillance interval or require surveillance every refueling interval or during shutdown would be changed to specify a 24-month interval. The CTS 4.0.A (ITS SR 3.0.2) provision to extend surveillances by 25 percent of the specified interval would extend the time limit for completing these surveillances from the CTS limit of 22.5 months to a maximum of 24 months. The NRC staff review of these items is covered in more detail in Section 3.0.G of this SE. These changes are consistent with the guidance established by the STS in consideration of the PINGP current licensing basis and, in view of the above, are acceptable.

Type 8 — Relaxation of CTS 3.0.C Requirements

CTS 3.0.C (ITS LCO 3.0.3) establishes actions that must be implemented when an LCO is not met and either an associated Required Action or Completion Time is not met and no other Condition applies, or the condition of the unit is not specifically addressed by the associated TS Actions. This specification delineates the time limits for placing the unit in a safe Mode or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its Actions. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. Unless otherwise stated, LCO 3.0.3 is always applicable to ITS LCO Actions. However, new Required Actions would be included within the Actions of ITS LCOs that provide guidance for placing the plant in a specified condition or applicable Mode in which the LCO does not apply without requiring entry into LCO 3.0.3 which would require a shutdown to Mode 5. These new remedial actions would require the licensee to place the plant in a safe condition in a controlled manner, thus reducing the likelihood that additional structures, systems, or components will be unavailable to mitigate operational occurrences or plant transients. Therefore, these proposed changes do not impact safe operation of the plant. These changes are consistent with the guidance established by the STS in consideration of the PINGP current licensing basis and, in view of the above, are acceptable.

Type 9 — Relaxation of SR 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 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 testing 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 would include relaxing both the acceptance criteria and the conditions of performance. Also, the ITS would permit the use of an actual, as well as a simulated, actuation signal to satisfy SRs for automatically actuated systems. This is acceptable because TS-required features cannot distinguish between an "actual" signal and a "test" signal. 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 STS in consideration of the PINGP current licensing basis.

For the reasons presented above, these less restrictive changes to the CTS are acceptable because they will not affect the safe operation of the plant. The ITS requirements are consistent with the current licensing basis, operating experience, and plant accident and transient analyses, and provide reasonable assurance that public health and safety will be protected.

Table L lists the less restrictive changes being made in the PINGP ITS conversion. Table L, which is organized in STS 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. The NRC staff reviewed all of the less restrictive changes proposed by the licensee and finds them acceptable because they are compatible with the STS, do not result in any change in operating requirements, and are consistent with the Commission's regulations.

D. Less Restrictive-Relocated Details

When requirements have been shown to give little or no safety benefit, their removal from the TSs 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 staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the owners groups' comments on STS. The NRC staff reviewed generic relaxations contained in the STS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The PINGP 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 STS and thus provide a basis for ITS. A significant number of changes to the CTS involved the removal of specific requirements and 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 USAR by 10 CFR 50.34. In addition, the quality assurance (QA) 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 QA plan (USAR Appendix C). 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 QA plan. The Technical Requirements Manual (TRM) is a general reference in the USAR and is subject to administrative controls that include the requirement to perform evaluations for changes made to the TRM consistent with the requirements specified in 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 from the CTS is acceptable because this information will be adequately controlled in the USAR in accordance with 10 CFR 50.59 or the ITS Bases, as appropriate. 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, "Administrative Controls," 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 or Plant Operation

The plans for normal and emergency operation of the facility are required to be described in the USAR by 10 CFR 50.34. Specifications 5.4.1.a and 5.4.1.e require written procedures to be established, implemented, and maintained for plant operating procedures recommended in Appendix A of Regulatory Guide (RG) 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, dated February 1978, and in all programs specified in Specification 5.5, respectfully. 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 ITS Bases. ITS 5.5.12 specifies controls for changing the Bases. It is acceptable to remove details of system operation from the TSs because this type of information will be adequately controlled in the USAR (which references the TRM), in the TS Bases, and in Specification 5.5, "Programs and Manuals," as appropriate. 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

Details for performing TS SRs are more appropriately specified in the plant procedures. Prescriptive procedural information in an ITS requirement is unlikely to contain all procedural considerations necessary for the plant operators to comply with TSs, and referral to plant procedures is therefore required in any event. Changes to procedural details include those associated with limits retained in the ITS. For example, Specification 5.4.1 requires that written procedures covering activities that include all programs specified in Specification 5.5 be established, implemented, and maintained. The Inservice Testing (IST) Program is required by Specification 5.5.7. ITS 5.5.7, "Inservice Testing Program," requires a program to provide controls for IST of American

Society of Mechanical (ASME) Code Class 1, 2, and 3 components. The program includes defining testing frequencies specified in Section XI of the ASME *Boiler and Pressure Vessel Code,* and applicable addenda. The CTS also contain requirements to test specific components such as pumps and valves, and which establish IST of Quality Group A, B, and C pumps and valves performed in accordance with the requirements for ASME Code Class 1, 2 and 3 components specified in Section XI of the applicable ASME *Boiler and Pressure Vessel Code* edition and addenda, subject to the applicable provisions of 10 CFR 50.55a. Therefore, it is acceptable to remove Type 3 details from the CTS and place them in licensee-controlled documents.

Type 4 - Relocated Redundant Requirements

Certain CTS administrative requirements are redundant to 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 CTS requirements that do not meet any of the criteria for mandatory inclusion in the TSs. Changes to the facility or to procedures as described in the USAR are made in accordance with 10 CFR 50.59. Changes made in accordance with the provisions of other licensee-controlled documents are subject to the specific requirements of those documents. For example, 10 CFR 50.54(a) governs changes to the QA Plan, and ITS 5.5.12 governs changes to the ITS Bases. Therefore, it is acceptable to remove Type 4 details from CTS and place them in licensee-controlled documents.

Table LR lists the less restrictive removal of detail changes being made in the PINGP ITS conversion. Table LR is organized in STS order by each LR-type DOC and includes the following:

(1) the DOC identifiers, formatted as DOC Type (e.g., LR), followed by the Chapter/Section number (e.g., 3.4), followed by a designator number (e.g., 74);
(2) a summary description of the relocated details and requirements;
(3) the name of the licensee-controlled document to contain the relocated details and requirements (location);
(4) the regulation (or ITS Specification) for controlling future changes to relocated requirements (change control process);
(5) the reference numbers of the associated CTS requirements; and
(6) a characterization of the type of change.

The NRC staff has concluded that these types of detailed information and specific requirements do not need to be included in the ITS to ensure the effectiveness of the ITS to adequately protect the health and safety of the public. Accordingly, these requirements may be moved to one of the following licensee-controlled documents for which changes are adequately governed by a regulatory or TS requirement:

- Bases controlled in accordance with ITS 5.5.12, “Technical Specifications (TS) Bases Control Program.”
- USAR (which references the TRM) controlled by 10 CFR 50.59.
- Programmatic documents required by ITS Section 5.5 and controlled by ITS Section 5.4.

- Inservice Inspection (ISI) and IST Programs controlled by 10 CFR 50.55a.
- Offsite Dose Calculation Manual (ODCM) controlled by ITS 5.5.1.
- COLR controlled by ITS 5.6.5.
- QA Plan, as approved by the NRC, referenced in the USAR, and controlled by 10 CFR Part 50, Appendix B, and 10 CFR 50.54(a).
- Site Emergency Plan controlled by 10 CFR 50.54(q).

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 CTS, 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.

E. Relocated Specifications

The Final Policy Statement 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 TSs (an NRC-controlled document) 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 a licensee-controlled document (i.e., TRM), as appropriate. The NRC staff has reviewed the licensee's submittals and finds that relocation of these requirements to a licensee-controlled document 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 lists the relocated changes that would be made in the PINGP ITS conversion and lists all specifications that are being relocated from the CTS to licensee-controlled documents. Table R includes:

(1) references to the DOCs;
(2) references to the relocated CTS requirements;
(3) summary descriptions of the relocated CTS requirements;
(4) names of the documents that will contain the relocated specifications (i.e., the new location); and
(5) the methods for controlling future changes to the relocated specifications (i.e., the regulatory control process).

The NRC staff's evaluation of each relocated specification listed in Table R is provided below, mostly in CTS order. New locations for relocated CTS are listed in Table R.

3.11 Core Surveillance Instrumentation

The core surveillance instrumentation would be relocated to the TRM. The CTS require that the moveable detector and core thermocouple instrumentation systems be operable. The moveable detector system is required to be operable following each refueling so that the power distribution can be confirmed. In addition, sufficient detectors, drives, and readout equipment to map these thimbles are required to be operable during recalibration of the excore axial offset detection system per the TSs. Instrumentation systems retained in TSs are relied upon to form a part of the primary success path 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. Credit for this instrumentation is not assumed in the safety analysis or in the Individual Plant Evaluation (IPE) (a site-specific probabilistic risk assessment) since it is an insignificant risk contributor. Accordingly, the moveable detector and core thermocouple instrumentation systems do not meet the requirements for inclusion in TSs and can be relocated to the TRM.

2.3.C, Control Rod Withdrawal Stops Instrumentation

The Control Rod Withdrawal Stops, P-2 interlock would be relocated to the TRM. This instrumentation is provided to prevent movement of rods using automatic rod controls. When the plant power level is less than 15 percent rated thermal power, the turbine impulse pressure (PT-485) input to the rod control system is blocked, preventing automatic rod withdrawal. This interlock enables automatic rod control when the power level reaches 15 percent. Instrumentation systems retained in TSs are relied upon to form a part of the primary success path 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. Credit for the P-2 interlocks is not assumed in the safety analysis. Therefore, these automatic stops do not meet the requirements for inclusion in TSs and can be relocated to the TRM.

Table 4.1-1C, Function 13, Containment Sump A, B, and C Level

The containment sump level instrumentation would be relocated to the TRM, except for containment sump B wide range indication, which would be included in ITS LCO 3.3.3. The containment sump level instruments are installed instrumentation that is capable of indicating in the control room a significant abnormal degradation of the reactor coolant pressure boundary. However, this instrumentation is less sensitive to reactor coolant pressure boundary leakage than other leakage instrumentation required by ITS LCO 3.4.16, and is not included in the safety analysis of reactor coolant system leakage detection systems. Credit for the containment sump level instrumentation is not assumed in the safety analysis or the IPE. Since containment sump level instruments are not required in ITS, the SRs on this instrumentation can be relocated to the TRM.

Table 4.1-1C, Function 16, Emergency Plan Radiation Instruments

The emergency plan radiation instrumentation would be relocated to the TRM. The emergency plan radiation instruments are used to gather environmental information following an accident that requires entry into the emergency plan. Instrumentation systems retained in TSs are relied upon to form a part of the primary success path 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. Credit for the emergency plan radiation instruments is not assumed in the safety analysis or the IPE. Since emergency plan radiation instruments are not required in ITS, the SRs on this instrumentation can be relocated to the TRM.

Table 4.1-1C, Function 17, Seismic Monitors

The seismic monitors (instrumentation) would be relocated to the TRM. The seismic monitors are used to record data for use in evaluating the effect of a seismic event. Instrumentation systems retained in TSs are relied upon to form a part of the primary success path 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. Credit for the seismic monitors is not assumed in the safety analysis or the IPE. Since seismic instruments are not required in ITS, the SRs on this instrumentation can be relocated to the TRM.

Table 4.1-1C, Function 19, CRDM Cooling Shroud Exhaust Air Temperature

The CRDM cooling shroud exhaust air temperature instrumentation would be relocated to the TRM. The CRDM cooling shroud exhaust air temperature instrumentation is installed instrumentation for indicating the cooling air temperature above the reactor pressure vessel. Instrumentation systems retained in TSs are relied upon to form a part of the primary success path 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. Credit for the CRDM cooling shroud exhaust air temperature instrumentation is not assumed in the safety analysis or the IPE. Since CRDM cooling shroud exhaust air temperature instruments are not required in ITS, the SRs on this instrumentation can be relocated to the TRM.

Table 4.1-1C, Function 20, Reactor Gap Exhaust Air Temperature

The reactor gap exhaust air temperature instrumentation would be relocated to the TRM. The reactor gap exhaust air temperature instrumentation is installed instrumentation for indicating the cooling air temperature around the reactor pressure vessel. Instrumentation systems retained in TSs are relied upon to form a part of the primary success path 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. Credit for the reactor gap exhaust air temperature instrumentation is not assumed in the safety analysis or the IPE. Since reactor gap exhaust air temperature instruments are not required in ITS, the SRs on this instrumentation can be relocated to the TRM.

Table 4.1-1C, Function 31, Turbine Overspeed Protection Trip Channel

The turbine overspeed protection trip instrumentation would be relocated to the TRM. The turbine overspeed protection trip instrumentation provides a means to detect a turbine overspeed condition and trip the turbine. Instrumentation systems retained in TSs are relied upon to form a part of the primary success path 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. Credit for the turbine overspeed protection trip instrumentation is not assumed in the safety analysis or the IPE. Since turbine overspeed protection trip channels are not required in ITS, the SRs on this instrumentation can be relocated to the TRM.

3.4.C, Table 3.5-1 Function 8, High Temperature in Ventilation Ducts
4.8.C and Table 4.1-1C Function 24, The Steam Exclusion System

The steam exclusion system (SES) actuation instrumentation and the associated setpoint would be relocated to the TRM. The SES is an installed system which monitors auxiliary building and turbine building ventilation duct temperatures and, upon a high temperature condition due to a high-energy line break, isolates the ducts and prevents steam from reaching safeguards equipment. The SES is a plant system. Credit for the SES actuation instrumentation and the associated setpoint is not assumed in the safety analysis or the IPE. Accordingly, the SES actuation instrumentation and the associated setpoint do not meet the requirements for inclusion in TSs and can be relocated to the TRM.

3.1.A.3 and 4.18 Reactor Vessel Head Vent System

The reactor vessel head vent system would be relocated to the TRM. The CTS require that the reactor shall not be made or maintained critical nor shall the reactor coolant system (RCS) average temperature exceed 200 degrees Fahrenheit unless the reactor head vent system paths from both the reactor vessel head and pressurizer steam space are OPERABLE and closed. These vents are designed to exhaust from the RCS noncondensable gases and steam that could inhibit natural circulation following an accident with an extended loss of offsite power. Credit for this vent system is not assumed in the safety analyses or in the IPE. Accordingly, the requirements of the reactor vessel head vent system LCO do not meet the requirements for inclusion in TSs and can be relocated to the TRM.

3.1.B.2 Pressurizer

The pressurizer heatup and cooldown limits would be relocated to the Pressure Temperature Limits Report (PTLR). The shutdown requirements associated with the pressurizer heatup and cooldown limitations would be relocated to the TRM. The CTS require specific limits on the maximum heatup and maximum cooldown in any 1-hour period. The pressurizer heatup and cooldown limits are not process variables, design features, or operating restrictions that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Credit for the pressurizer heatup and cooldown limits is not assumed in accident analyses. Therefore, pressurizer heatup and cooldown limits are not operational limits that are an initial assumption of any DBA or transient analysis, and may be relocated to licensee-controlled documents.

3.1.B.3 Steam Generator

The steam generator pressure/temperature (P/T) limits would be relocated to the PTLR. The shutdown requirements associated with the steam generator P/T limits would be relocated to the TRM. The CTS require that the secondary side of the steam generator not be pressurized above 200 psig if the temperature of the steam generator is below 70 degrees Fahrenheit. This operating restriction does not present a challenge to the integrity of a fission product barrier and these limits are not required for safe operation of the facility. Credit for the steam generator P/T limits is not assumed in accident analyses. Therefore, steam generator limits are not operational limits that are an initial assumption of any DBA or transient analysis, and may be relocated to the PTLR.

3.8.C, Small Spent Fuel Pool Restrictions

The small spent fuel pool restrictions would be relocated to the TRM. The requirement limits the number of recently discharged fuel assemblies stored in the small pool (Pool 1). Spent fuel storage limits retained in TSs are relied upon as process variables that are initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Credit for limits on the number of recently discharged fuel assemblies stored in the small pool are not assumed in the safety analysis. Therefore, limits on the number of recently discharged fuel assemblies stored in the small pool do not meet the requirements for inclusion in TSs, and can be relocated to the TRM.

CTS 3.12, 4.13 and Table 4.13-1, Snubber

The snubber requirements would be relocated to the TRM. Snubbers are passive devices that are designed to prevent unrestrained pipe motion under dynamic loads and allow normal thermal expansion of piping and nozzles to eliminate excessive thermal stresses during heatup or cooldown. The TS action statement for snubbers only requires that an inoperable snubber be replaced or repaired. The SRs for snubbers require that they be periodically examined under the ISI Program. Credit for snubber operability is not assumed in the safety analysis. Accordingly, limits for snubbers do not meet the requirements for inclusion in TSs and can be relocated to the TRM.

Table 4.1-2A, Function 11, Turbine Stop Valves, Governor Valves and Intercept Valves

Turbine stop valve, governor valve, and intercept valve requirements would be relocated to the TRM. Periodic testing of the turbine stop valves, governor valves, and intercept valves are performed to reduce the probability of a turbine missile ejection incident. These SRs are part of the turbine overspeed protection system. The turbine overspeed protection trip instrumentation provides a means to detect a turbine overspeed condition and trip the turbine. Instrumentation systems retained in TSs are relied upon to form a part of the primary success path 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. Credit for the turbine overspeed protection trip instrumentation is not assumed in the safety analysis or the IPE. Since turbine overspeed protection trip channels are not required in ITS, the SRs on turbine stop valves, governor valves and intercept valves can be relocated to the TRM.

4.19, Auxiliary Building Crane Lifting Devices

The TS governing auxiliary building crane lifting devices would be relocated to the TRM. The testing required for the equipment in this specification verifies that special lifting devices and slings used in conjunction with the auxiliary building crane are operable prior to their use in supporting heavy loads over safe shutdown equipment or spent fuel in the spent fuel pool. Credit for the auxiliary building crane lifting devices is not assumed in the safety analysis. Accordingly, the auxiliary building crane lifting devices do not meet the requirements for inclusion in TSs and can be relocated to the TRM.

The specifications relocated from the CTS discussed above are not required to be in the TSs because they do not fall within the criteria for mandatory inclusion in the TSs 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. In addition, the NRC staff has concluded that appropriate controls have been established for all of the current specifications and information that are being moved to the TRM, ODCM, or ISI or IST Programs. These relocations are the subject of a new license condition discussed in Section 5.0 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 that control these documents. Following implementation, the NRC may audit the removed provisions to ensure that an appropriate level of control has been achieved. The NRC staff has concluded that, in accordance with the Final Policy Statement, sufficient regulatory controls exist under the regulations, particularly 10 CFR 50.59 and 10 CFR 50.55a. Accordingly, the specifications and information, as described in detail in this SE, may be relocated from the CTS and placed in the licensee-controlled documents identified in the licensee’s submittals.

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 3.0.D and 3.0.E above. The facility and procedures described in the USAR and TRM can be revised only in accordance with the provisions of 10 CFR 50.59, which ensure that records are maintained, and establish appropriate control over 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. For example, the ODCM can be changed only in accordance with ITS 5.5.1, and the administrative instructions that implement the QA Plan can be changed only in accordance with 10 CFR 50.54(a) and 10 CFR Part 50, Appendix B. The documentation of these changes will be maintained by the licensee in accordance with the record retention requirements specified in the QA Plan 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 5.0 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 Changes) Included in the Application for Conversion to ITS

This section evaluates other TS changes included in the licensee's ITS application. These include items that deviate from both the CTS and the STS, do not fall clearly into a category, or are in addition to those changes that are needed to meet the overall purpose of the conversion. These changes are termed beyond-scope issues (BSIs), which have been identified by the licensee in its ITS application, and by the NRC staff during the course of its review. These BSIs were included in the Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Opportunity for a Hearing published in the *Federal Register* on June 25, 2002 (67 FR 42808).

G.1 BSI Changes Identified by the Licensee:

The changes discussed below are licensee-identified BSIs and are listed in the order of the applicable ITS specification or section, as appropriate. Also provided are references to the associated DOC to the CTS and JFD from the STS given in the licensee's application.

G.1.1 Extension of Surveillance Intervals (From 18 months to 24 months)

As part of this conversion process, the licensee has proposed to clarify the requirements for those surveillances which are intended to be performed during plant shutdown conditions or on an operating cycle interval.

The current fuel cycle at PINGP cannot exceed 24 months and the CTS does not include a provision for any extension of this surveillance interval. The CTS 4.0.A.2 states that "The intervals between tests scheduled for refueling shutdowns shall not exceed two years." The proposed ITS SR 3.0.2 retains this CTS requirement by specifying in part: "...The specified Frequency is met for each SR with a specified Frequency of 24 months if the Surveillance is performed within 24 months, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met...."

The licensee has categorized each ITS SR with a 24-month frequency into three groups, depending on their origin and CTS frequency requirements. The first group of SRs (Table 1 below) have such CTS SR Frequency statements as: "each refueling shutdown," "R (each refueling shutdown)," "once each refueling interval," "each reactor refueling shutdown," "each refueling outage," or "each refueling shutdown on STB [staggered test basis]," indicating operating cycle intervals. PINGP could operate on 24-month cycles and still meet the CTS required test interval for these SRs. Thus, no change to the CTS is required for this group of SRs.

The second group of SRs (Table 2 below) have such CTS SR Frequency statements as: "once per 18 months," "once each 18 months," "every 18 months," "once per operating cycle, or once each 18 months whichever occurs first," or similar words indicating 18 month intervals. (The provision to extend surveillances by 25 percent of the specified interval (TS 4.0.A.1) would extend the time limit for completing these surveillances to a maximum of 22.5 months.)

The third group of SRs (Table 3 below) are those that are not included in the CTS.

<u>GROUP 1 SRs</u> - TABLE 1

ITS SR	TYPE OF SR	ITS SR	TYPE OF SR
3.3.1.10	Channel Calibration	3.5.2.5	Verify ECCS valve position
3.3.1.11	Channel Calibration	3.5.2.6	Verify ECCS pump starts
3.3.1.12	Channel Calibration	3.6.3.7	Verify Containment valve position
3.3.1.13	Channel Operational Test	3.6.5.3	Verify Containment cooling flow
3.3.1.14	TADOT	3.6.5.5	Verify CS valve position
3.3.1.16	RTS Response Time Test	3.6.5.6	Verify CS pump starts
3.3.2.4	TADOT	3.6.5.7	Verify Containment cooling starts
3.3.2.5	TADOT	3.6.6.4	Verify spray additive
3.3.2.6	Channel Calibration	3.6.7.1	SFT for Hydrogen Recombiner
3.3.3.3	Channel Calibration	3.6.7.2	VT for Hydrogen Recombiner
3.3.4.3	Channel Calibration	3.6.7.3	ET for Hydrogen Recombiner
3.3.5.5	TADOT	3.6.8.2	Channel Calibration
3.3.5.6	Channel Calibration	3.6.9.4	Verify SBVS damper actuations
3.4.1.3	Verify RCS flow rate (COLR limit)	3.7.7.2	Verify CC valve actuations
3.4.12.5	Channel Calibration	3.7.7.3	Verify CC pump starts
3.4.13.6	Channel Calibration	3.7.8.5	Verify CL valve actuations
3.4.15.1	Verify PIV leakage	3.7.8.6	Verify DDCLP and MDCLP starts

<u>GROUP 2 SRs</u> - TABLE 2

ITS SR	TYPE OF SR	ITS SR	TYPE OF SR
3.4.9.3	Verify Pressurizer Heater power source	3.7.12.4	Verify ABSVS train actuates
3.4.11.2	Cycling of PORVs	3.7.13.3	Verify SFPSVS train actuates
3.5.2.7	Verify ECCS throttle valve position	3.7.13.4	Verify SFPSVS fan flow rate
3.6.9.3	Verify SBVS train actuation	3.8.1.7	EDG load reject test
3.7.5.3	Verify AFW valve actuation/position	3.8.1.8	Verify EDG trip bypass on SI signal
3.7.5.4	Verify AFW pump starts	3.8.1.9	EDG 24 hour run test
3.7.10.3	Verify CRSVS train actuation	3.8.1.10	LOOP/SI actuation test
3.7.10.4	Verify CRSVS fan flow rate		

GROUP 3 SRs - TABLE 3

ITS SR	TYPE OF SR	ITS SR	TYPE OF SR
3.3.1.13	Channel Operational Test	3.6.2.2	Verify air lock doors
3.3.2.7	Master relay test	3.7.2.2	Verify MSIV actuation
3.3.2.8	Slave relay test	3.7.3.2	Verify MFRV and bypass valve actuation
3.3.5.6	Master-slave relay test	3.7.4.2	SG PORV block valve
3.3.6.3	TADOT	3.7.11.1	Verify SCWS loop actuation
3.3.6.4	Master-slave relay test	3.8.1.11	DG LOOP LOCA test
3.3.7.3	Channel Calibration	3.8.4.2	Battery charger load test
3.4.16.3	Channel Calibration	3.8.4.3	Verify Battery charger capacity
3.4.16.4	Channel Calibration	3.9.3.2	Channel Calibration
3.5.2.8	Verify ECCS train sump suction	3.9.4.2	Verify actuation (Containment purge valve)

The NRC staff has reviewed and approved a number of requests for other licensees to extend SRs to accommodate a 24-month fuel cycle. As described in the preceding section, the licensee's request for PINGP is somewhat different than the previous cases because (1) PINGP's CTS allow surveillance intervals up to 24 months and (2) the proposed ITS retain this specification without the provision to extend surveillance intervals by 25 percent (i.e., up to 30 months). Proposed ITS SR 3.0.3 states in part the following:

> "The specified Frequency is met for each SR with a specified Frequency of 24 months if the Surveillance is performed within 24 months, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met."
>
> "For Frequencies specified as "once," the above interval extension (1.25 times the interval specified) does not apply."

Therefore, no change to the CTS is required for a majority of the 24-month interval SRs in the proposed ITS. Specifically, the Group 1 SRs (Table 1) which have such CTS SR Frequency statements as: "each refueling shutdown," "R (each refueling shutdown)," "once each refueling interval," "each reactor refueling shutdown," "each refueling outage," or "each refueling shutdown on STB," requires no change to the CTS. PINGP could operate on 24-month cycles and still meet the CTS-required test interval for these SRs. However, due to inconsistencies in the CTS, a number of similar SRs have such CTS SR Frequency statements as: "once per 18 months"; "once each 18 months;" "every 18-months;" "once per operating cycle, or once each 18 months whichever occurs first," or similar words indicating 18 month intervals (Table 2 - Group 2 SRs). The provision to extend surveillances by 25 percent of the specified interval would extend the time limit for completing these surveillances to a maximum of 22.5 months, versus the 24-month maximum interval for the ITS. For this reason, the NRC staff requested additional justification from the licensee based on GL 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," issued on April 2, 1991.

GL 91-04 provides staff guidance that identifies the types of information that should be addressed when proposing extensions of the fuel cycle to 24 months. The GL identified the following information to support conversion to a 24-month operating cycle:

1) Licensees should evaluate the effect on safety of an increase in 18-month surveillance intervals to accommodate a 24-month fuel cycle. This evaluation should support a conclusion that the effect on safety is small.

2) Licensees should confirm that historical plant maintenance and surveillance data support this conclusion.

3) Licensees should confirm that assumptions in the plant licensing basis would not be invalidated on the basis of performing any surveillance at the bounding surveillance interval limit provided to accommodate a 24-month fuel cycle.

4) In consideration of these confirmations, the licensees need not quantify the effect of the change in surveillance intervals on the availability of individual systems or components.

In accordance with the above guidance, the licensee evaluated the effect on safety of an increase in 18-month surveillance intervals to 24-month surveillance intervals, and concluded that the effect on plant safety is small. The licensee has also concluded that the assumptions in the plant licensing basis would not be invalidated based on the performance of any surveillance at the bounding surveillance interval limits (not to exceed 24 months). In addition, the licensee has reviewed historical plant maintenance and surveillance data. As a result of this review, the licensee identified a total of three failures in three different unrelated systems. With the exception of the three identified failures, all other surveillance tests were passed. Those systems for which the surveillance tests passed maintained a 100-percent reliability. The three systems which have had a failed surveillance test maintained a 84-percent reliability. These systems passed their surveillance test when minor repairs were completed and the failed surveillance was rerun.

Based on this information, the NRC staff concludes that the impact of the proposed change for the Group 2 SRs on plant safety is small and, therefore, the proposed change is acceptable.

Group 3 SRs (Table 3) represent new requirements in the ITS which the licensee has adopted as part of the ITS conversion process. Since these are new SRs that are not required in the CTS, these are considered more restrictive changes to the CTS, and are, therefore, acceptable.

G.1.2 <u>Extension of Allowed Outage Time for Emergency Core Cooling System Accumulators</u>

As part of this conversion process, the licensee has proposed to revise CTS 3.3, “Engineered Safety Features,” to extend the allowed outage time (AOT) of an inoperable accumulator from 1 hour to 24 hours. This proposal is based on the methodology described in Topical Report WCAP-15049-A, “Risk-Informed Evaluation of an Extension to Accumulator Completion Times.“

The purpose of the emergency core cooling system (ECCS) accumulators is to supply water to the reactor vessel during the blow-down phase of a loss-of-coolant accident (LOCA). The accumulators are large volume tanks, filled with borated water and pressurized with nitrogen. The cover-gas pressure is less than that of RCS so that when the RCS pressure decreases below the tank pressure, the accumulators inject borated water into the RCS cold legs. The CTS have an allowed outage time (AOT) of 1-hour for one accumulator. However, the Westinghouse Owners Group (WOG) determined that a 1-hour AOT is insufficient for responding to accumulator inoperability. Therefore, the WOG submitted Topical Report WCAP-15049-A, which generically evaluated the risk associated with extending accumulator AOTs from 1 hour to 24 hours for reasons other than boron concentration out of specification at Westinghouse plants. In the report, the WOG did not request an extension of the accumulator boron out-of-specification TS, which for most plants had been set to 72 hours. They determined that the 72 hours were an adequate amount of time to correct the boron-related problems. The NRC reviewed and approved WCAP-15409-A for referencing in licensing applications in its Safety Evaluation Report (SER), “Acceptance for Referencing of Westinghouse Owners Group Topical Report WCAP-15049-A, ‘Risk-Informed Evaluation of an Extension to Accumulator Completion Times’,” dated February 19, 1999.

CTS 3.3.A.2.e allows for one of the two accumulators to be inoperable for 1 hour during startup operation or power operation when the pressurizer pressure is greater than 1000 psig. With one accumulator inoperable, the remaining accumulator will be available to mitigate the consequences of a LOCA. The licensee’s request to increase the AOT for one accumulator from 1 hour to 24 hours does not change the design or the operating characteristics of the accumulators. Therefore, the current PINGP USAR safety analysis that evaluates the operation of the accumulators remains unchanged by the extension of the 1-hour AOT to a 24-hour AOT for one accumulator. However, the change in the AOT will affect the overall risk at the plant.

One portion of this AOT extension request is not covered by WCAP-15049-A because this extension also applies to the condition where an accumulator is inoperable due to the accumulator boron concentration being out of specification. As mentioned above, the WOG did not propose extending the boron out of specification AOT for an inoperable accumulator. The WOG did not request this extension because many Westinghouse plants have this AOT already set to 72 hours, which is adequate to correct the accumulator boron out-of-specification issues. PINGP, on the other hand, had only one TS AOT for all accumulator inoperability conditions. An increase to the AOT of the boron out-of-specification condition of the TS is acceptable because the boron concentration in the accumulators is considered only during the recirculation phase of the LOCA, and the impact of a single accumulator’s borated water volume is not significant when compared to the total borated water volume present during the recirculation phase. Therefore, applying the AOT extension that the NRC found acceptable in WCAP-15049-A to all cases of an inoperable accumulator (including boron out of specification) for PINGP would not invalidate the current safety analysis, and the NRC staff finds the proposed change acceptable.

With its current PRA model, the licensee estimates the average internal events core damage frequency (CDF) for PINGP to be 2.2E-05/yr. According to Topical Report WCAP-15049-A, which was prepared to support requests for increases in accumulator AOTs to 24 hours, the change in the average CDF for a two-loop plant with an increase in accumulator AOT to 24 hours is approximately 3E-07/yr. Based on these levels of risk, RG 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," dated July 1998, indicates that the proposed change can be considered on a risk-informed basis. Topical Report WCAP-15409-A presented the conclusion that the change in CDF (approximately 3E-07/yr) is small enough, compared to the criteria set forth in RG 1.174, to be acceptable, and the NRC staff’s SER approving this topical (which is bound into the approved version of the report) came to the same conclusion. The NRC staff has not reviewed the licensee's PRA nor the details of its numerical analysis, and has not performed an independent analysis of the proposed change. However, the licensee's CDF appears reasonable since it is not too different from the CDF reported in the PINGP IPE (5E-05/yr) considering improvements to the PRA model that the licensee listed in the amendment request, and the NRC staff has no basis for concluding the current PRA model is not adequate to support this request. With regard to WCAP-15049-A, as mentioned earlier, it was reviewed and the risk analysis was found to be acceptable by the NRC staff for use, by reference, in licensing applications extending accumulator AOTs to 24 hours provided the plant design and operating characteristics are consistent with the limitations stated in the WCAP and associated NRC SER. This review does not repeat review of matters described in the WCAP-15049-A except to ensure that PINGP is within the stated limitations of the report and associated NRC SER.

The licensee discussed the limitations of WCAP-15049-A (expressed in terms of assumptions and parameters used in the WCAP) with respect to its applicability to the PINGP design and operating characteristics. PINGP is a two-loop plant and the WCAP includes analysis of a two-loop plant. The licensee does not perform any accumulator test activities or preventative maintenance activities at power and the accumulator corrective maintenance frequency is 0.1/yr (from 1995 - 2001), which is an assumption incorporated in the WCAP. As in the PINGP PRA, the initiating events for which accumulators are modeled in the WCAP are large, medium, and small LOCAs; for all LOCA categories, the frequencies used in the WCAP bound those used in PINGP PRA (which are probably more realistic, being based on data from NUREG/CR-5750). The success criteria assumed in the WCAP analysis are the same as those used in the current plant-specific PRA model. For these reasons, WCAP-15049-A is considered applicable to PINGP.

Therefore, the NRC staff has concluded that the proposed changes to TS 3.3.A.2.e for extending the AOT for the PINGP ECCS accumulators from 1 hour to 24 hours is acceptable. Additionally, the applicability of this TS was changed from whenever the pressurizer pressure is greater than 1000 psig to whenever the RCS pressure is greater than 1000 psig. Since the RCS pressure and the pressurizer pressure are identical, this change is editorial in nature and has no effect on plant safety. Therefore, the NRC staff finds this change acceptable.

G.1.3 Missed Surveillance Consolidated Line-Item Improvement

As part of the ITS application, the licensee proposed to adopt TSTF-358, "Missed Surveillances" (Consolidated Line-Item Improvement). The proposed ITS Surveillance Requirement (SR) 3.0.3 would allow a longer period of time before entering a limiting condition for operation (LCO) in the event of a missed surveillance. The time would be extended from the CTS limit of "... up to 24 hours or up to the limit of the specified Frequency, whichever is less" to "...up to 24 hours or up to the limit of the specified Frequency, whichever is greater." In addition, the following requirement would be added to ITS SR 3.0.3: "A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed." TSTF-358 is one of the industry's initiatives under the Risk-Informed TS Program that has been approved by the NRC staff as documented in the staff's model SE published in the *Federal Register* on June 14, 2001 (66 FR 32400). The NRC staff's model SE is provided below:

> In response to NRC staff's comments of February 14, 2000, the Nuclear Energy Institute (NEI) TSTF submitted Revision 5 to TSTF-358, "Missed Surveillance Requirements," to

the NRC for review and approval on September 15, 2000 (Revisions 2 - 4 were only reviewed by the industry and were never submitted for NRC review). The NRC staff published a "Notice of Opportunity to Comment on Model Safety Evaluation on Technical Specification Improvement to Modify Requirements Regarding Missed Surveillances Using the Consolidated Line Item Improvement Process" (66 FR 32400, June 14, 2001) in the *Federal Register* for public comment. In response to public comments received, the NEI TSTF submitted Revision 6 to TSTF-358 to the NRC for review and approval on September 14, 2001, and it was approved by the staff on October 1, 2001. The NRC staff has since made minor editorial changes to the safety evaluation.

The regulations contained in 10 CFR 50.36, "Technical Specifications," require that TSs include SRs. SRs are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met. TSs require surveillance tests to be performed periodically (e.g., weekly or monthly). The periodic test interval defined in the TSs is called the surveillance frequency or surveillance interval. The majority of surveillance tests included in the TSs are designed to ensure that standby safety systems will be operable when they are needed to mitigate an accident. By testing these components, failures that may have occurred since the previous test can be detected and corrected.

Standard TS (STS) SR 3.0.1 states that SRs shall be met during the MODES or other specified conditions in the applicability for individual LCOs and that failure to perform a surveillance within the specified frequency shall be a failure to meet the LCO, except as provided in SR 3.0.3.

The current STS SR 3.0.3 requires that, if it is found that a surveillance test was not performed within its specified frequency, the associated LCO be declared not met (e.g., equipment be declared inoperable) unless the missed surveillance test is completed successfully within 24 hours or within the limit of the specified frequency, whichever is <u>less</u>, from the time it was discovered that the test was not performed. The requirements in STS SR 3.0.3 are based on NRC Generic Letter 87-09, "Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) of the Applicability of Limiting Conditions for Operation and Surveillance Requirements," dated June 4, 1987.

Generic Letter 87-09 was published to address three specific issues with the application of TSs. One of those issues was missed surveillances. The Generic Letter states, "The second problem involves unnecessary shutdowns caused by Specification 4.0.3 when surveillance intervals are inadvertently exceeded. The solution is to clarify the applicability of the Action Requirements, to specify a specific acceptable time limit for completing a missed surveillance in certain circumstances, and to clarify when a missed surveillance constitutes a violation of the Operability Requirements of an LCO. It is overly conservative to assume that systems or components are inoperable when a surveillance has not been performed because the vast majority of surveillances do in fact demonstrate that systems or components are OPERABLE. When a surveillance is missed, it is primarily a question of operability that has not been verified by the performance of a Surveillance Requirement. Because the allowable outage time limits of some Action Requirements do not provide an appropriate time for performing a missed surveillance before Shutdown Requirements apply, <u>the TS[s] should include a time limit that allows a</u>

delay of required actions to permit the performance of the missed surveillance based on consideration of plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and, of course, the safety significance of the delay in completing the surveillance. The staff has concluded that 24 hours is an acceptable time limit for completing a missed surveillance when the allowable outage times of the Action Requirements are less than this limit, or when time is needed to obtain a temporary waiver[1] of the Surveillance Requirement." [emphasis added]

The proposed change would extend the delay time for declaring the LCO not met and entering the required actions by allowing more time to perform the missed surveillance test. This will be achieved by modifying SR 3.0.3 to allow a delay period from 24 hours up to the surveillance frequency, whichever is greater, to perform a missed surveillance prior to having to declare the LCO not met. The change will add a sentence to SR 3.0.3 that states, "A risk evaluation shall be performed for any surveillance delayed greater than 24 hours, and the risk impact shall be managed."

The objective of the proposed change is to minimize the impact on plant risk resulting from the performance of a missed surveillance test by allowing flexibility in considering the plant conditions and other plant activities without compromising plant safety. In addition, implementation of the proposed change would reduce the need for the licensee to apply for regulatory relief to delay the performance of missed surveillances.

The basis for establishing the changes to requirements for missed surveillances in Generic Letter 87-09 continues to apply to the current proposed change to SR 3.0.3. As evidenced by the discussion in Generic Letter 87-09, the intent of the change proposed in the Generic Letter was to reduce the impact on plant risk resulting from the performance of a missed surveillance test by allowing some flexibility in the performance of missed tests. The delay time of 24 hours was selected using engineering judgement in the absence of suitable tools to determine a delay period on a case-by-case basis. In addition, the staff recognized in Generic Letter 87-09 that even a 24-hour delay period would not be sufficient in some cases and licensees would need to seek regulatory relief in those cases.

The recent revision to the Maintenance Rule to establish the requirement in 10 CFR 50.65(a)(4) to assess and manage the increase in risk that may result from maintenance activities provides a framework to allow a more risk-informed approach to addressing missed surveillances. This approach is consistent with the Commission's policy to increase the use of probabilistic risk assessment (PRA) technology in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data, and continues to support the objectives outlined by the staff in Generic Letter 87-09.

The staff believes that the proposed change to SR 3.0.3 is appropriate because: (1) the number of missed surveillance tests is a very small fraction of the total number of such tests performed at a nuclear plant each year; (2) the change applies to unintentionally

[1]The terminology "temporary waiver" was subsequently revised to refer to the practice as "enforcement discretion."

missed surveillance tests and is not intended to be used as an operational convenience to extend surveillance frequencies (as stated in the existing SR 3.0.3 Bases); and (3) missed surveillances will be placed in the licensee's corrective action program.

The staff has determined that the proposed change is applicable to all licensees. In Generic Letter 87-09, the staff concluded that the proposed modifications would result in improved TSs for all plants and no limitations were put on the applicability of the proposed changes. Because the basis for this proposed change is largely the same as for the change proposed in Generic Letter 87-09, the staff believes the same broad applicability is appropriate. In addition, every licensee is required to comply with the Maintenance Rule and, therefore, will have implemented programs to comply with 10 CFR 50.65(a)(4) to assess and manage risk associated with maintenance and other operational activities.

The proposed change would modify SR 3.0.3 to allow a delay period from 24 hours up to the surveillance frequency, whichever is greater, to perform a missed surveillance prior to having to declare the LCO not met. The change would add a sentence to SR 3.0.3 that states, "A risk evaluation shall be performed for any surveillance delayed greater than 24 hours, and the risk impact shall be managed."

The proposed change would not allow equipment known to be inoperable to be considered operable until the missed surveillance is performed. If it is known that the missed surveillance could not be met, SR 3.0.1 would require that the LCO be declared not met and the appropriate condition(s) entered. In addition, the Bases for SR 3.0.3 state that the use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend surveillance intervals, but only for the performance of missed surveillances.

The modification will also include changes to the Bases for SR 3.0.3 that provide details on how to implement the new requirements. The Bases changes provide guidance for surveillance frequencies that are not based on time intervals but are based on specified unit conditions, operating situations, or requirements of regulations. In addition, the Bases changes state that the licensee is expected to perform any missed surveillance test at the first reasonable opportunity, taking into account appropriate considerations, such as the impact on plant risk and accident analysis assumptions, consideration of unit conditions, planning, availability of personnel, and the time required to perform the surveillance. The Bases also state that the risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," dated May 2000, and that the missed surveillance should be treated as an emergent condition as discussed in Regulatory Guide 1.182. In addition, the Bases state that the degree of depth and rigor of the evaluation should be commensurate with the importance of the component and that missed surveillances for important components should be analyzed quantitatively. The Bases also state that, if the results of the risk evaluation determine that the risk increase is significant, the evaluation should be used to determine the safest course of action. Finally, the Bases state that all missed surveillances will be placed in the licensee's Corrective Action Program.

Key elements provided by the licensee to justify the proposed TS change are listed below. These elements were built into the process to ensure that every time a surveillance is missed, the risk will be properly assessed and managed. In addition, such elements facilitate regulatory oversight.

- A risk evaluation shall be performed for any surveillance test delayed longer than 24 hours and the risk impact shall be managed.

- Although the proposed change to SR 3.0.3 allows an increase of the delay time, the missed surveillance test should be performed at the "first reasonable opportunity."

- The "first reasonable opportunity" will be determined by taking into consideration the risk impact from delaying the surveillance test (including risk from changing plant configurations or shutting the plant down to perform the surveillance, whenever applicable) as well as the impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the surveillance.

- A missed surveillance will be treated as an emergent condition in the same fashion as other unplanned maintenance activities. The risk impact of the condition will be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, Regulatory Guide 1.182.

- A missed surveillance will be placed in the licensee's corrective action program, thus providing the staff with a means to verify that the number of missed surveillances continues to be very low.

- The NRC's operating reactor oversight process will provide the framework for inspectors and other staff to review missed surveillances and assess the licensee's actions and performance.

The staff finds that a process containing these key elements is appropriate in this case for the following reasons:

- 10 CFR 50.65(a)(4) requires licensees to implement programs to assess and manage increases in risk that may result from planned maintenance activities. This program is suitable to assess and manage the risk impact of missed surveillances because missed surveillances can be treated as emergent conditions and their risk impact will be assessed and managed in an integrated fashion with concurrent maintenance activities.

- Inspection procedures are in place which will allow the NRC staff to oversee the implementation of Maintenance Rule requirements, including the adequacy of risk assessments performed by licensees for maintenance configurations.

- The number of missed surveillance tests is a very small fraction of the total number of such tests performed at a nuclear plant each year. The proposed change is not intended to be used as an operational convenience to extend surveillance frequencies.

- This process is similar to other improvements that have been made to the TS that allow the use of a controlled decisionmaking process by licensees when the process has some high-level regulatory oversight. Two examples of this are the adoption of the Core Operating Limits Report and the Pressure/Temperature Limits Report. In each of these cases, the staff approved the methodology behind the calculation of certain TS parameter limits and then allowed the specific limits to be removed from TSs and controlled by the licensee using the approved methodology. Similarly, for this proposed change, the staff has already approved guidance that outlines a process for complying with 10 CFR 50.65(a)(4) and, therefore, can allow the licensee to use that guidance to determine the most prudent course of action in the case of a missed surveillance.

The guidance outlining an acceptable process for licensees to assess and manage increases in risk that may result from planned maintenance activities is found in Regulatory Guide 1.182. Regulatory Guide 1.182 endorses a revised Section 11 to NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 2, dated February 22, 2000, updated by NEI.

Section 11 of NUMARC 93-01 provides guidance for assessing and managing risk impact resulting from performance of maintenance activities, including guidance for establishing action thresholds based on qualitative and quantitative considerations as well as risk management actions. The objective of risk management is to control the temporary and aggregate risk increases from maintenance activities such that the plant's average baseline risk is maintained within a minimal range. This is accomplished by using the results of the risk assessment to plan and schedule maintenance such that the risk increases are limited, and to take additional actions beyond routine work controls to address situations where the temporary risk increase is above a certain threshold.

In order to gain additional insights into the proposed change, the staff referred to the regulatory guidance provided in Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, dated July 1998, and in Regulatory Guide 1.177, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications,” dated August 1998, although these Regulatory Guides do not specifically address the type of change in this proposal. Regulatory Guide 1.177 provides the staff’s recommendations for utilizing risk information to evaluate changes to nuclear power plant TSs by assessing the impact of such proposed changes on the risk associated with plant operation. The approach documented in Regulatory Guide 1.177 was taken into consideration by the staff in evaluating the risk information provided in support of the proposed changes in SR 3.0.3 to increase the time allowed to perform a missed surveillance.

One portion of the guidance in Regulatory Guide 1.177 includes the assessment of the risk impact of the proposed change for comparison to acceptance guidelines consistent with the Commission’s Safety Goal Policy Statement, as documented in Regulatory Guide 1.174. In addition, the approach outlined in the guidance aims at ensuring that the plant risk does not increase unacceptably at any time during the implementation of the proposed change (i.e., during the extended surveillance interval).

Another portion of the guidance addresses the need for identifying risk-significant configurations resulting from maintenance or other operational activities and taking appropriate compensatory measures to avoid such configurations. This type of evaluation is directly addressed by the requirement to perform a risk assessment for missed surveillances delayed longer than 24 hours.

The staff believes that insights from the guidance provided in Regulatory Guides 1.174 and 1.177 can be used to show how the proposed change is expected to result in, at most, an increase in risk which is small and consistent with the Commission's Safety Goal Policy Statement. The staff believes that in the majority of the cases of missed surveillances, implementation of the proposed change will result in a risk benefit due to the proposed requirement for the licensee to evaluate the risk impact for missed surveillances that would require a delay of longer than 24 hours.

The staff made a qualitative assessment of the risk impact of the proposed change for comparison with the intent of the acceptance guidelines documented in Regulatory Guide 1.174, consistent with the Commission's Safety Goal Policy Statement. Such risk impact is measured by the average (yearly) risk change. In addition, the staff took into consideration guidance in Regulatory Guide 1.177 aimed at ensuring that the plant risk does not increase unacceptably at any time during the implementation of the proposed change (i.e., during an extended surveillance interval in this case). The staff's qualitative assessment is summarized below.

The probability that a standby active component, such as a pump or a circuit breaker, will fail when demanded during an accident is based on the assumption that the component fails due to "standby" stresses (i.e., stresses which are present while the component is in standby, such as corrosion, dirt, lack of lubrication). This probability, also called the component's average "unavailability," is used in PRAs and is most frequently calculated by the following equation:

$$q = \tfrac{1}{2} * \lambda * T \qquad (1)$$

where:

q = the component's average unavailability,
λ = the component's failure rate (assumed constant) while in standby, and
T = the interval at which the component is tested for operability.

The average unavailability of a structure, system, or component (SSC), calculated by using the above equation, reflects the potential vulnerability of the component to "standby" stresses. Such vulnerability increases with time between operability checks (tests) assuming corrective action is taken to restore failed components identified by the test. Thus, the risk impact of a missed surveillance is reflected by the increased unavailability of the related SSCs due to the increase of the interval between surveillance tests. If the missed surveillance affects two or more components, some "standby" stresses may impact multiple components. In such a case, the missed surveillance would also increase the average common cause failure (CCF) unavailability of two or more components and this should be addressed in the risk assessment (CCF unavailabilities are calculated by adjusting the single component failure unavailability using standard PRA techniques, such as the beta factor or the Multiple Greek Letter method).

The thresholds of the aggregate risk impacts are based on the permanent change guidelines discussed in Regulatory Guide 1.174. The licensee will be expected to manage the risk from the proposed technical specification change in conjunction with the risk from other concurrent plant activities to ensure that any risk increase, in terms of core damage frequency (CDF) and large early release frequency (LERF), will be small and consistent with the Commission's Safety Goal Policy Statement.

Risk insights from existing PRAs and the low frequency of missed surveillances indicate that the proposed TS change is highly unlikely to lead to a significant increase in the average (yearly) risk, in terms of CDF or LERF. Significant risk increases can occur only under the following conditions:

- The number of missed surveillances is allowed to increase significantly;

- High risk configurations are allowed (e.g., by allowing certain combinations of multiple missed surveillances and/or outages); and

- Poor risk management of plant operational activities is allowed.

Any of these conditions would be in violation of the intent of the proposed SR 3.0.3 and could trigger a review by the NRC of the licensee's actions and performance. The implementation guidance found in the proposed SR 3.0.3 Bases is intended to ensure that such conditions would not occur. Licensees are already required to manage risk associated with online maintenance activities. Furthermore, the addition of missed surveillances (rather rare plant conditions) to the maintenance activities is not expected to increase risk. On the contrary, insights from existing risk assessments indicate that there are plant conditions during which it is preferable and safer not to have to complete missed surveillance tests for some SSCs. Therefore, the proposed TS change will allow the licensee to make informed decisions and take appropriate actions to control risk.

In addition to changes in the mean values of CDF and LERF, the incremental conditional core damage probability (ICCDP) and the incremental conditional large early release probability (ICLERP) are proposed by Regulatory Guide 1.177 as appropriate measures of the increase in probability of core damage and large early release, respectively, during the period of implementation of a proposed TS change (i.e., during the extended surveillance period in the case of a missed surveillance). Regulatory Guide 1.182 provides guidance for controlling temporary risk increases resulting from maintenance activities. Such guidance, which is consistent with guidance provided in Regulatory Guide 1.177, establishes action thresholds based on qualitative and quantitative considerations as well as risk management actions. The staff expects that the licensee will implement this guidance for assessing temporary risk increases from missed surveillances concurrently with maintenance and other operational activities.

Instantaneous and temporary risk increases from a missed surveillance are assessed by considering the time-dependent unavailability, most often calculated by the following equation:

$$q(t) = \lambda * t$$

where:

$q(t)$ = the component's unavailability at time t
λ = the component's failure rate (assumed constant) while in standby, and
t = time from end of surveillance frequency of a missed surveillance test.

If the missed surveillance affects two or more components, some "standby" stresses may impact multiple components. In such a case, the missed surveillance would increase also the time-dependent CCF unavailability of two or more components and this should be addressed in the risk assessment.

Significant temporary risk increases following a missed surveillance can occur only under the following conditions:

- High risk configurations are allowed (e.g., by allowing certain combinations of multiple missed surveillances and/or outages), and

- Poor risk management of plant operation activities is allowed.

Any of these conditions would be in violation of the intent of the proposed SR 3.0.3 and could trigger an NRC review of the licensee's actions and performance. The requirements associated with the proposed change are intended to ensure that such conditions would not occur. Thus, the proposed TS change is not expected to lead to significant temporary risk increases. Following the discovery of an unintentionally missed surveillance, the licensee will have to assess temporary risk increases, qualitatively or quantitatively depending on the importance of the component affected by the missed surveillance, if the surveillance cannot be performed within 24 hours from the time it has been discovered.

Regulatory Guide 1.177 addressed the need for identifying risk-significant configurations resulting from maintenance or other operational activities and taking appropriate compensatory measures to avoid such configurations. The objective of such guidance for this review is to ensure that plant safety will be maintained and monitored during the period of an extended surveillance testing interval (associated with an unintentionally missed surveillance). The licensee proposes to use the program in place to implement the Maintenance Rule to identify "high-risk" configurations resulting from missed surveillance tests in conjunction with outages associated with maintenance activities. It is worth noting that the guidance provided in Regulatory Guide 1.177 with regard to the Configuration Risk Management Program was used as the basis for developing the guidance contained in Regulatory Guide 1.182 for the 10 CFR 50.65(a)(4) provisions of the Maintenance Rule. This provides additional assurance that the proposed process for evaluating the risk impact of missed surveillances is consistent with guidance provided in Regulatory Guide 1.177.

Once a missed surveillance is discovered and the licensee determines that the surveillance cannot be performed within 24 hours, the licensee will have to use a risk assessment to determine the most prudent course of action. The risk assessment can be done qualitatively or quantitatively depending on the importance of the component affected by the missed surveillance (missed surveillances for risk-important components should be analyzed quantitatively). Such a risk assessment will be consistent with the program to implement the Maintenance Rule guidance to assess and account for both aggregate and temporary risk increases associated with “emergent” plant conditions as well as before undertaking online maintenance or other operational activities.

All licensees must have the capability to assess and manage increases in risk from maintenance activities as required by the Maintenance Rule. Risk assessments performed pursuant to 10 CFR 50.65(a)(4) may use qualitative, quantitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the complexity of the proposed configuration to be assessed. Section 11 of NUMARC 93-01 provides guidance for using qualitative, quantitative, or blended methods to assess risk. Current inspection programs allow the NRC staff to oversee licensee implementation of 10 CFR 50.65(a)(4) requirements, including the adequacy of pre-maintenance risk assessments performed by licensees.

For the reasons listed below, the staff finds that the same “quality” of PRA or PRA insights used to perform risk assessments pursuant to 10 CFR 50.65(a)(4) is also appropriate when assessing the impact of missed surveillances.

- The number of “emergent” conditions resulting from missed surveillances is very small (in both absolute terms and in comparison to the frequency of “emergent” conditions resulting from equipment failures). The licensee is expected to implement the proposed change to SR 3.0.3 in a manner that ensures that this statement remains valid.

- A missed surveillance is equivalent to a one-time surveillance frequency extension. Therefore, the risk exposure is limited to the duration of the surveillance frequency extension. Risk increases are small compared to similar increases associated with equipment failures. The average (conditional) risk increase, given a missed surveillance, may be comparable to the risk increase from equipment failures. However, due to the rarity of missed surveillances, the average (yearly) risk increase from missed surveillances is expected to be small compared to the risk increase from equipment failures.

- PRA insights indicate that the risk impact from missed surveillances is significant only for a relatively small set of standby equipment. This equipment, such as auxiliary feedwater, high pressure injection pumps, and emergency diesel generators, is located outside containment and generally can be easily tested in a short time, if necessary.

- NRC inspection programs allow the NRC staff to oversee the implementation of 10 CFR 50.65(a)(4) requirements, including the adequacy of pre-maintenance risk assessments performed by licensees.

The staff review finds that the process proposed by the licensee for addressing missed surveillance requirements meets Commission guidance for allowing TS SR changes. Key elements of the proposed change are listed below.

- A risk evaluation shall be performed for any surveillance delayed longer than 24 hours, and the risk impact shall be managed.

- The missed surveillance test should be performed at “the first reasonable opportunity.”

- The “first reasonable opportunity” will be determined by taking into consideration the risk impact from delaying the surveillance test as well as the impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the surveillance.

- A missed surveillance will be treated as an “emergent” condition in the same fashion as other unplanned maintenance activities. The risk impact of the condition will be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance (NRC Regulatory Guide 1.182). Rescheduling of missed surveillances pursuant to Regulatory Guide 1.182 will ensure the necessary provisions for managing the risk impact of performing the surveillance in conjunction with other ongoing plant configuration changes.

- The NRC’s operating reactor oversight process will provide the framework for inspectors and other staff to review missed surveillances and assess the licensee’s actions and performance. Inspection procedures are in place which will allow NRC staff to oversee the implementation of Maintenance Rule requirements, including the adequacy of pre-maintenance risk assessments performed by licensees.

- A missed surveillance will be placed in the licensee’s corrective action program, thus providing the staff with a means to verify that the number of missed surveillances continues to be very low.

- The number of missed surveillance tests is a very small fraction of the total number of such tests performed at a nuclear plant each year. The proposed change is not intended to be used as an operational convenience to extend surveillance frequencies.

- This process is similar to other improvements that have been made to the TSs that allow the use of a controlled decisionmaking process by licensees when the process has some high-level regulatory oversight. Two examples of this are the adoption of the Core Operating Limits Report and the Pressure/Temperature Limits Report. In each of these cases, the staff approved the methodology behind the calculation of certain TS parameter limits and then allowed the specific limits to be removed from TSs and controlled by the licensee using the approved methodology. Similarly, for this proposed change, the staff has already approved guidance that outlines a process for complying with 10 CFR 50.65(a)(4) and, therefore, can allow the licensee to use that guidance to determine the most prudent course of action in the case of a missed surveillance.

For these reasons, the staff finds that the proposed adoption of TSTF-358, to be implemented in accordance with the above listed key elements, is acceptable.

Since the licensee's request conforms to the foregoing, the NRC staff finds this change acceptable.

G.1.4 Revision to Ventilation Charcoal Adsorber Testing Program

By supplemental letter dated December 12, 2001, the licensee submitted a proposal to incorporate the recommendations delineated in GL 99-02, “Laboratory Testing of Nuclear-Grade Activated Charcoal,” dated June 3, 1999. By letters dated June 7 and June 25, 2002, the licensee provided additional information in response to an NRC staff RAI.

Safety-related air-cleaning units used in the engineered safety feature ventilation systems of nuclear power plants reduce the potential onsite and offsite consequences of a radiological accident by filtering radioiodine. When calculating offsite and control room operator doses for DBAs, estimated charcoal adsorption efficiencies are used as input assumptions. To ensure that the charcoal filters used in these systems will perform in a manner that is consistent with the licensing basis of a facility, licensees have requirements in their TSs to periodically perform a laboratory test to determine charcoal adsorption efficiency (in accordance with a test standard) of charcoal samples taken from these ventilation systems.

In GL 99-02, the NRC staff alerted licensees that testing nuclear-grade activated charcoal to standards other than American Society for Testing and Materials (ASTM) D3803-1989, “Standard Test Method for Nuclear-Grade Activated Carbon,” does not provide assurance for complying with their current licensing basis with respect to the dose limits of General Design Criterion 19 of Appendix A to 10 CFR Part 50 and Subpart A of 10 CFR Part 100.

In GL 99-02, the NRC staff requested that all licensees determine whether their TSs reference ASTM D3803-1989 for charcoal filter laboratory testing. Licensees whose TSs did not reference ASTM D3803-1989 were requested to either amend their TSs to reference ASTM D3803-1989 or propose an alternative test protocol.

The NRC received a letter from ASTM in response to a March 8, 2000, *Federal Register* notice (65 FR 12286) related to revising testing standards in accordance with ASTM D3803-1989 for laboratory testing of activated charcoal in response to GL 99-02. ASTM notified the NRC that the 1989 standard is out of date and should be replaced by ASTM D3803-1991(1998). The NRC staff acknowledges that the most current version of ASTM D3803 is ASTM D3803-1991 (reaffirmed in 1998). However, it was decided for consistency purposes to have all of the nuclear reactors test to the same standard (ASTM D3803-1989) because prior to GL 99-02 being issued, approximately one third of all nuclear reactors had TSs that referenced ASTM D3803-1989 and there are no substantive changes between the 1989 and 1998 versions.

The proposed use of ASTM D3803-1989 is acceptable because it provides accurate and reproducible test results. The proposed test temperature of 30 degrees Celsius and 95 percent relative humidity is acceptable for the control room special ventilation system (CRSVS), auxiliary building special ventilation system (ABSVS), spent fuel pool special ventilation system (SFPSVS), and shield building ventilation system (SBVS) because the temperatures are consistent with ASTM D3803-1989. This is consistent with the actions requested in GL 99-02.

The licensee stated that the credited removal efficiency for radioactive organic iodine for each of the four systems is 95 percent for CRSVS, 70 percent for ABSVS, 85 percent for SFPSVS, and 70 percent for SBVS. The proposed test penetration for radioactive methyl iodide for each system is as follows: less than 2.5 percent for CRSVS, 15 percent for ABSVS, 7.5 percent for

SFPSIPVS, and 15 percent for SBVS. The proposed test penetration was obtained by applying a safety factor of 2 to the credited efficiency. The proposed safety factor of 2 for all systems is acceptable because it ensures that the efficiency credited in the accident analysis is still valid at the end of the surveillance interval. This is consistent with the minimum safety factor of 2 specified in GL 99-02.

The August 23, 1999, errata to GL 99-02 clarified that if the maximum actual face velocity is greater than 110 percent of 40 feet per minute (fpm), then the test face velocity should be specified in the TSs. In its supplemental letter dated June 25, 2002, the licensee stated that the maximum test face velocity for each system was being evaluated, and that the licensee will commit to completing this evaluation by February 28, 2003. The licensee has also committed to submitting a TS amendment application by February 28, 2003, to specify the maximum test face velocity if the licensee's evaluation results in a maximum actual face velocity greater than 110 percent of 40 fpm. The licensee has requested that these commitments be incorporated into a license condition (See Section 5.0 of this SE).

Based on the above, and because the NRC staff considers ASTM D3803-1989 to be the most accurate and most realistic protocol for testing charcoal in safety-related ventilation systems, the NRC staff finds that the proposed TS changes satisfy the actions requested in GL 99-02, and are acceptable.

G.1.5 Required Shutdown Margin During Physics Tests

Proposed ITS 3.1.8, "Physics Tests Exceptions - Mode 2," requires that the shutdown margin (SDM) be within the limits specified in the COLR in order to suspend the requirements of certain LCOs associated with physics testing. For Mode 2 operation, the rod insertion limits ensure that the SDM is met.

However, CTS 3.10.D.2, "Rod Insertion Limits," states that "Insertion limits do not apply during PHYSICS TESTS or during periodic exercise of individual rods." The first objective of CTS 3.10, "Control Rod and Power Distribution Limits," is to assure that the core can reach subcriticality after a reactor trip. For physics testing, the CTS require that there be sufficient trippable reactivity to bring the reactor to a subcritical condition (i.e., an SDM greater than 0%Δk).

Since ITS 3.1.8 requires that the SDM be specified in the COLR, the licensee requested approval of a methodology for establishing the SDM. The licensee transmitted this request by letter dated January 25, 2002, as supplemented by letter dated June 25, 2002.

The regulation at 10 CFR Part 50, Appendix A, Criterion 26, "Reactivity Control System Redundancy and Capability," requires that the licensee provide two independent reactivity control systems of different design principles, and that one of the systems be capable of holding the reactor core subcritical under cold conditions. Licensees meet this requirement by establishing and staying within their SDMs. Maintaining the SDMs provides reactivity margin sufficient to ensure that the acceptable fuel design limits will not be exceeded for normal operation and anticipated operational occurrences. For Mode 1 and Mode 2 operation (power operation and startup), the TSs maintain these SDMs by establishing rod insertion limits. When establishing these rod insertion limits, the licensees take into account the individual rod worths.

However, 10 CFR Part 50, Appendix B, Section XI, “Test Control,” requires that licensees establish a test program to ensure that structures, systems, and components will perform satisfactorily in service. Therefore, licensees need to test all of the functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences. To accomplish this requirement, licensees perform physics testing for reload fuel cycles. One such test is control rod worth testing; but performing these physics tests may lead to violation of certain LCOs normally applicable in Modes 1 and 2, including the rod insertion limit LCO.

Because rod worth testing is necessary to ensure that specified design conditions are not exceeded during normal operation and analyzed abnormal operating conditions, TSs normally include an exception to the rod insertion limit LCOs for that testing, provided SDM is maintained through some other method. CTS 3.10 reflects this exception. In contrast, ITS 3.1.8 requires that the SDM during physics tests be specified in the COLR. Therefore, the licensee must specify an SDM for physics testing in Mode 2, calculated in accordance with an NRC-approved methodology applicable to the plant. This proposal establishes the licensee’s methodology for inclusion of the Mode 2 physics testing SDM in the COLR.

The licensee’s proposed methodology for establishing SDM during physics testing in Mode 2 is to set the value at 0.5%Δk. The licensee will use this value to determine the limiting reactor configuration allowed during physics tests. This limiting configuration will take into account rod cluster control assembly (RCCA) positions, core thermal power, and RCS loop average temperature. Also, as part of its proposal, the licensee stated that the reactor shall never be taken to a more limiting configuration during physics testing.

The NRC staff evaluated the licensee’s proposal by comparing it to CTS and USAR accident analysis requirements. The NRC staff has determined that if the proposed SDM methodologies were equivalent to or more restrictive than the current requirements, then it would continue to meet the intent of the applicable regulations. Therefore, it would be acceptable.

Currently, USAR section 3.1.2.5, “Reactivity Shutdown Capability,” states that the shutdown and control groups (control rods) make the reactor at least one percent subcritical following a trip from any credible operating condition to the hot zero power condition. It also states that the shutdown capability will maintain the core subcritical following the most severe anticipated cooldown transient. These assumptions are included in the safety analyses. However, the licensee states that these accident analyses are not applicable during low power physics testing. In order to perform this testing, the CTS do not require compliance with LCOs that normally ensure that the reactor operates within the assumptions used in the safety analyses, (e.g., moderator temperature coefficient and rod insertion limits). Upon review, the NRC staff agrees that for physics testing, the CTS do not require compliance with LCOs that would normally ensure that the reactor operates within the bounds of the safety analyses. Therefore, during low power physics testing, the USAR safety analyses’ 1%Δk SDM is not required.

The CTS also require that the rod insertion limits be maintained to ensure an SDM specified in the COLR is met. Currently, the Unit 1 and Unit 2 COLRs require an SDM of 2.0%Δk for Mode 2. However, CTS 3.10.D.3 states that “The shutdown margin specified in the Core Operating Limits Report must be maintained except for low power PHYSICS TESTING.” This TS requirement removes the SDM requirement for physics testing. The CTS also state that for 2 hours per year, the reactor may be critical with all but one high worth, full-length control rod inserted, provided that a rod drop test is performed on this rod prior to the particular physics test.

Both TS statements imply that an SDM greater than 0%Δk is required. Prior to the 2-hour period per year, the licensee must test their high-worth RCCA. This testing should show that the RCCA is capable of dropping into the core and, therefore, will be able to provide trippable negative reactivity to the reactor core. The licensee's conversion to ITS will clarify that a minimum SDM must be maintained during physics testing.

The NRC staff has reviewed the licensee's proposed methodology for establishing an SDM requirement for physics testing in Mode 2 for inclusion in the PINGP Units 1 and 2 COLRs. The NRC staff finds that the licensee's proposal to set the SDM value to 0.5%Δk is acceptable since it is more conservative than the CTS requirements. Using this more conservative value for the SDM requirement will ensure that the plant's current design basis is maintained. Therefore, the NRC staff concludes that the proposed SDM methodology is acceptable.

G.1.6 Heat Flux Hot Channel Factor Methodology

Proposed ITS 3.2.1, "Heat Flux Channel Factor, (F_Q(Z))," requires, in part, that:

(1) F_Q^W(Z), the elevation-dependent heat flux hot channel factor, be increased by an appropriate factor, F_Q^A, specified in the COLR;

or

(2) flux map measurements be taken every 7 effective full power days,

if measurements indicate that the maximum over z of [F_Q^C(Z)/K(Z)] has increased since the previous evaluation of F_Q^C(Z), the Heat Flux Hot Channel Factor, evaluated from an incore flux map.

By letter dated January 25, 2002, as supplemented by letter dated June 28, 2002, the licensee requested approval of a methodology for determining the factor, F_Q^A, for inclusion in the PINGP COLR. The licensee proposed to determine the F_Q^A factor using both historical and predictive (calculated) methods.

The regulation at 10 CFR Part 50, Appendix A, Criterion 10, "Reactor Design," specifies that the reactor core and associated coolant, control, and protection systems shall be designed to assure that the specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation and anticipated operational occurrences. Additionally, in order to limit fuel damage, 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for Light-Water Nuclear Power Reactors," states that the calculated peak cladding temperature shall not exceed 2200 ^{o}F during a LOCA.

To comply with these fuel design criteria, licensees place peak power limits on the reactors. The licensees verify that these limits are not exceeded by mapping the core power with flux detectors from the incore monitoring system. This core power map shows the localized differences in core power and helps to determine if the maximum heat flux is exceeded. However, the licensees typically only perform these measurements every 31 effective full power days (EFPDs). Therefore, to ensure that the reactor does not operate beyond these limits between measurements, licensees calculate predictive peaking factors.

One such peaking factor is the heat flux hot channel factor, $F_Q(Z)$. This factor is defined as follows:

$F_Q(Z)$ = maximum kw/ft (kilowatts per foot) at elevation z / average kw/ft in the core

Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core. The licensee approximates this peaking factor by using $F_Q^W(Z)$ and $F_Q^C(Z)$. The $F_Q(Z)$ limits are set in the licensee's COLR and the licensee's TSs require that the limits be met. $F_Q^W(Z)$ is the elevation dependent heat flux hot channel factor, which compensates for nonequilibrium conditions and worst case transient conditions within the core. But, to ensure that this factor has not changed to an unacceptable value, the licensee measures the hot channel factor every 31 days, which is called $F_Q^M(Z)$. However, to account for manufacturing tolerance issues and flux map measurement uncertainty associated with this measurement, the licensee must apply a factor to $F_Q^M(Z)$. With this factor, the licensee can then determine the calculated heat flux hot channel factor, $F_Q^C(Z)$, whereas:

$F_Q^C(Z) = F_Q^M(Z)$ x (fuel manufacturing tolerance and flux map measurement uncertainty factor)

Additionally, $F_Q^W(Z)$ is related to $F_Q^C(Z)$ by a cycle dependent factor, V(Z), which accounts for power distribution transients encountered during normal operation. V(Z) is included in the licensee's COLR. Consequently:

$F_Q^W(Z) = F_Q^C(Z) \times V(Z)$

Therefore, to determine if $F_Q^W(Z)$ has changed since the last core measurements, the licensee monitors $F_Q^C(Z)$. This is done by tracking $F_Q^C(Z)/K(Z)$, where K(Z) is a normalized $F_Q(Z)$ factor. K(Z) is used to account for height dependent differences in fuel damage during accidents. If $F_Q^C(Z)/K(Z)$ has increased since the last measurement, then the heat flux may have an increasing trend. This trend, if not accounted for, could cause $F_Q^W(Z)$ to exceed the $F_Q(Z)$ limits prior to taking the next flux map measurements (every 31 days). Therefore, during this time, the core may not be capable of meeting the ECCS Acceptance Criteria. To account for the increasing trend and to ensure that the heat flux hot channel factor remains limiting, the licensee must apply a multiplier, F_Q^A, to $F_Q^W(Z)$ or perform flux map measurements every 7 EFPDs.

The licensee proposes to determine F_Q^A by using the following steps:

1) Determine the bounding percentage increase of the maximum over z of [$F_Q^C(Z)/K(Z)$] with a 95%/95% level of confidence, based upon the historical data from at least three completed cycles at approximately 31 day intervals.

2) Calculate the percentage increase in the maximum predicted heat flux hot channel factor, $F_Q^P(Z)$. The licensee will calculate this factor based upon the computer models and reactor physics methodology described in NSPNAD-8108-A, Revision 2, "Qualification of Reactor Physics Methods for Application to Prairie Island." The NRC staff approved this methodology by letter dated, September 13, 2000. The $F_Q^P(Z)$ values may be taken from a predictive cycle depletion from the beginning of cycle to the end of cycle assuming nominal operating conditions.

3) Choose the maximum of numbers 1 and 2 above to be the value for F_Q^A. The licensee will then place the maximum F_Q^A factor into the COLR and use it to increase the value for $F_Q^W(Z)$ as necessary. However, the value of F_Q^A may vary, since historical data and the proposed predictive methodology could be a function of cycle exposure. The licensee plans to incorporate the F_Q^A variation into the COLR.

The NRC staff has evaluated the proposed method of determining F_Q^A to establish whether the methodology will ensure that the SAFDLs and 10 CFR 50.46 acceptance criteria will continue to be met. The NRC staff has concluded that these limits would continue to be met if the methods establish conservative values for $F_Q^W(Z)$ (i.e., they ensure that the possible change to $F_Q^W(Z)$ will remain below the $F_Q(Z)$ limits in the COLR).

If $[F_Q^C(Z) / K(Z)]$ changes between two flux maps, it may show an increasing trend in the heat flux of the core. This trend could cause $F_Q^W(Z)$ to exceed the $F_Q(Z)$ limits prior to taking the next map measurements (approximately every 31 days). The NRC staff has concluded that the licensee's choice to increase $F_Q^W(Z)$ by a factor, F_Q^A is conservative because it takes into account the possible increase in $F_Q^W(Z)$ between these flux map measurements. The NRC staff has also concluded that the licensee's procedure to establish F_Q^A as the more limiting of either the maximum over z of $[F_Q^C(Z)/K(Z)]$ from historical data (Step 1 above) or the maximum predicted $F_Q^P(Z)$ (Step 2 above), reasonably ensures that the value chosen to update $F_Q^W(Z)$ is a conservative value. Because 31 EFPDs is the maximum allowed period between taking flux map readings, the licensee's choice of this time period to account for the change in $F_Q^W(Z)$ will ensure that the maximum possible change has been chosen.

The licensee's proposal to apply a factor, F_Q^A to account for the change in $F_Q^W(Z)$ between successive flux maps is conservative. Also, the proposed methods of determining F_Q^A is conservative (i.e., using the greater of the historical data and the predicted data). Therefore, the NRC staff finds that applying this method to determine the bounding values for $F_Q^W(Z)$ is acceptable.

In conclusion, the NRC staff has reviewed the licensee's proposed methodology for determining the factor F_Q^A, for inclusion in the PINGP Units 1 and 2 COLRs, and has concluded, based on the above, that the licensee's proposal to apply the factor, F_Q^A, to account for the change in $F_Q^W(Z)$ between successive flux maps is conservative. Also, the proposed method of determining F_Q^A is conservative. Using these conservative values to determine the bounding values for $F_Q^W(Z)$ will ensure that the $F_Q(Z)$ limits are not exceeded. By staying within the $F_Q(Z)$ limits, this methodology ensures that the SAFDLs and the 10 CFR 50.46 ECCS acceptance criteria will continue to be met. Therefore, the NRC staff concludes that the proposed F_Q^A methodology is acceptable.

G.2 <u>Additional BSI Changes identified by the NRC staff:</u>

G.2.1 <u>Instrument Setpoint methodology and new allowable values</u>

In support of the conversion to ITS, which involves many changes to instrument allowable values, the licensee submitted its instrument setpoint methodology along with two sample calculations by supplemental letter dated March 6, 2002. Additional information was provided by the licensee by supplemental letter dated January 31, 2002, in response an NRC staff RAI.

Paragraph (c)(1)(ii)(A) of 10 CFR 50.36, “Technical Specifications,” requires, in part, that where a limiting safety system setting 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. RG 1.105, “Setpoints for Safety-Related Instrumentation,” Revision 3, and ANSI Standard ANSI/ISA-S67.04-1994, “Setpoints for Nuclear Safety-Related Instrumentation,” provide guidance and methodology for determining appropriate settings. The NRC staff has based its acceptance of the licensee’s setpoint methodology on these regulatory requirements and guidance.

The licensee submitted Revision 0 of Section 3.3.4.1 of the Engineering Manual, which is the Engineering Design Standard for Instrument Setpoint/Uncertainty Calculations and is used to calculate instrument allowable values. These instrument setpoint calculations resulted in some allowable values to be less restrictive and some to be more restrictive than the values provided in the CTS. The methodology used by the licensee is based on the general guidelines provided by ISA-S67.04-1987, "Setpoints for Nuclear Safety-Related Instrumentation", and the two-loop group setpoint methodology developed by Tenera, L. P. However, the licensee did not discuss any deviations from the ISA-S67.04 standard and the NRC staff has not endorsed the ISA-S67.04-1987 standard. ISA-S67.04-1982 and ISA-S67.04-1994 versions have been endorsed by the NRC staff in RG 1.105, Revisions 2 and 3, respectively. Therefore, the NRC staff, in its RAI, requested the licensee to provide a discussion on whether its setpoint methodology conforms with RG 1.105, Revision 2 or 3. In its response, the licensee stated that its setpoint methodology meets the provisions of the ISA-S67.04-1994 version. The licensee further stated that with the tacit assumption that vendor reference accuracy is provided with 95 percent probability and 95 percent confidence, it conforms to the guidance of RG 1.105, Revision 3.

In its RAI, the NRC staff requested the licensee to provide the basis for the acceptability of the vendor data and what steps are taken if the vendor data were not available. The licensee stated in its response that it has compared vendor data to Westinghouse-provided instrument channel component accuracies and have determined the vendor supplied accuracy values to be equal or more conservative. In addition to this, the licensee has performed drift analyses for various components and generally found that the vendor drift data bound plant-specific drift data for these components. Also, the licensee has stated that when vendor data are not available, a plant-specific drift analysis is performed to establish uncertainties for the affected components.

In its supplemental letter of March 6, 2001, the licensee also provided two sample calculations for PINGP which the NRC staff reviewed. The NRC staff verified that the licensee followed the setpoint methodology discussed in the engineering design standard. The proposed allowable values are intended to maintain acceptable margins between operating conditions and trip setpoints and do not significantly increase the likelihood of a false trip or failure to trip upon demand. Therefore, the existing licensing basis is not affected. Based on this review, the NRC staff finds the allowable value determination based on this methodology acceptable.

Based on the above, the NRC staff concludes that the licensee's setpoint methodology and the resulting allowable values incorporated in the ITS conversion package are consistent with the plant licensing basis and are, therefore, acceptable.

4.0 COMMITMENTS RELIED UPON

In reviewing the proposed ITS conversion for PINGP, 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 LR, "Less Restrictive-Relocated Details" (Attachment 5 to this SE) and Table R, "Relocated Specifications" (Attachment 6 to this SE). These tables 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 5.0 of this SE). Such commitments from the licensee are important to the ITS conversion because the acceptability of removing certain requirements from the TSs 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).

5.0 LICENSE CONDITIONS

License conditions to define the schedule to begin performing the new and revised SRs after implementation of the ITS are included in the Facility Operating Licenses. These conditions are:

(1) The schedule for performing SRs that are new or revised in License Amendments 158/149 shall be as follows:

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.

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.

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

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 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 an acceptable schedule. The licensee stated that their implementation date for the new ITS is no later than October 31, 2002. This implementation date is acceptable.

Also, a license condition is to be included that will enforce the relocation of requirements from the CTS to licensee-controlled documents. The relocations are described in Table LR (Attachment 5 to this SE), and Table R (Attachment 6 to this SE). The license condition states that the relocations would be completed no later than October 31, 2002. This schedule is acceptable.

In addition, a license condition is to be included that will require the licensee to complete evaluation of the maximum test face velocity for the ventilation systems included in ITS Section 5.5.9 by February 28, 2003. This license condition also requires the licensee to submit a license amendment request by February 28, 2003, to specify the maximum test face velocity if the maximum actual face velocity is greater than 110 percent of 40 feet per minute based on the licensee's evaluation.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 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 July 22, 2002 (67 FR 47868), for the proposed conversion of the CTS to ITS for PINGP. Accordingly, the Commission has determined that issuance of these amendments will not result in any significant environmental impacts other than those evaluated in the Final Environmental Statement.

8.0 CONCLUSION

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.

Attachments:
1. List of Acronyms
2. Table A - Administrative Changes
3. Table M - More Restrictive Changes
4. Table L - Less Restrictive Changes
5. Table LR - Less Restrictive Changes Relocated Details
6. Table R - Relocated Specifications
7. Table U - Unused Numbers

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LIST OF ACRONYMS

ABSV	Auxiliary Building Special Ventilation System
ASME	American Society of Mechanical Engineers
BSI	Beyond-Scope Issue
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CRSV	Control Room Special Ventilation System
CTS	Current Technical Specification
DBA	Design-Basis Accident
DOC	Discussion of Change (from the CTS)
ECCS	Emergency Core Cooling System
FHA	Fuel Handling Accident
FR	Federal Register
GDC	General Design Criteria
ISI	Inservice Inspection
IST	Inservice Testing
ITS	Improved Technical Specification
JFD	Justification for Deviation
LCO	Limiting Condition for Operation
LOCA	Loss-of-Coolant Accident
ODCM	Offsite Dose Calculation Manual
PINGP	Prairie Island Nuclear Generating Plant
P/T	Pressure/Temperature
PTLR	Pressure Temperature Limits Report
QA	Quality Assurance
RAI	Request for Additional Information
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SBVS	Shield Building Ventilation System
SE	Safety Evaluation
SER	Safety Evaluation Report
SFPSVS	Spent Fuel Pool Special Ventilation system
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SR	Surveillance Requirement
STS	Improved Standard Technical Specification, NUREG-1431, Rev. 1
CL	Service Water
TRM	Technical Requirements Manual
TS	Technical Specification
TSTF	Technical Specifications Task Force (re: generic changes to the STS)
USAR	Updated Safety Analysis Report

ATTACHMENT 2

Table A - Administrative Changes

ATTACHMENT 3

Table M - More Restrictive Changes

ATTACHMENT 4

Table L - Less Restrictive Changes

ATTACHMENT 5

Table LR - Less Restrictive Relocated Details

ATTACHMENT 6

Table R - Relocated Specifications

ATTACHMENT 7

Table U - Unused Numbers