Mr. David A. Christian
Sr. Vice President and
Chief Nuclear Officer
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

April 5, 2002

SUBJECT: NORTH ANNA POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS (TAC NOS. MB0799 AND MB0800)

Dear Mr. Christian:

The Commission has issued the enclosed Amendment Nos.231 and 212 to Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated December 11, 2000, as supplemented by letters dated May 30, June 18, July 16, July 20, August 13, August 27, September 27, October 10, October 17, November 8, November 19, November 29, December 3, December 7, December 12, and December 13, 2001, and January 2, January 25, January 31, February 11, February 18, February 22, February 27, March 7, March 18, March 22, and March 26, 2002.

These amendments convert the current TS (CTS) for North Anna Power Station, Units 1 and 2, to a set of improved Technical Specifications (ITS) based on NUREG-1431, Revision 1, "Standard Technical Specifications, Westinghouse Plants," dated April 1995.

The draft Safety Evaluation (SE) for the ITS conversion was sent to you by letter dated January 24, 2002, for your review to verify the accuracy of the draft SE. You provided comments by letter dated February 22, 2002. The comments you provided were reviewed and incorporated in the enclosed final SE for the amendments, as appropriate. The draft SE was also revised based on the staff's review after it was issued.

The ITS conversion will become effective as of its date of issuance and shall be implemented by September 2, 2002. If there is a request for amendment submitted prior to implementation of the ITS being completed, it will be necessary to submit separate TS pages for both CTS and ITS with the amendment request.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Stephen Monarque, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 231 to NPF-4
2. Amendment No. 212 to NPF-7
3. Safety Evaluation

cc w/encls: See next page

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Stephen Monarque, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures:
1. Amendment No. 231 to NPF-4
2. Amendment No. 212 to NPF-7
3. Safety Evaluation

cc w/encls: See next page

Letter Access.NO.ML021200265

OFFICE	PM:PDII/S1	LA:PDII/S2	SC:RORP	OGC	SC(A):PDII/S1
NAME	SMonarque	EDunnington	RDennig	RHoefling	RLaufer
DATE	04/02/02	04/02/02	04/03/02	04/03/02	04/04/02
COPY	Yes/No	Yes/No	Yes/No	Yes/No	Yes/No

OFFICIAL RECORD COPY

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 231
License No. NPF-4

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated December 11, 2000, as supplemented by letters dated May 30, June 18, July 16, July 20, August 13, August 27, September 27, October 10, October 17, November 8, November 19, November 29, December 3, December 7, December 12, and December 13, 2001, and January 2, January 25, January 31, February 11, February 18, February 22, February 27, March 7, March 18, March 22, and March 26, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 231, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

Also, the license is amended by adding the following paragraphs to 2.C(3) of Facility Operating License NPF-4:

d. The licensee is authorized to relocate certain Technical Specification requirements previously included in Appendix A to licensee-controlled documents, as described in Table R, Relocated Specifications and Removed Details, attached to the NRC staff's Safety Evaluation enclosed with Amendment No. 231. These requirements shall be relocated to the appropriate documents no later than September 2, 2002.

e. The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 231shall be as follows:

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

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

This license condition is effective as of its date of issuance.

3. This license amendment is effective as of its date of issuance and shall be implemented no later than September 2, 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachments:
1. Pages 3 and 3a of License NPF-4
2. Changes to the Technical Specifications

Date of Issuance: April 5, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 231

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following pages of the License and Appendix "A" Technical Specifications with the enclosed pages as indicated.

Remove Pages	Insert Pages
License Page 3	License Page 3
---	License Page 3a
Current TS (in their entirety)	Improved TS (in their entirety)

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 212
License No. NPF-7

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company et al., (the licensee) dated December 11, 2000, as supplemented by letters dated May 30, June 18, July 16, July 20, August 13, August 27, September 27, October 10, October 17, November 8, November 19, November 29, December 3, December 7, December 12, and December 13, 2001, and January 2, January 25, January 31, February 11, February 18, February 22, February 27, March 7, March 18, March 22, and March 26, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-7 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

Also, the license is amended by adding the following paragraphs to 2.C(3) of Facility Operating License NPF-7:

c. The licensee is authorized to relocate certain Technical Specification requirements previously included in Appendix A to licensee-controlled documents, as described in Table R, Relocated Specifications and Removed Details, attached to the NRC staff's Safety Evaluation enclosed with Amendment No. 212. These requirements shall be relocated to the appropriate documents no later than September 2, 2002.

d. The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 212 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 that 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 subject to the modified acceptance criteria is due at the end of the first surveillance interval that began on the date the surveillance was last performed prior to the implementation of this amendment.

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

This license condition is effective as of its date of issuance.

3. This license amendment is effective as of its date of issuance and shall be implemented no later than September 2, 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachments:
1. Pages 3 and 3a of License NPF-7
2. Changes to the Technical Specifications

Date of Issuance: April 5, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 212

TO FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the License and Appendix "A" Technical Specifications with the enclosed pages as indicated.

Remove Pages	Insert Pages
License Page 3	License Page 3
---	License Page 3a
Current TS (in their entirety)	Improved TS (in their entirety)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 231 TO FACILITY OPERATING LICENSE NO. NPF-4

AND AMENDMENT NO. 212 TO FACILITY OPERATING LICENSE NO. NPF-7

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By application dated December 11, 2000, as supplemented by letters dated May 30, June 18, July 16, July 20, August 13, August 27, September 27, October 10, October 17, November 8, November 19, November 29, December 3, December 7, December 12, and December 13, 2001, January 2, January 25, January 31, February 11, February 18, February 22, February 27, March 7, March 18, March 22, and March 26, 2002, Virginia Electric and Power Company (the licensee) requested amendments to the Facility Operating Licenses and Technical Specifications (TS) for the North Anna Power Station (NAPS), Units 1 and 2. The proposed amendments would convert the current TS (CTS) to improved TS (ITS).

NAPS has been operating with TS issued with the original Facility Operating Licenses on November 26, 1977 (for Unit 1), and April 11, 1980 (for Unit 2), as amended. The proposed conversion to the ITS is based upon:

- NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," Revision 1, dated April 1995;
- The current NAPS 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 improved TS for NAPS are referred to as the ITS, the current TS are referred to as the CTS, and the improved standard TS, 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. Several post-submittal letters of request for additional information (RAI) and a series of telephone conference calls were required during the course of this review. These RAIs 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 matters of a

generic nature that were not in the STS. The staff requested that the licensee submit such generic issues 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 NAPS ITS. Consistent with the Final Policy Statement, the licensee proposed transferring some CTS requirements to licensee-controlled documents (such as the NAPS Updated Final Safety Analysis Report (UFSAR), for which changes to the documents by the licensee are controlled by a regulation such as 10 CFR 50.59 and may be changed without prior NRC approval). NRC-controlled documents, such as the TS, may not be changed by the licensee without prior NRC approval. In addition, human factors principles were emphasized to add clarity to the CTS requirements being retained in the ITS, and to define more clearly the appropriate scope of the ITS. Further, significant changes were proposed to the CTS Bases to make each ITS requirement clearer and easier to understand.

The overall objective of the proposed amendments, consistent with the Final Policy Statement, is to rewrite, reformat, and streamline the TS for NAPS, while still satisfying the requirements of 10 CFR 50.36. Since the licensee submitted the December 11, 2000, application, a number of amendments to the NAPS operating license have been approved. The following table provides the subjects of the amendments and the dates of issuance.

Amendment Nos. Unit 1	Unit 2	Description of Change	Date
225	206	Increase Boron Concentration Limits in Reactor Coolant System during Refueling and Establish Boron Limits for Spent Fuel Pool.	3/20/01
226	207	Pressure-Temperature Limits, Low Temperature Overpressure Protection (LTOP) System Setpoints, and LTOP System Effective Temperature.	5/02/01
227	208	Increase Fuel Enrichment and Spent Fuel Pool Soluble Boron and Fuel Burnup Credit	6/15/01
228	209	Control Room Emergency Habitability Systems Increase Number of Compressed Air Bottles and Revise Differential Pressure Limit for Filter Assemblies	12/12/01
229	210	Elimination of Post Accident Sampling System Requirements	12/19/01
230		Delete Obsolete License Conditions for North Anna Unit 1	1/31/02
	211	Delete Obsolete License Conditions for North Anna Unit 2	3/19/02

The licensee has incorporated these amendments, as appropriate, into the ITS.

The license conditions included in the conversion amendment will make enforceable the following aspects of the conversion: (1) the relocation of requirements from the CTS and (2) the implementation schedule for new and revised surveillance requirements (SRs) in the ITS. The Commission's proposed action for the ITS conversion was published in the *Federal Register* on February 26, 2002 (67 FR 8827). During its review, the 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 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 NAPS. This SE also explains the staff's conclusion that the ITS, which are based on the STS as modified by plant-specific changes, are consistent with the NAPS current licensing basis and the requirements of 10 CFR 50.36.

The 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 NAPS. The staff approves the licensee's changes to the CTS with modifications documented in the licensee's supplemental submittals.

For the reasons stated *infra* in this SE, the 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 accord 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 TS. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and the mitigation of accident consequences. As recorded in the Statements of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports" (33 FR 18610, December 17, 1968), the Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories 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 TS.

For several years, NRC and industry representatives have sought to develop guidelines for improving the content and quality of nuclear power plant TS. On February 6, 1987, the Commission issued an interim policy statement on TS improvements, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, utility owners groups and the staff developed improved STS, such as NUREG-1431, that would establish models of the Commission's policy for each primary reactor type. In addition, the staff, licensees, and owners groups developed generic administrative and editorial guidelines in the form of a "Writer's Guide" for preparing TS, 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 was 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 (NSSS) 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 apply to NAPS .

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 TS and defined the guidance criteria to be used in determining which of the LCOs and associated SRs should remain in the TS. The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TS, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

> [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 TS; those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. The

Commission codified the four criteria in 10 CFR 50.36 (60 FR 36953, July 19, 1995). The four criteria are 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 safety assessment has shown to be significant to public health and safety.*

Part 3.0 of this SE explains the staff's conclusion that the conversion of the NAPS CTS to ITS based on STS, as modified by plant-specific changes, is consistent with the NAPS 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 NAPS ITS application, the staff evaluated five kinds of changes to the CTS as defined by the licensee. The 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. Following are the five types of CTS changes:

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

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

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

LA — Less Restrictive - Removal of Detail - Changes to the CTS that eliminate detail and relocate the detail to a licensee-controlled document. Typically, this involves details of system design and function, or procedural detail on methods of conducting a surveillance. This type of change is included with Relocated Specifications in Table R as described below.

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 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 staff's comments were documented as RAIs and forwarded in letters dated April 23, May 21, June 1, June 4, June 22, July 2, July 30, July 31, September 6, September 7, September 18, October 3, October 10, October 16, November 7, and December 7, 2001, and February 11, 2002. The licensee provided responses to the RAIs in supplemental letters dated June 18, July 16, July 20, August 13, August 27, September 27, October 10, October 17, November 8, November 19, November 29, December 3, December 7, December 12, and December 13, 2001, and January 2, January 25, January 31, February 11, February 18, February 22, February 27, March 7, March 18, March 22, and March 26, 2002. The letters clarified the licensee's basis for translating the CTS requirements into ITS. For items that have been reviewed by the staff as stated in this SE, the staff finds that the licensee's submittals, including the responses to the RAIs, provide sufficient detail to allow the 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 four tables attached to this SE:

- Table A - Administrative Changes
- Table M - More Restrictive Changes
- Table L - Less Restrictive Changes
- Table R - Relocated Specifications and Removed Details

These tables 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.

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

- Section A Administrative
- Section B More Restrictive
- Section C Less Restrictive
- Section D Less Restrictive Removal of 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

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 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 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 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 TS) but that do not involve a change in requirements;
- Wording changes and additions that are consistent with CTS interpretation and practice, and that more clearly or explicitly state existing requirements;
- Deletion of TS which 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 TS.

Table A lists the administrative changes being made in the NAPS 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 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. More Restrictive

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 to be operable, more restrictive requirements to restore inoperable equipment, and more restrictive SRs. Table M lists the more restrictive changes being made in the NAPS 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. Less Restrictive

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 TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of: (1) generic NRC actions, (2) new staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the owners groups' comments on the STS. The 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 NAPS design was also reviewed to determine if the specific design basis and licensing basis for NAPS are consistent with the technical basis for the model requirements in the STS, and thus provide a basis for the ITS.

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

- Relaxation of LCO Requirement (Category 1)
- Relaxation of Applicability (Category 2)
- Relaxation of Completion Time (Category 3)
- Relaxation of Required Action (Category 4)
- Deletion of SRs (Category 5)
- Relaxation of SRs Acceptance Criteria (Category 6)
- Relaxation of Surveillance Frequency (Category 7)
- Deletion of Reporting Requirements (Category 8)

The following discussions address why portions of various specifications within each of these eight categories of information or specific requirements are not required to be included in ITS:

1. Relaxation of LCO Requirement (Category 1)

   The CTS contain LCOs that are overly restrictive because they specify limits on operational and system parameters and on system operability beyond those necessary to meet safety analysis assumptions. The CTS also contain administrative controls that do not contribute to the safe operation of the plant. The ITS, consistent with the guidance in the STS, omit such operational limits and administrative controls. This category of change includes: (1) deletion of equipment or systems addressed by the CTS LCOs that are not required or assumed to function by the applicable safety analyses, (2) addition of explicit exceptions to the CTS LCO requirements consistent with the guidance of the STS and normal plant operations to provide necessary operational flexibility but without a significant safety impact, and (3) deletion of miscellaneous administrative controls sometimes contained in action requirements that have no effect on safety. Deletion of such administrative controls allows operators to more clearly focus on issues important to safety. The ITS LCOs and administrative controls resulting from these changes will continue to maintain an adequate degree of protection consistent with the safety analysis, while providing an improved focus on issues important to safety and necessary operational flexibility without adversely affecting the safe operation of the plant. Therefore, these less restrictive changes, which are consistent with STS and fall within Category 1, are acceptable.
2. Relaxation of Applicability (Category 2)

Reactor operating conditions are used in CTS to define when LCO features are required to be operable. CTS applicability requirements can be specifically defined terms of reactor conditions, such as hot shutdown, cold shutdown, reactor critical, or power operating conditions. CTS applicability requirements can also be more general. Depending on the circumstances, the CTS may require that an LCO be maintained within limits in “all modes” or “any operating mode.” Generalized applicability conditions are not contained in STS; therefore, ITS eliminate CTS requirements such as “all modes” or “any operating mode,” replacing them with ITS-defined modes or applicable conditions that are consistent with the application of the plant safety analysis assumptions for operability of the required features.

In another application of this category of change, CTS requirements may be eliminated during conditions for which the safety function of the specified safety system is met because the feature is performing its intended safety function. Deleting applicability requirements that are indeterminate or that are inconsistent with application of accident analyses assumptions is acceptable because when LCOs cannot be met, the TS are satisfied by exiting the specified LCO’s applicability, thus taking the plant out of the conditions that require the safety system to be operable. Therefore, these changes, which are consistent with STS and fall within Category 2, are acceptable.

3. Relaxation of Completion Time (Category 3)

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. 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 operable 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. Therefore, required action completion time extensions, which are consistent with STS and fall within Category 3, are acceptable.

4. Relaxation of Required Action (Category 4)

An LCO is 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, the ITS actions result in less restrictive requirements for taking the plant outside the applicability into shutdown conditions. For example, changes in this category 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. Therefore, these changes, which are consistent with STS and fall within Category 4, are acceptable.

5. Deletion of SRs (Category 5)

   The CTS require maintaining the LCO equipment operable by conducting SRs in accordance with the plant-specific equipment. The changes in this type relate to elimination of SRs in CTS that were no longer required because equipment had been replaced or the features that required surveillance actions had been replaced, or features with surveillance activities were duplicated by other new ITS requirements. Therefore, these changes, which fall within Category 5 and are consistent with the STS, are acceptable.

6. Relaxation of SRs Acceptance Criteria (Category 6)

   Relaxation of CTS SRs acceptance criteria provides operational flexibility, consistent with the guidance of the STS, but does not reduce the level of assurance of operability provided by the successful performance of the surveillance. Such revised acceptance criteria are acceptable because they remain consistent with the application of the plant safety analysis assumptions for the operability of the LCO-required features.

   Relaxation of CTS SRs performance conditions includes requiring de-energized equipment (e.g., instrumentation channel checks) and equipment that is already performing its intended safety function (e.g., position verification of valves locked in their safety actuation position). These changes are acceptable because the existing surveillances are not necessary to ensure the capability of the affected components to perform their intended functions. Another relaxation of SRs performance conditions is the allowance to verify the position of valves in high radiation areas by administrative means. This change is acceptable because licensee controls regarding access to high radiation areas make the likelihood of mispositioning such valves negligible. Therefore, these changes, which are consistent with STS and fall within Category 6, are acceptable.

7. Relaxation of Surveillance Frequency (Category 7)

   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 frequency thereafter, the CTS require verifying 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 include relaxing both the acceptance criteria and the conditions of performance. These CTS SR relaxations are consistent with the STS.

   Also, the ITS permits 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. Therefore, because of the reasons stated, less restrictive changes to CTS SRs falling within Category 7 are acceptable.

8. Deletion of Reporting Requirements (Category 8)

The CTS include requirements to submit special reports to the NRC when specified limits or conditions are not met. Typically, the time period for the report to be issued is within 30 days. However, the ITS eliminates the TS requirements for special reports and instead relies on the reporting requirements of 10 CFR 50.73. The ITS changes to the reporting requirements are acceptable because 10 CFR 50.73 provides adequate reporting requirements, and the special reports do not affect continued plant operation.

The CTS also include requirements for reports to be made to the NRC on data gathered as part of routine plant programs. These requirements are removed from the ITS. The requirement to report test frequency changes that occur due to consecutive SR failures has been deleted since the test schedule is already covered by the TS. In addition, a historical review has shown the SR has never failed. These changes are consistent with STS, are specified as Type 8, and are acceptable.

For the reasons presented above, these less restrictive requirements are acceptable because they will not affect the safe operation of the plant. The ITS requirements are consistent with current licensing practices, 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 NAPS 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 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. <u>Less Restrictive Removal of Details</u>

When requirements have been shown to give little or no safety benefit, their removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the owners groups comments on STS. The 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 NAPS 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 5 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 UFSAR 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 (UFSAR Chapter 17). 10 CFR 50.59 specifies controls for changing the facility as described in the UFSAR. 10 CFR 50.54(a) specifies criteria for changing the QA plan. The Technical Requirements Manual (TRM) is a general reference in the UFSAR and is subject to administrative controls that include the requirement to perform 10 CFR 50.59 evaluations for changes made to the TRM. The ITS Bases also contain descriptions of system design. ITS 5.5.13 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 UFSAR 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). ITS Section 5.6, Administrative Controls, includes the programmatic requirements for the COLR.

Type 2 - Removing Descriptions of System Operation

The plans for the normal and emergency operation of the facility are required to be described in the UFSAR by 10 CFR 50.34. ITS 5.4.1.a requires written procedures to be established, implemented, and maintained for plant operating procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978. Controls specified in 10 CFR 50.59 apply to changes in procedures as described in the UFSAR. The ITS Bases also contain descriptions of system operation. The NAPS CTS include instrumentation trip setpoints and Allowable Values. Trip setpoints are instrument field settings, and Allowable Values are the limiting values of the instrument trip setpoints before the LCO is exceeded. The relationship between the trip setpoints and the Allowable Values is determined through the setpoint methodology approved by the staff. Trip setpoints are system operation details that can be adequately controlled by licensee-controlled documents without adversely affecting safe operation of the plant. Allowable Values are specified in the ITS, while trip setpoints are relocated to the TRM. It is acceptable to remove details of system operation from the TS because this type of information will be adequately controlled in the UFSAR (which references the TRM) and the TS Bases, as appropriate.

Type 3 - Removing Procedural Details for Meeting TS Requirements and Related Reporting Requirements

Details for performing TS Actions and SRs are more appropriately specified in the plant procedures required by ITS 5.4.1, and described in the UFSAR and ITS Bases. For example, control of the plant conditions for surveillance testing is more appropriately governed by procedures and scheduling and has previously been determined to be unnecessary as a TS requirement. As indicated in GL 91-04, allowing this procedural control is consistent with the vast majority of other SRs that do not dictate plant conditions for surveillances. Prescriptive procedural information in an ITS action requirement is unlikely to contain all procedural considerations necessary for the plant operators to complete the actions required, and referral to plant procedures is therefore required in any event. Other changes to procedural details include those associated with limits retained in the ITS. The QA Program is approved by the NRC and contained in UFSAR Chapter 17, and changes to the QA Program are controlled by 10 CFR 50.54(a). The Offsite Dose Calculation Manual (ODCM) is required by ITS 5.5.1. The TRM is referenced in the UFSAR, and changes to the TRM are controlled by 10 CFR 50.59. The Inservice Testing (IST) program is required by ITS 5.5.7.

Type 4 - Removing Performance Requirements for Indication-Only Instrumentation and Alarms

Details for performance requirements for indication-only instrumentations and alarms are more appropriately specified in the plant procedures required by ITS 5.4.1, the UFSAR, and the Bases. Prescriptive procedural information in an action requirement is unlikely to contain all procedural considerations necessary for the plant operators to complete the actions required, and referral to plant procedures, based on TS Bases, is therefore required in any event. The removal of these kinds of procedural details from the CTS is acceptable because they will be adequately controlled by NRC requirements, the UFSAR, plant procedures, and the Bases, as appropriate. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Removal of requirements for indication-only instrumentation is acceptable because such instrumentation usually does not support system operability. Therefore, it is acceptable to remove Type 4 details from the CTS and place them in licensee-controlled documents.

Type 5 - Removal of Cycle-Specific Parameter Limits from the CTS to the COLR

Other changes to procedural details include those associated with limits retained in the ITS. For example, the ITS requirement may refer to programmatic requirements such as the COLR, included in ITS Section 5.6, which specifies the scope of the limits contained in the COLR and mandates NRC approval of the analytical methodology. Removal of requirements for programmatic requirements such as the COLR is acceptable because such a program usually does not support system operability. Therefore, it is acceptable to remove Type 5 details from the CTS and place them in licensee-controlled documents with references to ITS Chapter 5.0.

Table R lists the less restrictive removal of detail changes being made in the NAPS ITS conversion. Table R is organized in STS order by each LA- type and R-type DOC. It includes the following: (1) the DOC identifier (e.g., 3.1.1 followed by LA1 means STS 3.1.1, DOC LA1); (2) the reference numbers of the associated CTS requirements; (3) a summary description of the relocated details and requirements; (4) the name of the licensee-controlled document to contain the relocated details and requirements (location); (5) the regulation (or ITS Specification) for controlling future changes to relocated requirements (change control process); and (6) a characterization of the type of change (not applicable to R-type DOCs).

The 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 5.5.13, “Technical Specifications (TS) Bases Control Program.”
- UFSAR (which references 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.
- ODCM controlled by ITS 5.5.1.
- COLR controlled by ITS 5.6.4.
- QA Plan, as approved by the NRC and referenced in the UFSAR, 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 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 TS (an NRC-controlled document) to appropriate licensee-controlled documents. This section discusses the relocation of entire specifications in the CTS to licensee-controlled documents. These specifications include the LCOs, Action Statements (i.e., Actions), and associated SRs. In its application and its supplements, the licensee proposed relocating such specifications from the CTS to the TRM, and the ODCM, as appropriate. The NRC staff has reviewed the licensee's submittals and finds that relocation of these requirements to the TRM, and the ODCM, 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 the TRM, and the ODCM, will be adequately controlled by 10 CFR 50.59, 10 CFR 50.54(a), 10 CFR 50.55a, and ITS 5.5.1, 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 all specifications that are being relocated from the CTS to licensee-controlled documents. Table R is combined with type LA changes; however the relocated LA items are organized as described in Section 3.0.D above.

Table R lists the relocated changes being made in the NAPS 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 specifications; (3) summary descriptions of the relocated CTS specifications; (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 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.

1. 3.1.1.3.1 BORON DILUTION - Reactor Coolant Flow

   CTS 3.1.1.3.1 requires a minimum reactor coolant system (RCS) flow of 3000 gpm in all MODES. Various accident analyses assume adequate reactor coolant flow for heat removal and boron mixing. However, a specific flow rate is not assumed as an initial condition of any DBA or transient and is not credited for mitigation of any DBA or transient. The ITS contains adequate controls to ensure that RCS flow meets the general accident analysis assumption. In MODES 1, 2, and 3, at least one Reactor Coolant Pump (RCP) is required to be in operation, which provides flow in excess of 3000 gpm. In MODE 4, either an RCP or Residual Heat Removal (RHR) train is required to be in operation, and in MODES 5 and 6 at least one RHR train is required to be in operation. The ITS Bases state that when an RHR train is required to provide RCS flow, the flow rate must be sufficient for decay heat removal and boron mixing. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Boron Dilution - Reactor Coolant Flow LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

2. 3.1.2.1 FLOW PATHS – Shutdown

   CTS 3.1.2.1 provides requirements on the boration systems flow paths during shutdown. The boration systems are part of the Chemical and Volume Control System (CVCS) and provide the means to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the shutdown margin. The boration system is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of the boron dilution accident, the accident is addressed by preventing its occurrence or by terminating the event before the required shutdown margin is lost, not by boration. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Flow Paths – Shutdown LCO and Surveillances may be relocated to other plant controlled documents outside the ITS.

3. 3.1.2.2 FLOW PATHS – Operating

   CTS 3.1.2.2 provides requirements on the boration systems flow paths during operation. The boration systems are part of the CVCS and provide the means to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the shutdown margin. The boration system is not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. The Emergency Core Cooling System (ECCS) and Refueling Water Storage Tank (RWST) are credited in the accident analyses. In the case of the boron dilution accident, the accident is addressed by preventing its occurrence or by terminating the event before the required shutdown margin is lost, not by boration. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Flow Paths – Operating LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

4. 3.1.2.3 CHARGING PUMP – Shutdown

   CTS 3.1.2.3 provides requirements on the charging pumps during shutdown when used as part of the boration system. The charging pumps in the boration system are part of the CVCS and provide the means to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the shutdown margin. The charging pumps in the boration system are not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of the boron dilution accident, the accident is addressed by preventing its occurrence or by terminating the event before the required shutdown margin is lost, not by boration. OPERABILITY of the charging pumps is required as part of the ECCS, which is addressed in other specifications. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Charging Pump – Shutdown LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

5. 3.1.2.4 CHARGING PUMPS – Operating

CTS 3.1.2.4 provides requirements on the charging pumps during operation when used as part of the boration system. The charging pumps in the boration system are part of the CVCS and provide the means to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the shutdown margin. The charging pumps in the boration system are not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. The ECCS and RWST are credited in the accident analyses. In the case of the boron dilution accident, the accident is addressed by preventing its occurrence or by terminating the event before the required shutdown margin is lost, not by boration. OPERABILITY of the charging pumps is required as part of the ECCS, which is addressed in other specifications. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Charging Pumps – Operating LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

6. Unit 1; 3.1.2.5 BORIC ACID TRANSFER PUMPS – Shutdown

Unit 1 CTS 3.1.2.5 provides requirements on the boric acid transfer pumps during shutdown. The boric acid transfer pumps are part of the CVCS and provide the means to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the shutdown margin. The boric acid transfer pumps are not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of the boron dilution accident, the accident is addressed by preventing its occurrence or by terminating the event before the required shutdown margin is lost, not by boration. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Boric Acid Transfer Pumps – Shutdown LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

7. Unit 1; 3.1.2.6 BORIC ACID TRANSFER PUMPS – Operating

Unit 1 CTS 3.1.2.6 provides requirements on the boric acid transfer pumps during operation. The boric acid transfer pumps are part of the CVCS and provide the means to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the shutdown margin. The boric acid transfer pumps are not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. The ECCS and RWST are credited in the accident analyses. In the case of the boron dilution accident, the accident is addressed by preventing its occurrence or by terminating the event before the required shutdown margin is lost, not by boration. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Boric Acid Transfer Pumps – Operating LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

8. 3.1.2.7 BORATED WATER SOURCES – Shutdown

CTS 3.1.2.7 provides requirements on the borated water sources during shutdown. The borated water sources - shutdown are part of the CVCS and provide the means to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the shutdown margin. The borated water sources are not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. In the case of the boron dilution accident, the accident is addressed by preventing its occurrence or by terminating the event before the required shutdown margin is lost, not by boration. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Borated Water Sources – Shutdown LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

9. 3.1.2.8 BORATED WATER SOURCES – Operating

CTS 3.1.2.8 provides requirements on the borated water sources during operation. The borated water sources - operating are part of the CVCS and provide the means to control the chemical neutron absorber (boron) concentration in the RCS and to help maintain the shutdown margin. The borated water sources are not assumed to be OPERABLE to mitigate the consequences of a DBA or transient. The ECCS and RWST are credited in the accident analyses and are required by other specifications. In the case of the boron dilution accident, the accident is addressed by preventing its occurrence or by terminating the event before the required shutdown margin is lost, not by boration. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Borated Water Sources – Operating LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

10. 3.1.3.3 POSITION INDICATOR CHANNELS – Shutdown

CTS 3.1.3.3 provides requirements on the rod position indicator channels during shutdown (MODES 3, 4, and 5 with the reactor trip system (RTS) breakers in the closed position). The control rod position indicator channels provide indication of rod position to the operator. This indicator is used by the operator to verify that the rods are correctly positioned, and to verify the rods are inserted into the core following a reactor trip. The rod position indicator is also used during reactor startup. However, no DBA or transient initiated in MODES 3, 4, or 5 with the RTS breakers in the closed position assumes operator action to manually trip the reactor or to take some alternative action if an automatic reactor trip does not occur. With the reactor critical, the rod position indicator is used to verify that the insertion, sequence, and overlap limits are met. These are related to SHUTDOWN MARGIN and core power distribution limits. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Position Indicator Channels – Shutdown LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

11. 3.3.3.1 RADIATION MONITORING INSTRUMENTATION

CTS 3.3.3.1 states that the radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits. Portions of the Radiation Monitoring Instrumentation specification, as shown in the CTS markup, are addressed in ITS 3.4.15, RCS Leakage Detection Instrumentation, and ITS 3.3.3, Post-Accident Monitoring (PAM) Instrumentation. Those portions are not addressed in this change. The Radiation Monitoring Instrumentation monitors radiation levels in selected plant locations and indicates abnormal or unusually high radiation levels. The radiation monitors are not assumed in the accident analyses to provide signals to prevent initiation of a DBA or transient or to mitigate a DBA or transient. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Radiation Monitoring LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

12. 3.3.3.2 MOVABLE INCORE DETECTORS

CTS 3.3.3.2 provides requirements on the Movable Incore Detector Instrumentation when required to monitor the flux distribution within the core. The Movable Incore Detector System is used for periodic surveillance of the power distribution, and for calibration of the excore detectors. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Movable Incore Detectors LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

13. 3.3.3.3 SEISMIC INSTRUMENTATION

CTS 3.3.3.3 for Unit 1 states the Seismic Monitoring Instrumentation shown in Table 3.3-7 shall be OPERABLE. The Seismic Monitoring Instrumentation is used to record data for use in evaluating the effect of a seismic event. The Seismic Monitoring Instrumentation is not used to mitigate a DBA or transient. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Seismic Instrumentation LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

14. 3.3.3.4 METEOROLOGICAL INSTRUMENTATION

CTS 3.3.3.4 for Unit 1 states the Meteorological Monitoring Instrumentation shown in Tables 3.3-8 and 4.3-5 shall be OPERABLE. The Meteorological Monitoring Instrumentation is used to record meteorological data for use in evaluating the effect of an accidental radioactive release from the plant. The Meteorological Monitoring Instrumentation is not used to mitigate a DBA or transient. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Meteorological Instrumentation LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

15. 3.3.3.9 LOOSE PARTS MONITORING SYSTEM

Unit 1 CTS 3.3.3.9 requires the OPERABILITY of the loose parts detection instrumentation that can detect loose metallic parts in the RCS in order to avoid damage to the RCS components. The Unit 2 TS do not contain this requirement. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Loose Parts Monitoring System LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

16. 3.3.3.11 EXPLOSIVE GAS MONITORING INSTRUMENTATION

CTS 3.3.3.11 requires the Explosive Gas Monitoring Instrumentation be OPERABLE. The Explosive Gas Monitoring Instrumentation is used to ensure that the oxygen limits of the Waste Gas Holdup System are not exceeded. The oxygen concentration limit in the Waste Gas Holdup Tank ensures that the concentration of potentially explosive gas mixtures in the Waste Gas Holdup System is maintained below the flammability limits.

This instrumentation is not credited in preventing or mitigating any DBA or transient as the safety analysis concerning the Waste Gas Holdup System assumes a storage tank rupture with no mitigation. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Explosive Gas Monitoring Instrumentation LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

17. 3.4.6.3 PRIMARY-TO-SECONDARY LEAKAGE

CTS 3.4.6.3 provides limits on primary-to-secondary leakage in addition to the limits in CTS 3.4.6.2 and ITS 3.4.13. These additional limits lower the amount of allowed primary-to-secondary leakage when the reactor is operating above 50% power and were implemented to reduce the probability of a steam generator (SG) tube rupture (SGTR) following the Unit 1 SGTR event at NAPS Unit 1 on July 15, 1987. The CTS 3.4.6.2 leakage limits continued to be used in the accident analysis, not the additional limits in CTS 3.4.6.3. The NAPS Units 1 and 2 SGs have been replaced with models that are not susceptible to the fatigue-induced cracks that resulted in the tube rupture. As a result, these additional limits are not needed to lower the probability of an SGTR. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Primary-to-Secondary Leakage LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

18. 3.4.6.4 PRIMARY-TO-SECONDARY LEAKAGE DETECTION SYSTEMS

CTS 3.4.6.4 states requirements on primary-to-secondary leakage detection systems. These leakage detection systems are in addition to those systems required by CTS 3.4.6.1 and ITS 3.4.15 and were installed to monitor the stringent primary-to-secondary leakage limits in CTS 3.4.6.3. These additional primary-to-secondary leakage detection systems were added to the TS following the Unit 1 SGTR event at NAPS Unit 1 on July 15, 1987. Subsequently, the NAPS Units 1 and 2 SGs have been replaced and SG primary-to-secondary leakage is insignificant. As a result, the requirements in ITS 3.4.15 are sufficient to indicate significant abnormal RCS leakage. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Primary-to-Secondary Leakage Detection Systems LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

19. 3.4.7 CHEMISTRY

CTS 3.4.7 provides limits on the oxygen, chloride, and fluoride content in the RCS to minimize corrosion. Minimizing corrosion of the RCS will reduce the potential for RCS leakage or failure due to stress corrosion, and ultimately ensure the structural integrity of the RCS. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Chemistry LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

20. 3.4.9.2 PRESSURIZER

CTS 3.4.9.2 states that the pressurizer temperature shall be limited to a maximum heatup of 100°F or cooldown of 200°F in any 1-hour period and a maximum spray water temperature and pressurizer temperature differential of 320°F. The pressurizer temperature limits are placed on the pressurizer to prevent non-ductile failure. The limits meet the requirements given in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Appendix G. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Pressurizer LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

21. 3.4.10.1 STRUCTURAL INTEGRITY - ASME Code Class 1, 2 & 3 Components

CTS 3.4.10.1 provides requirements for the ASME Code Class 1, 2 and 3 components to ensure their structural integrity. These requirements are in addition to the requirements in CTS 4.0.5. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Structural Integrity - ASME Code Class 1, 2, and 3 Components LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

22. 3.4.11.1 REACTOR VESSEL HEAD VENT

CTS 3.4.11.1 provides requirements on the reactor vessel head vents. The reactor coolant head vents are provided to exhaust noncondensible gases or steam, which could inhibit core cooling, from the RCS. The reactor vessel head vents are not credited in any UFSAR accident analysis. They are included in the Emergency Operating Procedures for mitigation of beyond DBAs. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Reactor Vessel Head Vent LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

23. 3.5.4.2 HEAT TRACING

CTS 3.5.4.2 states, “At least two independent channels of heat tracing shall be OPERABLE for the boron injection tank and for the heat traced portions of the associated flow paths.” The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Boron Injection Tank Heat Tracing LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

24. 3.7.1.6 STEAM TURBINE ASSEMBLY

CTS 3.7.1.6 states that the structural integrity of the steam turbine assembly shall be maintained in MODES 1 and 2. The steam turbine assembly is used to provide the motive force for the main electrical generator. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Steam Turbine Assembly LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

25. 3.7.1.7 TURBINE OVERSPEED

CTS 3.7.1.7 states that at least one turbine overspeed protection system shall be OPERABLE in MODES 1, 2, and 3. The turbine overspeed protection system is used to prevent a turbine overspeed condition that could result in turbine damage and serves no accident mitigation function in any MODE. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Turbine Overspeed LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

26. 3.7.2.1 STEAM GENERATOR PRESSURE/TEMPERATURE (P/T) LIMITATION

CTS 3.7.2.1 states that the temperature of both the primary and secondary coolants in the SGs shall be greater than 70°F when the pressure of either coolant in the SG is greater than 200 psig at all times. The SG P/T Limitation serves no accident mitigation function in any MODE. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the SG P/T Limitation LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

27. 3.7.3.2 COMPONENT COOLING WATER (CCW) SUBSYSTEM - Shutdown

CTS 3.7.3.2 states that two CCW loops shall be OPERABLE. It is applicable when both units are in MODES 5 or 6. The primary function of the CCW System is to provide cooling water to the RHR heat exchangers, but does not warrant its own LCO. If insufficient CCW is available for RHR, RHR is declared inoperable and the Conditions and Actions for CCW in CTS are the same as those for RHR. Unlike other Westinghouse plants, RHR does not share components with the ECCS, and thus does not play a role in DBA mitigation in MODES 1, 2, 3, and 4. Other plants use CCW for DBA mitigation functions other than ECCS in MODES 1, 2, 3, and 4, but the CCW system at NAPS is not used in that manner. This makes the CCW System at NAPS different from the CCW System described in the NUREG STS, and retaining the CCW requirement for MODES 5 and 6 for supporting RHR or any other components not assumed in DBA analysis is inappropriate. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the CCW Subsystem - Shutdown LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

28. 3.7.4.2 SERVICE WATER SYSTEM - Shutdown

CTS 3.7.4.2 states that one service water (SW) loop shall be OPERABLE when both units are in MODES 5 or 6. The SW System in MODES 5 or 6 is used to provide cooling water to various safety- and nonsafety-related systems. Its principal safety function is to cool the recirculation spray heat exchangers that are not required to be OPERABLE in MODES 5 or 6. It also provides cooling water to the CCW system (which supports no accident loads), the main control room coolers, instrument air compressors, and charging pump gearbox coolers. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the SW System - Shutdown LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

29. 3.7.5.1.b ULTIMATE HEAT SINK - North Anna Reservoir

CTS 3.7.5.1.b states that one of the ultimate heat sinks (UHSs) that shall be OPERABLE is the North Anna Reservoir with a minimum water level at or above elevation 244 Mean Sea Level, USCG Datum, and average water temperature of ≤ 95° F as measured at the condenser inlet. The North Anna Reservoir provides makeup to the SW Reservoir for 30 days after a DBA as necessary to maintain cooling water inventory, ensuring a continued cooling capability. The SW Reservoir is credited as the UHS for the DBA. The SW Reservoir contains adequate water to provide at least 30 days of cooling to support simultaneous safe shutdown and cooldown of both units and their maintenance in a safe-shutdown condition. The SW Reservoir also provides sufficient cooling for at least 30 days in the event of an accident in one unit, to permit control of that accident and permit simultaneous safe shutdown and cooldown of the remaining unit and maintain them in a safe-shutdown condition. The North Anna Reservoir serves as a backup to the SW Reservoir. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the UHS - North Anna Reservoir LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS. Details of the staff's SE for the relocation of this CTS item is shown in Attachment 6 to this SE.

30. 3.7.6.1 FLOOD PROTECTION

CTS 3.7.6.1 states the maximum elevation of the North Anna Reservoir. If this limit is exceeded, flood control measures are required to protect safety-related equipment. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Flood Protection LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

31. 3.7.9.1 RESIDUAL HEAT REMOVAL SYSTEMS - (RHR) Operating

CTS 3.7.9.1 states that two RHR subsystems shall be OPERABLE in MODES 1, 2, and 3. The RHR System is used to remove decay heat from the reactor in MODES 4, 5, and 6. The RHR does not operate in MODES 1, 2 and 3 and must be isolated from the RCS in those MODES to prevent overpressurization of the RHR components. The RHR System serves no accident mitigation function in any MODE. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the RHR - Operating LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

32. 3.7.10 SNUBBERS

CTS 3.7.10 states that snubbers shall be OPERABLE. The OPERABILITY of snubbers ensures that the RCS and other safety-related fluid systems are adequately restrained and supported during an earthquake and are free to expand and contract during normal operation as the system temperature changes. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Snubbers LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

33. 3.7.11.1 SEALED SOURCE CONTAMINATION

CTS 3.7.11.1 states each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma-emitting materials or 5 microcuries of alpha-emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Sealed Source Contamination LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

34. 3.7.12.1 SETTLEMENT OF CLASS 1 STRUCTURES

CTS 3.7.12.1 and Table 3.7-5 provide limits on the total and differential settlement of Class 1 structures. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Settlement of Class 1 Structures LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

35. 3.7.13 GROUNDWATER LEVEL - SW Reservoir

CTS 3.7.13 requires periodic measurement of the groundwater level at locations around the SW Reservoir. The groundwater level of the SW Reservoir is used to monitor long-term performance of the SW Reservoir dike. Failure to meet the requirements of the LCO does not result in the inoperability of the SW System. The ACTIONS direct that evaluations be performed to determine cause and consequences of the high groundwater level. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Groundwater Level - SW Reservoir LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

36. 3.8.2.5 (Unit 2) CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Unit 2 CTS 3.8.2.5 states the primary and backup containment penetration conductor overcurrent protective devices associated with each containment electrical penetration circuit shall be OPERABLE. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Containment Penetration Conductor Overcurrent Protective Devices LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

37. 3.8.2.6 (Unit 2) MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION DEVICES

Unit 2 CTS 3.8.2.6 states the thermal overload protection devices, integral with the motor starter, of each valve in the safety system shall be OPERABLE. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Motor-Operated Valves Thermal Overload Protection Devices LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

38. 3.8.2.7 (Unit 2) NORMALLY DE-ENERGIZED POWER CIRCUITS

Unit 2 CTS 3.8.2.7 states that all circuits that have containment penetrations and are not required during reactor operations shall be de-energized. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Normally De-Energized Power Circuits LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

39. 3.9.3 DECAY TIME

CTS 3.9.3 states that the reactor must be subcritical for at least 150 hours prior to movement of irradiated fuel in the reactor pressure vessel (RPV). The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Decay Time LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

40. 3.9.5 COMMUNICATIONS

CTS 3.9.5 states that direct communications shall be maintained between the control room and personnel at the refueling station during CORE ALTERATIONS. This ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS. The prompt notification of the control room of a fuel handling accident is an assumption in the Fuel Handling Analysis. This prompt notification is used to ensure that the control room is isolated promptly and is necessary to meet the control room operator dose limits in General Design Criterion (GDC) 19. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Communications LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

41. 3.9.6 MANIPULATOR CRANE OPERABILITY

CTS 3.9.6 states that the manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE during movement of control rods or fuel assemblies within the RPV. This specification ensures that the lifting device on the Manipulator Crane has adequate capacity to lift the weight of a fuel assembly and a Rod Control Cluster Assembly, and that an automatic load limiting device is available to prevent damage to the fuel assembly during fuel movement. This specification also ensures that the auxiliary hoist on the Manipulator Crane has adequate capacity for latching and unlatching control rod drive shafts. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Manipulator Crane Operability LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

42. 3.9.7 CRANE TRAVEL - SPENT FUEL PIT

CTS 3.9.7 places restriction on movement of loads over irradiated assemblies in the spent fuel pit in excess of 2500 pounds. This represents the working load of the fuel assembly plus gripper. The LCO ensures that in the event this load is dropped, the activity release will be limited to that contained in a single fuel assembly, and any possible distortion of fuel in the storage racks will not result in a critical array. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Crane Travel - Spent Fuel Pit LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

43. 3.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

CTS 3.9.9 states requirements for the containment purge and exhaust isolation system, which automatically closes the containment purge and exhaust isolation valves in MODE 6. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Containment Purge and Exhaust System LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

44. 3.9.10.2 WATER LEVEL - Reactor Vessel Control Rods

CTS 3.9.10.2 states that the refueling cavity water level must be at least 23 feet above the fuel during MODE 6 during movement of control rods within the RPV. Movement of control rods is not an initiator of any UFSAR accident analysis. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Water Level - Reactor Vessel - Control Rods LCO and Surveillances may be relocated to other licensee-controlled documents outside the ITS.

The relocated specifications from the CTS discussed above are not required to be in the TS because they do not fall within the criteria for mandatory inclusion in the TS as stated in 10 CFR 50.36(c)(2)(ii). These specifications are not needed to obviate the possibility that an abnormal situation or event will give rise to an immediate threat to the public health and safety. In addition, the 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, 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 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 will be relocating specifications, requirements, and detailed information from the CTS to the licensee-controlled documents outside the CTS. This is discussed in Sections 3.0.D and 3.0.E above. The facility and procedures described in the UFSAR and TRM can only be revised in accordance with the provisions of 10 CFR 50.59, which ensure records are maintained and establish appropriate control over requirements removed from the CTS and over 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 in accordance with ITS 5.5.1, and the administrative instructions that implement the QA Plan can be changed 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 conversion 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 their submittal, and by the staff during the course of the staff 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 February 26, 2002 (67 FR 8827).

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.

1. ITS 3.3.2 ESFAS INTERLOCK P-12, CTS Table 3.3-3 ESF Interlock P-12 (DOC M.7, JFD 1)

The license proposed changes to the allowable values of the setpoint for the Engineered Safety Feature Actuation System (ESFAS) P-12 interlock (T_{AVG}-Low Low). These values changed from $\leq$545^{o}F and $\geq$541^{o}F to $\leq$545^{o}F and $\geq$542^{o}F. During increasing reactor power, the P-12 interlock automatically reinstates Safety Injection (SI) for High Steam Flow Coincident With Steam Line Pressure-Low or Coincident With T_{AVG}-Low Low. The associated value for this function, $\leq$545^{o}F, has not changed; therefore, this function of the interlock is not affected. However, during decreasing power, the P-12 interlock allows the operator to manually block SI for the ESFAS functions mentioned above, and the operators use this feature to prevent SI during controlled plant cooldowns. Currently, the CTS Allowable Value is $\geq$541^{o}F and the licensee proposes changing it to ≥542°F. The staff finds that the current value is still bounded by the proposed value, and that the proposed value is more restrictive; therefore, the proposed change is acceptable.

2. ITS LCO 3.3.2, ACTION E.1, CTS LCO 3.3.2.1 ACTION 16 requirement of Table 3.3-3 (DOC M7, JFD 2)

CTS LCO 3.3.2.1 requires the use of Table 3.3-3, ACTION Statement when an ESFAS instrument channel is inoperable. CTS Table 3.3-3, ACTION 16 requires the inoperable channel to be placed in a blocked condition within 72 hours. The proposed ITS LCO 3.3.2, ACTION E.1 for the containment pressure channels requires the inoperable channel to be placed in a bypass condition within 72 hours.

The term “blocked condition” of CTS, ACTION 16 is used for containment pressure channels, which energize to trip, to block the channel input in order to maintain the channel in the untripped state. Thus, compliance with ACTION 16 results in maintaining a blocked channel in the untripped state. The proposed ITS use the term “bypass” to encompass those actions required to maintain a channel in the untripped state, regardless of whether that untripped state would be energized or de-energized. In the proposed Bases of ITS 3.3.2, ACTION E.1, the licensee stated that the bypass action is intended to avoid the inadvertent actuation of containment spray. Thus, compliance with ACTION E.1 results in maintaining a bypassed channel in the untripped state, as does the CTS, Action 16 requirement. Therefore, the ITS term bypass condition used in the proposed ITS 3.3.2, ACTION E.1, is identical to the CTS LCO 3.3.2.1, ACTION 16, and the proposed change is acceptable.

3. ITS 3.3.2 ESFAS Functions 1.c, 1.d, 1.f, 2.c, 4.c, and 4.d, CTS Table 3.3-4 ESF Functions 1.c, 1.d, 1.f, 2.c, 4.c, and 4.d (DOC M.7, JFD 1)

The proposed changes will remove the Trip Setpoint settings from the CTS Table 3.3-4 ESF Functions 1.c, 1.d, 1.f, 2.c, 4.c, and 4.d, and modify the Allowable Values for the proposed ITS 3.3.2 ESFAS Functions 1.c, 1.d, 1.f, 2.c, 4.c, and 4.d. Furthermore, the Allowable Value specified in the proposed ITS will serve as the Limiting Safety System Setting (LSSS) as required by 10CFR50.36(c)(1)(ii)(A), and defined by the regulation as “....setting for automatic protective devices...so chosen that automatic protective action will correct the abnormal situation before a Safety Limit (SL) is exceeded.” The staff evaluated the proposed Allowable Values and their setpoint methodology and found that the licensee’s proposed Allowable Values to be based on their own plant-specific methodology described in the Technical Report EE-0116 and Westinghouse values for Safety Analysis Limits, Channel Statistical Allowance, which is the combination of various channel uncertainties derived by the square-root-of-the-sum-of-the-squares, and algebraic techniques. In addition, the staff defined the LSSS as the Allowable Values specified in the above proposed ITS, and finds the proposed Allowable Values for ITS 3.3.2 ESFAS Functions 1.c, 1.d, 1.f, 2.c, 4.c, and 4.d to be acceptable. Details of the staff evaluation for this BSI are attached to this SE as Attachment 7.

4. ITS 3.4.12, Low Temperature Overpressure Protection (LTOP) System, Condition C (DOC M.4 and JFD 6);

ITS 3.4.12 states for Condition C that when an accumulator is not isolated or power is available to one or more accumulator isolation valve operators, the accumulator must be isolated immediately and power removed from the affected accumulator isolation valve operator in 1 hour. A Note modifies the Condition to state that it is only applicable when accumulator pressure is greater than PORV lift setpoints.

This BSI is related to NUREG-1431, STS 3.4.12, “Low Temperature Overpressure Protection (LTOP) System” regarding the accumulator isolation requirement.   The licensee’s proposal (1) adds a note to ACTION C, which indicates that the accumulator isolation is only applicable when accumulator pressure is greater than the power-operated relief valve (PORV) setting; (2) adds REQUIRED ACTION C.2 to state “Remove power from affected accumulator isolation valve operators”; and (3) adds a note in the LCO section, which states that “Accumulator isolation with power removed from the isolation valve operators is only required when accumulator pressure is greater than the PORV lift setting.” STS 3.4.12 of NUREG-1431 has (1) a note in the APPLICABILITY section which states that “accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR [Pressure Temperature Limits Report],” and (2) ACTION C, which contains the similar words as the note in the APPLICABILITY section.

The primary purpose of the accumulator isolation during LTOP conditions is to prevent inadvertent injection of water from the accumulators into RCS, which may be a challenge to reactor vessel P/T limits during low temperature operating conditions. The PORVs serve as an LTOP system with their setpoints designed to protect reactor vessel P/T limits under the limiting mass addition and heat addition transients. The settings of these PORVs are lower than the P/T limits in various temperature regions.

The proposed ITS 3.4.12, in the area of accumulator isolation, will require that when the plant is operating in the LTOP conditions, and the accumulator pressures are above the PORV settings, the accumulators are required to be isolated with power removed from the isolation valve operators. The staff evaluated the licensee’s submittals and concludes that the licensee’s proposal regarding accumulator isolation is acceptable. The basis for staff acceptance is that the proposed ITS will only allow the accumulators to be connected to the RCS when the accumulator pressures are lower than the PORV settings. Since the PORVs are designed to mitigate the limiting mass addition from a charging pump, it is unlikely that the P/T limits will be challenged by water injecting to RCS from the accumulators. In addition, this proposal is more conservative than STS 3.4.12 in NUREG-1431 since the STS would allow the accumulators to be connected to the RCS when the accumulator pressures are below the P/T limits but above the PORV settings.   The proposed ITS will require power removal from the isolation valve operators for added assurance for accumulator isolation, and add plant operational restrictions to NAPS current licensing bases regarding the requirement of accumulation isolation. Currently, there is no such requirement in their CTS.

Based on the above review, the staff finds that the licensee-proposed ITS 3.4.12 in the area of requiring accumulator isolation during LTOP conditions is more conservative than that in CTS and STS 3.4.12 of NUREG-1431. Therefore, the proposed ITS 3.4.12 regarding accumulator isolation is acceptable.

5. <u>ITS 3.7.3 - Main Feedwater Isolation Valves (MFIVs), Main Feedwater Pump Discharge Valves (MFPDVs), Main Feedwater Regulating Valves (MFRVs), and Main Feedwater Regulating Bypass Valves (MFRBVs) (DOC M.1 and JFD 3)</u>;

The licensee proposed the adoption of Section 3.7.3 of the STS. Adoption of Section 3.7.3

presents several deviations to the standard format provided in NUREG-1431 and has therefore been identified as a BSI.

The NAPS feedwater system consists of three main feedwater pumps with associated MFPDVs that feed a common header. From this header are three lines feeding the three SGs. On each line is an MFIV in series with an MFRV. On a line which bypasses each MFIV and MFRV is a MFRBV. Each of these valves, MFPDVs, MFIVs, MFRVs, and MFRBVs, close on receipt of an SI or SG Water Level High-High Signal. The MFIVs and the MFRVs provide single-failure protection for each other. The MFPDVs and the MFRBVs provide single-failure protection for each other. Therefore, all four valve types are required to meet the safety analysis assumptions.

The most significant deviation in format to the STS is that the ITS 3.7.3 will include MFPDVs. The STS 3.7.3 (as written in NUREG-1431) addresses MFIVs and MFRVs and associated bypass valves but not MFPDVs. Since the Main Feedwater System includes MFPDVs, and because they provide single-failure protection for the MFRBVs (and therefore are required to meet safety analysis assumptions), it is appropriate that the MFPDVs be included in ITS 3.7.3.

Other changes being made are the inclusion of the plant-specific values and information, where appropriate, in place of those presented in Section 3.7.3 of the STS. An example of this is the isolation time for the MFIVs, MFRVs, and MFPDVs. The time presented in ITS SR 3.7.3.1 was changed to represent the requirement and this differs slightly from the isolation time presented in the STS SR 3.7.3.1. ITS SR 3.7.3.1 also adds the requirement to test the closure time of each MFPDV.

Based on our review, the staff finds the proposed change to adopt STS 3.7.3 to be acceptable.

6. ITS SR 3.7.11.1 - Main Control Room/Emergency Switchgear Room (MCR/ESGR) Air Conditioning System (DOC M.2, JFD 4)

The licensee proposed changing the frequency of SR 3.7.11.1 from “18 months” to “18 months on a Staggered Test Basis.”

An air conditioning system (ACS), with two independent 100% capacity trains for each unit which supplies the relay rooms and common control room, is designed for 75°F dry bulb at approximately 50% relative humidity during normal operation. For emergency conditions, there is sufficient cooling capacity to maintain the control room, computer room, and relay room space temperature well below the design maximum of 120°F. A third chiller is provided for each reactor unit as an alternative for either train. One 100% capacity cooling system, which supplies the relay rooms and common control room in order to meet the signal failure criterion, is installed for each reactor unit. The cooling systems cannot be cross-connected between the two reactor units. Only one train for each unit is used at a time.

The emergency ACS for the MCR/ESGR envelope consists of two independent 100% redundant subsystems, one chiller in one subsystem and two chillers in the other. Each subsystem consists of two air handling units, one for the MCR and one for the ESGR to provide the heat removal function during post-accident conditions as well as during normal operation. The licensee added a staggered test basis with the 18-month surveillance test frequency of chillers. The staff finds the proposed change acceptable because there are three chillers with 100% cooling operation capability, either of which can be used by the subsystem, and in staff’s judgment, changing the surveillance frequency to every 18 months on a staggered test basis provides an acceptable level of confidence that the system will function as assumed in the accident analysis, and therefore is acceptable.

7. ITS 3.7.12 LCO Note, CTS 3.7.8.1 (DOC M.2, JFD 4)

The licensee proposed to add the phrase “not open by design” to the ITS 3.7.12 LCO to convey that the ECCS pump room boundary openings not open by design may be opened. This additional wording is a BSI because it deviates from the NUREG NOTE, which states that the ECCS pump room boundary openings may be opened intermittently under administrative control.

The staff reviewed the proposed LCO Note [The ECCS pump room boundary openings not open by design may be opened intermittently under administrative control], and finds the proposed NOTE consistent with the NOTE in the STS for this LCO with a modification. The staff reviewed NAPS’ plant-specific design for the ECCS Pump Room Exhaust Air Cleanup System (PREACS) boundary for the charging pump cubicles associated with the auxiliary building central area exhaust fans, and concurred that the boundary of these areas are enclosed, but do not form an entire pressure boundary because they include certain openings that are left open by design. Based on this finding, the staff finds the proposed NOTE to be consistent with the STS and is also more accurately reflected NAPS’ plant specific conditions, and therefore is acceptable.

8. ITS SR 3.7.12.2 and 3.7.12.4, CTS 4.7.8.1.a.1 - ECCS PREACS (DOC M.1, JFD 7)

The licensee proposed adding the following SR as ITS SR 3.7.12.2, with a surveillance frequency of 31 days: “Actuate each ECCS PREACS train by aligning Safeguards Area exhaust flow and Auxiliary Building Central exhaust flow through the Auxiliary Building HEPA filter and charcoal adsorber assembly.”

The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a loss-of-coolant accident (LOCA). The ECCS PREACS, in conjunction with other normally operating systems, also provides environmental control of temperature in the ECCS pump room areas.

The licensee stated that ITS SR 3.7.12.2 is added to divert safeguards area exhaust flow and auxiliary building central exhaust system flow through the auxiliary building HEPA filter and charcoal adsorber assembly for the operating safeguards area fan from the control room every 31 days. ITS SR 3.7.12.2 requires certain dampers associated with the auxiliary building central exhaust system to be manually actuated and tested. This provides additional assurance that the exhaust flow can be diverted through the filters in case of a DBA that requires their actuation. The licensee also stated that the 31-day test frequency is based on the known reliability of the equipment and the availability of redundant trains.

This new SR is added to ensure that in the event of a postulated DBA, the ECCS PREACS train is operable to reduce the potential dose risk from a radiological event. The staff concludes that the proposed SR is a conservative addition and therefore finds it acceptable. With this proposed change, the STS SR 3.7.12.2 is then renumbered to become ITS SR 3.7.12.3. This is an administrative change and the staff finds it acceptable. Similarly, STS SR 3.7.12.3 is renumbered to become ITS SR 3.7.12.4. This is also an administrative change that the staff finds acceptable.

In addition, STS SR 3.7.12.3 requires verifying each ECCS PREACS train to actuate on an actual or simulated actuation signal. The licensee proposed a change to this SR by replacing "Verify each ECCS PREACS train actuates on an actual or simulated actuation signal" with "Verify Safeguards Area exhaust flow is diverted and each Auxiliary Building filter bank is actuated on an actual or simulated actuation signal" on a surveillance frequency of every 18 months. The staff finds this change acceptable because this SR verifies proper operation of the actuation signal and assures that each auxiliary building filter bank signal will actuate in case of an accident.

9. <u>ITS 3.7.15, CTS 3.9.12 (DOC L.2, JFD 5)</u>

This BSI is related to the Fuel Building Ventilation System (FBVS) - CTS SR 4.9.12.

The FBVS consists of dual exhaust fans and two-speed supply fans. One supply fan serves the spent fuel pit area and the other one serves the remote equipment space at elevation 249 ft. 4 in. Both take suction from a common plenum fitted with a combination roll and high efficiency filter (95% atmospheric dust spot efficiency) and steam coils for air tempering and space heating. The exhaust fans discharge through vent stack B and are arranged for selective alignment through the auxiliary building HEPA/charcoal filter bank. The area of the remote equipment room subject to radioactive contamination is exhausted by a branch from the decontamination building exhaust system.

The licensee proposed to eliminate the testing requirement for the fuel building filtration system from the ITS by deleting CTS SR 4.9.12 (a) and CTS SR 4.9.12 (c). The purpose of these SRs is to verify that the fuel building filters can perform as required. In the submittal, the licensee states that the deleted SRs are not necessary to verify that the equipment used to meet the LCO is consistent with the safety analysis and can perform its required functions. Thus, appropriate equipment continues to be tested in a manner and at a frequency necessary to give confidence that the equipment can perform its assumed safety function. Furthermore, the licensee stated that the deletion of the requirement for the FBVS filters is acceptable because the NAPS radiological analysis of the fuel handling accident (FHA) in the fuel building assumes that all of the radionuclides released from the fuel pool are released without credit for filtration of the released material.

In order to determine the acceptability of the deletion of requirements for the FBVS filters, the staff examined the licensee's design basis radiological analysis of the FHA as documented in the licensee's UFSAR, Chapter 15.4.5. The previous licensee analysis along with the resulting dose consequences were found to be acceptable by the staff. The staff verified that the current fuel building FHA radiological analysis does not take credit for filtration of the released material and that the analysis assumptions as listed in the UFSAR are consistent with RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." The dose consequences of the FHA were previously found by the staff to be well within the dose guidelines given in 10 CFR Part 100 for offsite doses and also meet the dose criteria in 10 CFR 50, Appendix A, GDC 19 for the control room. The staff finds the proposed changes to the TS that remove requirements for testing the FBVS filtration capability are consistent with assumptions used in the current design basis analysis found in the NAPS UFSAR.
The licensee proposes, in accordance with TSTF-51, to add the term "recently irradiated fuel" as fuel that has been part of a critical reactor core within a licensee-specified number of days. The proposed TS Bases state that until analyses are performed to determine a specific value, recently irradiated fuel is defined as any irradiated fuel. CTS 3.9.3 "Decay Time" is being relocated to the TRM. The required decay time is 150 hours before allowing movement of

irradiated fuel, which is longer than the assumed decay time of 100 hours in the UFSAR FHA radiological analysis. The staff finds that the licensee's proposed definition of recently irradiated fuel is consistent with the NAPS design basis analysis. Based on the above evaluation, the staff concludes that the proposed changes to SR 4.9.12 incorporated into the ITS are acceptable.

10. ITS 5.5.8-2 - CTS Table 4.4-2. Unit 1 CTS SG Tube Inspection Requirements (DOC L.22 and JFD 1)

The licensee proposed to delete the requirements in the Unit 1 CTS Section 4.4.5, Table 4.4-2, "Report to NRC and Obtain Approval Prior to Operation," in the event an additional SG is found to be in category C-3. The licensee stated that the requirement was not specified in the STS. The proposed deletion makes this table consistent with the corresponding Unit 2 table in the STS.

The proposed administrative TS retain the requirement to notify the NRC if inspection results fall into category C-3. This notification is to be made pursuant to 10 CFR 50.72, and the "approval" requirement was imposed on the licensee, prior to the replacement of the SGs, when tube leaks during operation were frequent. However, the licensee has since replaced the SGs, and the SG performance has significantly improved, thus the staff concurs that an "approval" requirement is no longer necessary. This deletion will make the NAPS TS consistent with the STS in NUREG-1431. The proposed change is not expected to have any affect on safety; therefore, the staff finds the proposed change acceptable.

G.2 Additional BSI Changes identified by the Staff:

1. ITS 3.3.1, (JFD 14, DOC A.24)

NAPS, Units 1 and 2, used the Westinghouse ITS and WCAP-14483-A "Generic Methodology for Expanded Core Operating Limits Report" to develop their ITS and new COLR. The COLR allows licensees to change cycle-specific technical values without NRC approval, provided that NRC-approved methodologies are used to determine the values. The staff reviewed the implementation of a COLR to ensure that the proper approved methodologies are being used. This SE discusses the review of the following two BSIs:

(1) The overtemperature ΔT and overpower ΔT formulas contained in Notes 1 and 2 of ITS Table 3.3.1-1 have been modified in the proposed ITS to reflect those used as the licensing basis in the North Anna CTS. The licensee stated that these changes reflect the plant-specific CTS formulas in the proposed ITS requirements.

(2) The licensee proposed to exclude the statement "with gains to be selected based on measured instrument response during plant startup tests such that:" in Table 2.2-1, Note 1 of the CTS, from the proposed ITS. This statement describes the methodology used to determine the gains used in the calculation of the overtemperature ΔT trip setpoints. The licensee's justification for deletion contends that this statement is for information only, and since the gains have not been adjusted without engineering evaluation and NRC approval since their initial calculation, the removal is administrative.

With regard to the BSI in item (1), the staff reviewed the formulas for the overtemperature and overpower ΔT functions in Notes 1 and 2 of the ITS Table 3.3.1-1, and found that they are identical to those in Notes 1 and 2, respectively, of the CTS Table 2.2-1. Since these formulas were previously approved by the NRC as the licensing basis in the NAPS CTS and have not been changed in the conversion to the ITS, the staff finds their use in the ITS acceptable.

In evaluating the second BSI, the staff reviewed the methodologies used by the licensee to calculate the allowable overtemperature ΔT gains and trip setpoints. The staff conducted this review to determine if it was acceptable for the licensee to exclude the statement "with gains to be selected based upon measured instrument response during plant startup tests such that:" from Note 1 of ITS Table 3.3.1-1. This statement appears in CTS Note 1 of Table 2.2-1 and describes how gains for the axial flux difference are determined and used in the calculation of the overtemperature ΔT trip setpoints. In two separate RAIs dated September 7 and November 7, 2001, the staff requested the licensee provide detailed information on the procedures and methodologies used to determine the Allowable Values for the gains and setpoints. The licensee provided responses dated October 10 and December 12, 2001, which indicate the procedures and NRC-approved methodologies used in determining the appropriate gains and trip setpoints. The licensee stated that they used the NRC-approved methodology contained in WCAP-8748-P-A, "Design Bases for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions." The staff has approved this topical report for calculation of the constants used in the overtemperature and overpower ΔT formulas. Since the licensee is using NRC-approved methodologies used for the calculation of the allowable overtemperature ΔT gains and trip setpoints, the staff finds it acceptable to exclude the identified statement from ITS Table 3.3.1-1, Note 1.

In reviewing the December 12, 2001, RAI response, the staff noted licensee statements to conditionally adopt WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," to allow relocation of overtemperature and overpower ΔT allowable values to the COLR. The staff reviewed the response to determine if an NRC-approved methodology was used in calculating the Allowable Values and gains for the purpose of acceptability to remove the statement on how gains are determined from the ITS. This SE did not review the response to determine acceptability of relocating values to the COLR for NAPS, Units 1 and 2, because no clear position on licensee use of WCAP-14483-A was established.

The staff reviewed the two BSIs and approves the use of the plant-specific ITS equations for the overtemperature and overpower ΔT equations shown in Table 3.3.1-1, Notes 1 and 2. The staff has concluded that these equations are identical to those previously approved in CTS Table 2.2-1, Notes 1 and 2. Secondly, the staff approves the exclusion of the statement "with gains to be selected based upon measure instrument response during plant startup tests such that:" from ITS Table 3.3.1-1, Note 1. The staff concluded that the licensee has used NRC-approved methodologies to calculate the allowable overtemperature ΔT gains and trip setpoints.

2. <u>ITS 3.3.1 - RTS Instrumentation; Relaxation of LCO Requirements, Allowable Values for the P-7 function come from the requirements of P-10 and P-13 (DOC L.8)</u>

The licensee proposed a change to the Allowable Values of the setpoints for the P-7 interlock (Low Power Reactor Trips Block) to a value not currently allowed by their current TS. The original Allowable Value for P-7 was <11 percent. The staff reviewed the proposed change and finds a change to the CTS that lists the Allowable Value as NA (Not Applicable). However, the P-7 interlock uses the P-10 and P-13 interlocks for inputs. The licensee proposed a new Allowable Value for P-10 and P-13 of ≤11 percent. This change effectively modifies the P-7 actuation from <11 percent to ≤11 percent, thus including 11 percent as an Allowable Value. The staff considers this change to be less restrictive, but considered it to have a negligible effect. Based on this review, the staff finds the proposed change acceptable.

3. ITS 3.3.1 - RTS Instrumentation; Relaxation of LCO Requirements, Allowable Value Changes for P-6, P-8, and P-13 interlocks (DOC L.14)

The licensee proposed changes to the Allowable Values for the P-6, P-8, and P-13 interlocks. The P-6 interlock function for increasing power (intermediate range above setpoints) allows the operators to manually block the source range channels trip capability. Securing the source range channels trip is not a safety function, but is an equipment protection function. The licensee proposed removing this P-6 setting from the improved TS. The staff reviewed the change and finds this removal acceptable. However, the P-6 interlock function while decreasing power (intermediate range below setpoints) is safety-related. This interlock activates the source range channels trip capability. The Allowable Value for the decreasing power P-6 interlock is listed as $>3x10^{-10}$ in the CTS. The proposed Allowable Value is listed as $\geq 3x10^{-10}$. This change is less restrictive, but is considered to have a negligible effect. Based on this review, the staff finds the proposed change acceptable.

When below the defined setpoints, the P-8 interlock prevents a reactor trip for the following conditions: low flow in a single loop, a single RCP breaker open, or a turbine trip. This function (power range below setpoints) is not a safety function and the associated setpoints have been removed from the proposed TS. The staff finds this removal acceptable. However, when above ITS setpoints, the P-8 interlock allows a reactor trip for the above conditions. The current TS list the Allowable Value for the setpoints as <31 percent on the power range channels. The licensee proposed changing the Allowable Value to ≤31 percent. This change is less restrictive, but is considered to have a negligible effect. Based on this review, the staff finds the proposed change acceptable.

The P-13 interlock (Turbine Impulse Pressure) is an input to the P-7 interlock. When above the setpoints, P-13 (in conjunction with P-10) allows a reactor trip under the following conditions in more than one loop: low flow, RCP breaker open, under voltage on the reactor coolant pump buses, and under frequency on the RCP buses. P-13 also allows a reactor trip on pressurizer low pressure or pressurizer high level when above the setpoints. The current TS list the Allowable Value as < 11 percent. The licensee proposed changing the Allowable Value to ≤11 percent. The inclusion of 11 percent is less restrictive, but it is considered negligible. Based on this review, the staff finds the proposed change acceptable.

When below the setpoints, P-13 (in conjunction with P-10) prevents a reactor trip when any of the following conditions occur: RCS low flow, RCP breakers open, RCP buses under voltage, RCP buses under frequency, pressurizer low pressure, and pressurizer high level. This function of P-13 is not assumed in the safety analyses. Therefore, the licensee proposed removing the setpoints and Allowable Values for this function of P-13 from their TS. Based on this review, the staff finds this removal acceptable.

4. ITS Table 3.3.2-1, ESFAS Instrumentation Function 7. Automatic Switchover to Containment Sump (DOC M.3)

Functional Unit 7, “Automatic Switchover to Containment Sump” is being included in TS 3/4.3.2, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation,” with allowed outage time and additional channel bypass. Changes include a statement in the Bases section to identify that a plant-specific risk assessment was performed to support the 72-hour proposed allowed outage and maintenance times and to allow an additional channel to be bypassed during surveillance for the instrumentation. The 72-hour Completion Time is justified in a plant-specific risk assessment, consistent with Reference 8 [WCAP-10271-P-A, Suppl. 2, Rev.1, June 1990, and WCAP-14333-P-Rev. 1, Oct 1998]. The total of 72 hours to reach Mode 3 and 12 hours for a second channel to be bypassed is acceptable based also on the results of a plant-specific risk assessment.

A plant-specific risk assessment was completed to include an assessment of Functional Unit 7, “Automatic Switchover to Containment Sump.” Functional Unit 7 had been included as a new unit in the ITS for consistency with NUREG-1431. The plant-specific risk evaluation assessed the change in core damage frequency (CDF) and the incremental conditional core damage probability as a result of the WCAP changes for the additional functions.

The licensee developed the CDF sensitivity for this function in the same manner as the WCAP-10271 and WCAP-14333 analyses. The automatic containment switchover function is similar to that of some of the WCAP channels and was estimated by comparison to similar functions. Once the channel failure impacts were quantified, these numbers were converted to a CDF impact by looking at the associated CDF sensitivity from the NAPS probable risk assessment model for the same function or a higher level function.

The automatic switchover to containment sump occurs when the RWST level drops to the established setpoint. Automatic switchover failure probability is estimated to increase by approximately 1.3E-4 as result of the proposed changes. This increase is based upon the assumption that the full allowed outage time will be used on a regular basis every year. The result of a plant-specific risk assessment for this function related to CDF impact is negligible (less than 0.01% of the CDF) based on the baseline CDF (3.3E-5/yr) at North Anna.

This risk assessment demonstrates that the effect on CDF and incremental conditional core damage probability is negligible for the potential unavailability changes associated with this function. Based on this review, the staff concludes that the licensee’s proposed Functional Unit 7 TS allowed outage and bypass times are acceptable.

5. ITS 3.7.11 Actions D and E, CTS 3/4.7.7.1 Action d (DOC M.1 and M.3, JFD 3)

The ITS proposes to only require entry into Action D, for one AC subsystem inoperable, as long as 100% ACS cooling equivalent to a single operable AC subsystem is available.

The emergency ACS for the MCR/ESGR envelope consists of two independent 100% redundant subsystems, one chiller in one subsystem, and two chillers in the other. Each subsystem consists of two air handling units, one for the MCR and one for the ESGR, to provide the heat removal function during post-accident conditions as well as during normal operation. An ACS with two independent 100% capacity subsystems for each unit, which supplies the relay rooms and common control room, are designed for 75°F dry bulb at approximately 50% relative humidity during normal operation. For emergency conditions, there is sufficient cooling capacity to maintain the control room, computer room, and relay room space temperature well below the design maximum of 120°F. A third chiller is provided for each reactor unit as an alternative for either train. The cooling systems cannot be cross-connected between the two reactor units. Only one train for each unit is used at a time.

The licensee stated that because the MCR/ESGR ACS includes a total of three chillers and flexibility in the use of system components, the description of system requirements, “Less than 100% of the MCR/ESGR ACS cooling equivalent to a single OPERABLE MCR/ESGR ACS subsystem available....” is proposed in the above ITS instead of a reference to two inoperable trains. The proposed ITS Conditions allow a variety of system configurations to be established that would provide sufficient cooling capacity to meet the design function and allows appropriate flexibility in operation of the system similar to ITS 3.5.2, ECCS. The licensee further stated that the Conditions D and E still require that when the design function cannot be met, that the appropriate Applicability (MODES 1,2, 3, and 4 and during movement of recently irradiated fuel assemblies) be exited.

The staff has reviewed the proposed change and agrees with the licensee that the proposed ITS change is consistent with the intent of STS 3.7.11, since there are three chillers with 100% cooling capability, one of which can be used by either subsystem. The staff finds the proposed ITS change acceptable because it provides the system flexibility in operation of the system, enables the various configurations to maintain the required cooling function, and provides an acceptable level of confidence that the system will function as assumed in the accident analysis. Based on the above evaluation, the staff concludes that the proposed changes to TS 3.7.11, Actions D and E, incorporated into the ITS, are acceptable.

4.0 <u>COMMITMENTS_RELIED_UPON</u>

In reviewing the proposed ITS conversion for NAPS, the staff has relied upon the licensee’s commitment to relocate certain requirements from the CTS to licensee-controlled documents as described in Table R, “Relocated Specifications and Removed Details” (Attachment 5 to this SE). This table reflects the relocations described in the licensee’s submittals on the conversion. The staff requested and the licensee submitted a license condition to make this commitment enforceable (see Section 5.0 of this SE). Such a commitment from the licensee is important to the ITS conversion because the acceptability of removing certain requirements from the TS is based on those requirements being relocated to licensee-controlled documents where further changes to the requirements will be controlled by regulations or other requirements (e.g., in accordance with 10 CFR 50.59).

5.0 <u>LICENSE_CONDITIONS</u>

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

(2) 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.

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

(4) 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 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 September 2, 2002. This implementation schedule 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 R (Attachment 5 to this SE), and Section 3.0.D, "Removed Details," and Section 3.0.E, "Relocated Specifications," above. The license condition states that the relocations would be completed no later than September 2, 2002. This schedule is acceptable.

6.0 STATE_CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified on February 27, 2002, of the proposed issuance of the ITS conversion amendment for NAPS. 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 March 29, 2002 (67 FR 15254), for the proposed conversion of the CTS to ITS for NAPS. Accordingly, the Commission has determined that issuance of these amendments will not result in any 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 R - Relocated Specifications and Removed Details
6. Staff Safety Evaluation for Relocation Item 29, Section 3.E
7. Staff Safety Evaluation for BSI 3, Section 3.G.1

Principal Contributors:	N. Le	S. Monarque	C. Schulten	B. Fu
	S. Peters	J. Golla	CY Liang	J. Raval
	A. Chu	C. Harbuck	R. Giardina	C. Doutt
	R. Tjader	S. Rhow	N. Iqbal	M. Hart

M. Hart R. Taylor S. Dinsmore R. Architzel

Date: April 5, 2002

LIST_OF_ACRONYMS

ACS	Air Conditioning System
ASME	American Society of Mechanical Engineers
BSI	Beyond-Scope Issue
CCW	Component Cooling Water
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CTS	Current Technical Specification
CVCS	Chemical and Volume Control System
DBA	Design-Basis Accident
DOC	Discussion of Change (from the CTS)
ECCS	Emergency Core Cooling System
ESFAS	Engineered Safety Features Actuation System
FBVS	Fuel Building Ventilation 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
LSSS	Limiting Safety System Setting
LTOP	Low Temperature Overpressure Protection
MCR/ESGR	Main Control Room/Emergency Switchgear Room
MFIV	Main Feedwater Isolation Valve
MFPDV	Main Feedwater Pump Discharge Valve
MFRBV	Main Feedwater Regulating Bypass Valve
MFRV	Main Feedwater Regulating Valve
NAPS	North Anna Power Station
NSSS	Nuclear Steam Supply System
ODCM	Offsite Dose Calculation Manual
PAM	Post-Accident Monitoring
P/T	Pressure/Temperature
PORV	Power Operated Relief Valve
PREACS	Pump Room Exhaust Air Cleanup System
PTLR	Pressure Temperature Limits Report
QA	Quality Assurance
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SE	Safety Evaluation

SI	Safety Injection

SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SR	Surveillance Requirement
STS	Improved Standard Technical Specification, NUREG-1431, Rev. 1
SW	Service Water
TRM	Technical Requirements Manual
TS	Technical Specification
TSTF	Technical Specifications Task Force (re: generic changes to the STS)
UHS	Ultimate Heat Sink
UFSAR	Updated Final 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 R - Relocated Specifications and Removed Details

ATTACHMENT 6

Staff Safety Evaluation for Relocation of

CTS 3.7.5.1.b, Ultimate Heat Sink - North Anna Reservoir to TRM

SAFETY EVALUATION

OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING RELOCATION OF TECHNICAL SPECIFICATION 3/4.7.5.1.b

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-338 AND 50-339

1.0 INTRODUCTION

By letter dated December 11, 2000, Virginia Electric and Power Company (the licensee) submitted a license amendment request to obtain NRC approval for conversion of the North Anna Power Station (NAPS), Units 1 and 2 current Technical Specifications (CTS) to Improved Standard Technical Specifications (STS). The proposed changes would revise the North Anna CTS to be consistent with NUREG-1431, "Standard Technical Specifications-Westinghouse Plants," Revision 1, and certain changes to the NUREG. The North Anna CTS, upon conversion, are referred to as "Improved Technical Specifications" (ITS).

This Safety Evaluation (SE) addresses only one portion of the amendment request presented in the licensee's letter dated December 11, 2000. This SE is specific to the licensee's request therein to remove CTS LCO 3.7.5.1.b from the CTS and relocate it to the Technical Requirements Manual (TRM). CTS Limiting Condition for Operation (LCO) 3.7.5.1 states, "The ultimate heat sinks shall be operable." This statement refers to both the Service Water Reservoir and the North Anna Reservoir. The Service Water Reservoir is credited as the ultimate heat sink (UHS) for the plant design-basis accident (DBA). The North Anna Reservoir serves as a backup to the Service Water Reservoir. The licensee is requesting to relocate only that portion of LCO 3.7.5.1 that refers to the North Anna Reservoir to the TRM. That specification is 3.7.5.1.b, which specifies the minimum water level and temperature for the North Anna Reservoir. The portion of LCO 3.7.5.1 that refers to the operability of the Service Water Reservoir is to be retained in the ITS as LCO 3.7.9. Relocating LCO 3.7.5.1.b to the TRM was identified as a "beyond scope" issue because the request involves the relocation of requirements.

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

For several years, NRC and industry representatives have sought to develop guidelines for improving the content and quality of nuclear power plant TS. On February 6, 1987, the Commission issued an interim policy statement on TS improvements, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, utility owners groups and the staff developed STS, such as NUREG-1431, that would establish models of the Commission's policy for each primary reactor type. In addition, the staff, licensees, and owners groups developed generic administrative and editorial guidelines in the form of a "Writer's Guide" for preparing TS, 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 apply to NAPS.

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 TS and defined the guidance criteria to be used in determining which of the LCOs and associated SRs should remain in the TS. The

Commission noted that in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TS, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

> [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 TS; those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36 (60 FR 36953, July 19, 1995). The four criteria are 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 safety assessment has shown to be significant to public health and safety.*

The licensee's rationale for relocating CTS 3/4.7.5 for the UHS source is that the LCOs do not meet any of the four criteria and that doing so would not compromise the safe operation of the plant. This SE will review the licensee's evaluation of LCO 3/4.7.5.1.b for the UHS source against the four criteria. The staff evaluation is discussed in the paragraphs below.

Proposed Relocation of CTS 3.7.5.1.b

The UHS provides a heat sink for processing and operating heat from safety-related components during a transient or accident, as well as during normal operation. This is done by utilizing the service water (SW) system and the component cooling water system. The UHS for a power reactor is comprised of a complex of water sources, including necessary retaining structures (e.g., a pond with a dam), and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident. A variety of complexes are used industry-wide to meet the requirements for a UHS. The basic performance requirements of the UHS are that a 30-day supply of water be available, and that the design-basis temperatures of safety-related equipment not be exceeded. Regulatory Guide (RG) 1.27, "Ultimate Heat Sink For Nuclear Power Plants" discusses technical considerations for the establishment of a UHS for a power reactor and promulgates four regulatory positions. The first regulatory position states that the UHS should be capable of providing sufficient cooling for at least 30 days (a) to permit simultaneous safe shutdown and cooldown of all nuclear reactor units that it serves and to maintain them in a safe shutdown condition, and (b) in the event of an accident in one unit, to limit the effects of that accident safely, to permit simultaneous and safe shutdown of the remaining unit(s), and to maintain them in a safe shutdown condition. This position further states that "procedures for ensuring a continued capability after 30 days should be available." The second regulatory position stipulates that the UHS should be capable of withstanding severe natural phenomena, site-related events, reasonably probable combinations of natural phenomena and site-related events, and a single failure of manmade structural features, and maintain its capability to function as described in the first regulatory position. The third regulatory position is that the UHS should consist of at least two sources of water, each with the capability to perform the safety functions specified in regulatory position 1, "unless it can be demonstrated that there is an extremely low probability of losing the capability of a single source." The fourth regulatory position stipulates that TS for the plant should include provisions for actions to be taken in the event that conditions threaten partial loss of the capability of the UHS or the plant temporarily does not satisfy regulatory positions 1 and 3 during operation.

As stated above, 10 CFR 50.36(c)(2)(ii) delineates four criteria that define the basis for which an LCO must be established for items meeting any one or more of the four criteria. The licensee's rationale for relocating the operability requirements residing in CTS LCO 3.7.5.1.b for the North Anna Reservoir is that these requirements do not meet any of the four criteria and that doing so would not compromise the safe operation of the plant.

3.0 EVALUATION

As indicated in the NAPS Updated Final Safety Analysis Report (UFSAR), the design of the UHS complies with RG 1.27, dated March 1974. The UHS at NAPS consists of the Service Water Reservoir and the North Anna Reservoir and their associated retaining structures. The normal source of SW for both units is the Service Water Reservoir, which is adequate to provide sufficient cooling for at least 30 days (a) to permit simultaneous safe shutdown and cooldown of two units, then maintain them in a safe-shutdown condition, and (b) in the event of an accident in one unit, to permit control of that accident safely and permit simultaneous safe shutdown and cooldown of the remaining unit and maintain both units in a safe-shutdown condition. After 30 days, makeup to the Service Water Reservoir is provided by the North Anna

Reservoir as necessary to maintain cooling water inventory, providing a continued cooling capability. This meets Regulatory Position 1 of RG 1.27.

The design of the Service Water Reservoir is described in section 3.8.4 of the NAPS UFSAR. It has been accepted previously by the NRC as meeting Regulatory Position 2 of RG 1.27 for the ability to withstand natural and man-made events without loss of function. The UHS is comprised of two sources of water each with the capability to perform the safety functions specified in Regulatory Position 1. This meets Regulatory Position 3. Finally, Regulatory Position 4 is met because plant TS have been established that ensure the minimum availability of the UHS.

The UHS for the plant is utilized primarily by the SW system. SW is used as cooling water for heat exchangers that remove heat from the component cooling system, the recirculation spray subsystem, and other station applications such as main control room air conditioning, and charging pump lubricating oil and instrument air compressors. In addition, SW is provided as a backup supply to the steam generator feed system, the fuel pit coolers, and the containment recirculation air cooling coils. Treated water from the Service Water Reservoir, or untreated water from the North Anna Reservoir as a backup supply, is circulated by pumps through the systems and components that require an ensured supply of SW under accident conditions. All systems that perform a safety function and that require cooling during an accident are cooled by the SW system normally utilizing the Service Water Reservoir. The Service Water Reservoir is in use during normal operation and during accident recovery. Review of the NAPS UFSAR indicates that the design functions of the North Anna Reservoir are not part of the safety sequence analysis for North Anna UFSAR Condition II, “Faults of Moderate Frequency,” Condition III, “LOCA Accidents,” and Condition IV, “Limiting Faults” events. The North Anna Reservoir only serves as a backup to the Service Water Reservoir for mitigation of UFSAR Chapter 6, “Engineered Safety Features” and Chapter 15, “Accident Analyses” Condition II, III, and IV events. Since the North Anna Reservoir does not involve assumptions for initiating events or affect any accident mitigation functions for North Anna Condition II, III, and IV events, then postulated loss of the North Anna Reservoir has no impact on safety margin from a deterministic point of view. Therefore, a postulated loss of the North Anna Reservoir does not affect the plant design basis or the limiting equipment availability assumptions used in the deterministic analyses to establish margins of safety.

Regarding the effect removing LCO 3.7.1.5.b from TS would have on other systems or components, there are no existing TS requirements for systems or components that rely on the North Anna Reservoir for operation. As stated above, the SW system can utilize the North Anna Reservoir as an untreated backup source but does not rely on it. The SW system relies on the Service Water Reservoir. If the SW system were to rely on the North Anna Reservoir, with the specifications for the North Anna Reservoir relocated to the TRM, the OPERABILITY requirements for the SW system would not be affected. The North Anna Reservoir, serving an “attendant support function” (as OPERABILITY is defined for plant TS) to the SW system, would itself have to be operable. If it were not, the SW system would have to be declared inoperable.

3.1 Technical Specification 3.7.5.1b Against Criteria 1, 2, and 3 of 10 CFR 50.36:

Regulation 10 CFR 50.36(c)(2)(ii) states, “A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:”

- *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.

The staff finds that the North Anna Reservoir does not meet this criterion as it is not installed instrumentation.

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

The staff finds that the North Anna Reservoir does not meet this criterion because it is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

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

The staff finds that the North Anna Reservoir does not meet this criterion because it is not an SSC that is a part of the primary success path and that 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. The North Anna Reservoir does serve as a backup system to the Service Water Reservoir, which meets this criterion. An NRC "Policy Issue" has been codified in SECY-93-067, "Final Policy Statement On Technical Specification Improvements," March 17, 1993, which illuminates the position taken by the NRC regarding the four criteria of 10 CFR 50.36(c)(2)(ii). Discussion therein concerning Criterion 3 clearly establishes the position that backup equipment is not included in the definition of the "primary success path," which functions or actuates to mitigate a DBA or transient. Therefore, the North Anna Reservoir, being a backup system, does not meet this criterion.

3.2 <u>Evaluation of Proposed Changes Against Criterion 4 of 10 CFR 50.36.</u>

- 10 CFR 50.36(c)(2)(ii)(D) Criterion 4 states that a TS LCO must be established for each "structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety." The Statements of Consideration for the NRC Final Rule "Technical Specifications," 60 FR 36955 (July 19, 1995) clarifies that it is the significance of the constraints, not the SSCs, that determine whether the LCO may be relocated from the TS to licensee-controlled documents.

  "Criterion 4 is intended to capture those constraints that probabilistic risk assessment [PRA] or operating experience show to be significant to public health and safety, consistent with the Commission's PRA Policies. The level of significance either would need to be such that it justified including the constraints in the technical specifications to ensure adequate protection of the public health and safety or that the addition of such constraints provides substantial additional protection to the public health and safety."

There are no guidelines provided for determining what is "significant" or "substantial." Rather, the Statements of Consideration for the NRC Final Rule "Technical Specifications," 60 FR 36956 (July 19, 1995) states in part:

> "To ensure consistent and appropriate decision-making that incorporates PRA methods and results, it is important that coherent and clear application guidelines are applied. As part of the PRA Implementation Plan, such guidelines will be established (incorporating safety goals and backfit rule considerations). . . . The NRC staff anticipates that, as it gains experience with the development and use of such PRA application guidelines, it will be better able to refine such phrases as "significant to public health and safety,'' and other phrases that are used in many of the Commission's regulations."

The guidance developed in RG 1.174 (Ref. 6) as part of the PRA Implementation Plan emphasizes that an estimate of the absolute change in risk arising from a proposed licensing change is the most desirable risk-informed criterion. As stated by the licensee in Reference 2, however, no changes in the treatment of the SSCs are anticipated as a result of the relocation of the specification to the TRM. Therefore, at this time there is no change in risk associated with this licensing amendment. However, these constraints in the future could be changed without the additional oversight and controls provided by a license amendment.

A recent Commission Order (Ref. 7) addresses the relocation of constraints that may in the future be changed without the additional oversight and controls provided by a license amendment. In a March 2001 decision (Ref. 8), the Atomic Safety and Licensing Board (ASLB) denied an intervenor group's petition challenging two licensing amendments approved by the staff. The licensing amendments relocated numerous detailed procedures for monitoring routine radioactive release from the TS to a licensee-controlled Radiological Effluent Monitoring and Offsite Dose Calculation Manual. In Reference 7, the Commission denied the intervenor group's appeal to overturn the ASLB decision.

In the *Background* discussion of the amendments, the Commission noted that the licensee in the future may make adjustments to monitoring practices, but they are not free to make any adjustment that could, in this case, lead to a violation of radiological effluent limits or related SRs [see Dominion Nuclear Connecticut, (Millstone Nuclear Power Station, Units 2 and 3), CLI-01-24, 349, 354 (2001)]. In its decision, the Commission rejected an implication that the amendments will lead to an increase in radiation exposures. *Id.* at 354. Therefore, when determining the significance of requirements to be relocated that may be changed in the future, it is consistent with the Commission's decision to consider that other TS, the Commission's regulations, or other constraints may achieve the same purpose as the requirement for which relocation is being considered.

The staff uses several methodologies and guidelines to incorporate risk implications derived from PRAs into regulatory decisions. These methods describe how to develop quantitative risk measures, and provide quantitative guidelines to determine whether a proposed change is "significant" insofar as the estimated risk measures may lie above the guideline values. These methods have not been applied to support the determination whether a TS constraint is, or is not, "significant to public health and safety" as used in Criterion 4. As discussed in this SE, the methods and guidelines are applied here to illustrate the significance of the TS requirements proposed for relocation.

There are currently three methodologies and associated guideline values that have been accepted and are utilized by the staff to determine the level of safety-significance of SSCs and of the inoperability of SSCs. The first methodology was accepted in the South Texas Project's (STP's) risk-informed exemption request (Ref. 9). The exemption methodology categorizes the safety-significance of SSCs based on their average contribution to risk and illustrates the safety-significance of the SSCs. The second methodology was developed to support the determination of the safety-significance of inspection findings. The significance determination process (SDP) methodology (Ref. 10) categorizes the safety significance of inoperable SSCs based on an observed plant configuration over some period of time. The third methodology was developed to provide guidance on acceptable allowed outage times (AOTs) for SSCs in the TS (Ref. 11). The AOT methodology provides guidance on the length of time that an SSC can be inoperable without causing an unacceptable risk increase.

State-of-the-art in PRA does not allow an estimate of the change in risk associated with exempting low-safety-significant (LSS) SSCs from special treatment requirements because the relationship between treatment and reliability has not been established. Consequently, in the STP exemption methodology, safety-significance was defined based on the factors that core damage frequency (CDF) and large early release frequency (LERF) would increase given the SSC was unavailable (the risk achievement worth (RAW)), coupled with a measure of the fraction of CDF and LERF to which the failure of the SSC contributes (the Fussel-Vesely measure (FV)). If either the CDF or the LERF RAW is greater than two, the SSC is not LSS and not exempted from the special treatment requirements. If the FV is greater than 0.005, the SSC is not LSS and not exempted from the special treatment requirements.

The SDP process was developed to provide NRC inspectors and management with a simple probabilistic framework for use in identifying potentially risk-significant issues caused by deficient licensee performance. In the SDP process, the boundary between a green (no- to low-safety-significance) and white (low- to moderate-safety-significance) is an equivalent CDF of 1E-6/year or an equivalent LERF of 1E-7/year. The equivalent CDF and LERF are the annual increased risk that would be expected if the increased risk during the time period the equipment was inoperable was averaged over a 1-year period.

A methodology and guideline values to be used to support the determination of acceptable extensions of AOTs were also developed as part of the PRA Implementation plan. Acceptability of an AOT extension is based, in part, on an incremental conditional core damage probability and incremental large early release probability of less than 5E-7 and 5E-8 respectively. The incremental conditional core damage probability and incremental large early release probability are the increased risk incurred during the time the SSC is inoperable and they are quantitatively identical to the SDP's equivalent CDF and LERF as long as the inoperable period does not extent beyond 1 year. Because the guidelines' values are ½ as large for AOT extensions as for SDP evaluations, and the measurement values are equal, use of the AOT extension guideline values results in shorter (exactly ½) inoperable time spans before the AOT guideline values would be exceeded than before a white SDP finding. The AOT safety-significance criteria are exceeded in a shorter time period than the SDP white guideline values because the AOT guideline values are used to approve permanent changes in planned configurations, while the SDP is used to evaluate singular and unplanned changes in configuration.

The STP exemption and the SDP methodologies include several safety-significance categories, generally low-, intermediate-, and high-safety-significance. It is reasonable to assume that “significant” in Criteria 4 would be something more significant than LSS. Rather than attempt to further interpret “significant” in Criteria 4, the medium- or moderate-safety-significance categories are not combined here into the LSS category. This affects the change in risk guideline values for the RAW, the FV, and the SDP. If the medium or moderate categories were combined into the low category, the RAW guideline value would increase from two to ten, the inoperable time required to yield a yellow (moderate- to high-safety significant) SDP finding would increase by a factor of ten compared to a white (low- to moderate-safety-significant) finding, and the FV guideline value would increase from 0.005 to 0.01 (given that the RAW was less than 2.0). The AOT extension method results support the approval or denial of an amendment request. In this case, changes in risk that would indicate that the request should be denied are considered significant.

In Reference 1, NAPS stated that the North Anna Reservoir does not meet any of the four criteria of 10 CFR 50.36 for inclusion of an LCO in the TS. Therefore, NAPS proposed to delete TS 3.7.5.1.b “Ultimate Heat Sink [North Anna Reservoir],” from the current TS and the ITS and to relocate the requirements to the TRM. The current LCO requires that the North Anna Reservoir must be operable. The two conditions that define operability are that the water level must be at or above elevation 244 feet mean sea level and the average water temperature must be less than or equal to 95 degrees Fahrenheit. If either condition exists, the plant must be in hot standby within 6 hours. There is an additional constraint regarding further transitions to other shutdown states that is outside of the scope of PRA analysis.

The current LCOs represents an acceptable risk. In this case, a defined water temperature and level of the North Anna Reservoir define the acceptable risk. This evaluation first estimates the change in risk if the SSCs identified in the TS are inoperable. If the change in risk is below the LSS guidelines, the SSC and the TS constraints are LSS according to the methodology and guideline values. Otherwise, the SSCs’ FV and RAW for operation in the configuration as identified in each of the TS constraints are estimated and compared to the LSS guidelines. If there are SSCs whose FV or RAW lie above the LSS guidelines, the time span that operation in violation of the TS constraint would still yield an acceptable risk increase as defined by the AOT extension guidelines and the SDP guidelines is estimated.

The North Anna Reservoir is the normal cooling water supply to the circulating water (CW) system that provides cooling to the condenser during normal power operation. It is modeled in the PRA as the water supply for the auxiliary service water system and for the CW system. The ASW pumps are a backup to the main service water system pumps that normally take water from the service water reservoir. In the PRA, the CW system is credited as a source of decay heat removal through the condenser for short-term decay heat removal following a steam dump to the condenser by opening of the turbine bypass valves. After a steam generator tube rupture (SGTR), both the condenser and the PORVs are normally available for short-term decay heat removal. If the North Anna Reservoir is unavailable, then the CW system is unavailable, and only the PORV decay heat removal path is modeled as available. Loss of the short-term decay heat removal from the condenser via the CW systems and failure of the PORVs decay heat removal path following an SGTR dominates the estimated RAW increases.

In Reference 3, the licensee estimated that, if the North Anna Reservoir is unavailable, the CDF would increase by a factor of 2.0 and the LERF would increase by a factor of 2.9. The system has negligible FV values. The RAW value for LERF is greater than two and the North Anna Reservoir would not be placed in the LSS category based on the STP exemption request guidelines.

The licensee estimated the total change in CDF and LERF, given the reservoir is inoperable for 1 year, is 1.3E-5/year and 3.4E-6/year, respectively. In order for the staff to develop a "white-finding" based on an observed, inoperable reservoir, the reservoir water supply would need to be inoperable for more than 11 days based on the more limiting LERF criteria. An AOT of greater than about 5 days would exceed the incremental large early release probability guideline.

Based on the current NAPS PRA model, the North Anna Reservoir exceeds the guidelines for LSS. It is not, however, possible to operate with the North Anna Reservoir unavailable, only to operate with the North Anna Reservoir slightly outside the physical characteristics specified in the CTS. In Reference 4, the licensee indicated that a turbine trip on low condenser vacuum restricts the North Anna Reservoir temperature and water level that can exist during power operation. Both the water temperature and level contribute to maintaining condenser vacuum. Reference 4 further indicates that the plant expects to have difficulty maintaining condenser vacuum when the water temperature in the North Anna Reservoir reaches 105 degrees Fahrenheit, somewhat higher than the CTS limit of 95 degrees. When the level falls below 244 feet, the screen wash pumps will lose suction, which may ultimately lead to the shutdown of the CW pumps as the differential level increases across the screens. The CW pumps must be shut down when the water level drops below 235 feet (Ref. 5), somewhat lower than the CTS limit of 244 feet. Reference 4 discusses an abnormal procedure that is be entered when the North Anna Reservoir level drops below 247 feet, so continuous operation with a low water level would require that the abnormal procedure be indefinitely followed. Reference 4 also states that it is very unlikely that the North Anna Reservoir temperature would exceed 95 degrees Fahrenheit. These operational constraints limit the time that the North Anna Reservoir would be operated with a water temperature or level in violation of the CTS constraints.

In References 4 and 5, the licensee discusses a plant system that is not modeled in the PRA, but that, if modeled, would bring the importance of the North Anna Reservoir below the LSS guideline values. The Decay Heat Release Valve (DHRV) can be used as an alternative to the power-operated relief valves (PORVs) for short-term decay heat removal. In the PRA, the licensee estimates that the probability of failure of the POVRs is approximately 1E-2/demand. The licensee discusses the similarities between the DHRV and the PORV SSCs and operating procedures and estimates that the DHRV unavailability should be similar to the PORVs and bounded by a 0.1 (1E-1) probability of failure on demand. The licensee identified all the cut sets in the PRA where the failure of the condenser cooling and the PORVs contributed to CDF and LERF. Taking credit for the DHRV with a probability of failure on demand of 0.1 in cut sets where it could be used, the RAW for LERF would decrease from 2.9 to 1.5 and the RAW for CDF would decrease from 2.0 to 1.8; and these values are below the LSS guidelines used in the STP exemption request.

Given the existence of the DHRV that is not modeled in the PRA, the inability to operate the plant with the North Anna Reservoir unavailable, the small likelihood that the water temperature would exceed the TS limit for an extended period of time, and the procedure that would recognize and monitor operation of the North Anna Reservoir in violation of the TS water level requirement, it is reasonable to conclude that the North Anna Reservoir TS constraints would not be significant if all these considerations were included in the estimated values.

4.0 CONCLUSION

Based on the above review, the staff finds that the proposed relocation of LCO 3/4.7.5.1.b from the CTS to the TRM (1) does not present a challenge to the plant design basis, and (2) does not negatively impact the limiting equipment availability assumptions used in the deterministic analysis to establish margins of safety (related to 10 CFR 50.36, criteria 1 through 4). The staff therefore concludes that the current plant-specific LCO 3/4.7.5.1.b at NAPS does not meet the 10 CFR 50.36 criteria for retention in the proposed NAPS ITS, and therefore the relocation of NAPS LCO 3/4.7.5.1.b from the CTS to the TRM will not adversely impact the safe operation of the plant, satisfies Section182a of the Atomic Energy Act, and is consistent with 10 CFR 50.36; thus, the proposed changes are acceptable.

Principal Contributors:
J. Golla
S. Dinsmore

REFERENCES

1. Letter, David A. Christian (Virginia Electric and Power Company), to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company, North Anna Power Station Units I and 2, Proposed Technical Specification Changes, Improved Technical Specifications*, dated December 11, 2000.

2. Letter, Leslie N. Hartz (Virginia Electric and Power Company), to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company (Dominion), North Anna Power Station Units I and 2, Proposed Improved Technical Specifications, Request for Additional Information, ISTS 3.77 and ITS 3.7.9, Beyond Scope Issues (TAC Nos. MB1439, MB1440, MB1451, and MB1452)*, dated November 19, 2001.

3. Letter, Leslie N. Hartz (Virginia Electric and Power Company), to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company (Dominion), North Anna Power Station Units I and 2, Proposed Improved Technical Specifications, Request for Additional Information, ISTS 3.77 and ITS 3.7.9, Beyond Scope Issues (TAC Nos. MB1439, MB1440, MB1451, and MB1452)*, dated January 25, 2002.

4. Letter, Leslie N. Hartz (Virginia Electric and Power Company), to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company (Dominion), North Anna Power Station Units I and 2, Proposed Improved Technical Specifications, Request for Additional Information, ITS 3.7.9, Beyond Scope Issues (TAC Nos. MB1451, and MB1452)*, dated February 18, 2002.

5. Letter, Leslie N. Hartz (Virginia Electric and Power Company), to U.S. Nuclear Regulatory Commission, *Virginia Electric and Power Company (Dominion), North Anna Power Station Units I and 2, Proposed Improved Technical Specifications, Request for Additional Information, ISTS 3.77 and ITS 3.7.9, Beyond Scope Issues (TAC Nos. MB1439, MB1440, MB1451, and MB1452)*, dated January 25, 2002.

6. NRC Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, July 1998.

7. *Dominion Nuclear Connecticut Inc. (Millstone Nuclear Power Station, Units 2 and 3)*, CLI-

01-24,slip op, December 5, 2001.

8. *Dominion Nuclear Connecticut Inc. (Millstone Nuclear Power Station, Units 2 and 3)* LBP-01-10, 53 NRC 273 (2001)

9. Letter, John A. Zwolinski (U.S. Nuclear Regulatory Commission) to William T. Cottle (STP Nuclear Operating Company), *South Texas Project, Units 1 and 2 - Safety Evaluation on Exemption Requests from Special Treatment Requirements of 10 CFR Parts 21, 50, and 100*, dated August 3, 2001.

10. NRC Inspection Manual, Manual Chapter 609, *Significance Determination Process*, August 16, 2001.

11. NRC Regulatory Guide 1.177, *An Approach for Plant-Specific Risk-Informed Decisionmakling: Technical Specifications*, August 1998.

ATTACHMENT 7

Staff Safety Evaluation for Beyond-Scope Item 3, Section G.1

“NAPS Safety Analysis and CSA Calculation

to Determine Allowable Values in ITS”

SAFETY_EVALUATION_BY_THE_OFFICE_OF_NUCLEAR_REACTOR_REGULATION

TECHNICAL_SPECIFICATION_CHANGES_BY_INCORPORATION_OF_IMPROVED

TECHNICAL_SPECIFICATIONS

VIRGINIA_ELECTRIC_AND_POWER_COMPANY

NORTH_ANNA_POWER_STATION_UNITS_1_AND_2

DOCKET_NOS._50-338_AND_50-339

1.0 INTRODUCTION

By letter dated December 11, 2000, Virginia Electric and Power Company (the licensee), proposed license amendments in the form of changes to the technical specifications to Facility Operating License Numbers NPF-4 and NPF-7 for North Anna Power Station Units 1 and 2. The proposed changes will revise the North Anna current technical specifications (CTS) to be consistent with NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 1, and certain generic changes to the NUREG. The licensee used the guidance of NEI 96-06, "Improved Standard Technical Specification Conversion (ITS) Guidance," dated August 1996, and NRC Administrative Letter 96-04, "Efficient Adoption of Improved Standard Technical Specifications," dated October 9, 1996 to prepare this submittal. The proposed changes will remove the trip setpoint settings from the TS and modify the allowable values. The allowable value specified in the ITS serves as the Limiting Safety System Setting (LSSS) as required by 10 CFR 50.36(c)(1)(ii)(A), defined by the regulation as "....setting for automatic protective devices...so chosen that automatic protective action will correct the abnormal situation before a Safety Limit (SL) is exceeded."

Following initial review of the submittal, the staff, by letter dated September 6, 2001, requested additional information regarding the setpoint methodology used to modify the allowable values in the ITS. By letter dated December 13, 2001, the licensee submitted Technical Report EE-0116, Revision 1, which provides the setpoint methodology and the calculation of the modified allowable values. The setpoint methodology was reviewed by the NRC when the staff approved the North Anna technical specification with the setpoints and the allowable values. This setpoint methodology is based on Westinghouse values for Safety Analysis Limits, Channel Statistical Allowance (CSA), which is the combination of various channel uncertainties derived by the square-root-of-the-sum-of-the-squares and algebraic techniques. The NRC defined the LSSS as the allowable values specified in the ITS.

2.0 EVALUATION

The licensee performed its own safety analysis and CSA calculations to determine allowable values in the ITS. The allowable values will be evaluated to ensure that they are bounded by

the CSA calculation of record and by the safety analysis assumption documented in Technical

Report NE-0994, Revision 7, “Safety Analysis Limits for Technical Specification Instrumentation - Companion to EE-0101 - Surry and North Anna Power Stations,” dated April 28, 1999. Based on this evaluation, the existing allowable values are revised for use in the North Anna ITS conversion.

1) Removal of Trip Setpoints from the TS Table

The licensee proposed to move the requirement for the trip setpoints from the TS to the Technical Requirements Manual (TRM). The NRC defined the LSSS as allowable values specified in the ITS and, therefore, the proposed removal of the trip setpoints from the TS is acceptable.

2) Functional Unit 1. C., Containment Pressure - High

The revised allowable value of 17.7 psia is based on maintaining a nominal trip setpoint value of 17.0 psia. The allowable value of 18.5 psia used in CTS does not conform to the methodology and does not reflect the rack uncertainties as detailed in CSA Calculation EE-0052, Revision 2, CSA for North Anna Containment Narrow Range Pressure ESFAS Trips. Therefore, the more conservative proposed allowable value of 17.7 psia is consistent with the calculated allowable value and is acceptable.

3) Function 1d. Pressurizer Pressure - Low-Low

The revised allowable value of >1770 psig is based on maintaining a nominal trip setpoint value of 1780.0 psig. The allowable value of >1755 psig used in CTS does not conform to the setpoint methodology used for arriving at the LSSS and does not reflect accurately the uncertainties detailed in CSA Calculation EE-0069, Revision 3, CSA for North Anna Pressurizer Pressure Protection. The actual calculated allowable value is >1771.7 psig. The proposed allowable value of >1770 psig is sufficiently close to the calculated value and the offset is bounded by the safety margin as documented in Technical Report EE-0101, Revision 3, Setpoint Basis Document - Analytical Limits, Setpoints and Calculations for Technical Specification Instrumentation At North Anna and Surry Power Stations, dated October 19, 1999. Therefore, the more conservative proposed allowable value is acceptable.

4) Function 1f. High Steam Flow in Two Steam Lines

The revised allowable value of 42.0% of nominal flow from 0 to 20% power increasing linearly to <111.0% of nominal flow at 100% power are based on maintaining a nominal trip setpoint of 40.0% of nominal flow from 0 to 20% power increasing linearly to 110.0% of nominal flow at 100% power. The proposed allowable values are accurately calculated by using setpoint methodology and the CSA rack error terms from Calculation EE-0736, Revision 0, CSA for North Anna Steam Flow, Steam Pressure and Feedwater Flow Protection. Therefore, the more conservative proposed allowable values are acceptable.

5) Function 2c. Containment Pressure High - High

The licensee proposes to change the allowable value from ≤29.25 psia to ≤28.45 psia. This modification causes the containment pressure signal to activate at a lower value than before, which will be earlier in an accident. The proposed allowable value of ≤28.45 psia is based on maintaining a nominal trip setpoint value of ≤27.75 psia. The TS allowable value of ≤29.25 psia used in North Anna's CTS does not conform to the setpoint methodology and does not reflect the rack uncertainties in detailed in CSA Calculation EE-0052, Revision 2, CSA for North Anna Containment Narrow Range Pressure ESFAS Trips. The proposed allowable value of ≤28.45 psia is approximately equal to the calculated allowable value and therefore is acceptable.

6) Function 4c. Containment Pressure - Intermediate High-High

The licensee proposes to change the allowable values from 19.3 psia to 18.5 psia. This modification causes the containment pressure signal to activate at a lower value than before, which will be earlier in an accident. The proposed allowable value of <18.5 psia is based on maintaining a nominal trip setpoint value of 17.8 psia. The allowable value of <19.3 psia used in North Anna's CTS does not conform to the setpoint methodology and does not reflect the rack uncertainties in detailed in CSA Calculation EE-0052, Revision 2, CSA for North Anna Containment Narrow Range Pressure ESFAS Trips. The proposed allowable value of 18.5 psia is approximately equal to the calculated value and therefore is acceptable

7) Function 4d. High Steam Flow in Two Steam Lines

The modified allowable value is the same as Function 1f, and is therefore acceptable.

3.0 CONCLUSION

Based on the above discussion, the staff concludes that the licensee's proposed changes to the TS are acceptable.

Principal Contributor: Sang Rhow

Date: April 5, 2002

cc:

Mr. C. Lee Lintecum
County Administrator
Louisa County
P.O. Box 160
Louisa, Virginia 23093

Ms. Lillian M. Cuoco, Esq.
Senior Nuclear Counsel
Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Building 475, 5th Floor
Rope Ferry Road
Rt. 156
Waterford, Connecticut 06385

Dr. W. T. Lough
Virginia State Corporation
Commission
Division of Energy Regulation
P.O. Box 1197
Richmond, Virginia 23209

Old Dominion Electric Cooperative
4201 Dominion Blvd.
Glen Allen, Virginia 23060

Mr. Stephen P. Sarver, Director
Nuclear Licensing & Operations Support
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

Office of the Attorney General
Commonwealth of Virginia
900 East Main Street
Richmond, Virginia 23219

Senior Resident Inspector
North Anna Power Station
U.S. Nuclear Regulatory Commission
1024 Haley Drive
Mineral, Virginia 23117

Mr. David A. Heacock
Site Vice President
North Anna Power Station
P.O. Box 402
Mineral, Virginia 23117-0402

Mr. Richard H. Blount, II
Site Vice President
Surry Power Station
Virginia Electric and Power Company
5570 Hog Island Road
Surry, Virginia 23883-0315

Robert B. Strobe, M.D., M.P.H.
State Health Commissioner
Office of the Commissioner
Virginia Department of Health
P. O. Box 2448
Richmond, Virginia 23218

Mr. William R. Matthews
Vice President - Nuclear Operations
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, Virginia 23060-6711