February 21, 2007

Mr. James H. Lash
Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: THE CONVERSION TO THE IMPROVED TECHNICAL SPECIFICATIONS WITH BEYOND-SCOPE ISSUES (TAC NOS. MC6285, MC6286, MC6579 - MC6612, MC6614 - MC6626, AND MC6783 - MC6792)

Dear Mr. Lash:

The Commission has issued the enclosed Amendment No. 278 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station Unit No. 1 and Amendment No. 161 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station Unit No. 2. These amendments consist of changes to the Beaver Valley Power Station Unit Nos. 1 and 2 (BVPS-1 and 2) Operating Licenses and Technical Specifications (TSs), in response to your application dated February 25, 2005, as supplemented by letters dated November 11, 2005, April 19, 2006, July 10, 2006, September 1, 2006, October 24, 2006, December 7, 2006, and February 1, 2007.

The amendment converts the current TSs (CTSs) to the improved TSs (ITSs) and relocates certain requirements to other licensee-controlled documents. The ITSs are based on NUREG-1431, “Standard Technical Specifications - Westinghouse Plants,” Revision 2; "NRC Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," dated July 22, 1993 (58 FR 39132); and 10 CFR 50.36, "Technical Specifications." Technical Specification Task Force changes were also incorporated to make the resulting BVPS-1 and 2 more consistent with Revision 3 of NUREG-1431. The license amendment also consolidate the BVPS-1 and 2 CTSs into a single set of ITS applicable to both units. The purpose of the conversion is to provide clearer and more readily understandable requirements in the TSs for BVPS-1 and 2 to ensure safe operation. In addition, the amendment includes a number of issues that were considered beyond the scope of NUREG-1431.

Included in the amendments are the following two conditions for the BVPS-1 and 2 operating licenses: (1) the requirement to relocate certain CTSs requirements into licensee-controlled documents during the implementation of the ITSs, and (2) the schedule for the first performance of new and revised surveillance requirements for the ITSs. These license conditions, which are discussed in the enclosed Safety evaluation (SE), are part of the implementation of the ITSs and constitute enforceable commitments that the NRC staff is relying upon in approving the amendment.

The ITSs will become the governing TSs for BVPS-1 and 2 upon the date of implementation. This means that until the implementation of the ITSs is complete, the CTSs shall remain in effect. Upon complete implementation of the ITSs, please submit a letter stating as such within 14 days of the date of completion.

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

Sincerely,

***/RA/***

Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures:
1. Amendment to DPR-66
2. Amendment to NPF-73
3. Safety Evaluation

cc w/o encls: See next page

The ITSs will become the governing TSs for BVPS-1 and 2 upon the date of implementation. This means that until the implementation of the ITSs is complete, the CTSs shall remain in effect. Upon complete implementation of the ITSs, please submit a letter stating as such within 14 days of the date of completion.

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

Sincerely,

***/RA/***

Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures:
1. Amendment to DPR-66
2. Amendment to NPF-73
3. Safety Evaluation

cc w/o encls: See next page

DISTRIBUTION:

| | | | |
|---|---|---|---|
| PUBLIC | RidsNrrLASLittle | RidsAcrsAcnwMailCenter | JYLee |
| LPLI-1 R/F | RidsNrrDirsItsb | RidsRgn1MailCenter | RTjader |
| RidsNrrDorlLpl1-1 | RidsNrrDorlDpr | GHill, OIS | PHearn |
| NMorgan | RidsOGCRp | RClark | CHarbuck |
| CSchulten | KDesai | RHernandez | LLois |
| SMazumdar | SUnikewicz | KWood | SSun |

Package Accession Number: ML070390063
Amendment Accession Number: ML070160593
Operating License pages Accession Number: ML070390530
Tech. Spec. pages Accession Number: ML070390284
Table A Accession Number: ML070390019
Table L Accession Number: ML070390024
Table LA Accession Number: ML070390051
Table M Accession Number: ML070390057
Table R Accession Number: ML070390061

| OFFICE | LPLI-1/PM | LPLI-1/LA | Tech/Branch | NRR/ITSB/BC | OGC | NRR/LPLI-1/BC (A) |
|---|---|---|---|---|---|---|
| NAME | NMorgan | SLittle | see note | TKobetz | MZobler | JBoska |
| DATE | 2/12/07 | 2/12/07 | 2/20/07 | 2/13/07 | 7/16/07 | 2/20/07 |

Note: Safety evaluation input on beyond-scope issues has been provided by the respective technical branches. The memos transmitting the SEs may be found in ADAMS under the TAC Nos. listed in the title above.

**OFFICIAL RECORD COPY**

Beaver Valley Power Station, Unit Nos. 1 and 2

cc:

Gary R. Leidich
President and Chief Nuclear Officer
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-19
76 South Main Street
Akron, OH 44308

Joseph J. Hagan
Senior Vice President of Operations
and Chief Operating Officer
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

Danny L. Pace
Senior Vice President, Fleet Engineering
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

Jeannie M. Rinckel
Vice President, Fleet Oversight
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
76 South Main Street
Akron, OH 44308

David W. Jenkins, Attorney
FirstEnergy Corporation
Mail Stop A-GO-18
76 South Main Street
Akron, OH 44308

Manager, Fleet Licensing
FirstEnergy Nuclear Operating Company
Mail Stop A-GHE-115
395 Ghent Road
Akron, OH 44333

Ohio EPA-DERR
ATTN: Zack A. Clayton
P.O. Box 1049
Columbus, OH 43266-0149

James H. Lash
Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

Director, Fleet Regulatory Affairs
FirstEnergy Nuclear Operating Company
Mail Stop A-GHE-315
395 Ghent Road
Akron, Ohio 44333

Manager, Site Regulatory Compliance
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-A
P.O. Box 4, Route 168
Shippingport, PA 15077

Richard Anderson
Vice President, Nuclear Support
FirstEnergy Nuclear Operating Company
Mail Stop A-GO-14
Akron, Ohio 44308

Commissioner James R. Lewis
West Virginia Division of Labor
749-B, Building No. 6
Capitol Complex
Charleston, WV 25305

Director, Utilities Department
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

Director, Pennsylvania Emergency
Management Agency
2605 Interstate Dr.
Harrisburg, PA 17110-9364

cc:

Dr. Judith Johnsrud
Environmental Coalition on Nuclear Power
Sierra Club
433 Orlando Avenue
State College, PA  16803

Director
Bureau of Radiation Protection
Pennsylvania Department of
  Environmental Protection
Rachel Carson State Office Building
P.O. Box 8469
Harrisburg, PA 17105-8469

Mayor of the Borough of Shippingport
P.O. Box 3
Shippingport, PA  15077

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA  19406

Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 298
Shippingport, PA  15077

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 278
License No. DPR-66

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee), dated February 25, 2005, as supplemented November 11, 2005, April 19, July 10, September 1, October 24, December 7, 2006, and February 1, 2007, 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. DPR-66 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 278, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. License condition, 2.C.(9), is deleted.

4. New license conditions are added in Appendix C to address the relocation of Technical Specification requirements and the performance of new and revised Surveillance Requirements (SRs):

   Relocation of Certain Technical Specification Requirements

   License Amendment No. 278 authorizes the relocation of certain Technical Specifications to other licensee-controlled documents. Implementation of this amendment shall include relocation of the requirements to the specified documents, as described in (1) Sections 4D and 4E of the NRC's staff's Safety Evaluation, and (2) Table LA, Removed Detail Changes, and Table R, Relocated Specifications, attached to the NRC staff's Safety Evaluation, which is enclosed in this amendment.

   Schedule for New and Revised Surveillance Requirements (SRs)

   The schedule for performing the new or revised SRs in Amendment No. 278 shall be as follows:

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

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

   For SRs that existed prior to this amendment, 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.

   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.

5. This license amendment is effective as of the date of its issuance and shall be implemented within 150 days.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

John P. Boska, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and
Technical Specifications

Date of Issuance: February 21, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 278

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of the Facility Operating License for Unit No. 1 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| Remove | Insert |
|---|---|
| 3 | 3 |
| 6 | 6 |
| 6a | 6a |
| ------ | Page 4 of Appendix C |

Replace the following pages of Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number.

**NOTE:** Only one copy of the Technical Specification pages is provided, which is applicable to both units.

| Remove | Insert |
|---|---|
| All | All |

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. NPF-73

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee), dated February 25, 2005, as supplemented November 11, 2005, April 19, July 10, September 1, October 24, December 7, 2006, and February 1, 2007, 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-73 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 161, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. License condition, 2.C.(12), is deleted.

4. New license conditions are added in Appendix D to address the relocation of Technical Specification requirements and the performance of new and revised Surveillance Requirements (SRs):

    Relocation of Certain Technical Specification Requirements

    License Amendment No. 161 authorizes the relocation of certain Technical Specifications to other licensee-controlled documents. Implementation of this amendment shall include relocation of the requirements to the specified documents, as described in (1) Sections 4D and 4E of the NRC's staff's Safety Evaluation, and (2) Table LA, Removed Detail Changes, and Table R, Relocated Specifications, attached to the NRC staff's Safety Evaluation, which is enclosed in this amendment.

    Schedule for New and Revised Surveillance Requirements (SRs)

    The schedule for performing the new or revised SRs in Amendment No. 161 shall be as follows:

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

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

    For SRs that existed prior to this amendment, 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.

    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.

5. This license amendment is effective as of the date of its issuance and shall be implemented within 150 days.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

John P. Boska, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and Technical Specifications

Date of Issuance: February 21, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 161

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

Replace the following pages of the Facility Operating License for Unit No. 2 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| Remove | Insert |
|---|---|
| 3a | 3a |
| 5 | 5 |
| ----- | Page 4 of Appendix D |

Replace the following pages of Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number.

**NOTE:** Only one copy of the Technical Specification pages is provided, which is applicable to both units.

| Remove | Insert |
|---|---|
| All | All |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 278 AND 161 TO FACILITY OPERATING

LICENSE NOS. DPR-66 AND NPF-73

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-334 AND 50-412

## 1.0 INTRODUCTION

By application dated February 25, 2005, Agencywide Documents Access and Management System (ADAMS) Accession No. ML050610339, and as supplemented by letters listed below, FirstEnergy Nuclear Operating Company (FENOC) requested a full conversion from the current Technical Specifications (CTS) to a set of Improved Technical Specifications (ITS) based on NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 2, for Beaver Valley Power Station Unit Nos. 1 and 2 (BVPS-1 and 2). In addition, Technical Specification Task Force (TSTF) changes were also incorporated to make the resulting BVPS-1 and 2 ITS more consistent with Revision 3 of NUREG-1431. The proposed license amendment request (LAR) would also consolidate the BVPS-1 and 2 Technical Specifications (TSs) into a single set of ITS applicable to both units.

The following supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination, and as well, updated the ITS conversion to incorporate TS changes due the approved LARs, TSTF changes, and resolutions to Nuclear Regulatory Commission (NRC) staff comments:

- Letter dated November 11, 2005 (ADAMS Accession No. ML053290071)
- Letter dated April 19, 2006 (ADAMS Accession No. ML061140093)
- Letter dated July 10, 2006 (ADAMS Accession No. ML061940177)
- Letter dated September 1, 2006 (ADAMS Accession No. ML062490197)
- Letter dated October 24, 2006 (ADAMS Accession No. ML063000086)
- Letter dated December 7, 2006 (ADAMS Accession No. ML063450214)
- Letter dated February 1, 2007 (ADAMS Accession No. ML070360546)

To expedite review of the application, the NRC staff posted its questions (Q's) related to the BVPS-1 and 2 application to a secure database through the BVPS ITS Conversion web page. The licensee then posted responses (A's) to the database, also through the web page. Access to the Q's&A's database is restricted so that only designated licensee and NRC staff can enter information into the database; however, the public can enter the database to read the questions and responses. To comply with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.4, for written communications for LARs and to have the database on the BVPS-1 and 2 dockets, the licensee submitted a copy of the complete database to the NRC (ADAMS Accession No. ML063000089) as an attachment to its October 24, 2006, supplement. The public can access the database through the NRC web site at www.nrc.gov by the following process: (1) click on the tab labeled "Nuclear Reactors" on the NRC home page along the upper part of the web page, (2) then click on the link to "Operating Reactors" which is under "Regulated Activities" on the left hand side of the web page, (3) then click on the link to "Improved Standard Technical Specifications," which is on the right hand side of the page, and (4) finally click on the link to "Comments on the application and responses by Beaver Valley Power Station," near the bottom of the web page, to open the database. The Q's&A's are organized by ITS Sections 1.0, 2.0, 3.0, 3.1 through 3.9, 4.0, and 5.0, which are listed first, and the beyond scope issues (BSIs) 1 through 30, which are listed later. For every listed ITS section or BSI, there is an Q&A which can be read by clicking on the ITS section or BSI number. The licensee's responses are shown by a solid triangle adjacent to the ITS section or BSI number, and, to read the response, click on the triangle. To page down through the ITS sections to the BSIs, click on "next" along the top of the page or on "previous" to return to the previous page.

## 2.0 BACKGROUND

BVPS-1 and 2 have been operating with the TSs issued with the original Facility Operating Licenses dated July 2, 1976 (for BVPS-1), and August 14, 1987 (for BVPS-2), as amended. The proposed conversion to the ITSs is based upon:

- NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," (ISTSs) Revision 2, dated April 30, 2001, with additional changes to make the resulting BVPS-1 and 2 ITS more consistent with Revision 3 of NUREG-1431.
- The BVPS-1 and 2 CTSs;
- "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (Final Policy Statement), published on July 22, 1993 (58 FR 39132); and
- 10 CFR 50.36, “Technical specifications,” as amended July 19, 1995 (60 FR 36953).

Hereinafter, the proposed TSs for BVPS-1 and 2 are referred to as the ITSs, the existing TSs are referred to as the CTSs, and the improved standard TSs, given in NUREG-1431, are referred to as the ISTSs. The corresponding Bases are ITS Bases, CTS Bases, and ISTS 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 ITSs on the ISTSs, the Final Policy Statement, and the requirements in 10 CFR 50.36, the licensee retained portions of the CTSs as a basis for the ITSs. During the course of its review, the NRC staff utilized the Q's&A's database, conducted a series of telephone conference calls, and held meetings with the licensee. The Q's&A's database, meetings, and conference calls served to clarify the ITSs with respect to the guidance in the

Final Policy Statement and the ISTSs. The NRC staff requested that the licensee docket the Q's&A's database in a sworn statement with regards to its accuracy, as well as docket all requests for additional information (RAIs) and responses under oath and affirmation, in a supplement to the license amendment. The RAIs are a subset of the Q's&A's database and are used to support the SEs for BSIs. The licensee also proposed changes of a generic nature that were not in the ISTSs. The NRC staff requested that the licensee submit such generic changes as proposed changes to the ISTSs through the industry TSTF. These generic issues were considered for specific applications in the BVPS-1 and 2 ITS. Consistent with the Commission's Final Policy Statement and 10 CFR 50.36, the licensee proposed transferring some CTS requirements to licensee-controlled documents (such as the BVPS-1 and 2 Updated Final Safety Analysis Reports (UFSARs)), for which changes to the documents by the licensee are controlled by a regulation (e.g.,10 CFR 50.59) and which may be made without prior NRC approval. NRC-controlled documents, such as the TSs, may not be changed by the licensee without prior NRC approval. In addition, human factors principles were emphasized to add clarity to the CTS requirements being retained in the ITS, and to define more clearly the appropriate scope of the ITS. Further, significant changes were proposed to the CTS Bases to make each ITS requirement clearer and easier to understand.

The overall objective of the proposed amendments, consistent with the Final Policy Statement, is to rewrite, reformat, and streamline the BVPS-1 and 2 CTSs to provide clearer, more readily understandable requirements to ensure safer operation of the units, while still satisfying the requirements of 10 CFR 50.36. During its review, the NRC staff relied on the Final Policy Statement and 10 CFR 50.36, and the ISTSs as guidance for acceptance of CTS changes. This SE provides a summary basis for the NRC staff's conclusion that use of the licensee's proposed ITSs based on ISTSs, as modified by plant-specific changes, is acceptable for continued operation of BVPS-1 and 2. This SE also explains the NRC staff's conclusion that the ITSs are consistent with the BVPS-1 and 2 current licensing bases and the requirements of 10 CFR 50.36.

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

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

## 3.0 REGULATORY REQUIREMENTS

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

> [S]uch technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization . . . of

> special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

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

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

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 ISTSs in NUREG-1431 were established as a model for developing the ITSs for Westinghouse plants, in general. The ISTSs reflect the results of a detailed review of the application of the Interim Policy Statement criteria which have been incorporated in 10 CFR 50.36(c)(2)(ii), to generic system functions, which were published in a "Split Report" issued to the nuclear steam supply system vendor owners groups in May 1988. ISTSs also reflect the results of extensive discussions concerning various drafts of ISTSs 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 ISTSs present requirements that are necessary to protect public health and safety. The ISTSs in NUREG-1431, Revision 2, as modified, apply to BVPS-1 and 2.

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 ISTSs and encouraged licensees to use the ISTSs as the basis for plant-specific TS amendments and for complete conversions to ITSs based on the ISTSs. In addition, the Final Policy Statement gave guidance for evaluating the required scope of the TSs and defined the guidance criteria to be used in determining which of the LCOs and associated SRs should remain in the TSs. The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TSs, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

> [T]here is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

By this approach, existing LCO requirements that fall within or satisfy any of the criteria in the Final Policy Statement should be retained in the TSs; those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36 (60 FR 36953, July 19, 1995). The four criteria are stated as follows:

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

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

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

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

Part 4.0 of this SE explains the NRC staff's determination that the conversion of the BVPS-1 and 2, CTSs to ITSs based on ISTSs, as modified by plant-specific changes, is consistent with the BVPS-1 and 2, current licensing bases, the requirements and guidance of the Final Policy Statement, and 10 CFR 50.36.

## 4.0 EVALUATION

In its review of the BVPS-1 and 2 ITS application, the NRC staff evaluated five kinds of CTS changes, as defined by the licensee. The NRC staff's review also included an evaluation of whether existing regulatory requirements are adequate for controlling future changes to requirements that are removed from the CTS and placed in licensee-controlled documents. The following are the five types of CTS changes:

| | |
|---|---|
| A | Administrative - Changes to the CTS that do not result in new requirements or change operational restrictions and flexibility. |
| M | More Restrictive - Changes to the CTS that result in added restrictions or reduced flexibility. |
| L | Less Restrictive - Changes to the CTS that result in reduced restrictions or added flexibility. |
| LA | Removed Details - Changes to the CTS that eliminate detail and relocate the detail to a licensee-controlled document. Typically, this involves details of system design and system description including design limits, description of system operation, procedural details for meeting TS requirements or reporting requirements, and cycle-specific parameter limits and TS requirements redundantly located in other licensee-controlled documents. |
| R | Relocated Specifications - Changes to the CTS that relocate LCO requirements that do not meet the selection criteria of 10 CFR 50.36(c)(2)(ii). |

The ITS application included a detailed discussion of change (DOC) for each proposed change to the CTS. The DOCs are identified by an alpha-numeric designation formatted in accordance with the type of change listed above followed by a sequential number (e.g., A.1, M.1, L.1, LA.1, R.1, etc.). In addition, the ITS application also included an explanation of the difference between each ITS and ISTS requirements in a numbered justification for deviation (JFD). The JFDs are necessary in order to make the ISTS document specific to BVPS-1 and 2 design bases.

A summary of the DOCs as presented in the ITS application, are listed and described in the following five tables attached to this SE:

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

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 CTSs or ITSs are provided in the licensee's application and supplemental letters.

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

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

The control of specifications, requirements, and information relocated from the CTSs to licensee-controlled documents is described in Section F. The review of BSIs is contained in Section G.

A. Administrative Changes to the CTS

Administrative changes are intended to incorporate human factors principles into the form and structure of the ITSs 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 ITSs reflects this type of change. In order to ensure consistency, the NRC staff and the licensee have used the ISTSs as guidance to reformat and make other administrative changes. Among the changes proposed by the licensee and found acceptable by the NRC staff are:

- Identifying plant-specific wording for system names, etc.;
- Splitting up requirements currently grouped under a single current specification and moving them to more appropriate locations in two or more specifications of the ITS;
- Combining related requirements currently presented in separate specifications of the CTS into a single specification of ITS;
- Presentation changes that involve rewording or reformatting for clarity (including moving an existing requirement to another location within the TSs) but that do not involve a change in requirements;
- Wording changes and additions that are consistent with CTS interpretation and practice and that more clearly or explicitly state existing requirements;
- Deletion of TSs that no longer apply;
- Deletion of details that are strictly informational and have no regulatory basis;
- Deletion of redundant TS requirements that exist elsewhere in the TSs;
- Changes necessary to combine the BVPS-1 and 2 TS into a single document.

Table A attached to this SE lists the administrative changes being made in the BVPS-1 and 2 ITS conversion. Table A is organized in ITS section order which includes the following:

1. ITS/CTS Section;
2. DOC identifier;

3. Summary description of the administrative changes; and
4. Reference to ITS and CTS requirements.

The NRC staff reviewed all of the administrative and editorial changes listed in Table A and finds them acceptable because they are compatible with the Writer's Guide and the ISTSs, do not result in any change in operating requirements, and are consistent with the Commission's regulations.

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

The licensee, in electing to implement the specifications of the ISTSs, proposed a number of requirements that are more restrictive than those in the CTSs. The ITS requirements in this category include requirements that are either new, more conservative than corresponding requirements in the CTSs, or have additional restrictions that are not in the CTSs, but are in the ISTSs. Examples of more restrictive requirements are placing an LCO on plant equipment that is not required by the CTS, more restrictive requirements to restore inoperable equipment, and more restrictive SRs. Table M attached to this SE lists the more restrictive changes being made in the BVPS-1 and 2 ITS conversion. Table M is organized in ITS section order which includes the following:

1. ITS/CTS Section;
2. DOC identifier;
3. Summary description of each more restrictive change adopted; and
4. Reference to ITS and CTS requirements.

The NRC staff concludes that the more restrictive changes listed in Table M are acceptable. The proposed changes are additional restrictions on plant operation that enhance plant safety and are consistent with the BVPS-1 and 2 licensing bases and the Commission's regulations.

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

Less restrictive requirements include deletions and relaxations to portions of the CTS requirements that are being retained in the ITS. When requirements have been shown to give little or no safety benefit, their relaxation or removal from the TSs may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new NRC staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups' comments on ISTSs.

The less restrictive changes to the CTS requirements were grouped in the following eight categories:

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

The NRC staff's evaluation of each category is as follows:

Category 1 — Relaxation of LCO Requirement

Certain CTS LCOs specify limits on operational and system parameters beyond those necessary to ensure meeting safety analysis assumptions and, therefore, are considered overly restrictive. The CTS also contain operating limits that have been shown to give little or no safety benefit to the operation of the plant. The ITSs, consistent with the guidance in the ISTSs, would delete or revise such operating limits. CTS LCO changes of this type include: (1) redefining operating modes, including mode title changes; (2) deleting or revising operational limits to establish requirements consistent with applicable safety analyses; (3) deleting requirements for equipment or systems that establish system capability beyond that assumed to function by the applicable safety analyses, or that are implicit to the ITS requirement for systems, components, and devices to be operable; and (4) adding allowances to use administrative controls on plant devices and equipment during times when automatic control is required, or to establish temporary administrative limits, as appropriate, to allow time for systems to establish equilibrium operation. TSs changes represented by this type allow operators to more clearly focus on issues important to safety. The resultant ITS LCOs maintain an adequate degree of protection consistent with the safety analysis. They also improve focus on issues important to safety and provide reasonable operational flexibility without adversely affecting the safe operation of the plant. Changes involving the relaxation of LCO's are, therefore, consistent with the guidance established by the ISTSs, taking into consideration the BVPS-1 and 2 current licensing bases.

Category 2 — Relaxation of Applicability

The CTS require compliance with the LCO during the applicable Mode(s) or other conditions specified in the Specification's Applicability statement. When CTS Applicability requirements are inconsistent with the applicable accident analyses assumptions for a system, subsystem, or component specified in the LCO, the licensee proposed to change the LCO to establish a consistent set of requirements in the ITSs. These modifications or deletions are acceptable because, during the operational or other conditions specified in the ITSs applicability requirements, the LCOs are consistent with the applicable safety analyses. Changes involving relaxation of applicability requirements are, therefore, consistent with the guidance established by the ISTSs, taking into consideration the BVPS-1 and 2 current licensing bases.

Category 3 — Relaxation of Completion Time

Upon discovery of a failure to meet an LCO, the TSs specify time limits for completing Required Actions of the associated TS Conditions. Required Actions establish remedial measures that must be taken within specified Completion Times. Completion Times specify limits on the duration of plant operation in a degraded condition. Incorporating longer Completion Times is acceptable because such Completion Times will continue to be based on the operability status of redundant TSs required features, the capacity and capability of remaining TS-required features, providing a reasonable time for repairs or replacement of required features, vendor-developed standard repair times, and the low probability of a DBA occurring during the repair period. Changes involving relaxation of

Completion Times are, therefore, consistent with the guidance established by the ISTSs, taking into consideration the BVPS-1 and 2 current licensing bases.

Category 4 — Relaxation of Required Action

LCOs specify the lowest functional capability or performance level of equipment that is deemed adequate to ensure safe operation of the facility. When an LCO is not met, the CTSs specify actions to restore the equipment to its required capability or performance level, or to implement remedial measures providing an equivalent level of protection. Compared to CTS-required actions, certain proposed ITS actions would result in extending the time period during which the licensee may continue to operate the plant with specified equipment inoperable. (Upon expiration of this time period, further action, which may include shutting down the plant, is required.) Changes of this type include providing an option to (1) isolate a system, (2) place equipment in the state assumed by the safety analysis, (3) satisfy alternate criteria, (4) take manual actions in place of automatic actions, (5) "restore to operable status" within a specified time frame, (6) place alternate equipment into service, or (7) use more conservative TS instrumentation actuation setpoints. The resulting ITS actions provide measures that adequately compensate for the inoperable equipment, and are commensurate with the safety importance of the inoperable equipment, plant design, and industry practice. Therefore, these action requirements will continue to ensure safe operation of the plant. Changes involving relaxations of action requirements are, therefore, consistent with the guidance established by the ISTSs, taking into consideration the BVPS-1 and 2 current licensing bases.

Category 5 — Deletion of SR

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

Deleting the SRs, including acceptance criteria and/or conditions for performing the SRs, for these items provides operational flexibility, consistent with the objective of the ISTSs, without reducing confidence that the equipment is operable. For example, the CTS contain SRs that are not included in the ISTSs for a variety of reasons. This includes deletion of SRs for measuring values and parameters that are not necessary to meet ISTS LCO requirements. Also, the ISTSs may not include reference to specific acceptance criteria contained in the CTS, because these acceptance criteria are not necessary to meet ISTS LCO requirements, or are defined in other licensee-controlled documents. The changes to SR acceptance criteria are acceptable because appropriate testing standards are retained for determining that the LCO-required features are operable as defined by the ISTSs.

Deleting conditions for performing SRs includes not requiring testing of de-energized equipment (e.g., instrumentation channel checks) or equipment that is already performing its intended safety function (e.g., position verification of valves locked in their safety actuation position). Also included is allowing verification of the position of valves in high radiation areas by administrative means. ITS administrative controls (ITS 5.7) regarding access to high radiation areas make the likelihood of mispositioning such valves small. Waiving performance of surveillance under these conditions is acceptable because the equipment is already performing its intended safety function.

These deletions of CTS SRs optimize test requirements for the affected safety systems and increase operational flexibility. Changes involving relaxations of SRs, as described, are consistent with the guidance established by the ISTSs, taking into consideration the BVPS-1 and 2 current licensing bases.

Category 6 — Relaxation of SR Acceptance Criteria

Prior to placing the plant in a specified operational Mode or other condition stated in the applicability of an LCO, and in accordance with the specified SR time interval thereafter, the CTS require establishing the operability of each LCO-required component by meeting the SRs associated with the LCO. This usually entails performance of testing to demonstrate the operability of the LCO-required components, or the verification that specified parameters are within LCO limits. A successful demonstration of operability requires meeting the specified acceptance criteria, as well as any specified conditions, for the conduct of the test. Relaxations of CTS SRs would include relaxing both the acceptance criteria and the conditions of performance. Also, the ITSs would permit the use of an actual, as well as a simulated, actuation signal to satisfy SRs for automatically actuated systems. This is acceptable because TS-required features cannot distinguish between an “actual” signal and a “test” signal. These relaxations of CTS SRs optimize test requirements for the affected safety systems and increase operational flexibility.

These CTS SR relaxations are, therefore, consistent with the guidance established by the ISTSs, taking into consideration the BVPS-1 and 2 current licensing bases.

Category 7 — Relaxation of Surveillance Frequency

Prior to placing the plant in a specified operational Mode or other condition stated in the applicability of an LCO, and in accordance with the specified SR time interval (frequency) thereafter, the CTS require establishing the operability of each LCO required component by meeting the SRs associated with the LCO. This usually entails performance of 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, at a specified frequency based on the reliability and availability of the LCO-required components. Relaxations of CTS SRs would include extending the interval between the SRs. This interval is the surveillance test interval (STI). These relaxations of CTS SR frequencies (or extending the STI) optimize test requirements for the affected safety systems and increase operational flexibility. These CTS SR frequency relaxations (or extending the STI) are, therefore, consistent with the guidance established by the ISTSs, taking into consideration the BVPS-1 and 2 current licensing bases.

Category 8 — Deletion of Reporting Requirements

The CTS contains requirements that are redundant to reporting regulations in 10 CFR. For example, CTSs include requirements that a "Reportable Event" is any of those conditions specified in 10 CFR 50.73. However, consistent with the ISTSs, the ITSs would omit many of the CTS reporting requirements because the reporting requirements in the regulations cited do not need repeating in the TSs to ensure timely submission to the NRC. Therefore, Category 8 changes have no impact on the safe operation of the plant. Deletion of these requirements is beneficial because it reduces the administrative burden on the licensee and in turn allows increased attention to plant operations important to safety.

Table L attached to this SE lists the less restrictive changes to the BVPS-1 and 2 ITS conversion. Table L is organized in ITS section order which includes the following:

1. ITS/CTS Section;
2. DOC identifier;
3. Summary description of each less restrictive change adopted;
4. Reference to ITS and CTS requirements; and
5. Type of change.

The NRC staff reviewed the less restrictive changes described in Table L and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The BVPS-1 and 2 designs were also reviewed to determine if the specific design bases and licensing bases are consistent with the technical basis for the model requirements in the ISTSs, and thus provide an acceptable basis for approval of the ITSs.

D. Removed Details

When requirements have been shown to give little or no safety benefit, their removal from the TSs may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new NRC staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the owners groups' comments on ISTSs.

A significant number of changes to the CTSs involved the removal of specific requirements and detailed information from individual specifications. These changes were grouped in the following categories:

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 UFSARs 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 Quality Assurance Program Description (QAPD). The regulations at 10 CFR 50.59, specify controls for changing the facility as described in the UFSARs. The regulations at 10 CFR 50.54(a), specify criteria for changing the QAPD. The Licensing Requirements Manual (LRM) is a general reference in the UFSARs, in Appendix 15A for Unit 1 and Appendix 16A for Unit 2, and changes to it are accordingly subject to 10 CFR 50.59. The ITS Bases also contain descriptions of system design. ITS 5.5.10 specifies controls for changing the Bases. Removing details of system design is acceptable because the associated CTS requirements which are being retained without these details are adequate to ensure safe operation of the facility. In addition, retaining such details in TSs is unnecessary to ensure proper control of changes. For example, cycle-specific design limits are contained in the Core Operating Limits Report (COLR) in accordance with Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," dated October 3, 1988.

Since the CTS requirements retained are adequate to ensure safe operation of the facility, the NRC staff concludes that it is acceptable to remove Type 1 details from the CTS and place them in licensee-controlled documents.

Type 2 - Removing Descriptions of System Operation

The plans for normal and emergency operation of the facility are required to be described in the UFSARs by 10 CFR 50.34. ITSs 5.4.1.a and 5.4.1.e will require written procedures to be established, implemented, and maintained for plant operating procedures recommended in Appendix A of Regulatory Guide (RG) 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, dated February 1978, and in all programs specified in ITS Section 5.5, respectfully. The ITS Bases also contain descriptions of system operation. Controls specified in 10 CFR 50.59 apply to changes in procedures as described in the UFSARs and LRM. ITS 5.5.10 specifies controls for changing the Bases. Removing details of system operation is acceptable because the associated CTS requirements being retained without these details are adequate to ensure safe operation of the facility. In addition, retaining such details in TSs is unnecessary to ensure proper control of changes. Therefore, it is acceptable to remove Type 2 details from the CTS and place them in licensee-controlled documents.

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

Details for performing TS SRs or for regulatory reporting are more appropriately specified in the plant procedures. Prescriptive procedural information in a TS requirement is unlikely to contain all procedural considerations necessary for the plant operators to comply with TSs and all regulatory reporting requirements, and referral to plant procedures is therefore required in any event. Changes to procedural details include those associated with limits retained in the ITS. For example, ITS 5.4.1 requires that written procedures covering activities that include all programs specified in ITS 5.5 be established, implemented, and maintained. ITS 5.5.4, “Inservice Testing Program,” requires a program to provide controls for inservice testing (IST) of American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Class 1, 2, and 3 components. The program includes defining testing frequencies specified in the ASME Operation and Maintenance Standards and Codes (OM Codes), and applicable addenda. The ITS also contain requirements to test specific components such as pumps and valves, and establish IST of Quality Group A, B, and C pumps and valves performed in accordance with the requirements for ASME Code, Class 1, 2, and 3 components specified in the ASME OM Codes and addenda, subject to the applicable provisions of 10 CFR 50.55a. ITS Section 5.5.13, “Battery Monitoring and Maintenance Program” contains programmatic requirements for battery restoration and maintenance.

Since the CTS requirements retained are adequate to ensure safe operation of the facility, the NRC staff concludes that it is acceptable to remove Type 3 details from the CTS and place them in licensee-controlled documents.

Type 4 - Removal of Administrative Requirements Redundant to Regulations

Certain CTS administrative requirements are redundant to regulations and thus are relocated to the UFSARs or other appropriate licensee-controlled documents, including the LRM, ODCM, QAPD, or ISI Plan (IIP). The Final Policy Statement allows licensees to relocate to licensee-controlled documents, CTS requirements that do not meet any of the criteria for mandatory inclusion in the TSs. Changes to the facilities or to procedures as described in the UFSARs are made in accordance with 10 CFR 50.59. Changes made in accordance with the provisions of other licensee-controlled documents are subject to the specific requirements of those documents. For example, 10 CFR 50.54(a) governs changes to the QAPD, and ITS 5.5.10 governs changes to the ITS Bases. Based on the above, the NRC staff has determined that it is acceptable to remove Type 4 details from CTS and place them in licensee-controlled documents.

Type 5 - Removing Performance Requirements for Indication-Only Instrumentation and Alarms

Certain CTS requirements are for instruments and alarms that are not required for operability of the LCO-required equipment, and thus may be relocated to the UFSAR or other appropriate licensee-controlled documents. Changes to the facility or to procedures as described in the UFSARs are made in accordance with 10 CFR 50.59. Changes made in accordance with the provisions of other licensee-controlled documents are subject to the specific requirements of those documents. For example, 10 CFR 50.54(a) governs changes to the QAPD, and ITS 5.5.10 governs changes to the ITS Bases. Based on the above, the NRC staff has determined that it is acceptable to remove Type 5 details from CTS and place them in licensee-controlled documents.

Table LA attached to this SE lists the proposed removed detail changes to the BVPS-1 and 2 ITS conversion. Table LA is organized in ITS section order which includes the following:

1. ITS/CTS Section;
2. DOC identifier;
3. Reference to CTS requirements;
4. Summary description of the relocated details and requirements;
5. Name of the licensee-controlled document to contain the relocated details and requirements (location);
6. Regulation (or ITS Specification) for controlling future changes to relocated requirements (change control process); and
7. Type of change.

The NRC staff concludes that the detailed information and specific requirements described in Table LA do not fall within any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii). Nor is the detailed information and specific requirements needed to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to public health and safety. Accordingly, these requirements may be moved to one or more of the following licensee-controlled documents for which changes are adequately governed by a regulatory or TS requirement:

- Bases controlled in accordance with ITS 5.5.10, “Technical Specifications (TS) Bases Control Program.”
- UFSARs (which reference the LRM) controlled by 10 CFR 50.59.
- Programmatic documents required by ITS Section 5.5 and controlled by ITS Section 5.4.
- ISI and IST Programs controlled by 10 CFR 50.55a.
- ODCM controlled by ITS 5.5.1.
- COLRs controlled by ITS 5.6.3.
- QAPD, as approved by the NRC, referenced in the UFSARs, and controlled by 10 CFR Part 50, Appendix B, and 10 CFR 50.54(a).
- Site Emergency Plan controlled by 10 CFR 50.54(q).

E. <u>Relocated Specifications</u>

The Final Policy Statement states that LCOs and associated requirements that do not satisfy or fall within any of the four specified criteria may be relocated from existing TSs (an NRC-controlled document) to appropriate licensee-controlled documents as noted in Section D above. These specifications generally would include LCOs, Required Action Statements (i.e., Actions), and associated SRs. In its application and supplements, the licensee proposed relocating such specifications from the CTS to a licensee-controlled document (i.e., LRM), as appropriate.

Table R attached to this SE lists the relocated changes that would be made in the BVPS-1 and 2 ITS conversion and lists all specifications that are being relocated from the CTSs to licensee-controlled documents. Table R includes the following information:

1. ITS/CTS section;
2. DOC identifier;
3. References to the relocated CTS requirement;
4. Summary descriptions of the relocated CTS requirement;
5. Names of the document that will contain the relocated specifications (i.e., the new location);
6. Methods for controlling future changes to the relocated specifications (i.e., the regulatory change control process); and
7. Type of change.

The NRC staff's evaluation of each relocated specification listed in Table R is provided below.

E.1 Reactivity Control Systems

LCO: CTS 3.1.1.3, DOC R.1

CTS 3.1.1.3, “Boron Dilution,” requires that "The flow rate of reactor coolant through the core shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made." CTS 3.1.1.3 is applicable in "All Modes." This CTS contains SRs that verify at least one reactor coolant pump (RCP) is in operation or that a specific residual heat removal (RHR) system flow is maintained during dilution operations. The ISTS does not contain a corresponding requirement.

Discussion

CTS 3.1.1.3 is based on standard TS 3.1.1.3, “Boron Dilution and Addition” that was contained in the original Standard TS for Westinghouse Plants (Revision 0) of NUREG-0452, dated March 15, 1975. The BVPS-1 TSs are based on Revision 0 of NUREG-0452. The TSs for BVPS-2 are based on the BVPS-1 TSs in order to make the TSs for both units as similar as possible. In NUREG-0452, Revision 2, dated July 1979, TS 3.1.1.3, “Boron Dilution and Addition,” was eliminated from the Standard TS for Westinghouse Plants. This TS does not appear in any subsequent revisions of NUREG-0452, nor in any revisions of NUREG-1431.

In Modes 1-3, the BVPS-1 and 2 CTS (and the ISTS) contain other requirements for reactor coolant system (RCS) flow (RCS Loops in Section 3.4) that require at least one RCS loop in service with the RCP in operation. The flow of a single RCP exceeds the flow requirements of CTS 3.1.1.3. In addition to the RCS loop requirements, the departure from nucleate boiling (DNB) limits applicable in Mode 1 also specify an RCS flow far greater than that required by CTS 3.1.1.3. As such, in Modes 1-3, CTS 3.1.1.3 does not contribute any new or more restrictive RCS flow requirements than already exist in the CTS and the ISTS. The existing and proposed TS requirements for the RCS loops and DNB limits specify more restrictive RCS flow requirements in Modes 1-3 than CTS 3.1.1.3.

In Modes 4, 5, and 6, the BVPS-1 and 2 CTS (and the proposed BVPS-1 and 2 ITS) require that unborated water source isolation valves be secured in the closed position such that the possibility of an inadvertent boron dilution accident is precluded. BVPS-1 and 2 do not assume a design-basis boron dilution accident in Modes 4, 5, or 6. In Mode 4, the CTS (and the proposed BVPS-1 and 2 ITS) contain other requirements (RCS Loops, Mode 4 in Section 3.4) for RCS flow that ensure an RCS or RHR loop is in service to provide sufficient RCS flow to remove decay heat. In Modes 5 and 6, the BVPS-1 and 2 CTS (and the proposed BVPS-1 and 2 ITS) contain other requirements for RCS flow (RCS Loops, Mode 5 in Section 3.4, and RHR and coolant circulation in Section 3.9) that ensure an RHR loop is in operation with sufficient flow to remove decay heat and prevent boron and thermal stratification. However, the specific flow rate requirement of CTS 3.1.1.3 (3,000 gpm) is not an assumption or initial condition, of any DBA analysis applicable for Modes 4, 5, and 6.

Comparison to Screening Criteria

Criteria 1 The RCS flow requirement of CTS 3.1.1.3 is not an instrument used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.

Criteria 2 The RCS flow requirement of CTS 3.1.1.3 is not a process variable or operating restriction required in Modes 1-3 to preserve or support any safety analysis assumptions due to other more restrictive RCS flow TS requirements applicable in these Modes. In Modes 4, 5, and 6, the RCS flow requirement of CTS 3.1.1.3 is not a specific assumption of any DBA described in the BVPS-1 and 2 UFSARs. Therefore, in Modes 4, 5, and 6, the RCS flow requirement of CTS 3.1.1.3 is not a process variable 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.

Criteria 3 The RCS flow requirement of CTS 3.1.1.3 is not used as part of a primary success path in the mitigation of a DBA or transient.

Criteria 4 As documented in the Individual Plant Examinations and the associated PRA Update Reports for both units, the RCS flow requirement of CTS 3.1.1.3 for Modes 4, 5, and 6 are not modeled in the BVPS-1 and 2 PRA since these currently only reflect at-power operating Modes. As such, the CTS 3.1.1.3 requirement was not identified as being a "constraint of prime importance in limiting the likelihood or severity of accident sequences that are commonly found to dominate risk". Therefore, the CTS 3.1.1.3 requirement has not been shown to be significant to public health and safety.

Conclusion

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the boron dilution TS and associated SR may be relocated from the TSs to licensee-controlled documents. The boron dilution TS will be relocated to the LRM, and changes to the LRM will be controlled pursuant to the provisions of 10 CFR 50.59.

E.2 Post-Accident Monitoring (PAM) System

LCO CTS 3.3.3.8, DOC R.1

The following BVPS-1 and 2 CTS PAM functions are proposed for relocation to the LRM:

- RCS Subcooling Margin Monitor
- Power-Operated Relief Valve (PORV) Limit Switch Position Indicator
- PORV Block Valve Limit Switch Position Indicator
- Safety Valve Position Indicator (BVPS-2), and
- Safety Valve Acoustical Detector Position Indicator (BVPS-1)

Discussion

The purpose of the PAM instrumentation included in the TSs is to function in a post-accident environment to provide the following:

- Primary indications necessary for operators to take manual actions (for which no automatic control is provided) to mitigate the consequences of an accident (i.e, RG 1.97, Type A variables), and

- Key indications (i.e, RG 1.97, Category 1 variables) that may be deemed risk significant because they are used to:

  - Determine whether a system important to safety is performing its intended function,

  - Determine the likelihood of a gross breach of a barrier to radioactive release, or

  - Determine the need to initiate action to protect the public and to estimate the magnitude of the threat.

10 CFR 50.36(c)(2)(ii), Criterion 1, applies to instrumentation used to detect RCS leakage and is satisfied by the instrumentation included in the RCS Leakage Detection Instrumentation TS. 10 CFR 50.36(c)(2)(ii), Criterion 2, applies to a process variable, design feature, or operating restriction that must be maintained within limits by a TS requirement to preserve an initial condition assumed in a DBA. Individual TSs for process variables such as boron concentration and operating limits such as Rod Insertion Limits address items that satisfy 10 CFR 50.36 (c)(2)(ii), Criterion 2. Based on the description of the PAM functions above, a required PAM TS indication may satisfy either Criterion 3 (primary indication to initiate an action) or Criterion 4 (risk) of 10 CFR 50.36(c)(2)(ii) when evaluating individual indications for retention in the PAM TS. Each BVPS-1 and 2 indication proposed for relocation is evaluated below.

<u>Comparison to Screening Criteria 3 and 4</u>

1. RCS Subcooling Margin Monitor

   The RCS subcooling indication provides information to the control room operators regarding the core cooling safety function and is used to satisfy a safety injection (SI) termination criteria. The inputs to the RCS subcooling margin monitor are the core exit thermocouples for RCS temperature and the wide range RCS pressure indication for RCS pressure. Since both of these indications are independently available in the control room and are also included in proposed ITS PAM TS 3.3.3, the RCS subcooling margin monitor only provides a verification of these other primary indications. The BVPS-2 UFSAR clearly identifies the RCS subcooling margin monitor as backup instrumentation (BVPS-2 UFSAR, Table 7.5-4). Based on the inclusion in the PAM ITS of the primary instruments for this indication (i.e., RCS temperature and pressure), the RCS subcooling margin monitor is not the primary indication for this variable or the key indication in terms of risk. The RCS pressure and temperature indications included in the proposed PAM ITS are classified as RG 1.97, Category 1 variables, the RCS subcooling margin monitor is not classified as a RG 1.97, Category 1 instrument. Therefore, the RCS subcooling margin monitor does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii), and should not be included in the PAM ITS.

2. PORV Limit Switch Position Indicator

   The PORV Limit Switch Position Indicators provide information to the control room operators related to the position of the pressurizer PORVs. This indication could be used to diagnose a high RCS pressure or a stuck open PORV (loss-of-cooling accident

(LOCA)) at lower RCS pressures. However, the PORV Limit Switch Position Indicator does not provide an indication for operator actions for which no automatic control is provided and it is not identified as a key indication from a risk perspective (i.e., it is not classified as an RG 1.97, Category 1 variable). The DBA analysis of an inadvertent opening of the PORV does not rely on operator diagnosis and closure of the PORV or block valve; the DBA analysis assumes that automatic SI actuation will provide adequate protection. Therefore, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM TS.

3. PORV Block Valve Limit Switch Position Indicator

   The PORV Block Valve Limit Switch Position Indicator provides information to the control room operators on the position of the pressurizer PORV block valves. It could be used to diagnose the availability of the pressurizer PORVs for use in depressurizing the RCS or to indicate the isolation of a stuck open PORV (LOCA) at lower RCS pressures. However, the PORV Block Valve Limit Switch Position Indicator does not provide an indication for operator actions for which no automatic control is provided and it is not identified as a key indication from a risk perspective (i.e., it is not classified as an RG 1.97, Category 1 variable). Therefore, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM ITS.

4. Safety Valve Position Indicator (BVPS-2)

   For BVPS-2, the Safety Valve Position Indicators are driven by magnetic reed switches which provide the control room operators information on the position of the pressurizer safety valves. It could be used to diagnose high RCS pressure or a stuck open safety valve (LOCA) at lower RCS pressures. However, the Position Indicator does not provide an indication for operator actions for which no automatic control is provided and it is not identified as a key indication from a risk perspective (i.e., it is not classified as a RG 1.97, Category 1 variable). Therefore, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in the PAM ITS.

5. Safety Valve Acoustical Detector Position Indicator (BVPS-1)

   For BVPS-1 the Acoustical Detector Position Indicators are use to provide the control room operators information on the position of the pressurizer safety valves. It could be used to diagnose high RCS pressure or a stuck open safety valve (LOCA) at lower RCS pressures. However, the Position Indicator does not provide an indication for operator actions for which no automatic control is provided and it is not identified as a key indication from a risk perspective (i.e., it is not classified as an RG 1.97, Category 1 variable). Therefore, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in the PAM ITS.

<u>Conclusion</u>

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the BVPS-1 and 2 CTS PAM functions and associated surveillances may be relocated from the TSs. The above requirements will be relocated to the LRM, and changes to the LRM will be controlled in accordance with 10 CFR 50.59.

E.3 Radiation Monitoring

LCO CTS 3.3.3.1, DOC R.2

The BVPS-1 and 2 CTS 3.3.3.1 contain requirements that address the Containment Area Radiation Monitor alarm and indication function for each unit. The BVPS-2 CTS 3.3.3.1 also contains requirements that address the alarm and indication functions of the BVPS-2 Main Steam Discharge Effluent Radiation Monitors. The BVPS-1 and 2 Containment Area Radiation Monitor alarm functions (not the indication function) and the BVPS-2 Main Steam Discharge Effluent Radiation Monitors (both the alarm and indication functions) including all associated LCO, Applicability, Action, and SRs are proposed to be relocated from the TS to the LRM and offside dose calculation manual (ODCM), respectively. It should be noted that the Containment Area Radiation Monitor indication function is retained in the proposed PAM ITS. Only the alarm function (and all associated LCO, Actions, etc.,) of the Containment Area Radiation Monitors is proposed for relocation to the LRM. The following discussion provides information regarding the BVPS-1 and 2 Containment Area Radiation Monitor alarms and the BVPS-2 Main Steam Discharge Radiation Monitors TSs which are proposed for relocation.

Discussion

10 CFR 50.36(c)(2)(ii), Criterion 1, applies to instrumentation used to detect RCS leakage and is satisfied by the instrumentation included in the RCS Leakage Detection Instrumentation TS. 10 CFR 50.36(c)(2)(ii), Criterion 2, applies to a process variable, design feature, or operating restriction that must be maintained within limits by a TS requirement to preserve an initial condition assumed in a DBA. Individual TSs for process variables such as boron concentration and operating limits such as Rod Insertion Limits address items that satisfy 10 CFR 50.36 (c)(2)(ii), Criterion 2. Based on the description of the PAM functions above, a required PAM TS indication may satisfy either Criterion 3 (primary indication to initiate an action) or Criterion 4 (risk) of 10 CFR 50.36(c)(2)(ii) when evaluating individual indications for retention in the PAM TSs.

Comparison to Screening Criteria 3 and 4

1. BVPS-1 and 2 Containment Area Radiation Alarms and BVPS-2 Main Steam Discharge Effluent Radiation Alarm

   The containment area radiation monitors and main steam discharge radiation monitors provide alarms and indications to alert plant personnel of high radiation conditions and to assist in evaluating and trending plant effluents. The TS Actions applicable if these monitors are inoperable require that the channel be restored to Operable status within 72 hours, or a preplanned alternate method of monitoring the parameter be initiated and the channel to be restored to Operable status within 30 days or that an explanation be provided in the next Annual Effluent Release Report why the channel was not restored to Operable status in a timely manner. The TS Actions do not impact or reference the operability of other systems or require a unit shutdown. Additionally, the alarm function of the BVPS-1 and 2 containment area radiation monitors and all functions of the BVPS-2 main steam discharge radiation monitors proposed for relocation do not:

- Provide an automatic initiation function assumed in the safety analysis for any DBA described in Chapter 14 of the BVPS-1 UFSAR, or in Chapter 15 of the BVPS-2 UFSAR.

- Provide indication or alarm functions relied on by operators to take manual actions that are assumed in the safety analyses for any DBA described in Chapter 14 of the BVPS-1 UFSAR, or in Chapter 15 of the BVPS-2 UFSAR.

- Provide the primary indication that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary, or
- Monitor variables which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Based on the above discussions, the alarm function of the BVPS-1 and 2 containment area radiation monitors and the BVPS-2 main steam discharge radiation monitors do not satisfy any of the 10 CFR 50.36(c)(2)(ii) criteria for retention in the TSs. Therefore, the proposed change to relocate the TS requirements for the above alarms is acceptable.

2. BVPS-2 Main Steam Discharge Effluent Radiation Indication

The BVPS-2 Main Steam Discharge Radiation indication may be used for the diagnosis of a steam generator tube rupture (SGTR) accident, which prompts an operator action for which no automatic actuation is provided. However, with the low fuel rod leakage history of current operating plants, secondary side radiation is not a reliable indicator of an SGTR accident. The history of diagnosis and response to an SGTR accident has typically been based on increased RCS inventory losses (e.g., decreasing pressurizer level and RCS pressure) and increasing water level in the affected steam generator (SG). These indications provide the most reliable diagnosis of an SGTR accident to prompt the appropriate operator actions and these indications are included in the PAM TS. In addition, the more sensitive radiation monitors (N16, SG blowdown, and condenser air ejector) are used for early detection of SG tube leakage. As such, the BVPS-2 Main Steam Discharge Radiation indication is not the primary or key indication relied on to diagnose or mitigate an SGTR accident.

<u>Conclusion</u>

Therefore, based on the discussions above, the BVPS-1 and 2 Containment Area Radiation alarms and the BVPS-2 Main Steam Discharge Radiation alarm and indication functions do not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in the PAM TS. The proposed changes to relocate the TS requirements for the above alarm and indication functions are acceptable.

E.4 <u>Containment Purge and Exhaust Isolation and BVPS-1 Radiation Monitoring</u>

LCO <u>BVPS-1 CTS 3/4.3.3.1, 3/4.9.9, DOC R.1</u>

- BVPS-1 CTS 3/4.9.9 Containment Purge and Exhaust Isolation System
- BVPS-1 CTS 3/4.3.3.1 Table 3.3-6, Radiation Monitoring, Instrument 1.b.i Purge &

Exhaust Isolation

- BVPS-1 CTS 3/4.3.3.1 Table 4.3-3, Instrument 1.b.i Purge & Exhaust Isolation

The above listed CTS LCOs contain the requirements for the automatic and manual isolation of the Containment Purge and Exhaust System for BVPS-1. The Purge and Exhaust Isolation radiation monitors specified in CTS 3/4.3.3.1, Table 3.3-6 (RM-1VS 104 A & B), function to automatically isolate the Containment Purge and Exhaust Valves on high radiation. The LCOs, Actions and SRs associated with the above equipment are proposed to be relocated to the LRM.

Discussion

The proposed ITS 3.3.6, “Containment Purge and Exhaust Isolation Instrumentation” does not contain requirements for the BVPS-1 automatic or manual Purge and Exhaust System isolation. ITS 3.3.6 is only applicable to BVPS-2. The CTS is revised to conform to the ITS. This changes the BVPS-1 CTS Purge and Exhaust System requirements for automatic isolation on high radiation and manual isolation by moving the CTS requirements to the BVPS- 1 LRM.

The current BVPS-1 and 2 design-basis fuel-handling accidents (FHAs) of record do not credit any automatic actuation to mitigate an FHA when moving non-recently irradiated fuel assemblies or fuel over assemblies that are not recently irradiated. Recently irradiated fuel is defined in the TS Bases as "…fuel that has occupied part of a critical reactor core within the previous 100 hours." Although BVPS-1 and 2 do not currently have safety analyses that support moving recently irradiated fuel assemblies, TS requirements have been retained to address the condition of moving recently irradiated fuel assemblies. These TS requirements are retained because decay time limits for moving irradiated fuel are in the LRM and the NRC staff has determined that fuel handling limits should be retained in the TSs should the licensee develop future capability to move recently irradiated fuel.

The retained TS requirements which are applicable when moving recently irradiated fuel or fuel assemblies over recently irradiated fuel assemblies in containment include Containment Purge and Exhaust System isolation for BVPS-2 and Containment Purge and Exhaust System effluent filtration for BVPS-1. Proposed ITS 3.9.3, "Containment Penetrations," contains these BVPS-1 and 2 unit-specific requirements for the Containment Purge and Exhaust System. The current FHA analysis for moving non-recently irradiated fuel and the CTS requirements for moving recently irradiated fuel were approved by the NRC in Amendment Nos. 241 for BVPS-1 and 121 for BVPS-2, issued August 30, 2001.

The relocation of the BVPS-1 requirements for Containment Purge and Exhaust System isolation on high radiation and manual isolation to the LRM is acceptable because BVPS-1 can not credit Containment Purge and Exhaust System isolation to mitigate the consequences of an FHA in containment. Instead, BVPS-1 must rely on filtration of the effluent by an operable train of the Supplemental Leakage Collection and Release System (SLCRS) when necessary to mitigate the consequences of an FHA inside containment. BVPS-1 must rely on filtration of the effluent instead of isolation because the Containment Purge and Exhaust System ductwork where the radiation monitors are located is not designed to withstand a seismic event. Although the radiation monitors provide an isolation signal to the purge and exhaust valves to close, no credit for the isolation signal may be taken in the BVPS-1 design-basis FHA. The

NRC staff concurred with this position in the NRC SE for BVPS-1 Amendment No. 23, issued December 12, 1979 (which added the TS requirement for the containment air to be exhausted through the SLCRS), as stated below;

> However, since the purge exhaust ductwork inside the containment containing the radiation monitors is non-seismic we have made dose calculations assuming the ductwork and monitors are damaged during a seismic event. In such an event we have assumed there is no containment isolation.

Therefore, based on the above SE applicable to the BVPS-1 Containment Purge and Exhaust System, any BVPS-1 safety analysis performed to support the movement of recently irradiated fuel would credit filtration instead of isolation. The proposed ITS reflect the BVPS-1 Containment Purge and Exhaust System specific design and licensing bases.

Comparison to Screening Criteria

Criteria 1 and 2 are not applicable to the Containment Purge and Exhaust Isolation System or the associated radiation monitors. Based on the design and licensing bases for the BVPS-1 Containment Purge and Exhaust System and associated radiation monitors discussed above, Criterion 3 is not met either. The proposed BVPS-1 ITS rely on filtration of the Containment Purge and Exhaust System effluent not system isolation as described in the NRC SER for BVPS-1 Amendment No. 23. Nor is the isolation function of the Containment Purge and Exhaust Isolation System and associated radiation monitors during refueling operation modeled in the BVPS PRA as documented in the Individual Plant Examinations (IPE) and the associated PRA Update Reports for both units. In addition, the actuation instrumentation for this isolation function is not significant to risk because it is not involved in any accident initiation sequences. As such, the Containment Purge and Exhaust Isolation System and associated radiation monitors were not identified as being a "constraint of prime importance in limiting the likelihood or severity of accident sequences that are commonly found to dominate risk." Since these CTS requirements have not been shown by risk to be significant to public health and safety, Criterion 4 is not met.

Conclusion

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the Containment Purge and Exhaust Isolation and BVPS-1 Radiation Monitoring and associated surveillance may be relocated from the TSs. The above specifications will be relocated to the LRM, and changes to the LRM will be controlled pursuant to provisions of 10 CFR 50.59.

E.5 Control Room Radiation Monitors

LCO CTS 3.3.3.1, DOC R.1

CTS 3.3.3.1, Radiation Monitoring, Function 1.c for control room area radiation monitors requires that two channels be OPERABLE. These radiation monitors are use to automatically initiate the Control Room Emergency Ventilation System (CREVS) in Modes 1, 2, 3, and 4. All Mode 1, 2, 3, and 4 Applicability requirements for CTS 3.3.3.1 including the LCO, Actions and SRs are proposed to be relocated to the LRM. However, the LCO requirements for these radiation monitors will be retained in ITS 3.3.7 for fuel movement involving recently irradiated fuel.

Discussion

The applicable safety analyses for all DBAs considered in MODES 1-4 (except LOCA) that require control room isolation and pressurization allow sufficient time for manual initiation of the emergency pressurization mode of operation of control room ventilation (i.e., control room ventilation isolation, filtered makeup, and pressurization). The safety analyses assume a 30-minute delay for control room isolation and pressurization to allow for manual action. The LOCA analysis assumes the control room ventilation system is automatically isolated on a Containment Isolation Phase B (CIB) signal and subsequently pressurized with filtered air by manual initiation of a CREVS fan and alignment to a filtered flow path. Although the CIB signal will automatically start a CREVS fan and filtered flow path, a 30-minute delay to allow for manual initiation of a CREVS fan and filtered flow path is specifically assumed in all analyses. The 30-minute allowance is required to permit the use of a BVPS-1 CREVS fan and filtration flow path which require manual operator action to place in service. The proposed BVPS ITS 3.3.7 continues to assure the assumptions of the safety analysis are met by specifying requirements for the manual system level CREVS initiation switches for each unit in Modes 1 through 4. The requirements for the CIB signal continue to be specified in ITS 3.3.2, "ESFAS Instrumentation" consistent with the ISTS.

The current safety analyses do not assume the control room area radiation monitors provide a CREVS actuation signal for any DBA. However, requirements for the radiation monitors to be OPERABLE are retained in case the monitors are required to support the assumptions of an FHA analysis involving the movement of recently irradiated fuel or the movement of fuel over recently irradiated fuel. The retention of requirements for fuel movement involving recently irradiated fuel is consistent with the guidance (STS) provided in NUREG -1431.

The BVPS-1 and 2 specific safety analyses assumptions for manual actuation of the CREVS results in a different bases for these requirements than described in the ISTS. Due to the BVPS-1 and 2 safety analysis reliance on manual operation, the BVPS-1 and 2 radiation monitors do not serve as backup for a required automatic initiation for all DBAs. The BVPS-1 and 2 safety analysis reliance on manual actuation reduces the importance of the automatic function provided by the BVPS-1 and 2 control room radiation monitors. For example, the ISTS Actions for inoperable CREVS instrumentation in Modes 1-4 require CREVS equipment to be run continuously and could result in a unit shutdown. In addition, the continuous operation of the filter system will eventually expend the filter media and result in additional equipment unavailability. The ISTS Actions are more appropriate for plants that rely on automatic CREVS Actuation to mitigate all DBAs. Considering the BVPS-1 and 2 specific safety analyses reliance on manual CREVS operation, the additional equipment wear and potential system unavailability, as well as the potential for a unit shutdown introduced by the ISTS Actions are overly conservative for inoperable radiation monitor(s). Therefore, the licensee is proposing to relocate the Mode 1 through 4 CTS requirements for the control room area radiation monitors to the BVPS-1 and 2 LRM as appropriate. The control room area radiation monitors will continue to be maintained operable within a more appropriate licensee controlled document consistent with the NRC recommendations in the “Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors,” July 22, 1993 (58 FR 39132).

Comparison to Screening Criteria

Criteria 1 and 2 are not applicable to the control room area radiation monitors. Based on the BVPS-1 and 2 safety analysis reliance on manual operation of the CREVS, Criterion 3 is not met either. Nor is the CREVS actuation function of the control room area radiation monitors modeled in the BVPS PRA as documented in the IPE and the associated PRA Update Reports for both units. In addition, the radiation monitoring actuation instrumentation for CREVS is not significant to risk because it is not involved in any accident initiation sequences. As such, the control room area radiation monitors were not identified as being a "constraint of prime importance in limiting the likelihood or severity of accident sequences that are commonly found to dominate risk." As such, these CTS requirements have not been shown by risk to be significant to public health and safety. Therefore, Criterion 4 is not met.

Conclusion

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the Control Room Area Radiation Monitors and associated surveillance may be relocated from the TSs. The above specifications will be relocated to the LRM, and changes to the LRM will be controlled pursuant to the provisions of 10 CFR 50.59.

E.6 Supplemental Leak Collection and Release System

LCO CTS 3.7.8.1, DOC R.1

CTS 3/4.7.8, "Supplemental Leak Collection and Release System (SLCRS)," requires that two SLCRS exhaust air filter trains be OPERABLE. CTS 3/4.7.8 is applicable in MODES 1, 2, 3, and 4 and contains SRs that verify the Operability of the SLCRS exhaust air filter train. The requirements of CTS 3.7.8.1 for SLCRS in Modes 1-4 are proposed to be relocated to the LRM. However, SLCRS operability requirements are retained in the ITS to address a potential FHA involving recently irradiated fuel assembles.

Discussion

The bases for including the requirements for SLCRS in the CTS was the need to filter airborne radioactivity, prior to release to the environment, from the areas of active Engineered Safeguards Features (ESF) components outside of the reactor containment building during the recirculation phase of a DBA LOCA. This ensures ESF leakage following the postulated DBA LOCA will not cause the resulting dose to exceed 10 CFR 50.67 limits. The CTS SLCRS surveillance and acceptance criteria verify the SLCRS filtration capability to assure it is adequate to mitigate the limiting dose consequences of a DBA LOCA.

In addition, SLCRS performs the secondary functions of heat removal from areas containing active ESF components and serves to minimize the accumulation of radiation in these areas to help support equipment qualification (EQ) requirements.

Amendment Nos. 257 (BVPS-1) and 139 (BVPS-2) issued on September 10, 2003, approved changes related to "Selective Implementation of Alternative Source Term and Control Room Habitability." In this amendment, the alternative source term applied to the DBA LOCA analyses was approved. The result of this revised LOCA analysis was that the filtration capability of SLCRS was no longer credited to maintain the resulting dose to within the limits of 10 CFR 50.67. The BVPS-1 and 2 extended power uprate licensing report (EPULR) submitted with LAR 302 (BVPS-1) and 173 (BVPS-2), approved by NRC Amendment Nos. 275 and 156 on July 19, 2006, confirms that the revised LOCA analyses no longer credit the filtration capability of the SLCRS to maintain dose to within the limits of 10 CFR 50.67. As such, the bases for the CTS requirement that two SLCRS exhaust air filter trains be maintained operable in MODES 1, 2, 3, and 4 is no longer supported by the post Alternative Source Term/EPULR LOCA safety analyses.

Although SLCRS is no longer credited in the safety analyses for MODES 1, 2, 3, and 4, SLCRS operability requirements are retained in the ITS to address a potential FHA involving recently irradiated fuel assemblies. The requirements necessary to address this FHA scenario have been proposed in ITS 3.7.12.

Based on the revised DBA LOCA safety analyses no longer crediting the SLCRS to maintain dose within the 10 CFR 50.67 limits, the CTS requirements for SLCRS in MODES 1, 2, 3, and 4 are proposed for relocation to the LRM.

Comparison to Screening 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 SLCRS is not installed instrumentation that would be used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. The SLCRS performs a ventilation/filtration function and does not include instrumentation that meets Criterion 1.

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.

The SLCRS is not a process variable, design feature, or operating restriction required in Modes 1, 2, 3, or 4 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 SLCRS is a system with components that function to ventilate and filter the exhaust from ESF areas outside of containment.

Criterion 3. An SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The SLCRS is not an SSC that is part of the primary success path (of a safety sequence analysis) 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 current safety analyses no longer credit the SLCRS to limit the radiological consequences of a DBA. The SLCRS functions regarding ESF component area heat removal and EQ concerns are not part of the primary safety analysis success path for DBA mitigation. The capability of SLCRS to perform these secondary functions may be adequately assured by controls outside of the TSs (i.e., the LRM as described below).

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

The SLCRS is not an SSC which operating experience or PRA has shown to be significant to public health and safety.

Conclusion

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the SLCRS and associated surveillance for Modes 1-4 may be relocated from the TSs. The above specifications will be relocated to the LRM, and changes to the LRM will be controlled pursuant to the provisions of 10 CFR 50.59.

Summary

The NRC staff finds that relocation of the requirements specified in Table R to a licensee-controlled document is acceptable in that the LCOs and associated requirements were found not to fall within the scope of 10 CFR 50.36(c)(2)(ii). The relocated specifications are not needed to obviate the possibility that an abnormal situation or event will give rise to an immediate threat to the public health and safety. In addition, the NRC staff has concluded that changes to licensee-controlled documents will be adequately controlled as discussed in Section 4.F.

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

In the ITS conversion, the licensee proposes to relocate specifications, requirements, and detailed information from the CTSs to licensee-controlled documents. This is discussed in Sections 4.D and 4.E of this SE. The facility and procedures described in the UFSARs, TS Bases, and the LRM can be revised only in accordance with the provisions of 10 CFR 50.59, which ensure that records are maintained, and that appropriate controls are established for requirements removed from the CTSs. Other licensee-controlled documents contain provisions for making changes consistent with applicable regulatory requirements. For example, the ODCM can be changed only in accordance with ITS 5.5.1, and the administrative instructions that implement the QAPD can be changed only in accordance with 10 CFR 50.54(a) and 10 CFR Part 50, Appendix B.

The NRC staff finds that the BVPS-1 and 2 ITS provide clearer, more readily understandable requirements to ensure safer operation of the units. Further, based on the considerations discussed above, the NRC staff finds that the BVPS-1 and 2 ITS satisfies the Commission's Final Policy Statement and 10 CFR 50.36. Based on these findings, the NRC staff has determined and concludes that the proposed ITS for BVPS-1 and 2, as documented in the licensee's application and supplemental letters, is acceptable.

G. Evaluation of Other TS Changes (Beyond-Scope Issues) Included in the Application for Conversion to ITS

This section evaluates other TS changes identified in Volume 1 of the licensee's ITS conversion application. These changes include items that deviate from both the CTSs and the ISTSs. These changes are termed BSIs. They were either identified by the licensee in its ITS application, or by the NRC staff during the course of its review. The 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 March 22, 2006 (71 FR 14554).

This section of the SE is divided into BSIs identified by the licensee (Section G.1) and those identified by the NRC staff (Section G.2).

G.1 BSI Changes Identified by the Licensee

G.1.1 BSIs-1 and 2, "Changes to BVPS-1 Analog Rod Position Indication System"

BSIs-1 and 2, propose changes to the BVPS-1 analog Rod Position Indication (RPI) system. BVPS-2 uses a digital RPI system and the proposed change does not apply to BVPS-2. The proposed changes would modify the CTS 3.1.3.2 notes to apply the 1-hour thermal soak time to all power levels instead of only to power levels below 50 percent, and to apply the exception to the ± 12 step-requirement during rod insertion and withdrawal (provided by the Mode 2 footnote) to any time "during rod motion." The CTS 3.1.3.1 notes would be moved directly to the ITS 3.1.4 LCO. The LCO for CTS 3.1.3.1 (ITS 3.1.4) refers to CTS 3.1.3.2 (ITS 3.1.7.1) for determining rod position. CTS 3.1.3.2 contains notes that provide allowances for a ±12-step accuracy limit when using the analog RPI system to determine rod position.

BSIs-1 and 2 identified the following deviations from the Westinghouse ISTS in NUREG-1431, Revision 2:

(1) Relocation of the affected BVPS-1 CTS 3.1.3.2 notes to ITS 3.1.4 LCO and SRs where the ±12-step accuracy limit is required;

(2) Modifications of the BVPS-1 notes that provide allowances for the analog RPI System. The modifications for the affected notes contain:

- application of the 1-hour thermal soak allowance to all power levels instead of only to power levels below 50 percent, and
- application of the exception to the ±12-step requirement to any time during rod motion not just during rod withdrawal or insertion as limited by CTS 3.1.3.2 Mode 2 footnote.

G.1.1.1 Regulatory Evaluation

Unit 1 is designed in accordance with AEC General Design Criteria (GDC) Criterion 6 published July of 1967. This criterion specified that: "The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine-generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power."

10 CFR 50.36 specifies the Commission's regulatory requirements related to the content of TSs. Specifically, 10 CFR 50.36(c)(2)(ii)(B) requires that TS LCOs be established for process variables, design features, and operating restriction for which a value is assumed as initial condition of a design basis accident in the licensee's safety analyses.

NUREG-1431 was developed based on the criteria in 10 CFR 50.36(c)(2)(ii). Shutdown and control rod operability and alignment are directly related to power distribution and shutdown margin that are initial conditions assumed in the safety analyses. In accordance with Criterion 3 of 10 CFR 50.36(c)(2)(ii)(B) discussed above, a TS LCO is required for the shutdown and control rod operability and alignment.

Since BVPS-1 is a Westinghouse pressurized-water reactor (PWR) and BSIs-1 and 2 are TS changes deviating from NUREG-1431, the NRC staff's review of the proposed TS changes will be based on the compliance with AEC GDC 6 requirements and consistency with similar TSs previously approved by the NRC for Westinghouse plants.

G.1.1.2 <u>Technical Evaluation</u>

The NRC staff reviewed BSIs-1 and 2 based on the proposed TS changes, associated JFDs, and DOCs as described in the licensee's application. The staff's evaluation is provided below.

<u>ITS 3.1.4 - Notes Added to LCO 3.1.4 and SR 3.1.4.1</u>

Westinghouse ISTS LCO 3.1.4 states that "[a]ll shutdown and control rods shall be OPERABLE and individual indicated rod positions shall be within 12 steps of their group step counter demand position." A note was added to this LCO in the proposed ITS 3.1.4. Specifically, the added note states that "...verification of rod operability and that the individual indicated rod positions are within the 12 step limit is not required during rod motion and for the first hour following rod motion." A similar note allowing exception to the LCO requirements was added to the associated SR 3.1.4.1. The added notes are only applicable to BVPS-1.

The licensee proposed that the added notes be relocated from CTS 3.1.3.2 to ITS 3.1.4 with modifications. In CTS 3.1.3.2 for BVPS-1, the LCO requirements for the individual RPI system accuracy and the corresponding SR are modified by a note (Note 1) that states that "[d]uring the first hour following rod motion, the group demand counter is the primary indicator of precise rod position information, with analog channels displaying general movement information. For power levels below 50 percent, a 1-hour thermal soak time is allowed before the analog channels are required to perform within the specified accuracy." The Mode 2 applicability requirements for CTS 3.1.3.2 include another note (designated # ) that specifies an exception to meeting the $\pm$12-step requirement during reactor startup and shutdown operations when the rods are being withdrawn or inserted.

When relocating the CTS notes to ITS 3.1.4, three changes were proposed to the CTS notes as follows.

(1) Relocate the notes from the RPI LCO (CTS 3.1.3.2, "Position Indication System - Operating) requirements to the rod alignment LCO (ITS 3.1.4, "Rod Group Alignment Limits.")

The NRC staff finds that the relocation of TS notes is editorial in nature and does not change the TS requirements. Therefore, it is acceptable.

(2) Remove inconsistencies between the notes in CTS 3.1.3.2 LCO (Note 1) and the footnotes in APPLICABILITY requirements for Mode 2 Operation.

CTS 3.1.3.2 APPLICABILITY requirements indicated that for core physics testing in Mode 2, the existing Mode 2 footnote allows an exception to meeting the ±12-step requirement up to 1-hour during reactor startup and shutdown operations when the rods are being withdrawn or inserted. The licensee indicated that the allowance provided by the Mode 2 footnote is not consistent with the more general requirements in CTS 3.1.3.2 Note 1. CTS LCO 3.1.3.2 lists the specific operability requirements for the RPIs and includes Note 1 that provides an exception for the 1-hour allowance following any rod motion below 50 percent power levels (this includes Mode 2 startup and shutdown operations). Therefore, the licensee indicated, and the NRC staff agrees that Note 1 in CTS 3.1.3.2 provides a broader exception that is applicable to both Modes 1 and 2. When relocating the Note 1 exception to ITS 3.1.4 LCO, the licensee removed the Mode 2 footnote that is bounded by Note 1. The staff has determined that the proposed TS modifications do not reduce the CTS requirements and thus, concludes that the proposed TS modifications are acceptable.

(3) Increase the applicable power range from below 50 percent to all power levels for the 1-hour thermal soak allowance.

CTS 3.1.3.2 Note 1 provides for up to a 1-hour allowance after control rod motion to verify satisfaction of the alignment limit for individual rod positions. The 1-hour allowance is based on the time necessary for the control rod shaft to reach thermal equilibrium under which individual rod alignment limit can be accurately verified using the analog position indicators. The licensee indicated that it initially requested the 1-hour thermal soak time for power levels below 50 percent, as previously approved by the NRC and implemented in the CTS. The licensee originally considered that rod motion above 50 percent was too limited and not expected to induce significant thermal transients in the analog rod position indication instrumentation. However, the licensee's operating experience showed that rod movement and subsequent indication drift at power levels above 50 percent did occur (e.g., the monthly freedom of movement tests performed on the rods). Therefore, the licensee proposed to expand the allowances for rod motion and thermal soak time from the applicable range of power levels below 50 percent to all power levels.

Based on the results of its review, the NRC staff noted that the change only provides a short delay for analog indicators to stabilize after rod motion before the requirement of the LCO and surveillance are applied. The LCO and SRs are not revised. During the requested time, RPI continues to be available from both the analog system and demand counter.

The NRC staff also noted that the NRC had previously approved similar TS changes for other Westinghouse plants with the analog RPI. In the TSs of Point Beach Nuclear Plant (PBNP), a Westinghouse plant, the allowances for rod motion and 1-hour thermal soak time applicable up to all power levels were included in the rod alignment LCO. In the PBNP TSs, as in the BVPS-1 and 2 ITS, the rod alignment LCO contains the requirements for rod operability and requires that the indicated individual rod positions are maintained within an alignment limit of the demand position indicators. Failure to meet the rod operability or alignment limit results in the application of a 1-hour Action to restore operability or alignment followed by a plant shutdown. As indicated in the SE for the PBNP license amendment, issued May 8, 2001, the NRC's approval of allowances for rod motion and 1-hour thermal soak time is based on the following reasons: (1) during the 1-hour thermal soak following rod motion the likelihood of an accident to occur during this period is small; and (2) in general, the rods will not be misaligned, only indicated to be misaligned. The staff determines that because of the similarity in the Westinghouse plant designs and TS changes, the bases for approval of the thermal soak allowance for the PBNP are applicable to BVPS-1 and 2.

In addition, the licensee indicated that to further assure the reliability and accuracy of the group demand counters, a CTS surveillance (SR 4.1.3.2.1.b) beyond the Westinghouse ISTS requirements is included in the proposed BVPS-1 and 2 ITS (SR 3.1.7.1.1). The additional surveillance requires verification of the accuracy of the bench board demand counters by comparing them to the logic solid state indicators in the logic cabinet. The retention of this CTS surveillance will provide additional assurance of the demand counter accuracy and reliability for primary RPI during the first hour following rod movement.

Therefore, based on the evaluation discussed for item (3) above, the NRC staff concludes that it is acceptable to extend the allowances for rod motion and 1-hour thermal soak time from power levels below 50 percent to all power levels.

ITS 3.1.7.1 Actions - Add a new Actions Condition A

The BVPS-1 ITS 3.1.7.1 Actions are modified by the addition of a new Actions Condition A. The added Action A.1 requires that if the RPI system indicates one or more potentially misaligned rods, the affected rod position shall be verified within 15 minutes by measuring the RPI primary voltage. Further, the added Action A.2.1 requires that if the rod position measurement determines that any rod is misaligned, ITS 3.1.4 shall be applied within 15 minutes. Alternatively, the added Action A.2.2 requires that the affected RPI shall be declared inoperable and the applicable Conditions and required Actions of ITS 3.1.7.1 shall be entered within 15 minutes.

The NRC staff finds that: (1) the added Actions Condition A (Actions A.1 and A.2.1) discussed above, is consistent with the CTS 3.1.3.2 Action a; (2) the CTS Action was modified by adding Action A.2.2 to give more complete directions for indication of potentially misaligned rods. Therefore, the staff has determined that the added Actions Condition A does not reduce the CTS requirements and thus, concludes that the addition of the new Actions Condition A is acceptable.

Furthermore, CTS 3.1.3.2 Action "A" was modified by adding a note stating that entry into Condition "A" is "not required for misalignment indications during rod motion and for up to 1-hour following rod motion." This new note, that provides an allowance for rod motion and thermal soak time, was also incorporated into the corresponding Action Condition A in ITS 3.1.7.1. The note providing the allowance for rod motion and thermal soak time was also added to SR 3.1.7.1.2 that requires verification that each RPI agree "within 12 steps of group demand position for the full indicated range of rod travel" at least "once prior to criticality after each removal of the reactor head." The added SR 3.1.7.1.2 note is based on BVPS-1 CTS 4.1.3.2.2.b, Note 1 (that is the same note as CTS 3.1.3.2 Note 1).

As the results of its review, the NRC staff finds that the added note provides the same allowances as the note added to ITS LCO 3.1.4. The addition of the note assures consistent application of the allowances to both ITS LCO 3.1.4, "Rod Group Alignment Limits" and ITS 3.1.7.1, "Rod Position Indication." Therefore, the staff has determined that the bases discussed above for approving the allowances for ITS LCO 3.1.4, Note 1, are applicable to the added ITS 3.1.7.1 note. Thus, the staff concludes that the added note is acceptable.

G.1.1.3 Conclusion

The NRC staff has reviewed BSIs-1 and 2 dealing with changes in ITSs 3.1.4, "Rod Group Alignment Limits," and 3.1.7.1, "Unit 1 Rod Position Indication." Based on the evaluation discussed above, the staff has determined that the TS changes do not reduce CTS requirements and are consistent with the similar TSs previously approved by the NRC. Therefore, the NRC staff finds that the proposed TS changes identified in BSIs-1 and 2 are acceptable.

G.1.2 BSIs-3 and 4

BSI-3 proposes changes to the ISTS time limit and power level specified in the note modifying SR 3.3.1.3. The proposed time limit would change from 1 to 7 days and the proposed power level would change from ≥15 percent rated thermal power (RTP) to ≥50 percent RTP (ITS 3.3.1 and SR 3.3.1.3 note, DOC M.12, JFDs 4 and 6). BSI-4 proposes changes to ISTS SR 3.3.1.6 (ITS SR 3.3.1.9) to change the time allowed to perform the surveillance from 24 hours after RTP is ≥50 percent, to 7 days. Additionally, BSI-4 proposes to change the requirement to perform SR 3.3.1.9 every 92 effective full-power days (EFPD) thereafter, to perform the surveillance "once per fuel cycle" (ITS 3.3.1, SR 3.3.1.9 note, DOC M.19, JFD 7).

G.1.2 1, BSI-3, ISTS SR 3.3.1.3, Note 2

The licensee proposes to revise ISTS SR 3.3.1.3, Note 2, to specify that SR 3.3.1.3 be performed 7 days after the reactor reaches or exceeds 50 percent of RTP. This requirement does not affect the regular surveillance interval of 31 EFPD.

- Surveillance Time After Ascent to Power

  ISTS SR 3.3.1.3, Note 2, provides a time limit for performing surveillance after a specified power level is exceeded. The corresponding CTS SR in Note 3 of Table 4.3-1 has a 31-EFPD surveillance frequency but does not require surveillance after refueling or ascent to power. The ISTS surveillance frequency in SR 3.3.1.3 Note 2 is revised from a bracketed 24 hours to 7 days.

  SR 3.3.1.3 is performed every 31 EFPD to ensure that the excore channel ΔI input to the axial flux difference (AFD) and the f(ΔI) input to the Overtemperature-ΔT (OTΔT) trip function are maintained and calibrated in accordance with the incore detector measurements.

- Power Level to Conduct SR 3.3.1.3

  ISTS SR 3.3.1.3, Note 2, also specifies a power level of greater than or equal to 15 percent of RTP to conduct the surveillance. The proposed change specifies conducting the surveillance at 50 percent of RTP. The change to increase the specified surveillance power level to 50 percent RTP was previous approved in License Amendment Nos. 274 (BVPS-1) and 155 (BVPS-2) issued February 27, 2006.

  The 7 days requested to perform the calibration is a reasonable time interval because it is the last step in a series of actions that begin with a full-power core flux-map or a series of maps at partial power. The proposed time interval is more conservative than the CTS time interval which specifies conducting a routine surveillance every 31 EFPD. The proposed interval conforms with the 7-day SR 3.2.3.1 for the AFD that specifies a calibrated excore channel ΔI. Finally, this is consistent with previous NRC staff precedent, i.e., the interval specified in and approved by the NRC staff during the Joseph M. Farley Nuclear Plant, Units 1 and 2, ITS conversion. For the above reasons, the staff finds that the proposed 7-day surveillance interval is acceptable.

  The proposed change to conduct the surveillance at 50 percent of RTP is acceptable because: (1) measurements are more accurate at higher power levels; and (2) is consistent with ITS SR 3.3.1.9 which also requires an incore/excore normalization to be performed at power levels greater than or equal to 50 percent of RTP during plant startup.

- Summary

  The proposed change to the time limit for performing SR 3.3.1.3 and when reactor power is greater than or equal to 50 percent of RTP, is consistent with the AFD and ΔI adjustments that require incore/excore normalization. In addition, this change is also consistent with the ITS conversion previously approved by the NRC staff for the Joseph M. Farley Nuclear Plant, Units 1 and 2. Therefore, the NRC staff finds the proposed changes to ISTS SR 3.3.1.3 acceptable.

<u>G.1.2.2, BSI-4, ITS SR 3.3.1.9, Note</u>

The licensee proposed to add a new SR for the nuclear instrumentation system (NIS) that requires the excore NIS channels be calibrated to agree with the incore detector measurements once per refueling cycle, and that the surveillance be performed within 7 days after exceeding 50 percent of RTP.

The ISTS SR 3.3.1.6 is required to be performed within [24] (bracketed) hours after power exceeds 50 percent of RTP and every [92] (bracketed) EFPD, thereafter. The BVPS-1 and 2 CTSs do not contain an SR that corresponds to the ISTS SR 3.3.1.6. The proposed change differs from CTS practice at BVPS-1 and 2 and differs from the bracketed values in ISTS.

- Surveillance Frequency

  As stated above, the ISTS requirement is to perform the surveillance within 24 hours after the RTP exceeds 50 percent. The licensee proposes to perform the surveillance once per fuel cycle (18 months). The CTSs (Note 3 in Table 4.3-1) requires that the excore nuclear instrumentation system (NIS) be recalibrated if the absolute difference between the incore and excore readings differ by more than 3 percent. The monthly verification and normalization of the excore channels (required by the CTSs) address the flux redistribution every 31 EFPD. This feature of the CTSs is retained in ITS SR 3.3.1.3.

The ISTS requirement to perform the calibration every 92 EFPDs does not add needed information to the NIS.

The added surveillance in ITS SR 3.3.1.6 to be performed once every refueling cycle at the beginning of core service to setup the core NIS, is sufficient because the ITS SR 3.3.1.3 will maintain the incore and excore NIS agreement with the 31 EFPD normalization and verification adjustment. The beginning of cycle incore/excore calibration includes the AFD input to the OTΔT function to match the cycle-specific core power distributions.

Consistent with the ISTS format (numbering the longer surveillance requirements in ascending time order), SR 3.3.1.6 is renumbered to SR 3.3.1.9. The subsequent SRs 3.3.1.7/8/9 are renumbered accordingly. This is an editorial change.

The proposed change is acceptable to the NRC staff because it continues to provide assurance that the incore/excore NIS will remain in agreement within the existing 3-percent band.

- Time to Perform the Surveillance

  The ISTS recommends that the surveillance be completed within 24 hours after exceeding 50 percent of RTP. The licensee proposed that the surveillance be completed up to 7 days after exceeding 50 percent of RTP. The proposed time interval is reasonable because at the beginning of cycle the surveillance is performed at several power levels and followed by NIS calibration. In addition, this 7 day limit has to conform with the AFD surveillance that requires calibrated excore NISs and plant chemistry work at the beginning of the cycle. This change is acceptable to the NRC staff because it sets the foundation for continued assurance that the surveillance is performed on a timely manner and is consistent with concurrent plant activities. This change is consistent with that requested in the review of BSI-3 above, and is consistent with the changes requested during the review and approval of the Joseph M. Farley Nuclear Plant, Units 1 and 2, ITS conversion.

- Summary

  The ISTS SR 3.3.1.6 surveillance is required to be performed within 24 hours after power exceeds 50 percent of RTP and every 92 EFPD, thereafter. The licensee proposes to perform the surveillance within 7 days after power exceeds 50 percent of RTP and once per fuel cycle. The frequency was justified because of the monthly core measurements and associated adjustments to the NIS required by CTS. The time after exceeding 50 percent RTP is justified in terms of the associated core requirements. Both are acceptable because they assure continued agreement of incore/excore NIS.

G.1.3 BSIs-5, 6, 7, 8, 9, 10, and 11

BSI-5 proposes a change to ITS SR 3.3.4.2 frequency for verifying the operability of the Remote shutdown System control and transfer switches from 18 months to 36 months. CTS 3.3.3.5 currently does not have operability or SRs for these control and transfer switches (ITS 3.3.4, SR 3.3.4.2, DOC M.4, JFD 1).

BSI-6 proposes a change to the ISTS note that modifies the precision heat balance SR to require the surveillance to be performed within 30 days of reaching the specified power level in lieu of 24 hours of reaching the specified power level (CTS 4.2.5.2 and its note 2 do not contain a specified time limit in which to perform the heat balance) (ITS 3.4.1, SR 3.4.1.4 note, DOC M.1, JFD 1).

BSIs 7-11 propose revising the ISTS note for RCP and RHR pump standby pump breaker alignment and power availability every 7 days (and within 24 hours after the pump is not in operation) to remove the requirement for performing the surveillance within 24 hours after the pump is not in operation and considering the SR to be met for a pump just removed from operation and to clarify that the starting time for the 7-day SR begins “when the pump is removed from operation” instead of when the pump “is not in operation.” The CTS SRs do not have a note containing the 24-hour requirement for the RCPs and RHR pumps (ITS SR 3.4.5.3, DOC L.3, JFD 2, SR 3.4.6.3, DOC L.4, JFD 2, SR 3.4.7.3, DOC L.5, JFD 4, SR 3.4.8.2, DOC L.4, JFD 3, and SR 3.9.5.2, DOC M.1, JFD 2).

G.1.3.1 BSI-5, Remote Shutdown System Instrumentation

The licensee proposed to change the control and transfer switches surveillance frequency from 18 months to 36 months. BVPS-1 and 2 CTS SR 3.3.3.5 states that, “The remote shutdown monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE with readouts displayed external to the control room.”

Westinghouse ISTS SR 3.3.4.2 states that, “Verify each required control circuit and transfer switch is capable of performing the intended function.” The surveillance frequency is [18] months. The number in the bracket is a plant-specific number.

BVPS-1 and 2 CTS SR for the remote shutdown system do not include any operability or SR for control and transfer switches. The CTS requirements only address monitoring instrumentation. Currently, BVPS-1 and 2 CTSs verify the operability of these switches every 54 months using existing plant procedures which are outside the scope of the TSs. Based on the operating experience, the control and transfer switches verified by this surveillance are designed to be reliable and are not subject to excessive wear from daily use or being in a harsh environment. In addition, the inclusion of this instrumentation in the TS provides additional assurance that adequate post-maintenance testing will be performed to assure operability after modifications or design changes. Thus, the proposed surveillance frequency is different than the ISTS bracketed surveillance frequency, it is more conservative than the BVPS-1 and 2 CTS SR for this instrumentation and will provide additional assurance that the required control and transfer switches are maintained operable. Based on the plant operating experience and adequate post-maintenance testing procedures, the NRC staff finds that although the proposed SR frequency change is a deviation from the ISTS, it is an improvement as compared to the CTSs.

Therefore, the NRC staff concludes that the proposed surveillance frequency for control and transfer switches every 36 months is acceptable.

G.1.3.2 BSI-6, Precision Heat Balance

The licensee proposed to change the precision heat balance SR frequency from 24 hours to 7 days after RTP reaches greater than or equal to [95] percent.

BVPS-1 and 2 CTS SR 4.2.5.2 states that, "The Reactor Coolant System total flow rate shall be determined to be within its limits by measurement at least once per 18 months." The associated Note 2 includes that the provisions of Specification 4.0.4 are not applicable for RCS total flow rate to allow a calorimetric flow measurement and the calibration of the RCS total flow rate indicators.

Westinghouse ISTS SR 3.4.1.4 Note states that, "Not required to be performed until 24 hours after greater than or equal to [90]% RTP."

BVPS-1 and 2 CTSs do not contain a specific time limit in which to perform the heat balance surveillance but include an exception to Specification 4.0.4 (Note 2) that allows the performance to be delayed longer than 24 hours. Initially, the licensee proposed to perform precision heat balance SR within 30 days after reaching 90 percent RTP versus the 24-hour frequency stated in the ISTS. In response to an NRC staff RAI, the licensee revised its proposal from within 30 days of reaching 90 percent RTP to within 7 days after reaching greater than or equal to 95 percent RTP and provided operating data to support its precision heat balance SR.

In its response to the NRC staff RAI, the licensee stated that it performs the BVPS-1 and 2 flow measurement test as close to 100 percent power as possible in order to minimize flow measurement uncertainty. It can take up to 3 days to reach 100 percent power from 90 percent RTP, depending upon the plant conditions during startup. BVPS-1 and 2 operation allows another 3 days to achieve stable power operation after reaching a power as close to 100 percent as possible. Feedwater flow requires time to stabilize by performing feedwater heater level adjustments and SG chemistry optimization. The performance of the testing including the analysis of data takes another day. Thus, it takes approximately 7 days to complete the surveillance after reaching 90 percent power. This permits little to no contingency for any potential delays during the plant startup. Therefore, the licensee revised the Note to require performance of the surveillance to be within 7 days of reaching 95 percent RTP to allow time for contingencies. The NRC staff finds that this change will provide a reasonable period of additional time for any potential delays during plant startup, and is therefore, acceptable.

In addition, ISTS SR 3.3.1.2 requires that the power range indication be verified daily after reaching 15 percent RTP. This ISTS SR compares the power range indication with a secondary side calorimetric heat balance calculation, and requires that the power range channels be adjusted if the secondary side calorimetric heat balance calculations results exceed the power range channel output by more than +2 percent. RCS flow is not used in this SR for determining the correct power range indication. Thus, the power is continuously verified daily by ISTS SR 3.3.1.2 after reaching 15 percent power. This SR provides assurance that the plant will be maintained within its analyzed conditions during this 7-day period. Furthermore, this proposed change is consistent with other Westinghouse plant LARs (Farley Units 1/2, Braidwood Units 1/2, R. E. Ginna Nuclear Power Plant and, Wolf Creek Unit 1) that were previously approved by the NRC staff. Therefore, for the above reasons, the NRC staff concludes that this proposal is acceptable.

G.1.3.3 BSIs-7, 8, 9, 10, and 11-Verification of Standby Pump (RCP or RHR) is not in operation-correct breaker alignments and power availability.

The licensee proposed to deviate from the ISTS Note that states that, "Not required to be performed until 24 hours after a required pump is not in operation." The licensee proposed to change "24 hours" in the note to 7 days. BVPS-1 and 2 CTS SR 4.4.1.2.2 states that, "With the rod control system incapable of rod withdrawal, at least two cooling loops, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability." BVPS-1 and 2 CTS SR 4.4.1.3.2 states that, "The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability."

Westinghouse ISTS SR 3.4.5.3, 3.4.6.3, SR 3.4.7.3 and SR 3.4.8.2 contain a Note that states that the surveillances are, "Not required to be performed until 24 hours after a required pump is not in operation." ISTS SR 3.9.6.2 does not have this Note. BVPS-1 and 2 CTS surveillances do not include a similar Note. The NRC staff RAI requested additional justification to support the licensee proposal, "Not required to be performed until seven days after a required pump is not in operation." In response to the staff RAI, the licensee withdrew its proposal and agreed to comply with the Westinghouse ISTS NOTE (BSIs-7, 8, 9, and 10 were withdrawn by the licensee's supplemental letter dated April 19, 2006 (ADAMS Accession No. ML061140093)). Therefore, BSIs-7, 8, 9, and 10 are no longer subjects of this review. In addition, the licensee added the same NOTE in its SR 3.9.5.2 that is consistent with other surveillances in TS 3.4.1 and with other Westinghouse plant LARs (North Anna and DC Cook) previously approved by the NRC staff. Therefore, the previous BSI-11 change is no longer judged by the NRC staff to be "beyond scope" and is acceptable.

G.1.3.4 Conclusion

The NRC staff has reviewed BSIs 5, 6, 7, 8, 9, 10, and 11 dealing with deviation from various ISTS surveillances. Based on its evaluation above, the NRC staff has determined that BSIs-5 and 6 are acceptable. The NRC staff also finds that BSIs 7, 8, 9, 10, and 11 are no longer BSI items, since the licensee has revised its application to comply and be consistent with the ISTS requirements. Therefore, the NRC staff determines that the licensee proposal for these TS changes is acceptable.

G.1.4 BSI-12, "RCS Isolated Loop Startup" - ITS 3.4.18

BSI-12 proposes to change the ISTS 3.4.18, "Isolated Loop Startup," LCO and SRs related to the isolated loop temperature to be more consistent with the BVPS-1 and 2 safety analyses assumptions and CTS RCP start restrictions. The improved STS requires that the isolated loop temperature be no greater than 20 °F below the operating loop temperature before the cold leg isolation valve can be opened. The licensee proposes to change this requirement to, "the cold leg temperature must be ≥ the minimum reactor coolant system (RCS) temperature assumed in the analysis before the cold leg isolation valve can be opened." In addition, new temperature requirements are added similar to the temperature restrictions for starting an RCP in ITS 3.4.7, "RCS Loops-Mode 5" (ITS 3.4.18, DOC M.1, JFDs 1 and 2).

By letter dated July 10, 2006 (ADAMS Accession No. ML061940177), the licensee withdrew this proposed change.

G.1.5 BSIs-13 and 14, "MFIV/MSIV Stroke Time" - SRs 3.7.2.1 and 3.7.3.1

BSIs-13 and 14 proposes to remove the valve isolation times from SR 3.7.2.1 for the main steam isolation valves (MSIVs), and from SR 3.7.3.1 for the main feedwater isolation valves (MFIVs), main feedwater regulating valves and associated bypass valves and replace the times with a specific reference that the isolation time of each valve is "within limits." The valve isolation times would be relocated to the LRM and future changes would be controlled pursuant to 10 CFR 50.59. The licensee states that this is consistent with the previously approved relocation of other valve response times such as the relocation of containment isolation valve response times. The CTS SR 4.7.1.5 for MSIVs would thus be changed; however, the licensee has no CTS for MFIVs.

MSIVs

One MSIV is located in each main steam line outside of the containment. Closing the MSIVs isolates each SG from the others and isolates the turbine, steam bypass system and other auxiliary steam supplies from the SG. The MSIVs close on a main steam isolation signal generated by either low SG pressure or high containment pressure. The MSIVs fail close on loss of control air pressure (BVPS-1) and loss of control or actuation power (BVPS-2). By isolating the steam flow from the secondary side of the SG, the MSIVs prevent over cooling the reactor core following a high energy line break (HELB). By preventing core overcooling, the MSIVs protect the reactor core from being damaged.

The licensee is proposing to relocate the required closure times for the MSIVs to the LRM. Changes to the LRM are subject to the 10 CFR 50.59 process. The 10 CFR 50.59 criteria provide adequate assurance that prior staff review and approval will be requested by the licensee for changes to the LRM requirements with the potential to affect the safe operation of the plant. Furthermore, the MSIVs are subject to periodic testing and acceptance criteria in accordance with the Inservice Testing (IST) Program. Compliance with the IST Program is required by Section 5.5.4 of the BVPS-1 and 2 CTS, ITS and 10 CFR 50.55a. The IST Program includes specific baseline reference value operating times for valves that are not subject to arbitrary changes. 10 CFR 50.36 requires the inclusion of the periodic testing of the MSIVs in the SRs, but not the actual closure time of the valves. The BVPS-1 and 2 ITS contain the periodic testing requirements for MSIVs in accordance with 10 CFR 50.36.

Based on the requirements of 10 CFR 50.36, 10 CFR 50.59 and the BVPS-1 and 2 IST Program, the NRC staff has determined that relocating the MSIV closure times to the LRM is acceptable.

MFIVs, Main Feedwater Regulation Valves (MFRVs), and Associated Bypass Valves (BVs)

The purpose of MFIVs is to isolate the nonsafety-related portions from the safety-related portions of the main feedwater system. In the event of a secondary side pipe rupture inside containment, these valves limit the quantity of high energy fluid that enters the containment through the break and provide a pressure boundary for the controlled addition of auxiliary feedwater to the intact loops. The MFIVs, the MFRVs, and BVs close on receipt of an SI signal or SG water level - high high signal. By isolating the feedwater flow from the affected SG, the MFIVs, MFRVs and BVs prevent overcooling the reactor core and over pressurizing of the containment due to feedwater pump runout.

The BVPS-1 and 2 CTS do not contain MFIV, MFRV, and BV requirements. The licensee is proposing to adopt the applicable TS from the Section 3.7.3 of NUREG-1431, Revision 2.0. As with the MSIVs, the licensee is also proposing to relocate the required closure times for the MFIVs, MFRVs and BVS to the LRM. Changes to the LRM are subject to the provisions of 10 CFR 50.59. The 10 CFR 50.59 criteria provide adequate assurance that prior NRC staff review and approval will be requested by the licensee for changes to the LRM requirements that have the potential to affect the safe operation of the plant. Furthermore, the MFIVs, MFRVs, and BVs are subject to periodic testing and acceptance criteria in accordance with the IST Program. Compliance with the IST Program is required by Section 5.5.4 of the BVPS-1 and 2 CTS, ITS, and 10 CFR 50.55. The IST Program includes specific baseline reference value operating times for valves that are not subject to arbitrary changes. 10 CFR 50.36 requires the inclusion of the periodic testing of the MFIVs, MFRVs, and BVs in the SRs, but not the actual closure time of the valves. The BVPS-1 and 2 ITS contain the periodic testing requirements for MFIVs, MFRVs, and BVs, in accordance with 10 CFR 50.36.

Based on the requirements of 10 CFR 50.36, 10 CFR 50.59 and the BVPS-1 and 2 IST Program, the NRC staff has determined that relocating the MFIVs, MFRVs and BVs closure times to the LRM is acceptable.

G.1.6 BSIs-15, 16, and 17, "Inoperable Component Cooling Water (CCW)/SW Train" - ITS 3.7.7/3.7.8 Action C

BSIs-15-17 propose changes to the ISTS 3.7.7 and 3.7.8 to provide a new Action Condition C, rather than the application of LCO 3.0.3, for the case where 2 CCW (ISTS 3.7.7) or 2 service water (SW) (ISTS 3.7.8) trains are inoperable resulting in insufficient cooling capacity for decay heat removal in Mode 4 such that the plant cannot cool down to Mode 5 (ITS 3.7.7 and 3.7.8, DOC L.3, JFD 2).

In converting to BVPS-1 and 2 CTS 3.7.7, "Component Cooling Water" and 3.7.8, "Service Water," to the ISTS, the licensee proposes to include a new Condition C for the case where 2 trains are inoperable. The ISTS does not explicitly address these conditions and therefore, LCO 3.0.3 would apply for the case of 2 inoperable trains of CCW or SW. LCO 3.0.3 requires that the plant be placed in Mode 5 within the specified time. The CCW system at BVPS-1 and 2 functions to supply the RHR heat exchangers with cooling water for cooling the units from the point of acceptable RHR entry conditions to Mode 5. Without the cooling capacity of the CCW system, the plant is in a seriously degraded condition and is unable to transition from Mode 4 to Mode 5 and remain in Mode 5. Likewise, the SW system supplies the CCW heat exchanger with cooling water which in turn provides RHR cooling for cooling the units from the point of acceptable RHR entry conditions to Mode 5. Without the cooling capacity of the SW system, the plant is in a seriously degraded condition and is unable to transition from Mode 4 to Mode 5 and remain in Mode 5.

The proposed new Condition C will be reflected in BVPS-1 and 2 ITS 3.7.7 and ITS 3.7.8, and will apply for 2 inoperable CCW trains and 2 inoperable SW trains, respectively, in Mode 4. In response to the NRC staff's RAI, the licensee stated that the proposed Condition C will require immediate action to restore one train of CCW or SW, as applicable, to OPERABLE status. LCO 3.0.3 will still be applicable for operation in Modes 1, 2, and 3, if 2 trains of CCW or SW, as applicable, become inoperable. The proposed change recognizes that it is not possible to transition from Mode 4 to Mode 5 without having the RHR system available to remove the decay heat.

The SW system and the CCW system provide support for systems that are required to maintain the capability to safely operate the plant, such as RCP seal cooling. Therefore, BVPS-1 and/or 2 cannot be maintained in Modes 1-3 if no SW or CCW is available and LCO 3.0.3 will require the affected plant to be shut down and placed in Mode 4. Once in Mode 4, proposed ITS 3.7.7.C or ITS 3.7.8.C, as applicable, will be in effect and will prohibit any further Mode changes until at least one train of either the CCW or SW system, as applicable, is restored to operable status thereby enabling the affected plant to be cooled down to Mode 5. Furthermore, the proposed new Condition C is similar to previously approved ITS 3.7.5, “Auxiliary Feedwater System”, Condition D, which suspends any action to change Modes when all 3 trains of auxiliary feedwater (AFW) are inoperable until at least one AFW train is restored to operable status. Similar to the situations that are being addressed by the proposed changes to ITS 3.7.7 and ITS 3.7.8, AFW is the assured source of SG makeup for cooling the units to RHR entry conditions and the proposed ITS changes are consistent with ISTS requirements that have been approved for the AFW system.

Based on the evaluation discussed above, the NRC staff has determined that requiring the plant to stay in Mode 4 whenever the CCW and/or SW systems are not operable is necessary and appropriate. Therefore, the staff finds the proposed changes to ITS 3.7.7 and ITS 3.7.8 to be acceptable.

G.1.7 <u>BSI-18, “Ultimate Heat Sink Temperature”</u>

BSI-18 proposes changes to ITS 3.7.9, Ultimate Heat Sink [UHS],” Action Condition B, such that the proposed Action does not include the ISTS upper and lower temperature limits, but will require more frequent monitoring of the UHS temperature when the BVPS-1 and 2 limit is exceeded rather than an immediate unit shutdown, and would require a unit shutdown when the UHS temperature averaged over the previous 24 hours exceeds the limit (ITS 3.7.9 Action A, DOC L.1, JFD 2).

This change was withdrawn by the licensee’s letter of April 19, 2006 (ADAMS Accession No. ML061140093).

G.1.8 <u>BSI-19, “Diesel Generator Monthly Startup Requirements”</u>

BSI-19 proposes to modify the notes in ISTS SRs 3.8.1.2 and 3.8.1.3 to add the words “or based on operating experience,” to supplement the phrase “as recommended by the manufacturer” (ITS SR 3.8.1.2 and SR 3.8.1.3, DOC L.19, JFD 17).

This change was withdrawn by the licensee’s letter of April 19, 2006 (ADAMS Accession No. ML061140093).

G.1.9 BSI-20, “Diesel Generator Fuel Oil Day Tank Level,” SR3.8.1.5.1

BSI-20 proposes to modify ISTS SR 3.8.1.5 by changing the requirement to “Check for and remove accumulated water from each day tank [and engine mounted tank]” to “Check and remove water from each engine mounted tank.” A note has been added to indicate that this is applicable to BVPS-1 only (ITS SR 3.8.1.5.1, DOC L.18, JFD 10).
This change was withdrawn by the licensee’s letter of April 19, 2006 (ADAMS Accession No. ML061140093).

G.1.10 BSI-21, “Diesel Generator Load Sequencing (Shutdown),” SR3.8.2.1 Note 2, and BSI-22, “Diesel Generator Load Sequencing (delete SI auto-start signal),” SR 3.8.2.1 Note 3

BSI-21 proposes a note to ITS SR 3.8.2.1 to address the surveillances (SRs 3.1.8.13 and 3.8.1.14) used to verify the capability of the automatic load sequencer function of the emergency diesel generators (EDGs). The note states that the load sequencer function SRs only include the verification of loads applicable (necessary for operability) in the shutdown modes of operation (Modes 5 and 6) addressed by ITS 3.8.2 (ITS SR 3.8.2.1 Note 2, DOC L.3, JFD 5).

BSI-22 proposes to revise improved STS SR 3.8.2.1 by the addition of Note 3. Proposed Note 3 to ITS SR 3.8.2.1 states, “SR 3.8.1.14 is only required to be met with the use of an actual or simulated loss of offsite power signal.” SR 3.8.1.14 verifies the response of the emergency bus and EDG to an engineered safety features (ESF) signal in conjunction with a loss-of-offsite power (LOOP). The proposed note is intended to clarify that in the shutdown modes addressed by SR 3.8.2.1, there are no required ESF actuation signals. The ESF actuation instrumentation specified in ITS 3.3.2 is only required to be operable in Modes 1-4, and ITS 3.8.2, “AC Sources Shutdown,” is only applicable in Modes 5 and 6 (ITS SR 3.8.2.1 Note 3, DOC L.3, JFD 6).

BSI-21

CTS SR 4.8.1.2 specifies the SRs applicable to alternating current (AC) power sources during shutdown conditions. CTS 4.8.1.2 states: “The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2.a.6.” The single exception is CTS SR 4.8.1.1.2.a.6 which corresponds to proposed ITS SR 3.8.1.3. Like CTS 4.8.1.2, ITS SR 3.8.2.1 specifies the SRs applicable to AC sources during shutdown conditions. However, SR 3.8.2.1 has 3 notes that modify the AC source surveillance required during shutdown conditions and relax the surveillance acceptance criteria applicable to AC sources during shutdown condition. BSI-21 relates to ITS SR 3.8.2.1, Note 2. Note 2 reads: “The verification of load sequencer functions in SRs 3.8.1.13 and 3.8.1.14 is only required to be met for those loads required in the applicable modes of LCO 3.8.2.”

The SRs referred to in Note 2 address the capability to automatically sequence loads onto the EDG. During the shutdown conditions to which ITS 3.8.2.1 applies, the loads required to maintain the unit in a safe shutdown condition are small compared to the total number of loads that must normally be sequenced on the emergency bus. Note 2 clarifies that the load sequencer function is required only for those loads necessary for the shutdown modes of operation addressed by ITS 3.8.2. The Bases for the proposed note explains that the required loads referred to by the note consist of the equipment required operable by TS 3.8.2 as well as the equipment required to support the operability of the TS-required equipment. The proposed note is consistent with the intent of the ITS note it replaces. The NRC staff finds that the addition of ITS Note 2 is acceptable, because it clarifies that SR 3.8.2.1 applies only to the capability to sequence only those loads required for the safe operation of the unit during shutdown conditions.

BSI-22

BSI-22 relates to proposed Note 3, which clarifies that certain ESF actuation signals are not applicable in the shutdown modes addressed by SR 3.8.2.1. The ESF actuation instrumentation (specified in ITS 3.3.2) is only required operable in Modes 1 through 4 and is not applicable in Modes 5 and 6 and during fuel movement. Therefore, these ESF actuation (i.e., SI) signals are not required for the emergency bus or EDG during shutdown conditions. The only applicable actuation signal specified in SR 3.8.1.14 is the loss of voltage (offsite power) actuation signal. The proposed change continues to assure the verification of the required system response to a loss of voltage. The change merely clarifies that the verification of the system response to certain ESF actuation signals is not required during the shutdown conditions applicable under ITS 3.8.2. The NRC staff finds the proposed Note 3 acceptable, because it clarifies that the only applicable actuation signal specified in SR 3.8.1.14 during shutdown conditions applicable under ITS 3.8.2 is the loss of voltage actuation signal.

On the basis of its review, the NRC staff finds that the proposed Notes 2 and 3 provide a clarification for the verification of load sequencer function and certain ESF actuation signals addressed by ITS 3.8.2. The load sequencer function is required only for those loads necessary for the shutdown modes of operation. The applicable actuation signal specified in SR 3.8.1.14 during shutdown conditions is the loss of voltage signal. Therefore, the changes proposed by BSIs-21 and 22 are acceptable.

G.1.11 BSI-23, "Containment Purge and Exhaust System," ITS LCO 3.9.3.c.2, 3.9.3.c.3 and ITS 3.7.12, "Supplemental Leak Collection and Release System" (SLCRS)

BSI-23 proposes to revise ISTS 3.9.3 by making changes to ITS 3.9.3.c.2 and adding ITS 3.9.3.c.3. ITS 3.7.12 was also revised to include supporting requirements for the changes to ITS 3.9.3. The requirements of ITS 3.9.3 and 3.7.12 are only applicable during fuel movement involving recently irradiated fuel. The changes are intended to make the containment penetration requirements of ITS 3.9.3 more consistent with the design and licensing basis for BVPS-1 and 2. The LCO requirement that specifies that each penetration providing direct access from the containment atmosphere to the outside atmosphere be capable of being closed by an OPERABLE containment purge and exhaust isolation system and its associated surveillances are made applicable to BVPS-2 only, and a provision is added for BVPS-1 only (ITS 3.9.3.c.3) that allows the purge and exhaust system penetrations to be open when the system air is exhausted to an OPERABLE SLCRS train (ITS 3.9.3, DOC L.1, JFD 3). Corresponding changes are made to ITS 3.7.12 to require one train of SLCRS OPERABLE when required to support ITS 3.9.3.c.3 for BVPS-1 and to provide an appropriate Action in ITS 3.7.12 if the required train was inoperable (ITS 3.7.12, JFD 1 and ITS 3.9.3 DOC L.1).

The current BVPS-1 and 2 design-basis FHA of record do not credit containment isolation or filtration of containment ventilation exhaust to mitigate an FHA involving non-recently irradiated fuel in containment. Recently irradiated fuel is defined in the TS Bases as "… fuel that has occupied part of a critical reactor core within the previous 100 hours." Although BVPS-1 and 2 do not currently have safety analyses that support moving recently irradiated fuel assemblies, TS requirements have been retained to address the condition of moving recently irradiated fuel assemblies. These TS requirements are retained because decay time limits for moving irradiated fuel have been relocated to the LRM (see Table LA, page 47 of 53) and the NRC staff has determined that fuel handling limits should be retained in the TSs should the licensee develop future capability to move recently irradiated fuel.

Movement of recently irradiated fuel can only be done if the dose analysis required by 10 CFR 50.59 indicates that the calculated dose increase is less than or equal to 10% of the difference between the current calculated dose value for an FHA involving non-recently irradiated fuel and the regulatory guideline value (10 CFR 50.67 or GDC 19, as applicable), and the increase in dose does not exceed the current SRP guideline value for the particular design basis event. The dose analyses for recently irradiated fuel will take credit for the SLCRS filtration for Unit 1 and containment isolation for Unit 2. If the dose analysis indicates a calculated dose increase greater than 10% as described above, the licensee must submit a license amendment request for NRC review and approval.

The retained TS requirements which are applicable when moving recently irradiated fuel or fuel assemblies over recently irradiated fuel include the containment penetration requirements of ITS 3.9.3. The current FHA analysis for moving non-recently irradiated fuel and the CTS requirements retained for moving recently irradiated fuel were approved by the NRC in Amendment Nos. 241 for BVPS-1 and 121 for BVPS-2, issued August 30, 2001.

BVPS-1 does not rely on the containment purge and exhaust system isolation to mitigate the radiological consequences of an FHA in containment. For BVPS-1, the mitigation of the radiological consequences of an FHA in containment would require filtration of containment air exhaust by the SLCRS instead, consistent with the design and licensing basis as described in Amendment No. 23, issued December 12, 1979 (ADAMS Accession No. ML003766458), and in Section 14.2.1.4 of the BVPS-1 UFSAR, Revision 18. In addition, the proposed changes include requirements in ITS 3.7.12 to assure an OPERABLE train of SLCRS is available when required by ITS 3.9.3 to support filtration of the BVPS-1 containment exhaust. Because the BVPS-1 FHA involving recently irradiated fuel would rely upon the SLCRS, unlike BVPS-2, and appropriate requirements for an OPERABLE train of SLCRS for BVPS-1 have been included in ITS 3.7.12, the NRC staff finds that this proposed BSI change accurately reflects the BVPS-1 design basis and is, therefore, acceptable.

As stated above, BVPS-2 would rely on the containment purge and exhaust isolation system to mitigate the radiological consequences of an FHA in containment, as described in Section 15.7.4.1 of the BVPS- 2, UFSAR, Revision 11. Therefore, this requirement in the ITS is retained consistent with the ISTS and made applicable to BVPS-2 only for fuel movement involving recently irradiated fuel. The NRC staff finds that this proposed BSI change is appropriate for the BVPS-2 design and is also acceptable.

G.1.12 BSI-24, “RHR-Low Water Level,” ITS 3.9.4 and 3.9.5

BSI-24 proposes to incorporate a note into ITS 3.9.5, “RHR and Coolant Circulation - Low Water Level,” and ITS 3.9.4, “RHR and Coolant Circulation - High Water Level.” NRC-approved TSTF-21 Revision 0, incorporated a Bases change to ITS 3.9.5 that provides an exception to the requirement for the RHR loop to be circulating reactor coolant to allow both RHR pumps to be aligned to the refueling water storage tank (RWST) to support filling or draining of the refueling cavity or for performance of required testing. This exception was incorporated into NUREG-1431, Revision 3. In a letter dated April 29, 1999, from W. D. Beckner, NRC, to J. Davis, Nuclear Energy Institute, the NRC recommended that TSTF-21, Revision 0 be revised to include an LCO exception note to remove the RHR loop from operation (i.e., from circulating coolant) to support cavity fill and drain or to support required testing. The licensee’s note incorporates this NRC recommendation which was not incorporated into TSTF-21, Revision 0 or NUREG-1431, Revision 3 (ITS 3.9.4, LCO Note 3 and ITS 3.9.5, LCO Note 3, DOC L.4, JFD 3).

This change was withdrawn by the licensee’s letter of July 10, 2006 (ADAMS Accession No. ML061940177).

G.1.13 BSI-25, “ASME Test Frequencies,” ITS 5.5.4.b

BSI-25 proposes to revise ISTS 5.5.4.b which states, “The provisions of SR 3.0.2 are applicable to the above required Frequencies [improved STS 5.5.4.a] for performing inservice testing activities.” The licensee states that the list in ISTS 5.5.4.a lists some of the test intervals referenced in the IST requirements but is not a comprehensive list. The licensee proposes to revise ITS 5.5.4.b to state, “The provisions of SR 3.0.2 are applicable to the above required Frequencies and other normal and accelerated Frequencies specified in the Inservice Testing Program for performing inservice testing activities.” This would expand the applicability of SR 3.0.2 provisions to all inservice testing requirements intervals and not just those listed in ITS 5.5.4.a (ITS 5.5.4.b, DOC L.4, JFD 34).

BSI-25 involves a change to CTS 4.0.5.c, described in ITS 5.5.4.b, which changes the applicable surveillance intervals listed in CTS 4.0.5.b (or ITS 5.5.4.a), for which the 1.25 factor provided in SR 3.0.2 would apply. The licensee provided additional information in its submittal dated April 19, 2006 (ADAMS Accession No. ML061140093), which revised proposed ITS 5.5.4.b. Specifically, in the proposed ITS 5.5.4.b, the provisions of SR 3.0.2 would be applicable to, not only the intervals listed in the ITS 5.5.4.a table, but to “other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program” for performing IST activities.

The licensee’s proposed change to the BVPS-1 and 2 CTS 4.0.5.c, described in ITS 5.5.4.b, would apply the 25-percent time extension provided for in SR 3.0.2 to other normal and accelerated frequencies specified as 2 years or less in the IST Program, in addition to those periods listed in the CTS 4.0.5.b table. The NRC staff finds that applying the extension to these time periods, up to and including 2 years, is acceptable. The purpose of the table is to more precisely define the meaning of “monthly”, “quarterly”, and other terms for time periods, such that surveillances are not delayed due to lack of precision of the stated time periods.
For example, the term “yearly” is defined by the table as 366 days, which prevents a scenario where testing is performed at any time during 1 year and testing is performed again at any time

during the next year, which could result in an approximate doubling of the 366-day period. With the proposed ITS 5.5.4.b, if a test frequency (e.g., monthly) is augmented and increased (e.g., to a 2-week test frequency, which is not listed in the table), it could also have the 25-percent extension applied. This is consistent with the intent of the 25-percent extension as described in the Bases for SR 3.0.2, in that the extension would provide operational flexibility, but would not significantly degrade the quality of systems and components that results from performing the surveillance at the specified frequency. Further, the proposal to limit the applicability to frequencies of 2 years or less, limits the maximum incremental time period between surveillances, which could be added by the 25-percent extension, and is consistent with guidance provided in NUREG-1482, Revision 1. Without this limitation, some components, such as safety and relief valves, which may be tested at surveillance intervals significantly greater than 2 years, could have extensions applied which would be much greater than that needed for operational flexibility.

The NRC requirements for TS SRs are provided in 10 CFR 50.36(c)(3). The NRC staff has determined that the proposed change to CTS 4.0.5.b, described in ITS 5.5.4.b, meets the requirements of 10 CFR 50.36(c)(3), since the change would not significantly degrade the quality of systems and components that results from performing the surveillance at the specified frequency. Therefore, the staff finds that the proposed ITS 5.5.4.b is acceptable.

G.2 BSI Changes Identified by the Staff:

The licensee stated that five proposed ITS changes were in accordance with pending TSTFs. Because these TSTFs were pending and had not been approved by the NRC staff, the staff determined that inclusion of the proposed TS changes based on the pending TSTFs constituted BSIs until such time as the staff approved the TSTS from which the proposed ITS changes were developed. The staff, therefore, assigned these proposed changes BSI numbers 26-30.

G.2.1 BSI-26, "Turbine-Driven Auxiliary Feedwater (TDAFW) Pump," ITS 3.7.5

BSI-26 proposes to incorporate pending TSTF-412, Revision 0, which would provide actions and clarify the operability status when one steam supply to a TDAFW pump is inoperable. TS LCOs for AFW systems are required in accordance with 10 CFR 50.36(c)(2)(ii)(C). The ISTS are reflective of policies and practices that the NRC considers to be acceptable with respect to TS LCOs, Completion Times, and Action requirements. The BVPS-1 and 2 AFW systems consist of two motor-driven AFW (MDAFW) pumps and one steam TDAFW pump configured into three trains. Any two of the three AFW pumps are capable of supplying the required feedwater flow assumed in the accident analysis. Each MDAFW train feeds one of the two AFW supply headers, and each AFW supply header feeds all three SGs. The TDAFW pump is aligned to automatically provide flow to one of the two AFW supply headers upon AFW actuation. The TDAFW pump turbine is aligned receive steam flow from at least two of the three main steam lines from upstream of the MSIVs. Each of the TDAFW pump steam supply lines is sized to provide 100 percent of the steam flow that is necessary for operating the TDAFW pump at full capacity.

BVPS-1 and 2, ITS 3.7.5, Proposed Condition A

The licensee proposes to adopt this ISTS requirement, with Condition A modified to refer to the inoperability of a TDAFW train due to an inoperable steam supply, instead of referring to the inoperability of a TDAFW pump steam supply as reflected in the ISTS. The proposed ITS change maintains the train-oriented language in the BVPS-1 and 2 ITS for Condition A, which is consistent with the approach that is used for the other Conditions that are specified by ISTS 3.7.5. The licensee's proposed use of train-oriented language is editorial in nature and is preferred by the NRC staff because it tends to maintain clarity and consistency of the BVPS-1 and 2 ITS, while still reflecting the ISTS requirement that is specified. Therefore, the proposed change is considered to be acceptable.

BVPS-1 and 2, ITS 3.7.5, Proposed Condition C

A new Condition C is proposed for when the TDAFW train is inoperable due to one inoperable steam supply at the same time that one MDAFW train is inoperable. Proposed Action C.1 requires restoration of the affected steam supply to operable status within 24 hours. Alternatively, proposed Action C.2 requires restoration of the inoperable MDAFW train to operable status within 24 hours.

The ISTS typically allows a 72-hour Completion Time for Conditions where the remaining operable equipment is able to mitigate postulated accidents as long as a concurrent single active failure is not imposed. In this particular case, a 24-hour Completion Time is proposed for a situation where the AFW system would be able to perform its function for most postulated events, and would only be challenged by situations such as a main steam line break or a feedwater line break that render the remaining operable steam supply to the TDAFW pump inoperable. The 24-hour Completion Time is based on the remaining operable steam supply to the TDAFW pump and the continued functionality of the TDAFW train, the remaining operable MDAFW train, and the low likelihood of an event occurring during this 24-hour period that would challenge the capability of the AFW system to perform its function. The proposed Completion Time for this particular situation is consistent with what was approved for Waterford Unit 3 by Amendment No. 173 for a similar Condition (ADAMS Accession No. ML012840538), and it is commensurate with the ISTS, in that the proposed Completion Time is much less than the 72 hours that is allowed for the situation where full accident mitigation capability is maintained. Therefore, the NRC staff agrees that the proposed 24-hour Completion Time is acceptable.

BVPS-1 and 2, ITS 3.7.5, Proposed Condition D

The current Condition C is renamed Condition D and is modified to accommodate the requirements that are specified by the proposed new Condition C. Specifically, the first part of Condition D is modified to apply to the situation where the Required Actions and associated Completion Times of Conditions A, B, or the proposed new Condition C are not met. The requirement specified by Condition D, to place the plant in a Mode that does not apply, within the specified time period when the Required Actions and Completion Times of proposed new Condition C are not met, is appropriate and consistent with the existing ISTS 3.7.5 requirements. Therefore, the proposed change is acceptable.

The second part of Condition D following the first "OR" is modified from "Two AFW trains inoperable in MODE 1, 2, or 3" to "Two AFW trains inoperable in MODE 1, 2, or 3 for reasons other than Condition C." This change is necessary to recognize the situation specified by the proposed new Condition C where 2 AFW trains are allowed to be inoperable as a special case (i.e., one MDAFW train is inoperable at the same time that the TDAFW train is inoperable due solely to an inoperable steam supply to the pump turbine). Therefore, the proposed change is acceptable.

The original ITS LAR also proposed to add a second "OR" to Condition D that adds a note that states: "This Condition is only applicable when the turbine driven AFW train is inoperable solely due to one inoperable steam supply." The note was followed by a new proposed third provision of Condition D: "Three AFW trains inoperable." In the case where all three AFW trains are inoperable, with the TDAFW train inoperable due solely to one steam supply being inoperable, this proposed provision of Condition D required the licensee to place the plant in Mode 4 using the TDAFW pump with only one steam supply available. In an RAI, the NRC requested that the licensee provide confirmation (such as through confirmatory testing or operational data) that Mode 4 conditions can be achieved within the specified time period using the TDAFW pump with only one steam supply available for providing SG makeup water. Because the licensee was not able to provide the confirmation that was requested, this proposed new third provision of Condition D was withdrawn by letter dated September 1, 2006 (ADAMS Accession No. ML062490197). Therefore, no further NRC evaluation is required of this part of the proposed change as it is no longer applicable.

<u>BVPS-1 and 2, ITS 3.7.5, Proposed Conditions E and F</u>

The current Conditions D and E are renamed as Conditions E and F, respectively. These changes are purely editorial as no other changes are involved. Therefore, the proposed change is acceptable.

Based on a review of the information that was provided by the licensee as discussed above, the NRC staff has determined that the proposed changes to BVPS-1 and 2, ITS 3.7.5, are acceptable. The proposed changes are consistent with NRC practices and policies as generally reflected in the ISTS and as reflected by applicable precedents that have been approved. Therefore, the staff has determined that the proposed changes to BVPS-1 and 2, ITS 3.7.5, as described in BSI-26, are acceptable.

G.2.2 <u>BSI-27, "Battery Monitoring and Maintenance Program," SR 3.8.4.2 (Bases) and ITS 5.5.13</u>

BSI-27 proposes to incorporate pending TSTF-451-T, Revision 0, which would provide corrections to the battery monitoring and maintenance program (Section 5.0) and the Bases of SR 3.8.4.2 (Section 3.8).

The licensee requests the NRC staff's approval to use the changes made in TSTF 451-T, Revision 0, for BVPS-1 and 2 ITS. The TSTF revises the Bases of ITS SR 3.8.4.2 to be consistent with the surveillance. The sentence "The other option requires that each battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur)" is revised to read, "The other option requires that each battery charger be capable of recharging the battery after a service test coincident with supplying the largest <u>combined</u> demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur)" (emphasis added).

RG 1.32, "Criteria for Safety-Related Electric Power System for Nuclear Power Plant," Revision 2, Regulatory Position 1.b states:

> Battery Charger Supply. The provisions of Section 5.3.4 of IEEE Standard 308-1974 should be construed to mean that the capacity of the battery charger supply should be based on the largest combined demands of the various steady-state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charge state, irrespective of the status of the plant during which these demands occur.

The proposed wording change of SR 3.8.4.2 Bases is consistent with the RG. The change to the SR 3.8.4.2 Bases also makes the Bases consistent with the associated SR. On this basis, the NRC staff finds the proposed change to make the Bases consistent with the specification to be acceptable.

G.2.3 BSI-28, Rod Withdrawal From Subcritical or Low Power Conditions

BSI-28 proposes to incorporate pending TSTF-453-T, Revision 2, which would provide a new specification in Section 3.1 and revise existing requirements in Section 3.3 to more completely address a rod withdrawal from subcritical conditions (RWFS) event. The TSTF adds new boron concentration operating restrictions during conditions when the power range nuclear instrumentation may not be able to provide the necessary trip function to protect against an RWFS event.

The ISTS Section 3.1 is revised by the addition of a new TS developed to address issues raised in Westinghouse Nuclear Safety Advisory Letter (NSAL)-00-016 "Rod Withdrawal from Subcritical Protection in Lower Modes." NSAL-00-016 discusses the reactor trip protection function for an uncontrolled Rod Cluster Control Assembly (RCCA) RWFS event. The BVPS-1 and 2 ITS conversion submittal based this change on TSTF-453, that has not yet been approved. TSTF-453 could not be used as a basis for approval, therefore, this review is based on the NSAL-00-016.

In the current plant configuration, when the control rod drive mechanisms (CRDMs) are powered, there exists the potential for RCCAs to be withdrawn due to operator error or equipment malfunction. If the reactor is at low power or subcritical, primary protection for this transient is provided by the power range (PR) neutron flux, low trip setpoint. The source range (SR) flux trip function is implicitly credited as the primary trip function in modes 3, 4, and 5 because the PR flux, low power trip is not required to be operable in these modes. The PR monitors are adequate for temperatures at or above the hot-zero-power (HZP) of 500 °F for which the instruments are calibrated. At low RCS temperatures, the Power Range Neutron Flux-Low Trip Function may not provide the required protection. This limitation of the power range instrumentation is due to calibration issues associated with shielding caused by cold water in the downcomer region of the reactor vessel and is discussed in more detail in the NSAL and TSTF-453.

The RWFS event is initiated at HZP, with the reactor just critical in Mode 2. Analysis of this event is done to ensure that the primary acceptance criterion of DNB ratio design basis is satisfied. A positive reactivity insertion rate corresponding to simultaneous withdrawal of two RCCAs is assumed. In Mode 2, all RCPs are required to be operable, however, in the analysis only two RCPs are assumed operating, which is conservative with respect to DNB ratio. The event is terminated by the PR neutron flux low setpoint trip function. In Modes 1 and 2, the PR low setpoint trip function is operable, but not in Modes 3, 4, and 5. It is assumed that the RWFS is bounded by the analysis performed for Mode 2, based on crediting an SR reactor trip function.

The problem as stated in NSAL-00-016, is that the SR flux trip function provides the primary protection, however, the SR reactor trip function is neither time response tested nor seismically qualified, and therefore, cannot be considered operable to provide protection for RWFS transients at coolant temperatures less than 500 °F. In the bases of the SR for reactor trip function, the ISTS states: "In Modes 3, 4 or 5 with the reactor shut down, the Source Range Neutron Flux trip function must also be OPERABLE. If the CRD system is capable of rod withdrawal, the Source Range Neutron Flux trip must be OPERABLE to provide core Protection against a rod withdrawal accident." The SR trip function is required to be operable, it has neither time response nor seismic qualification requirements. Therefore, the assumption (as presented in the BVPS-1 and 2 UFSAR analyses) that the Mode 2 analysis bounds the lower Modes is questionable.

NSAL-00-016 proposes two fundamental types of solutions, either credit the SR trip function (recommends a 0.5 sec time delay) or give no credit to the SR trip function, with the reactor in Modes 3, 4, or 5 borate the core to the "all-rods-out" (ARO) level to avoid criticality in a RWFS accident. The BVPS-1 and 2 licensee chose boration to the ARO level.

To implement this solution for BVPS-1 and 2, the licensee proposed a new ITS 3.1.10 that addresses the potential of an RWFS event at low temperatures. ITS 3.1.10 is applicable when operating with any RCS cold leg temperature less than 500 °F and with the rod control system capable of withdrawing rods in Modes 2 (with $k_{eff} < 1.0$), 3, 4, and 5. In these modes, ITS 3.1.10 requires that the RCS be borated to greater than the ARO boron concentration. This level of boron concentration, at temperatures less than 500 °F, assures that if an RWFS were to occur, there will be sufficient shutdown margin to avoid criticality.

Applicability: ITS 3.1.10 provides protection against RWFS for the Modes and conditions as follows:

- In Mode 2, $k_{eff} < 1.0$, any cold leg temperature less than 500 °F, and the control rod system capable of rod withdrawal.

- In Mode 3, any cold leg temperature less than 500 °F, and the control rod system capable of rod withdrawal.

- In Modes 4 and 5, and the control rod system capable of rod withdrawal.

In Mode 1, ITS 3.1.10 is not applicable because the PR neutron flux high trip function is available to mitigate the event. In Mode 2 with $k_{eff} \geq 1.0$, in Mode 2 with $k_{eff} < 1.0$, all cold leg temperature ≥500 °F and the rod control system capable of rod withdrawal, and in Mode 3 with all cold leg temperature ≥500 °F and the rod control system capable of rod withdrawal, the PR flux low trip function provides protection against RWFS.

Actions: Actions A.1 and A.2 are to be implemented immediately in case the boron concentration is not within the specified limits, i.e., bring the RCS boron concentration to greater than the ARO boron concentration, or make the rod control system incapable of rod withdrawal, respectively. Action A.3 offers an alternative to restore all cold leg temperatures to $\geq$500 °F where the PR flux low trip function is operable. Action A3, is modified by a note that it is applicable to Modes 2 and 3, where the temperature can readily be increased to $\geq$500 °F.

Surveillance Requirement: This surveillance ensures using chemical analysis that the boron concentration is within the required limits. Frequency is every 24 hours and is based on the estimated time to cause significant boron dilution in the RCS.

Summary and Limitations

The licensee proposed an additional ITS 3.1.10 to protect against RWFS. The proposed solution is based on Westinghouse's NSAL-00-016 which identified the shortcoming in the CTSs. The problem identification is clear and the proposed ITS is responsive to the problem and conservative, therefore, the NRC staff finds the proposed ITS acceptable. The NRC staff has not as part of this review, completed its review of TSTF-453.

G.2.4 BSI-29, "TSTF-283, Bases Errors," SR 3.8.1.3 (Bases)

BSI-29 proposes to incorporate pending TSTF-472-T, Revision 0, which corrects a Bases error introduced by implementation of NRC-approved TSTF-283 (approved in November 2000). This affects Section 3.8.

TSTF-283, Revision 3, modified the Mode restrictions on the performance of SRs 3.8.1 and 3.8.4. The MODE restriction Notes were modified to allow performance of the SR in prohibited Modes in order to reestablish OPERABILITY. However, the Bases change for ITS SR 3.8.13 (ISTS SR 3.8.1.18) was inadvertently omitted from the NUREG-1431 markup. The Bases change was included in the Bases markup for the other ISTS NUREGs i.e., NUREG-1430, Babcock and Wilcox Plants, NUREG-1432, Combustion Engineering Plants, NUREG-1433, General Electric Plants BWR-4, Plants, and NUREG-1434, General Electric Plants BWR-6 Plants. TSTF-472-T was issued to correct this problem.

Proposed ITS Bases Changes

SR 3.8.1.13 is normally performed during refueling outages to verify that the automatic load sequence timers for each emergency diesel generator is within ± 10 percent of the required value. However, the Note for SR 3.8.1.13 states:

> This Surveillance shall not normally be performed in MODE 1, 2, 3 or 4. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR.

The above Note was incorporated into NUREG-1431, Revision 2, per TSTF-283, Revision 3, but the reasons for the above TS changes were not amplified in the Bases. The Bases should have stated that the note was meant only for the purpose of reestablishing OPERABILITY following post modification testing due to corrective maintenance, corrective modification, deficient or incomplete SRs, and other unanticipated OPERABILITY concerns. It should have also stated that prior to performing the SR in the restricted MODES, an assessment must be performed to ensure plant safety is maintained or enhanced. This assessment should, as a minimum, consider the potential outcomes and transients associated with a failed SR, a successful SR, and perturbation of the offsite or onsite system when they are tied together or operated independently for the SR; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the SR is performed in Modes 1, 2, 3, or 4. Risk insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

TSTF-283, Revision 3, was approved by the NRC on April 13, 2000. TSTF-283 modified the mode restrictions on the performance of SRs in ITS Sections 3.8.1 and 3.8.4. The MODE restriction Notes for the above SRs were modified to allow performance of the SRs in the prohibited Modes in order to reestablish OPERABILITY provided that an assessment determined that plant safety would be maintained or enhanced. TSTF-472-T does not change the TS requirements for removing the MODE restrictions described in TSTF-283, it only adds information that was inadvertently omitted from the Bases of NUREG-1431, Revision 2.

The proposed changes described in TSTF-472-T have been incorporated into Revision 3.1 of NUREG-1431, and all future changes to the Bases will be controlled in accordance with the program requirements of ITS Section 5.5.10, "Technical Specifications (TS) Bases Control Program." This program allows licensee to make changes to the Bases without prior NRC approval provided that the change do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59, changes, tests and experiments.

Since the above changes do not effect the ITS, nor do they require prior NRC approval, the NRC staff finds that the proposed changes to the BVPS-1 and 2 ITS Bases are acceptable.

G.2.5 <u>BSI-30, "LCO 3.0.6 Bases Provisions"</u>

BSI-30 proposes to incorporate pending TSTF-482, Revision 0, which would provide editorial enhancements to the Bases for LCO 3.0.6. TSTF-482, Revision 0, was issued to correct a bases error.

TSTF-482, Revision 0, revised the LCO 3.0.6 Bases from "LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS)" to "LCO 3.0.6 establishes an exception to LCO 3.0.2 for supported systems that have a support system LCO specified in the Technical Specifications (TS)." In addition, two editorial corrections are made to the LCO 3.0.6 Bases to make the sentences grammatically correct.

TSTF-482, Revision 0, was approved by the NRC on December 6, 2005, and modifies the Bases for LCO 3.0.6. LCO 3.0.6 states, "When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system." However, the LCO 3.0.6 Bases states "LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the TS." The Bases also do not specify that this is only true if the support system has an LCO in the TS. This is inconsistent with the Specification and incorrect. LCO 3.0.2 states that when an LCO is not met, the Conditions and Required Actions must be entered. LCO 3.0.6 requires entering the Conditions and Required Actions for support systems when those support systems have an LCO in the TS. This change makes the Bases consistent with the Specifications.

The proposed changes described in TSTF-482, Revision 0, have been incorporated into Revision 3.1 of NUREG-1431, and all future changes to the Bases will be controlled in accordance with the program requirements of ITS Section 5.5.10, “TS Bases Control Program.” This program allows licensee to make changes to the Bases without prior NRC approval provided that the change do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59, changes, tests and experiments.

Since the above changes do not affect the ITS, nor do they require prior NRC approval, the NRC staff has determined that the proposed changes to the BVPS-1and 2 ITS Bases are acceptable.

## 5.0 PROPOSED DELETION OF BVPS-1 AND 2 LICENSE CONDITIONS

In Revision 4 of the application, the licensee proposed that the BVPS-1 License Condition C.(9), “Steam Generator Surveillance Interval Extension” and BVPS-2 License Condition (C.12), “Steam Generator Surveillance Interval Extension” be deleted from the facility operating licenses. These existing License Conditions provided a limited surveillance extension for the BVPS-1 and 2 SGs based on specific dates which have expired. Therefore, BVPS-1 License Condition C(.9) and BVPS-2 License Condition (C.12) are no longer applicable. The NRC staff finds the proposed changes to be administrative in nature and, therefore, acceptable.

## 6.0 LICENSEE COMMITMENTS

In reviewing the proposed ITS conversion for BVPS-1 and 2, the NRC staff has relied upon the licensee’s commitment to relocate certain requirements from the CTS to licensee-controlled documents as described in Table LA, “Removed Details” (Attachment 4 to this SE) and Table R, “Relocated Specifications” (Attachment 6 to this SE). These tables, and Sections 4.D and 4.E of this SE, reflect the relocations described in the licensee’s submittals on the conversion. The NRC staff requested and the licensee submitted a set of license conditions to make these commitments enforceable (see Section 7.0 of this SE). Such commitments from the licensee are important to the ITS conversion because the acceptability of removing certain requirements from the TSs is based on those requirements being relocated to licensee-controlled documents where further changes to the requirements will be controlled by applicable regulations or other requirements (e.g., 10 CFR 50.59).

In addition, the licensee proposed that the current SRs to measure battery state-of-charge using specific gravity be replaced with float current monitoring. In its December 7, 2006, supplemental letter, the licensee provided letters from the battery manufacturers (C&D Technologies, and EnerSys), confirming that float current monitoring is an acceptable method for determining the state-of-charge of the BVPS-1 and 2, batteries. More specifically, C&D Technologies stated that a float current value ≤ 2 amps is both a reliable and an accurate parameter to use to ascertain a state-of-full charge for the 125-V DC Station Batteries installed at BVPS-1 (batteries 1-3 & 1-4). C&D Technologies also stated that the accuracy and reliability of this reading will hold true over the expected life of these batteries (i.e., 20 years). EnerSys also confirmed that float current monitoring is an acceptable method for determining the battery state-of-charge, but stated that a float current value of ≤ 2 amps indicates 95 percent available capacity for the 2GN-13 model batteries (BVPS-2 Batteries 2-3 and 2-4) and 98 percent available capacity for the 2GN-21 model battery (BVPS-1 Batteries 1-1 and 1-2 and BVPS-2 Batteries 2-1 and 2-2).

The licensee also stated in its December 7, 2006, supplemental letter that instrumentation with the appropriate range and accuracy for the expected current reading will be used to monitor the battery charging current. The use of instrumentation with the appropriate range and accuracy will assure a valid determination of the battery state-of-charge. Recognizing that the 2-amp float current value is an indication that the EnerSys batteries are either 95 percent or 98 percent charged, the licensee provided regulatory commitments to maintain at least 5-percent or 2-percent design margin, as applicable, for the appropriate batteries. The licensee also stated that it will list these values in the BVPS-1 and 2, TS Bases for ITS 3.8.4.

The NRC staff finds that the use of float current monitoring in lieu of specific gravity measurements will not have a significant impact on safety or the ability to accurately determine the operability of the BVPS-1 and 2, batteries for the following reasons:

1. Battery manufacturers confirmed that float current monitoring is a reliable and accurate parameter for assessing battery state-of-charge.

2. Float current monitoring will use instrumentation with appropriate range and accuracy.

3. Licensee’s commitment to maintain either 2-percent or 5-percent design margin, as applicable, for the appropriate EnerSys batteries.

The NRC staff considers the regulatory commitment to maintain adequate design margin for the EnerSys batteries and that these design margins would be specified in the TS Bases is sufficient.

## 7.0 LICENSE CONDITIONS

In its letter dated October 24, 2006, the licensee agreed to license conditions to define the schedule to begin performing the new and revised SRs after implementation of the ITS. The following license conditions are included in the Facility Operating Licenses:

1. Schedule for New and Revised SRs

   The schedule for performing the new or revised SRs in Amendment No. 278 (BVPS-1) and Amendment No. 161 (BVPS-2) shall be as follows:

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

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

   For SRs that existed prior to this amendment, 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.

   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.

2. Relocation of Certain Technical Specification Requirements

   Amendment No. 278 (BVPS-1) and Amendment No. 161 (BVPS-2) authorize the relocation of certain TSs to other licensee-controlled documents. Implementation of these amendments shall include relocation of the requirements to the specified documents, as described in (1) Sections 4D and 4E of the NRC's staff's SE, and (2) Table LA, Removed Detail Changes, and Table R, Relocated Specifications, attached to the NRC staff's SE, which is enclosed with these amendments.

## 8.0 STATE CONSULTATION

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

## 9.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact was published in the *Federal Register* on February 12, 2007 (72 FR 6611), for the proposed conversion of the CTS to ITS for BVPS-1 and 2. Accordingly, the Commission has determined that issuance of these amendments will not result in any significant environmental impacts other than those evaluated in the Final Environmental Statement for BVPS-1 and 2 dated July 1973 and September 1985, respectively. The Commission also issued a Notice of Consideration of Issuance of Amendment to Facility Operating Licenses and Opportunity for a Hearing on March 22, 2006 (71 FR 14554). There have been no comments or requests for hearing.

## 10.0 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 Standard Acronyms and Abbreviations
2. Table A - Administrative Changes
3. Table L - Less Restrictive Changes
4. Table LA - Removed Details
5. Table M - More Restrictive Changes
6. Table R - Relocated Specifications

Principal Contributors: R. Tjader, P. Hearn, C. Schulten, R. Clark, C. Harbuck, T. Colburn

Date: February 21, 2007

## LIST OF STANDARD ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| AC | Alternating Current |
| ASME | American Society of Mechanical Engineers |
| AFD | Axial Flux Difference |
| AFW | Auxiliary Feedwater |
| AOT | Allowable Outage Time |
| ARO | All Rods Out |
| BSI | Beyond-Scope Issue |
| BVPS | Beaver Valley Power Station |
| CCW | Component Cooling Water |
| CDF | Core Damage Frequency |
| CFR | Code of Federal Regulations |
| CFT | Channel Functional Test |
| CIB | Containment Isolation Phase B |
| COLR | Core Operating Limits Report |
| COT | Channel Operational Test |
| CRAC | Control Room Air Conditioning |
| CRDS | Control Rod Drive System |
| CRDM | Control Rod Drive Mechanism |
| CREVS | Control Room Emergency Ventilation System |
| CSS | Containment Spray System |
| CT | Completion Time (in the ITSs) |
| CTS | Current TSs |
| CVCS | Chemical and Volume Control System |
| DBA | Design-Basis Accident |
| DG | Diesel Generator |
| DIS | Distributed Ignition System |
| DNB | Departure from Nucleate Boiling |
| DOC | Discussion of Change (from the CTS) |
| DVR | Degraded Voltage Relays |
| ECCS | Emergency Core Cooling System |
| EDG | Emergency Diesel Generator |
| EFPD | Effective Full-Power Days |
| EPRI | Electric Power Research Institute |
| EQ | Equipment Qualification |
| ESF | Engineered Safety Features |
| ESFAS | Engineered Safety Feature Actuation System |
| ESFVS | Engineered Safety Feature Ventilation System |
| ESW | Essential Service Water |
| FHA | Fuel Handling Accident |
| FHAEVS | Fuel Handling Area Exhaust Ventilation System |
| FR | Federal Register |
| FSAR | Final Safety Analysis Report |
| GDC | General Design Criterion (of Appendix A to 10 CFR Part 50) |
| GL | Generic Letter |
| HELB | High Energy Line Break |

| | |
|---|---|
| HEPA | High Efficiency Particulate [filter] |
| HZP | Hot-Zero-Power |
| IIP | Inservice Inspection Program |
| IPE | Individual Plant Examination |
| ISI | Inservice Inspection |
| IST | Inservice Testing |
| ISTS | Improved Standard Technical Specifications |
| ITS | Improved TSs |
| JFD | Justification for Deviation |
| LAR | License Amendment Request |
| LBB | Leak Before Break |
| LCO | Limiting Condition for Operation |
| LOCA | Loss-of-Coolant Accident |
| LOOP | Loss-of-Offsite Power |
| LOV | Loss of Voltage |
| LRM | Licensing Requirements Manual |
| MFIV | Main Feedwater Isolation Valve |
| MFRV | Main Feedwater Regulation Valve |
| MOL | Minimum Operating Limit |
| MSIV | Main Steam Isolation Valve |
| MSLB | Main Steamline Break |
| NIC | Nuclear Instrumentation Channels |
| NIS | Nuclear Instrumentation System |
| NRC | Nuclear Regulatory Commission |
| NSAL | Nuclear Safety Advisory Letter |
| NTSP | Nominal Trip Setpoints |
| ODCM | Offsite Dose Calculation Manual |
| OPΔT | Overpower Delta Temperature |
| OTΔT | Overtemperature Delta Temperature |
| PAM | Post Accident Monitoring |
| PBNP | Point Beach Nuclear Plant |
| PORV | Power Operated Relief Valve |
| PRA | Probabilistic Risk Assessment |
| P/T | Pressure/Temperature |
| PTLR | Pressure Temperature Limits Report |
| PWR | Pressurized-Water Reactor |
| QA | Quality Assurance |
| QAPD | Quality Assurance Program Description |
| RAI | Request for Additional Information |
| RCS | Reactor Coolant System |
| RCCA | Rod Cluster Control Assembly |
| RCP | Reactor Coolant Pump |
| RG | Regulatory Guide |
| RHR | Residual Heat Removal |
| RPI | Rod Position Indication |
| RPV | Reactor Pressure Vessel |
| RTP | Rated Thermal Power |
| RTS | Reactor Trip System |
| RWST | Refueling Water Storage Tank |
| SAL | Safety Analysis Limit |

| | |
|---|---|
| SBVS | Shield Building Ventilation System |
| SE | Safety Evaluation |
| SFPSVS | Spent Fuel Pool Special Ventilation system |
| SG | Steam Generator |
| SGTR | Steam Generator Tube Rupture |
| SI | Safety Injection |
| SLCRS | Supplemental Leakage Collection and Release System |
| SR | Surveillance Requirement |
| SSCs | Structures, Systems, and Components |
| SSPS | Solid State Protection System |
| STI | Surveillance Test Interval |
| SW | Service Water |
| TADOT | Trip Actuating Device Operational Test |
| TMD | Transient Mass Distribution |
| TS | Technical Specification |
| TSs | Technical Specifications |
| TSTF | Technical Specification Task Force |
| UFSAR | Updated Final Safety Analysis Report |
| UHS | Ultimate Heat Sink |
| WEC | Westinghouse Electric Company |

# Table A

## Administrative Changes

ML070390019

# Table L

## Less Restrictive Changes

ML070390024

Attachment 3

## Table LA

## Removed Details

ML070390051

Attachment 4

## Table M

## More Restrictive Changes

ML070390057

# Table R

## Relocated Specifications

ML070390061

Attachment 6