July 3, 2002

Mr. Michael Kansler
Sr. Vice President and Chief
Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - AMENDMENT RE: CONVERSION TO IMPROVED TECHNICAL SPECIFICATIONS (TAC NO. MA5049)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 274 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power (FitzPatrick or JAFNPP). The amendment consist of changes to the Technical Specifications (TSs) in response to the application by the Power Authority of the State of New York (PASNY), the former licensee, dated March 31, 1999, as supplemented by letters dated June 1, July 14 and October 14, 1999, February 11, April 4, April 13, June 30, July 31, September 12, September 13, and October 23, 2000. On November 21, 2000, PASNY's ownership interest in FitzPatrick was transferred to Entergy Nuclear FitzPatrick, LLC, to possess and use FitzPatrick and to Entergy Nuclear Operations, Inc. (ENO) to possess, use and operate FitzPatrick. By letter dated January 26, 2001, ENO requested that the Nuclear Regulatory Commission (NRC) continue to review and act on all requests before the Commission which had been submitted by PASNY before the transfer. Accordingly, the NRC staff continued its review of PASNY's responses concerning the issue of the conversion of the current TSs for JAFNPP to a set of improved Technical Specifications (ITSs). Supplements to the application by ENO were submitted by letters dated May 31 and October 18, 2001, February 6, March 27, April 26, June 11, and June 12, 2002 (two letters).

This amendment converts the current TS (CTS) for JAFNPP to a set of ITS based on NUREG-1433, Revision 1, "Standard Technical Specifications for General Electric Plants, BWR/4," dated April 1995 and NUREG-1434, Revision 1, "Standard Technical Specifications for General Electric Plants, BWR/6," dated April 1995.

The draft Safety Evaluation (SE) for the ITS conversion was sent to you by letter dated November 7, 2001, for your review to verify the accuracy of the draft SE. You provided comments by letter dated December 14, 2001. However, by letter dated February 6, 2002, you informed the staff that the final submittal required further engineering analysis and was to be submitted by April 30, 2002. The comments you provided in your December 14, 2001, letter and revised information submitted by your April 26 and June 10, 2002, letters were reviewed and incorporated in the enclosed final SE for the amendments, as appropriate. The enclosed SE also incorporates revisions resulting from the staff's review subsequent to the transmission of the draft SE to you in November 2001.

The ITS conversion will become effective immediately and shall be implemented within 120 days. If there is a request for amendment submitted prior to implementation of the ITS being completed, it will be necessary to submit separate TS pages for both CTS and ITS with the amendment request. The NRC staff is currently reviewing your request for amendment dated January 9, 2002, relating to a proposed TS change regarding allowable main steam isolation valve leakage. You are requested to submit separate TS pages for the ITS for that amendment request.

You are requested to submit a letter stating that the ITS are implemented within 14 days of the date of the completed action.

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

Sincerely,

/RA/

Guy S. Vissing, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures: 1. Amendment No. 274 to DPR-59
2. Safety Evaluation

cc w/encls: See next page

The ITS conversion will become effective immediately and shall be implemented within 120 days. If there is a request for amendment submitted prior to implementation of the ITS being completed, it will be necessary to submit separate TS pages for both CTS and ITS with the amendment request. The NRC staff is currently reviewing your request for amendment dated January 9, 2002, relating to a proposed TS change regarding allowable main steam isolation valve leakage. You are requested to submit separate TS pages for the ITS for that amendment request.

You are requested to submit a letter stating that the ITS are implemented within 14 days of the date of the completed action.

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

Sincerely,

/RA/

Guy S. Vissing, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures: 1. Amendment No. 274 to DPR-59
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

PUBLIC	OGC	NLe	GThomas	CHarbuck
PDI-1 R/F	ACRS	BMarcus	PGills	BThomas
SRichards	GHill (2)	JLazevnick	HLi	KKavanagh
RLaufer	WBeckner	RGoel	ETomlinson	
GVissing	BPlatchek, RGNI	AChu	RTjader	
SLittle	CSchulten	JFoster	RGiardina	

Package No.: ML021980178
TSs: ML021970028, ML021970071, ML021970084, ML021970095, ML021970138, ML021970142, ML021970092, ML021970102
Accession No.: ML021620504 *See previous concurrence

OFFICE	PM/PDI-1	LA/PDI-1	DR/RORP	OGC*	SC/PDI-1
NAME	GVissing	TClark for SLittle	WBeckner	RWeisman	RLaufer
DATE	7/3/02	7/3/02	7/3/	July/2 /02	7/3/02

OFFICIAL RECORD COPY

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 274
License No. DPR-59

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

   A. The application for amendment by the Power Authority of the State of New York (PASNY) (the former licensee) dated March 31, 1999, as supplemented by letters dated June 1, July 14, and October 14, 1999, February 11, April 4 and 13, June 30, July 31, September 12 and 13, and October 23, 2000, and letters from Entergy Nuclear Operations, Inc. (ENO), dated May 31 and October 18, 2001, and February 6, March 27, April 26, and June 11 and 12 (two letters), 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission’s rules and regulations set forth in 10 CFR Chapter 1.*

   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

---

*On November 21, 2000, the operating license for the James A. FitzPatrick Nuclear Power Plant (FitzPatrick) was transferred to Entergy Nuclear FitzPatrick, LLC, which is authorized to possess and use FitzPatrick, and to ENO, which is authorized to possess, use, and operate FitzPatrick. By letter dated January 26, 2001, ENO requested that the NRC continue to review and act on all requests before the Commission which had been submitted by PANSY before the transfer. Accordingly, the NRC staff continued its review of the application for this amendment, as supplemented.

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-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 274, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

Also, the license is amended by modifications to 2.C(3) of Facility Operating License DPR-59 as follows:

(4) Systems Integrity
Deleted by Amendment No. 274

(5) Iodine Monitoring
Deleted by Amendment No. 274

(6) New or Revised ITS Surveillance Requirements Appicability

The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 274 shall be as follows:

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

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

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

(d) For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance 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.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of the date of issuance. In implementing the amendment, the licensee shall relocate the Technical Specification requirements identified in Table LA- "Removal

of Details Matrix" and Table R - "Relocated Specifications" to licensee-controlled documents, as described in the application as supplemented on June 12, 2002, and the NRC staff's Safety Evaluation enclosed with Amendment No. 274, dated July 3, 2002. Further, relocations to the updated Final Safety Analysis Report (UFSAR) shall be reflected in the next UFSAR update required by 10 CFR 50.71(e) following implementation of this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachments: 1. Pages 4 and 4a to License No. DPR-59
2. Changes to the Technical Specifications

Date of Issuance: July 3, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 274

TO FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Replace the following pages of the License and Appendix "A" Technical Specifications with the enclosed pages as indicated. Marginal lines indicate changed areas on the license pages.

Remove Pages	Insert Pages
License Page 4	License Page 4
---	License Page 4a
Current TSs (in their entirety)	Improved TS (in their entirety)

(4) Systems Integrity
Deleted by Amendment No. 274

(5) Iodine Monitoring
Deleted by Amendment No. 274

(6) New or Revised ITS Surveillance Requirements Applicability:

The schedule for performing Surveillance Requirements (SRs) that are new or revised in Amendment No. 274 shall be as follows:

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

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

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

(d) For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance is die 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.

D. Physical Protection

ENO shall fully implement and maintain in effect all provisions of the Commission-approved physical security guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements.

Revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the autority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "FitzPatrick Modified Amendment Security Plan," with revisions submitted through March 7, 1988; "FitzPatrick Modified Amended Security Force Training and Qualification Plan," with revisions submitted through April 10, 1985; and "FitzPatrick Security Contingency Plan," with revisions submitted through June 20, 1980. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

E. Power Uprate License Amendment Implementaton

The licensee shall complete the following actions as a condition of the approval of the power uprate license amendment.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 274 TO FACILITY OPERATING LICENSE NO. DPR-59

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-333

1.0 INTRODUCTION

The James A. FitzPatrick Nuclear Power Plant (FitzPatrick or JAFNPP) has been operating with Technical Specifications (TSs) issued with the full power operating license (DPR-59) on October 17, 1974, as amended. By application dated March 31, 1999, as supplemented by letters dated June 1, July 14, and October 14, 1999, February 11, April 4, April 13, June 30, July 31, September 12, September 13, and October 23, 2000, the Power Authority of the State of New York (PASNY), the former licensee, proposed to convert the current Technical Specifications (TSs) to improved TSs. On November 21, 2000, PASNY's ownership interest in FitzPatrick was transferred to Entergy Nuclear FitzPatrick, LLC, to possess and use FitzPatrick and to Entergy Nuclear Operations, Inc. (ENO) to possess, use and operate FitzPatrick. By letter dated January 26, 2001, ENO requested that the NRC continue to review and act on all requests before the Commission which had been submitted by PASNY before the transfer. Accordingly, the Nuclear Regulatory Commission (NRC) staff continued its review of PASNY's responses concerning the requested conversion of the current TSs for the JAFNPP to a set of improved TSs. Supplements to the application by ENO were submitted by letters dated May 31 and October 18, 2001, and February 6, March 27, April 26, June 11, and June 12, 2002 (two letters). The conversion to the improved TS is based upon:

- NUREG-1433, "Standard Technical Specifications for General Electric Plants, BWR/4," Revision 1, dated April 1995,

- NUREG-1434, "Standard Technical Specifications for General Electric Plants, BWR/6," Revision 1, dated April 1995,

- Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (Final Policy Statement), published on July 22, 1993 (58 FR 39132),

- 10 CFR 50.36, "Technical Specifications," as amended July 19, 1995 (60 FR 36953).

- The current FitzPatrick TS.

Hereinafter, the proposed TS for FitzPatrick (or JAFNPP) are referred to as the improved TS (ITS), the current TS are referred to as the current TS (CTS), and the improved standard TS, such as in NUREG-1433 or NUREG-1434, are referred to as the STS. The corresponding TS

Bases are ITS Bases, CTS Bases, and STS Bases, respectively. For convenience, a list of acronyms used in this safety evaluation (SE) is provided in Attachment 1.

In addition to basing the ITS on the STS, the Final Policy Statement, and the requirements in 10 CFR 50.36, the licensee retained portions of the CTS as a basis for the ITS. Plant-specific issues, including design features, regulatory requirements, and operating practices, were discussed with the licensee during a meeting on April 14, 1999. During the review of the application, the NRC staff requested additional information on these issues in writing and in a series of follow-up telephone conference calls. Information discussed with the licensee during follow-up phone calls has been incorporated in the staff's requests for additional information and the licensee's supplemental submittals. These plant-specific changes clarify the ITS with respect to the guidance in the Final Policy Statement and STS.

Also, based on these discussions, the licensee proposed generic editorial and format changes that differed from the STS. The NRC staff requested that the licensee address these matters by submitting them as proposed changes to STS through the NRC/Nuclear Energy Institute's Technical Specifications Task Force (TSTF). These matters were considered for specific application in the ITS.

Consistent with the Final Policy Statement, the licensee proposed transferring some CTS requirements to licensee-controlled documents, such as the updated final safety analysis report (UFSAR) for JAFNPP. Licensee changes to such documents are controlled by regulation, such as 10 CFR 50.59, and a licensee may make such changes without prior NRC approval, provided that they meet specified criteria. (NRC-controlled documents, such as the TS, may not be changed by the licensee without prior NRC approval.) In addition, human factors principles were emphasized to add clarity to the CTS requirements being retained in the ITS, and to define more clearly the appropriate scope of the ITS. Further, significant changes were proposed to the CTS Bases to make each ITS requirement clearer and easier to understand.

The overall objective of the proposed amendment, consistent with the Final Policy Statement, is to rewrite, reformat, and streamline the CTS to be in accordance with 10 CFR 50.36.

Since the licensee submitted the March 31, 1999, application, a number of amendments to the JAFNPP operating license were approved. Table 1 provides the subjects of the amendments and the dates of issuance.

TABLE 1

Amendment No.	Description of Change	Date
252	Revise Section 6.0	04/12/99
253	Extend Allowed Outage Time (AOT) for Emergency Diesel Generators (EDGs) from 7 to 14 days	07/30/99

Amendment No.	Description of Change	Date
254	Revise Section 6.0 and Appendix B	09/13/99
255	Revisions to actions to be taken in event multiple control rods are inoperable	09/21/99
256	Provide additional storage racks to increase spent fuel capacity	11/10/99
257	Revise calibration requirements for Local Power Range Monitor (LPRM)	11/22/99
258	Pressure and Temperature Limits	11/29/99
259	Extend the AOT time for Residual Heat Removal (RHR) Service Water System	01/28/00
260	Delete Section 4.7.D.1.e to eliminate requirement for partial stroking the plant Main Steam Isolation Valves (MSIVs)	02/24/00
261	Changes Standby Gas Treatment (SBGT) filter efficiency	04/14/00
262	Preclude Applicability of TSs 3.0.D and 4.0.D	09/29/00
263	Trip Level Setting for RHR, Core Spray (CS), and Automatic Depressurization System (ADS) Pumps Start Timers	10/04/00
264	Revise reactor water level set point for Anticipated Transient Without Scram (ATWS), Recirculation pump trip, and alternate rod insertion functions	10/10/00
265	Revise MSIV closures scram trip level setting	10/10/00
266	Minimum Critical Power Ratio Safety Limit	10/30/00
267	Leakage and Hydrostatic Testing Condition	11/03/00
268	Transfer of License to Entergy Nuclear Operations, Inc	11/21/00

Amendment No.	Description of Change	Date
269	Revise TS Surveillance testing requirements of the Charcoal adsorbors to meet GL 99-02	02/05/01
270	Relocate “Offgas Treatment System Explosive Gas Mixing Instrumentation” to Administrative Section 6 of the TS	04/18/01
271	One time change of out of service time allowance for the residual heat removal system service water system	07/27/01
272	One time change to out-of-service time for one incoming reserve AC power line inoperable from 7 days to 14 days commencing September 9, 2001	09/15/01
273	Revise ATWS Recirculation Pump Trip instrumentation setpoint	05/08/02

ENO has incorporated these amendments, as appropriate, into the ITS by supplemental submittal letters dated May 31, August 6, October 18, 2001, and June 11, 2002.

The NRC staff's evaluation of the licensee’s application dated March 31, 1999 (JNP-99-008), as supplemented by letters dated June 1, July 14, and October 14, 1999, February 11, April 4, April 13, June 30, July 31, September 12, September 13, and October 23, 2000, May 31 and October 18, 2001, and February 6, March 27, April 26, June 11, and June 12, 2002 (two letters), is presented in this SE. The NRC staff requested additional information from the licensee in RAIs dated December 10, 1999, February 9, 2000, and June 14, 2000.

Two license conditions for implementing the conversion will make enforceable the following aspects of the conversion: (1) relocation of requirements from the CTS to licensee-controlled documents and (2) an implementation schedule for new and revised surveillance requirements (SRs) in the ITS.

In addition, two license conditions are deleted because the first condition’s requirements have been incorporated into the ITS, and the second condition’s requirements need not be located in the TS pursuant to 10 CFR 50.36; rather they may be relocated to plant-controlled documents.

The Commission issued notices of the proposed action on the JAFNPP application for an amendment dated March 31, 1999, which were published in the *Federal Register* on November 8, 1999 (64 FR 60854), December 13, 1999 (64 FR 69574), and on November 28, 2001 (66 FR 59595). The *Federal Register* notices also addressed changes outsite the scope of converting to ITS (beyond-scope issues) identified in the licensee’s supplemental submittals and by the staff during its review of the submittals. The letters subsequent to the *Federal Register* of

November 28, 2001, did not change the technical content of the *Federal Register* notices, and did not change the scope of the proposed action.

During its review, the NRC staff relied on the Final Policy Statement and the STS as guidance for acceptance of CTS changes. This SE provides a summary basis for the NRC staff's conclusion that the licensee can develop ITS based on the STS, as modified by plant-specific changes, and that the use of the ITS is acceptable for continued operation. The SE also explains the NRC staff's conclusion that the ITS are consistent with the JAFNPP current licensing basis and the requirements of 10 CFR 50.36.

The NRC staff acknowledges that, as indicated in the Final Policy Statement, the conversion to STS is a voluntary process. Therefore, it is acceptable that the ITS differ from the STS to reflect the current licensing basis for JAFNPP.

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

2.0 BACKGROUND

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

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

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

For several years, NRC and industry representatives sought to develop guidelines for improving the content and quality of nuclear power plant TS. On February 6, 1987, the Commission issued an interim policy statement on TS improvements, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, the utility owners groups and the NRC staff developed improved STS, such as NUREG-1433 for GE BWR/4's or NUREG-1434 for GE BWR/6's, that would establish models of the Commission's policy for each primary reactor type. In addition, the NRC staff, licensees, and owners groups developed generic administrative and editorial guidelines in the form of a "Writer's Guide" for preparing TS, which gives greater consideration to human factors principles and was used throughout the development of licensee-specific ITS.

In September 1992, the Commission issued NUREG-1433 and NUREG-1434, which were developed using the guidance and criteria contained in the Commission's Interim Policy Statement. The STS in NUREG-1433 and NUREG-1434 were established as models for developing the ITS for GE BWR/4 and GE BWR/6 plants in general. The STS reflect the results of a detailed review of the application of the interim policy statement criteria to generic system functions, which were published in a "Split Report" issued to the nuclear steam system supplier owners groups in May 1988. STS also reflect the results of extensive discussions concerning various drafts of STS, so that the application of the TS criteria and the Writer's Guide would consistently reflect detailed system configurations and operating characteristics for all reactor designs. As such, the generic Bases presented in NUREG-1433 and NUREG-1434 provide an abundance of information regarding the extent to which the STS present requirements that are necessary to protect public health and safety. Because the JAFNPP design includes both BWR-4 and BWR-6 design elements, the STS in NUREG-1433 and NUREG-1434 apply to it.

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 (58 FR 39132). The Final Policy Statement described the safety benefits of the STS, and encouraged licensees to use the STS as the basis for plant-specific TS amendments, and for complete conversions to ITS based on the STS. Further, the Final Policy Statement gave guidance for evaluating the required scope of the TS and defined the guidance criteria to be used in determining which of the LCOs and associated SRs should remain in the TS. The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TS, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

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

which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

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

Criterion 1

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

Criterion 2

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

Criterion 3

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

Criterion 4

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

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

3.0 EVALUATION

In its review of the application, the NRC staff evaluated the proposed changes to the JAFNPP CTS, which the licensee divided into five categories, as set forth below. As part of the review, the NRC staff evaluated whether existing regulatory requirements would be adequate for controlling future changes to requirements removed from the CTS and placed in licensee-controlled documents.

During the review, the NRC staff also identified the need for clarifications and additions to the March 31, 1999, application in order to establish an appropriate regulatory basis for translation of CTS requirements into ITS. Each change proposed in the amendment request is identified as either a discussion of change (DOC) to the CTS or a justification for difference from the STS. The NRC staff's questions regarding the application were documented as RAIs and forwarded to the licensee in letters dated December 10, 1999, February 9, 2000, and June 14, 2000. The licensee provided responses to the RAIs in letters dated February 11, 2000, April 4, 2000, April 13, 2000, June 30, 2000, July 31, 2000, September 12, 2000, September 13, 2000, December 1, 2000, May 31, 2001, August 6, 2001, and October 18, 2001. The letters clarified the licensee's bases for translating the CTS requirements into ITS. The NRC staff finds that the licensee's submittals, including the responses to the RAIs, provide detail sufficient to allow the staff to reach a conclusion regarding the adequacy of the licensee's proposed changes to the CTS.

The license amendment application was organized such that changes were included in each of the following CTS change categories, as appropriate:

(1) Administrative Changes (A) are changes to the CTS that do not result in new requirements or change operational restrictions or flexibility;

(2) Technical Changes - More Restrictive, (M) are changes to the CTS that establish a new requirement, require new or more frequent testing, or reduce operational flexibility;

(3) Technical Changes - Less Restrictive (specific), (L) are changes that eliminate existing requirements, require less or less frequent testing, or increase operational flexibility;

(4) Technical Changes - Less Restrictive (removal of details), (LA) are changes that relocate details out of the CTS and into the Bases, UFSAR, or other appropriate licensee-controlled document. These changes are less restrictive because they result in a less restrictive change control process and a reduced level of regulatory oversight;

(5) Relocated Specifications, (R), i.e., relaxations in which whole specifications are removed from the CTS (based an application of the four criteria specified in 10 CFR 50.36) and placed in licensee-controlled documents.

The changes that are in the ITS conversion for JAFNPP for each of the above categories are listed in the following five tables (matrixes) attached to this SE:

- Table A of Administrative Changes to Current Technical Specifications
- Table M of More Restrictive Changes to Current Technical Specifications
- Table L of Less Restrictive Changes to Current Technical Specifications
- Table LA of Relocated Details from Current Technical Specifications
- Table R of Relocated Specifications from Current Technical Specifications

The tables are only meant to summarize the changes being made to the CTS. The details, as to what the actual changes are and how they are being made to the CTS or ITS, are provided in the licensee’s application and supplemental letters.

The general categories of changes to the licensee's CTS requirements and STS differences may be better understood as follows:

A. Administrative Changes

Administrative (non-technical) changes are intended to incorporate human factors principles into the form and structure of the ITS so that plant operations personnel can use them more easily. These changes are editorial in nature or involve the reorganization or reformatting of CTS requirements without affecting technical content or operational restrictions. Every section of the ITS reflects this type of change. In order to ensure consistency, the NRC staff and the licensee have used the STS as guidance to reformat and make other administrative changes. The licensee proposed, and the NRC staff found acceptable, changes:

(1) Identifying plant-specific wording for system names, etc.;

(2) Splitting up requirements currently grouped under a single current specification to more appropriate locations in two or more specifications of ITS;

(3) Combining related requirements currently presented in separate specifications of the CTS into a single specification of ITS;

(4) Rewording or reformatting TSs for clarity (including moving an existing requirement to another location within the TS) but which do not involve a change in substantive requirements;

(5) Rewording STS and adding ITS that are consistent with CTS interpretation and practice, and that more clearly or explicitly state existing requirements;

(6) Deleting TS whose applicability has expired; and

(7) Deleting redundant TS requirements.

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

B. Technical Changes - More Restrictive

The licensee, in electing to implement the specifications of the STS, proposed a number of requirements more restrictive than those in the CTS. The ITS requirements in this category include requirements that are new, are more conservative than corresponding requirements in the CTS, or have additional restrictions that are not in the CTS but are in the STS. Examples of more restrictive requirements include (1) LCOs on plant equipment which is not required by the CTS to be operable, (2) more restrictive requirements for restoring inoperable equipment, and (3) more restrictive SRs. Table M lists the more restrictive changes being made in the JAFNPP ITS conversion. Table M is organized in ITS order by each M-type DOC to the CTS and provides a summary description of each proposed more restrictive change, and the CTS and ITS references. These changes are additional restrictions on plant operation that enhance safety and are acceptable.

C. Technical Changes - Less Restrictive

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

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

Type 1 — Relaxation of LCO Requirement

Type 2 — Relaxation of Applicability

Type 3 — Relaxation of Surveillance Requirement

Type 4 — Relaxation of Required Action Detail

Type 5 — Relaxation of Required Actions to Exit Applicability

Type 6 — Relaxation of Completion Time

Type 7 — Allow Mode Changes When LCO Not Met

Type 8 — Elimination of the Requirement to Lock the Reactor Mode Switch in Shutdown or Refuel

Type 9 — Elimination of CTS Reporting Requirement

The following discussion addresses why the various types of changes are acceptable.

Type 1 - Relaxation of the LCO Requirement

Certain CTS LCOs contain limits on operational and system parameters beyond those necessary to meet safety analysis assumptions and therefore are considered overly restrictive. CTS also contain limits that have been shown to give little or no safety benefit to the operation of the plant. The ITS, consistent with the guidance in the STS, delete or revise operating limits of this type. CTS LCO changes included in this type are: (1) revising setpoints to be consistent with instrument setpoint methodologies; (2) deleting or revising operational limits to establish requirements consistent with applicable safety analyses; (3) deleting requirements for equipment or systems which establish system capability beyond that assumed to function by the applicable safety analyses or which are implicit in 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.

This type of TS change allows operators to focus more clearly on issues important to safety, consistent with the STS. 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. Accordingly, these changes are acceptable.

Type 2 - Relaxation of Applicability

The CTS require compliance with an LCO during the Operational Mode(s) or other conditions specified in the LCO Applicability statement. Five Operating Modes are defined by ITS according to average reactor coolant temperature, the position of the reactor mode switch located in the control room, and reactor vessel head closure bolt tensioning. These are: Power Operation, Startup, Hot Shutdown, Cold Shutdown, and Refueling. When CTS Applicability requirements are inconsistent with the applicable accident analyses assumptions for a system, subsystems, or component specified in the LCO, the LCO is changed in the ITS to establish a consistent set of requirements. These modifications or deletions are acceptable because, during the conditions referenced in the ITS, the operability requirements are consistent with the applicable safety analyses. These changes are also consistent with STS.

Type 3 — Relaxation of Surveillance Requirements

CTS require maintaining LCO equipment operable by meeting SRs in accordance with the specified SR Frequency. This requires conducting tests to demonstrate equipment is operable, or that LCO parameters are within specified limits. When the test acceptance criteria and any specified conditions for the conduct of the test are met, the equipment is deemed operable. The changes in this type relate to relaxation of CTS SR acceptance criteria and/or the conditions for performing the SR.

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

Relaxing conditions for performing SRs include, for example, 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). The changes also include the allowance to verify the position of valves in high radiation areas by administrative means. For such valves, ITS administrative controls (ITS 5.7) regarding access to high radiation areas make the likelihood of mispositioning small. These changes are acceptable because the changes do not affect the licensee’s ability to determine whether equipment is capable of performing its intended safety function.

These relaxations of CTS SRs optimize test requirements for the affected safety systems and increase operational flexibility. These changes are also consistent with the STS. Based on the above, these changes are acceptable.

Type 4 — Relaxation of Required Action Detail

LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, CTS specify actions to be taken until the equipment is restored to its required capability or performance level, or remedial measures are established. In revising the Required Actions, details are deleted or options are added such that resulting ITS actions continue to provide measures that conservatively compensate for the inoperable equipment consistent with the STS. Furthermore, adopting STS action requirements results in simpler, more concise and more direct action requirements. This allows more effective use of operator resources for placing and maintaining the reactor in a safe condition when the LCO is not met. Accordingly, these changes are acceptable.

Type 5 — Relaxation of Required Actions to Exit Applicability

LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is not met, CTS specify actions to be taken until the equipment is restored to its required capability or performance level, or remedial measures are established. Compared to CTS required actions, the ITS actions result in extending the time period before the licensee is required to shut down the plant. For example, changes of this type include providing an option to isolate a system, place equipment in the state assumed by the safety analysis, satisfy alternate criteria, take manual actions in place of automatic actions, “restore to operable status” within a specified time frame, place alternate equipment into service, or use more conservative TS setpoints consistent with the STS. The resulting ITS actions continue to provide measures that conservatively compensate for the inoperable equipment. The ITS actions are commensurate with the safety importance of the inoperable equipment, plant design, and industry practice and do not compromise safe operation of the plant. Accordingly, these changes are acceptable.

Type 6 — Relaxation of Completion Time

Upon discovery of a failure to meet an LCO, TS specify times for completing Required Actions of the associated TS conditions. Required Actions establish remedial measures that must be taken within specified completion times (allowed outage times). These times define limits during which operation in a degraded condition is permitted.

Incorporating completion time extensions is acceptable because completion times take into account the operability status of the redundant systems of TS required features, the capacity and capability of remaining features, a reasonable time for repairs or replacement of required features, vendor-developed standard repair times, and the low probability of a design-basis accident (DBA) occurring during the repair period, consistent with the STS. Accordingly, these allowed outage time extensions are acceptable.

Type 7 — Allow Mode Changes When LCO Not Met

CTS 3.0.D (ITS LCO 3.0.4) precludes entry into the applicable Mode or other specified conditions while relying on the Actions, even though the Actions are designed to provide for safe operation of the plant. Unless otherwise stated, ITS LCO 3.0.4 is always applicable to action requirements associated with ITS LCOs. However, the ITS add a Note to certain Actions stating “LCO 3.0.4 is not applicable.” The addition of this Note allows transition between Modes or other specified conditions with the LCO not met (i.e., relying on the Actions) even though the Actions may require plant shutdown. The addition of “LCO 3.0.4 is not applicable” notes does not impact normal operation of the plant for the specified LCO features and would not provide additional initiators for plant transients during the Mode or other specified conditions consistent with the STS. This exception to ITS 3.0.4 is acceptable due to the passive function or the installed redundancy of the features, the plant conditions that apply to the Notes, and the low probability of an event requiring the inoperable features. Accordingly, these changes are acceptable.

Type 8 — Elimination of the Requirement to Lock the Reactor Mode Switch in Shutdown or Refuel

Some CTS LCOs and action requirements specify "lock" the mode switch in "Shutdown" (shutdown position) or "Refuel" (refueling position). Other CTS action requirements also specify placing the reactor in the shutdown or refueling Mode without requiring the mode switch to be "locked." The requirement to "lock" the mode switch in Shutdown or Refuel is not retained in the ITS. CTS Definitions 1.0.C, 1.0.D, 1.0.I, and 1.0.O (ITS Table 1.1-1) defines reactor operational Modes based on the reactor mode switch position, among other things. Moving the reactor mode switch from Shutdown into a position other than Shutdown or Refuel may cause a Mode change as defined by TS, and results in associated TS compliance requirements for the LCOs that become applicable in the new Mode. CTS 3.0.D (ITS LCO 3.0.4) precludes changes in reactor Modes without all TS required equipment being operable consistent with the STS. Thus, ITS LCO 3.0.4 is an administrative requirement put in place to prevent movement of the reactor mode switch between positions without first ensuring TS required equipment is operable, and changing the mode switch from the required position is adequately controlled by ITS Table 1.1-1 without adding a requirement to "lock" the mode switch. Accordingly, these changes are acceptable.

Type 9 — Elimination of CTS Reporting Requirement

CTS include requirements to submit special reports to the NRC when specified limits or conditions are not met. Typically, the time period for the report to be issued is "within 30 days." However, the ITS eliminates the TS requirements for special reports and instead relies on the reporting requirements of 10 CFR 50.73 consistent with the STS. The changes to the reporting requirements are acceptable because 10 CFR 50.73 provides adequate reporting requirements, and the special reports do not affect continued plant operation. CTS also include requirements for reports to be made to the NRC on data gathered as part of routine plant programs. These requirements have no impact on the safe operation of the plant and can be removed from the ITS.

Deleting TS reporting requirements reduces unnecessary regulatory burden on the licensee and allows the licensee to concentrate its efforts on maintaining operation within TS required limits. Based on the above, these changes are acceptable.

D. Technical Changes — Less Restrictive Removal of Details (LA)

When requirements have been shown to give little or no safety benefit, their removal from the TS may be appropriate. These are grouped as LA changes. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups comments on STS. The NRC staff reviewed generic relaxations contained in the STS, as applied at JAFNPP, and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The JAFNPP design was also reviewed to determine if the specific design basis and licensing basis are consistent with the technical basis for the model requirements in the STS and thus provide a basis for ITS. A significant number of changes to the CTS involved the removal of specific requirements and detailed information from individual specifications evaluated to be Types 1 through 3 that follow:

Type 1 — Details of System Design and System Description including Design Limits

Type 2 — Descriptions of System Operation

Type 3 — Procedural Details for Meeting TS Requirements, Relocated Reporting Requirements, and Relocated Specification Requirements

The following discussion addresses why each of the three types of information or requirements is not required to be included in ITS.

Type 1 — Details of System Design and System Description Including Design Limits

The design of the facility is required to be described in the Updated Final Safety Analysis Report (UFSAR) by 10 CFR 50.34. In addition, the quality assurance (QA) requirements of Appendix B to 10 CFR Part 50 require that plant design be documented in controlled procedures and drawings and maintained in accordance with an NRC-approved QA Program (UFSAR Chapter 17). In 10 CFR 50.59, controls are specified for changing the facility as described in the UFSAR (including the Technical Requirements Manual (TRM)); in 10 CFR 50.54(a), criteria are specified for changing the QA Program. The ITS Bases also contain descriptions of system design. ITS 5.5.11 specifies controls for changing the Bases. Removing details of system design from the CTS is acceptable because this information will be adequately controlled in the UFSAR (including TRM) in accordance with 10 CFR 50.59 or the ITS Bases, as appropriate. In addition, cycle-specific design limits are contained in the Core Operating Limits Report (COLR). ITS Section 5.6, Administrative Controls, includes the programmatic requirements for the COLR which have not been changed in substance from the CTS.

Type 2 — Descriptions of System Operation

The plans for the normal and emergency operation of the facility are required to be described in the UFSAR by 10 CFR 50.34. ITS 5.4.1.a requires written procedures to be established, implemented, and maintained for plant operating procedures, including procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978. Controls specified in 10 CFR 50.59 apply to changes in procedures as described in the UFSAR. The ITS Bases also contain descriptions of system operation.

It is acceptable to remove details of system operation from the TS because this type of information will be adequately controlled in the UFSAR (including TRM) and the TS Bases, as appropriate.

Type 3 — Procedural Details for Meeting TS Requirements, Relocated Reporting Requirements, and Relocated Specification Requirements

Details for performing TS Actions and SRs are more appropriately specified in the plant procedures required by ITS 5.4.1, and described in the UFSAR and ITS Bases. For example, control of the plant conditions appropriate to perform a surveillance test is an issue for procedures and scheduling and, as described in Generic Letter (GL) 91-04, has been determined to be unnecessary as a TS restriction. As set forth in GL 91-04 the vast majority of SRs do not dictate plant conditions for surveillances; rather such plant conditions are controlled by procedures. Further, prescriptive procedural information in an ITS action requirement is unlikely to contain all procedural considerations necessary for the plant operators to complete the actions required, and referral to plant procedures is therefore required in any event.

Other changes to procedural details include those associated with limits retained in the ITS. For example, the ITS requirement may refer to programmatic requirements such as that for the COLR, included in ITS Section 5.6, which specifies the scope of the limits contained in the COLR and mandates NRC approval of the analytical methodology; the QA Program, which is approved by the NRC and contained in UFSAR Chapter 17, and changes to which are controlled by 10 CFR 50.54(a); the Offsite Dose Calculation Manual (ODCM), which is required by ITS 5.5.1; the TRM, which is incorporated by reference into the UFSAR, and changes to which are controlled by 10 CFR 50.59; and the Inservice Testing (IST) Program which is required by ITS 5.5.7.

CTS specification requirements, including LCO, action, and surveillance requirements, have been relocated in adopting the STS. For example, certain power operated isolation valves that do not receive an automatic isolation signal and for which the closure time is not assumed in the safety analysis are not required to be TS under 10 CFR 50.36, and requirements for their periodic testing are moved to the procedures that implement the IST Program (controlled by 10 CFR 50.55a) or the TRM (controlled by 10 CFR 50.59). In addition, support system specification requirements for other equipment with its own specifications are moved to the TRM. The definition of operability, as applied to the supported features, provides sufficient assurance that the supporting system can perform its required support function.

> The removal of these kinds of procedural details from the CTS is acceptable because they will be adequately controlled in the UFSAR (including TRM), ITS Bases, and ITS administrative control and reporting requirements (e.g., COLR), as appropriate. This approach provides an effective level of regulatory control and provides for a more appropriate change control process.

Table LA describes the information that is removed from individual CTS requirements and relocated to ENO-controlled documents. Table LA is organized by ITS section and includes the following: a DOC identification number referenced to ITS Section; a CTS reference; a summary description of the requirement; the document that retains the relocated CTS requirements; and the specific change type, as discussed above.

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

(1) TS Bases controlled by ITS 5.5.11, "Technical Specifications Bases Control Program."

(2) UFSAR (includes TRM by reference) controlled by 10 CFR 50.59.

(3) ODCM controlled by ITS 5.5.1, “Offsite Dose Calculation Manual.”

(4) QA Program controlled by 10 CFR 50.54(a).

(5) Inservice Testing Program controlled by ITS 5.5.7, “Inservice Testing Program,” and 10 CFR 50.55a and 50.59.

(6) Plant Procedures controlled by the FitzPatrick procedure control process.

To the extent that requirements and information have been relocated to ENO-controlled documents, such information and requirements are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, where such information and requirements are contained in LCOs and associated requirements in the CTS, the NRC staff has concluded that they do not fall within any of the four criteria in 10 CFR 50.36 (discussed in Part 2.0 of this SE). Accordingly, existing detailed information and specific requirements, such as generally described above, may be relocated from the CTS.

E. Relocated CTS Specifications

The Final Policy Statement states that LCOs and associated requirements that do not satisfy or fall within any of the four specified criteria (now contained in 10 CFR 50.36) may be relocated from existing TS (an NRC-controlled document) to appropriate licensee-controlled documents.

This section of the SE discusses the relocation of entire specifications in the CTS to licensee-controlled documents. These specifications include LCOs, Action Statements (i.e., Actions), and associated SRs. In its application and its supplements, the licensee proposed relocating such specifications from the CTS to the UFSAR, which includes the TRM, the Process Control Program (PCP), and the ODCM, as appropriate. The staff has reviewed the licensee's submittals, and finds that relocation of these requirements to the UFSAR, TRM, PCP, and ODCM is acceptable in that changes to the UFSAR, TRM, PCP, and ODCM will be adequately controlled by 10 CFR 50.59 and ITS 5.5.1 as applicable. These provisions will continue to be implemented by appropriate station procedures (i.e., operating procedures, maintenance procedures, surveillance and testing procedures, and work control procedures).

Table R lists all specifications that are being relocated from the CTS to licensee-controlled documents. Table R includes: (1) references to the DOCs, (2) references to the relocated CTS specifications, (3) summary descriptions of the relocated CTS specifications, (4) names of the documents that will contain the relocated specifications (i.e., the new location), and (5) the methods for controlling future changes to the relocated specifications (i.e., the regulatory control process).

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

1. 3/4.2.C CONTROL ROD BLOCK ACTUATION - Average Power Range Monitor (APRM)

   The Average Power Range Monitor (APRM) control rod blocks function to limit control rod withdrawal errors during power range operations utilizing local power range monitor (LPRM) signals to create the APRM rod block signal. APRMs provide information about the average core power and APRM rod blocks are not used to mitigate a DBA or transient. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Control Rod Block LCO and Surveillances applicable to APRM Instrumentation may be relocated to other plant controlled documents outside the ITS.

2. 3/4.2.C CONTROL ROD BLOCK ACTUATION - Intermediate Range Monitor (IRM)

   The Intermediate Range Monitor (IRM) control rod blocks function to limit control rod withdrawal errors during reactor startup utilizing IRM signals to create the rod block signal. IRMs are provided to monitor the neutron flux levels during refueling, shutdown, and startup conditions. No DBA or transient analysis takes credit for rod block signals initiated by IRMs. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Control Rod Block LCO and Surveillances applicable to IRM Instrumentation may be relocated to other plant controlled documents outside the ITS.

3. 3/4.2.C CONTROL ROD BLOCK ACTUATION - Source Range Monitor (SRM)

The Source Range Monitor (SRM) control rod blocks function to limit control rod withdrawal errors during reactor startup utilizing SRM signals to create the rod block signal. SRM signals are used to monitor neutron flux during refueling, shutdown and startup conditions. No DBA transient analysis takes credit for rod block signals initiated by the SRMs. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Control Rod Block LCO and Surveillances applicable to SRM Instrumentation may be relocated to other plant controlled documents outside the ITS.

4. 3/4.2.C CONTROL ROD BLOCK ACTUATION - Scram Discharge Instrument Volume High Water Level (SDVHWL)

The Scram Discharge Instrument Volume High Water Level (SDVHWL) control rod block functions to prevent control rod withdrawals, utilizing SDVHWL signals to create the rod block signal if water is accumulating in the scram discharge instrument volume. The purpose of measuring the scram discharge instrument volume water level is to ensure that there is sufficient volume to contain the water discharged by the control rod drives during a scram, thus ensuring that the control rods will be able to insert fully. This rod block signal provides an indication to the operator that water is accumulating in the scram discharge instrument volume and prevents further rod withdrawals. With continued water accumulation, a reactor protection system initiated scram signal will occur. Thus, the SDVHWL rod block signal provides an opportunity for the operator to take action to avoid a subsequent scram. No DBA or transient takes credit for rod block signals initiated by the SDVHWL instrumentation. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Control Rod Block LCO and Surveillances applicable to SDVHWL Instrumentation may be relocated to other plant controlled documents outside the ITS.

5. 3/4.2.H ACCIDENT MONITORING INSTRUMENTATION

Each individual accident monitoring parameter has a specific purpose; however, the general purpose for all accident monitoring instrumentation is to provide sufficient information to confirm an accident is proceeding as anticipated, i.e., automatic safety systems are performing properly, and deviations from expected accident course are minimal. The NRC's position on application of the screening criteria to post-accident monitoring instrumentation is documented in a letter dated May 7, 1988, from T.E. Murley (NRC) to R.F. Janecek (BWROG). The position stated in the letter was that the post-accident monitoring instrumentation table should list, on a plant-specific basis, all Regulatory Guide (RG) 1.97 Type A instruments specified in the plant's Safety Evaluation Report (SER) on RG 1.97, and all

RG 1.97 Category 1 instruments. Accordingly, this position has been applied to the JAFNPP RG 1.97 instruments. Those instruments meeting these criteria have been retained in TSs.

In addition to the above instrumentation in CTS, the licensee also proposed to reclassify 5 variables [Core Spray Flow, Core Spray Discharge Pressure, low-pressure core injection (LPCI), residual heat removal (RHR)] Flow, and RHR Service Water Flow,) from “Type A and Category 1" to “Type D and Category 2" to be included in the above group of instrumentation. The staff reviewed the licensee’s justification for this reclassification, and finds the reclassification acceptable as follows:

The staff has determined that for the proposed instrumentation to be relocated, the screening criteria of 10 CFR 50.36 have not been satisfied, and thus their associated LCO and related Surveillances may be relocated to other plant controlled documents outside the ITS. The TSs to be relocated are those for the following instruments:

- Stack High Range Effluent Monitor
- Turbine Building Vent High Range Effluent Monitor
- Radwaste Building Vent High Range Effluent Monitor
- Safety/Relief Valve Position Indicator
- Torus Water Level (narrow range)
- Drywell — Torus Differential Pressure
- Core Spray Flow
- Core Spray Discharge Pressure
- LPCI (RHR) Flow
- RHR Service Water Flow

6. 3/4.6.F STRUCTURAL INTEGRITY

The structural integrity of the reactor coolant system shall be maintained at the level required by the original acceptance standards throughout the life of the plant. The inspection programs for Ameican Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained throughout the components’ life. Other Technical Specifications require important systems to be Operable (for example, emergency core cooling system (ECCS) 3/4.5.A) and in a ready state for mitigative action. This specification is directed toward prevention of component degradation and continued long-term maintenance of acceptable structural conditions. Hence, it is not necessary to retain this specification to ensure continuous operability of safety systems.

Further, this specification prescribes inspection requirements which are performed during plant shutdown. It is, therefore, not directly important for responding to DBAs, and the specification requirement is currently covered by 10 CFR 50.55a and the plant's ISI Program.

The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Structural Integrity LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

7. 3/4.7.A.3 PRIMARY CONTAINMENT PURGE

The containment shall be purged through the Standby Gas Treatment System (SGTS) whenever primary containment integrity is required. If this requirement cannot be met, then purging shall be discontinued without delay. The drywell vent and purge system is used primarily to control drywell-to-suppression chamber differential pressure during reactor operation, to reduce drywell airborne radioactivity levels before personnel entry, and to purge the nitrogen from the drywell for personnel safety. This LCO is intended to provide reasonable assurance that releases from normal drywell purging operations will not exceed the annual dose limits of 10 CFR Part 20 for unrestricted areas. In addition, venting and purging through the SGTS during normal operation is not part of a primary success path in the mitigation of a DBA or transient; and thus is a non-significant risk contributor to core damage frequency and offsite releases. These limits are not related to protection of the public from the consequences of any DBA or transient. The relocation of this Specification from the plant TSs is consistent with Generic Letter (GL) 89-01 for removal of Radiological Effluent Technical Specification and relocation to the ODCM. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Primary Containment Purge LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

8. 3/4.8 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material, shall have removable contamination of less than or equal to 0.005 microcuries. The limitations on miscellaneous radioactive materials sources are intended to ensure that the total body or individual organ irradiation doses do not exceed allowable limits in the event of ingestion or inhalation. This is done by imposing a maximum limitation of $\leq$ 0.005 microcuries of removable contamination on each sealed source. This requirement and the associated SRs bear no relation to the conditions or limitations which are necessary to ensure safe reactor operation. The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Miscellaneous Radioactive Materials Sources LCO and Surveillances may be relocated to other plant controlled documents outside the ITS.

9. RETS 2.1 LIQUID EFFLUENT MONITORS

The radioactive liquid effluent monitoring instrumentation is neither a safety system nor is it connected to the reactor coolant system. This instrumentation is used for the purpose of showing conformance to the discharge limits of 10 CFR

Part 20. It is not installed to detect excessive reactor coolant leakage. The radioactive liquid effluent monitors are used routinely to provide a continuous check on the release of radioactive liquid effluent from the normal plant liquid effluent flow paths. These specifications require the licensee to maintain operability of various liquid effluent monitors and establish setpoints in accordance with the ODCM. The alarm/trip setpoints are established to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. Plant DBA analyses do not assume any action, either automatic or manual, resulting from radioactive liquid effluent monitors. Accordingly, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Radioactive Liquid Effluent Monitoring Instrumentation LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

10. RETS 2.2 CONCENTRATION OF LIQUID EFFLUENTS

The current requirement for concentrations of radioactive materials released to unrestricted areas is that they shall not exceed the values specified in 10 CFR Part 20, Appendix B, Table II, Column 2, and for dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcurie/ml. 10 CFR Part 20, BII(2) refers to releases to an unrestricted area of radioactive material in concentrations that exceed the specified limits. No screening criteria apply because the process variable of the LCO is not an initial condition of a DBA or transient analysis. Neither does the system comprise a part of the safety sequence analysis or a part of the primary coolant pressure boundary. Effluent control is for protection against radiation hazards from licensed activities, not accidents.

Based on the above, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Concentration LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

11. RETS 2.3 DOSE FROM LIQUID EFFLUENTS

Limitations of the quarterly and annual projected doses to members of the public which result from cumulative liquid effluent discharges during normal operation over extended periods is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are directed to maintaining doses as low as is reasonably achievable, and are not related to protection of the public from any DBA or transient.

The staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Dose from Liquid Effluents LCO and Surveillances may be relocated to other plant controlled documents outside the ITS.

12. RETS 2.4 LIQUID RADIOACTIVE WASTE TREATMENT SYSTEM

The requirement for a liquid waste treatment system in 10 CFR Part 50, Appendix A, General Design Criteria GDC-60, pertains to controlling the release of site liquid effluents during normal operational occurrences. No loss of primary coolant is involved; neither is an accident condition assumed or implied. The limits for release in 10 CFR Part 50, Appendix I, Sec. II.A, for liquids are design objectives for operation.

Based on the above, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Liquid Radioactive Waste Treatment System LCO and Surveillances may be relocated to other plant controlled documents outside the ITS.

13. RETS 3.1 GASEOUS EFFLUENT MONITORS

The radioactive gaseous effluent monitors are neither a safety system nor are they connected to the reactor coolant system. The primary function of this instrumentation is to show conformance to the discharge limits of 10 CFR Part 20. This instrumentation is not installed to detect excessive reactor coolant leakage. The radioactive gaseous effluent monitors are used routinely to provide a continuous check on the releases of radioactive gaseous effluents from the normal plant gaseous effluent flow paths. These TSs require the licensee to maintain operability of various effluent monitors and establish setpoints in accordance with the ODCM. The alarm/trip setpoints are established to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. Plant DBA analyses do not assume any action, either automatic or manual, resulting from radioactive effluent monitor (except as indicated in the Discussion of Changes for ITS 3.3.6.2, Secondary Containment Instrumentation). (The Refuel Floor and Reactor Building exhaust monitor are retained in ITS 3.3.6.2.)

Based on the above, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Gaseous Effluent Monitors LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

14. RETS 3.2 GASEOUS DOSE RATE

This LCO limits the dose rate due to gaseous effluents in unrestricted areas at any time to a value less than the yearly dose limit of 10 CFR Part 20. This provides reasonable assurance that no member of the public is exposed to annual average concentrations which exceed the limits of 10 CFR Part 20 Appendix B, Table-II. This is a limit which applies to normal operation of the plant. It is not assumed as an initial condition of any DBA or transient and is not relied upon to limit the consequences of such events.

Based on the above, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Gaseous Dose Rate LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

15. RETS 3.3 AIR DOSE, NOBLE GASES

This LCO limits the air dose to areas at or beyond the site boundary from noble gases released from the plant in gaseous effluents. Limitation of the quarterly and annual air doses from noble gases in plant gaseous effluents during normal operation over extended periods is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from the consequences of any DBA or transient.

Based on the above, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Air Dose, Noble Gases LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

16. RETS 3.4 DOSE DUE TO IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

This LCO limits the dose to a member of the public at or beyond the site boundary from Iodine-131, Iodine-133, Tritium, and radionuclides in particulate form with half-lives greater than 8 days released from the plant in gaseous effluents. Limitation of the quarterly and annual projected doses to members of the public from radionuclides other than noble gases during normal operation over extended periods is intended to assure compliance with the dose objectives of 10 CFR Part 50, Appendix I. These limits are not related to protection of the public from the consequences of any design basis accident or transient.

Based on the above, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Dose due to Iodine-131, Iodine-133, Tritium, and radioactive material in particulate form LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

17. RETS 3.5.b SJAE RADIATION MONITORS

The Steam Jet Air Ejector (SJAE) radiation monitors are neither a safety system nor are they connected to the reactor coolant system. The primary function of this instrumentation is to show conformance to the discharge limits of 10 CFR Part 20. This instrumentation is not installed to detect excessive reactor coolant leakage. The SJAE System monitors are used routinely to provide a continuous check on the release of radioactive gaseous effluents from the Main Condenser Steam Jet Air Ejector. These TSs require the licensee to maintain Operability of the SJAE radiation monitors and established setpoints. The alarm/trip setpoints are established to ensure that the alarm/trip will occur to prevent exceeding the limits of 10 CFR Part 20. Plant DBA analyses do not assume any action, either automatic or manual, resulting from these monitors.

Based on the above, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the SJAE Radiation Monitors LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

18. RETS 3.6 OFFGAS TREATMENT SYSTEM

This LCO requires that the Offgas Treatment System be used to reduce the concentration of radioactive materials in gaseous effluents prior to release from the plant within 24 hours after the start-up of the second turbine driven feedwater pump. The Offgas Treatment System reduces the activity level of the non-condensible fission product gases from fuel defects removed from the main condenser prior to their release to the environs. The Operability of the Offgas Treatment System is required to meet the requirements of 10 CFR 50.36a and GDC-60 of Appendix A to 10 CFR Part 50 (i.e., releases of radioactive materials in gaseous effluents will be kept "as low as reasonably achievable"). The Operability of the Offgas Treatment System is not assumed in the analysis of any DBA or transient. However, offgas activity is an initial condition of a DBA and is being retained in ITS LCO 3.7.5. Therefore, there is no need to retain this requirement.

The staff has determined that to the extent the screening criteria of 10 CFR 50.36 have been satisfied with respect to this system, requirements are retained in the ITS in LCO 3.7.5. In addition, for the reasons set forth above, the Offgas Treatment System LCO and Surveillances is not required by 10 CFR 50.36, and may be relocated to other plant-controlled documents outside the ITS.

19. RETS 4.0/4.1 SOLID RADIOACTIVE WASTE - PROCESS CONTROL PROGRAM (PCP)

The solid radwaste system is used in accordance with the PCP to process wet radioactive wastes to meet shipping and burial ground requirements. The Solid Radwaste System is a logical continuation of the liquid radwaste system. It operates on the same requirement for effluent control, identified as 10 CFR Part 50, Appendix A, GDC-60. The system serves to control operational release of solid waste, not accidental release.

Based on the above, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Solid Radioactive Waste-Process Control Program LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

20. RETS 5.0/5.1 TOTAL DOSE - TOTAL DOSE FROM URANIUM FUEL CYCLE

This LCO limits the dose or dose commitment to any member of the public due to releases of radioactivity and radiation from uranium fuel cycle sources. This LCO also limits the annual doses to individual members of the public from all plant sources. This is intended to assure that normal operation of the plant is in compliance with the provisions of 40 CFR Part 190. These limits are not related to protection of the public from any DBA or transient.

Based on the above, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Total Dose from Uranium Fuel Cycle LCO and Surveillances may be relocated to other plant controlled documents outside the ITS.

21. RETS 6.1 RADIOLOGICAL ENVIRONMENTAL MONITORING - MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public resulting from station operations. This program monitors the long term impact of normal plant operations and are not related to the protection of the public from any DBA or transient.

Thus, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Monitoring Program LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

22. RETS 6.2 LAND USE CENSUS PROGRAM

A land use census is conducted to identify the locations of all milch animals, the nearest residence, and all gardens of greater than 50 square meters producing fresh leafy vegetables, in each of the 16 meteorological sectors within a distance of 5 miles from the site. The land use census required by this specification supports the measurement of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public resulting from station operations. This program ensures that changes in the use of areas at or beyond the site boundary are identified and changes made to the radiological environmental monitoring program, if required and are not related to the protection of the public from the public from any DBA or transient.

Thus, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Land Use Census Program LCO and Surveillances may be relocated to other plant-controlled documents outside the ITS.

23. RETS 6.3 INTERLABORATORY COMPARISON PROGRAM

The interlaboratory comparison program required by this specification confirms the accuracy of the measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures for members of the public resulting from station operation. This program ensures independent checks on the precision and accuracy of the instrumentation used in the measurements of radioactive material for the radiological environmental monitoring program are performed and are not related to the protection of the public from any DBA or transient.

Thus, the staff has determined that the screening criteria of 10 CFR 50.36 have not been satisfied, and thus the Interlaboratory Comparison Program LCO and Surveillances may be relocated to other plant controlled documents outside the TSs.

The relocated specifications from the CTS discussed above are not required to be in the TS because they do not fall within the criteria for mandatory inclusion in the TS as stated in 10 CFR 50.36(c)(2)(ii). These specifications are not needed to obviate the possibility that an abnormal situation or event will give rise to an immediate threat to the public health and safety. In addition, the NRC staff has concluded that appropriate controls have been established for all of the current specifications and information that are being moved to the UFSAR, TRM, ODCM, or PCP. These relocations are the subject of a new license condition discussed in Section 5.0 of this SE. Until incorporated in licensee-controlled documents, changes to these specifications and information will be controlled in accordance with the current applicable procedures and regulations that control these documents. Following implementation, the NRC may audit the relocated provisions to ensure that an appropriate level of control has been achieved. The NRC staff has concluded that, in accordance with the Final Policy Statement, sufficient regulatory controls exist under the regulations, particularly 10 CFR 50.59. Accordingly, the specifications and information, as described in detail in this SE, may be relocated from the CTS and placed in the licensee-controlled documents identified in the licensee's application dated March 31, 1999, as supplemented by letters dated June 1, July 14, and October 14, 1999, February 11, April 4, April 13, June 30, July 31, September 12, September 13, and October 23, 2000, May 31 and October 18, 2001, February 6, March 27, April 26, 2002, June 11, and June 12, 2002.

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

In the ITS conversion, the licensee will be relocating specifications, requirements, and detailed information from the CTS to licensee-controlled documents outside the CTS. This is discussed in Sections 3.D and 3.E above. The facility and procedures described in the UFSAR and TRM, which is a part of the UFSAR, can only be revised in accordance with the provisions of 10 CFR 50.59, which ensures records are maintained and establishes appropriate control over requirements removed from the CTS and over future changes to the requirements. Other licensee-controlled documents contain provisions for making changes consistent with applicable regulatory requirements. For example, the ODCM can be changed in accordance with ITS 5.5.1, and the administrative instructions that implement the QA Program can be changed in accordance with 10 CFR 50.54(a) and 10 CFR Part 50, Appendix B. The documentation of these changes will be maintained by the licensee in accordance with the record retention requirements specified in the licensee's QA Program for JAFNPP and such applicable regulations as 10 CFR 50.59.

The license condition for the relocation of requirements from the CTS, which is discussed in Section 5.0 of this SE, will address the implementation of the ITS conversion, and the schedule for the relocation of the CTS requirements into licensee-controlled documents. The relocations to the UFSAR, which includes the TRM, shall be included in the next required update of this document in accordance with 10 CFR 50.71(e).

G. Evaluation of Other TS Changes Included in the Application for Conversion to ITS

This section evaluates other TS changes included in JAFNPP's ITS conversion application. These include items which deviate from both the CTS and the STS, do not fall clearly into a category, or are in addition to those changes that are needed to meet the overall purpose of the conversion. These changes are termed as the beyond-scope issues and were addressed in the notice of consideration of amendment published in the *Federal Register* on November 8, 1999 (64 FR 60854), December 13, 1999 (64 FR 69574), and November 28, 2001 (66 FR 59495).

The changes discussed below are listed in the order of the applicable ITS specification or section, as appropriate.

(1) <u>ITS Table 3.3.1.1-1 Function 8 (DOC L14), Turbine Stop Valve Closure</u>
<u>ITS Table 3.3.1.1-1 Function 9 (DOC L14), Turbine Control Valve Fast Closure, EHC Oil Pressure - Low</u>

Change Allowable Value, CTS Table 2.1.A, Item 3 and Table 3.1-1, Item 15, Turbine Stop Valve Closure, from "$\leq$ 10% valve closure" to "$\leq$ 15% closed."

Change Allowable Value, CTS Table 2.1.A, Item 4 and Table 3.1-1, Item 14, Turbine Control Valve Fast Closure, from "$500 < p < 850$ psig" to "$\geq$ 500 psig and $\leq$ 850 psig."

The proposed allowable values have been established consistent with the New York Power Authority (NYPA) Engineering Standards Manual, IES-3A, "Instrument Loop Accuracy and Setpoint Calculation Methodology." The methodology used to determine the allowable values is consistent with the methodology described in ISA-S67.04-1994, Part I, "Setpoints for Nuclear Safety-Related Instrumentation." The licensee has not committed to complying with RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," which endorses ISA-S67.04-1994, Part I with exceptions and clarifications. Nonetheless, the licensee has provided a comparison of IES-3A with RG 1.105, Revision 3. In this comparison, the licensee adequately addressed each of the RG 1.105 exceptions and clarifications.

The proposed allowable values were calculated by applying calibration based errors to the trip setpoints; thereby establishing an operability limit associated with the entire loop of each instrumentation function. The licensee has stated that any changes to the safety analysis limits applied in the methodologies were evaluated, and that it has confirmed that applicable design requirements of the associated systems are maintained. The use of this methodology for establishing Allowable Values and Trip Setpoints ensures design or safety analysis limits are not exceeded in the event of transients or accidents and accounts for uncertainties and environmental conditions. Therefore, the proposed allowable value changes do not affect the existing licensing basis, and are, therefore, acceptable.

(2) ITS 3.3.4.1 (DOCs A3), ATWS-RPT Instrumentation

The proposed change revises the STS channel configuration. In STS 3.3.4.2, Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation, LCO 3.3.4.2 requires two channels per trip system for each ATWS-RPT function to be operable. The STS ATWS-RPT Bases describes a system that consists of two independent trip systems, with two channels of Reactor Pressure - High and two channels of Reactor Vessel Water Level - Low Low in each trip system. Either two Reactor Pressure - High or two Reactor Vessel Water Level - Low Low signals are needed to trip a trip system. The outputs are combined in a logic so that either trip system will trip both recirculation pumps.

The JAFNPP logic configuration in CTS Table 3.2-7, ATWS-RPT, is different from the configuration described in the STS Bases. The JAFNPP configuration includes one trip system with four channels of Reactor Pressure - High and four channels of Reactor Vessel Water Level - Low Low. Two channels of each function are powered by Division 1 and the other two channels of each function are powered by Division 2. The outputs of each channel provide input into the trip system which is one-out-of-two taken twice logic for each function. One channel from each division of the same function must trip to complete the logic. The trip system is arranged so that each function will trip both recirculation pumps.

In this proposed ITS 3.3.4.1, ATWS-RPT Instrumentation, LCO 3.3.4.1 requires four channels for each ATWS-RPT function to be operable. The staff also reviewed plant-specific ATWS-RPT configuration as described in the ITS Bases B 3.3.4.1, and finds the requirement for four channels of each function being operable provides appropriate protection for this plant-specific application, and, therefore, is acceptable.

(3) ITS Table 3.3.5.1-1 (DOCs M2, M6, L6):

The licensee proposed changes to the following Allowable Values:

- Function 1.c, Core Spray System, Reactor Pressure - Low (Injection Permissive) and Function 2.c, Low Pressure Coolant Injection System, Reactor Pressure - Low (Injection Permissive)

  Change Allowable Value, CTS Table 3.2-2 Item 9, Reactor Low Pressure - from "≥ 450 psig" to "≥ 410 psig and ≤ 490 psig."

- Function 1.e, Core Spray System, Core Spray Pump Discharge Flow - Low (Bypass)

  Set Allowable Value as "≥ 510 gpm and ≤ 980 gpm."

- Function 1.f, Core Spray System, Core Spray Pump Discharge Pressure - High (Bypass)

  Set Allowable Value as "≥ 90 psig and ≤ 110 psig."

- Function 2.d, Low Pressure Coolant Injection System, Reactor Pressure - Low (Recirculation Discharge Valve Permissive)

  Change Allowable Value, CTS Table 3.2-2 Item 24, Reactor Low Pressure - from "285 to 335 psig" to "≥ 295 psig."

- Function 2.e, Low Pressure Coolant Injection System, Reactor Vessel Shroud Level (Level 0)

  Change Allowable Value, CTS Table 3.2-2 Item 5, Reactor Low Level (inside shroud) - from "≥ 0 in above TAF" to "≥ 1.0 inch."

- Function 2.g, Low Pressure Coolant Injection System, Low Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)

  Set Allowable Value as "≥ 1040 gpm and ≤ 1665 gpm."

- Function 2.h, Low Pressure Coolant Injection System, Containment Pressure - High

  Change Allowable Value, CTS Table 3.2-2 Item 6, Containment High Pressure - from "1 < p < 2.7 psig" to "≥1 psig and ≤ 2.7 psig."

- Function 3.f, High Pressure Coolant Injection System, High Pressure Coolant Injection Pump Discharge Flow - Low (Bypass)

  Set Allowable Value as "$\geq$ 475 gpm and $\leq$ 800 gpm."

- Function 3.g, High Pressure Coolant Injection System, High Pressure Coolant Injection Pump Discharge Pressure - High (Bypass)

  Set Allowable Value as "$\geq$ 25 psig and $\leq$ 80 psig."

The above proposed allowable values have been established consistent with the JAFNPP Engineering Standards Manual, IES-3A, "Instrument Loop Accuracy and Setpoint Calculation Methodology." The methodology used to determine the allowable values is consistent with the methodology described in ISA-S67.04-1994, Part I, "Setpoints for Nuclear Safety-Related Instrumentation." The licensee has not committed to complying with RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," which endorses ISA-S67.04-1994, Part I with exceptions and clarifications. Nonetheless, the licensee has provided a comparison of IES-3A with RG 1.105, Revision 3. In this comparison the licensee adequately addressed each of the RG 1.105 exceptions and clarifications.

The proposed allowable values were calculated by applying calibration based errors to the trip setpoints; thereby establishing an operability limit associated with the entire loop of each instrumentation function. The licensee has stated that any changes to the safety analysis limits applied in the methodologies were evaluated, and that it has confirmed that applicable design requirements of the associated systems are maintained. The use of this methodology for establishing Allowable Values and Trip Setpoints ensures design or safety analysis limits are not exceeded in the event of transients or accidents and accounts for uncertainties and environmental conditions. Therefore, the proposed allowable value changes do not affect the existing licensing basis, and are, therefore, acceptable.

(4) ITS Table 3.3.5.1-1 (DOC A6), Function 3.e, Low Pressure Coolant Injection Suppression Pool Water Level - High

The license proposed change to Allowable Value for CTS Table 3.2-2 Item 18, Suppression Chamber High Level from "$\leq$ 6 in. above normal level" to "$\leq$ 14.5 feet".

The staff reviewed the licensee response to the staff question and concurred that normal suppression pool water level is 13.88 to 14.00 feet. The CTS requires an allowable value of $\leq$ 6 inches above normal. The ITS allowable value of $\leq$ 14.5 is equivalent to the CTS value and is consistent with the format of the STS. Therefore, this change is editorial in nature since there is no technical change, and is acceptable.

(5) ITS Table 3.3.6.1-1 (DOC L16) Function 3a, High Pressure Coolant Injection - System Isolation HPCI Steam Line Flow - High

The licensee proposed a change to the Allowable Value for CTS Table 3.2-1, HPCI Turbine Steam Line High Flow from "$\leq$ 160 in. H_2O dp" to $\leq$ "168.24 inches of water dP".

The proposed allowable value has been established consistent with the JAFNPP Engineering Standards Manual, IES-3A, "Instrument Loop Accuracy and Setpoint Calculation Methodology." The methodology used to determine the allowable values is consistent with the methodology described in ISA-S67.04-1994, Part I, "Setpoints for Nuclear Safety-Related Instrumentation." The licensee has not committed to complying with RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," which endorses ISA-S67.04-1994, Part I with exceptions and clarifications. Nonetheless, the licensee has provided a comparison of IES-3A with RG 1.105, Revision 3. In this comparison the licensee adequately addressed each of the RG 1.105 exceptions and clarifications.

The proposed allowable value was calculated by applying calibration based errors to the trip setpoints; thereby establishing an operability limit associated with the entire loop of each instrumentation function. The licensee has stated that any changes to the safety analysis limits applied in the methodologies were evaluated, and that it has confirmed that applicable design requirements of the associated systems are maintained. The use of this methodology for establishing Allowable Values and Trip Setpoints ensures design or safety analysis limits are not exceeded in the event of transients or accidents and accounts for uncertainties and environmental conditions. The proposed allowable value change does not affect the existing licensing basis, and is, therefore, acceptable.

(6) ITS Table 3.3.6.1-1 (DOC M14):

The licensee proposed changes to the following Allowable Values:

- Function 3.d, High Pressure Coolant Injection System Isolation, HPCI Steam Line Penetration (Drywell Entrance) Area Temperature - High, and Function 3.e, High Pressure Coolant Injection System Isolation, HPCI Steam Line Torus Room Area Temperature - High

  Change Allowable Value, CTS Table 3.2-1, ITS marked up Item 16, HPCI Steam Line/Area Temperature, from "$\leq$ 40^{o}F above max. ambient" to "$\leq$ 160^{o}F."

- Function 3.f, High Pressure Coolant Injection System Isolation, RHR Heat Exchanger A Area Temperature - High and Function 3.g, High Pressure Coolant Injection System Isolation, RHR Heat Exchanger B Area Temperature - High

  Change Allowable Value, CTS Table 3.2-1, ITS marked up Item 16, RHR Heat Exchangeers A and B Temperature- High, from "$\leq$ 40^{o}F above max. ambient" to "$\leq$ 170^{o}F".

- <u>Function 3.h, High Pressure Coolant Injection System Isolation, RB Southwest Area of Elevation 272' Temperature - High, Function 3.i, High Pressure Coolant Injection System Isolation, RB Southeast Area of Elevation 272' Temperature - High, and Function 3.j, High Pressure Coolant Injection System Isolation, HPCI Equipment Area Temperature - High</u>

  Change Allowable Value, CTS Table 3.2-1, ITS marked up Item 16, RB Elevation 272' and HPCI Equipment Area Temperature - High, from "≤ 40°F above max. ambient" to "≤ 144°F".

- <u>Function 4.d, Reactor Core Isolation Cooling System Isolation, RCIC Steam Line Penetration (Drywell Entrance) Area Temperature - High and Function 4.e, Reactor Core Isolation Cooling System Isolation, RCIC Steam Line Torus Room Area Temperature - High</u>

  Change Allowable Value, CTS Table 3.2-1, ITS marked up Item 20, RCIC Steam Line/Area Temperature, from≤ 40°F above max. ambient" to "≤ 160°F".

- <u>Function 4.f, Reactor Core Isolation Cooling System Isolation, RCIC Equipment Area Temperature - High</u>

  Change Allowable Value, CTS Table 3.2-1, ITS marked up Item 20, RCIC Equipment Area Temperature High, from "≤ 40°F above max. ambient" to "≤ 144°F".

- <u>Function 5.a, Reactor Water Cleanup System Isolation, RWCU Suction Line Penetration Area Temperature - High</u>

  Change Allowable Value, CTS Table 3.2-1, ITS marked up Item 11, RWCU System Isolation Suction Line Penetration Area Temperature High, from "≤ 40°F above max. ambient" to "≤ 144°F".

- <u>Function 5.b, Reactor Water Cleanup System Isolation, RWCU Pump Area Temperature - High Pump A</u>

  Change Allowable Value, CTS Table 3.2-1, ITS Marked up Item 11, RWCU System Isolation Pump Area Temperature High Pump A, from "≤ 40°F above max. ambient" to "≤ 165°F".

- <u>Function 5.b, Reactor Water Cleanup System Isolation, RWCU Pump Area Temperature - High Pump B</u>

  Change Allowable Value, CTS Table 3.2-1, ITS Marked up Item 11, RWCU System Isolation, Pump Area Temperature High Pump B, from "≤ 40°F above max. ambient" to "≤ 175°F".

- Function 5.c, Reactor Water Cleanup System Isolation, RWCU Heat Exchanger Room Area Temperature - High

  Change Allowable Value, CTS Table 3.2-1, ITS Marked up Item 11, Reactor Water Cleanup System Temperature Isolation, RWCU Heat Exchanger Room Area Temperature High, from "$\leq$ 40^{o}F above max. ambient" to "$\leq$ 155^{o}F".

- Function 1.e, Main Steam Line Isolation, Main Steam Tunnel Area Temperature - High

  Change Allowable Value, CTS Table 3.2-1, ITS Marked up Item 10, Main Steam Line Leak Detection High Temperature, from "< 40^{o}F above max. ambient" to "$\leq$ 195^{o}F".

- Function 3.b, High Pressure Coolant Injection System Isolation, HPCI Steam Supply Line Pressure - Low

  Change Allowable Value, CTS Table 3.2-1 I, TS Marked up Item 14, HPCI Steam Line Low Pressure, from "100 > P > 50 psig" to "$\geq$ 61 psig and $\leq$ 90 psig."

- Function 3.c, High Pressure Coolant Injection System Isolation, HPCI Turbine Exhaust Diaphragm Pressure - High

  Change Allowable Value, CTS Table 3.2-1, ITS Marked up Item 15, HPCI Turbine High Exhaust Diaphragm Pressure, from "$\leq$ 10 psig" to "$\leq$ 9.9 psig."

- Function 4.a, Reactor Core Isolation Cooling System Isolation, RCIC Turbine Steam Line Flow - High

  Change Allowable Value, CTS Table 3.2-1 I, ITS Marked up Item 17, RCIC Turbine Steam Line High Flow, from "$\leq$ 282 in H_2O dp" to "$\leq$ 272.26 inches of water dP."

- Function 4.b, Reactor Core Isolation Cooling System Isolation, RCIC Steam Supply Line Pressure - Low

  Change Allowable Value, CTS Table 3.2-1, ITS Marked up Item 18, RCIC Steam Line Low Pressure, from "100 > P > 50 psig" to "$\geq$ 58 psig and $\leq$ 93 psig."

- Function 4.c, Reactor Core Isolation Cooling System Isolation, RCIC Turbine Exhaust Diaphragm Pressure - High

  Change Allowable Value, CTS Table 3.2-1, ITS Marked up Item 19, RCIC Turbine High Exhaust Diaphragm Pressure, from "$\leq$ 10 psig" to "$\leq$ 5 psig."

- <u>Function 6.a, Shutdown Cooling System Isolation, Reactor Pressure - High</u>

  Change Allowable Value, CTS Table 3.2-1 I, ITS Marked up Item 3, Reactor High Pressure (Shutdown Cooling Isolation), from "≤ 75 psig" to "≤ 74 psig."

The above proposed allowable values have been established consistent with the JAFNPP Engineering Standards Manual, IES-3A, "Instrument Loop Accuracy and Setpoint Calculation Methodology." The methodology used to determine the allowable values is consistent with the methodology described in ISA-S67.04-1994, Part I, "Setpoints for Nuclear Safety-Related Instrumentation." The licensee has not committed to complying with RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," which endorses ISA-S67.04-1994, Part I with exceptions and clarifications. Nonetheless, the licensee has provided a comparison of IES-3A with RG 1.105, Revision 3. In this comparison, the licensee adequately addressed each of the RG 1.105 exceptions and clarifications.

The proposed allowable values were calculated by applying calibration based errors to the trip setpoints; thereby establishing an operability limit associated with the entire loop of each instrumentation function. The licensee has stated that any changes to the safety analysis limits applied in the methodologies were evaluated, and that it has confirmed that applicable design requirements of the associated systems are maintained. The use of this methodology for establishing Allowable Values and Trip Setpoints ensures design or safety analysis limits are not exceeded in the event of transients or accidents and accounts for uncertainties and environmental conditions. Therefore, the proposed allowable value changes do not affect the existing licensing basis, and are, therefore, acceptable.

(7) <u>ITS Table 3.3.6.1-1 Function 1.c (DOC L17), Main Steam Line Isolation, Main Steam Line Flow-High:</u>

The licensee proposed a change to the following Allowable Value for CTS Table 3.2-1 Item 9, Main Steam Line High Flow, from "≤ 140% of Rated Steam Flow" to "≤ 125.9 psid."

The above proposed allowable value has been established consistent with the JAFNPP Engineering Standards Manual, IES-3A, "Instrument Loop Accuracy and Setpoint Calculation Methodology." The methodology used to determine the allowable values is consistent with the methodology described in ISA-S67.04-1994, Part I, "Setpoints for Nuclear Safety-Related Instrumentation." The licensee has not committed to complying with RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," which endorses ISA-S67.04-1994, Part I with exceptions and clarifications. Nonetheless, the licensee has provided a comparison of IES-3A with RG 1.105, Revision 3. In this comparison, the licensee adequately addressed each of the RG 1.105 exceptions and clarifications.

The proposed allowable value was calculated by applying calibration based errors to the trip setpoints; thereby establishing an operability limit associated with the entire loop of each instrumentation function. The licensee has stated that any changes to the safety analysis limits applied in the methodologies were evaluated, and that it has confirmed that applicable design requirements of the associated systems are maintained. The use of this methodology for establishing Allowable Values and Trip Setpoints ensures design or safety analysis limits are not exceeded in the event of transients or accidents and accounts for uncertainties and environmental conditions. Therefore, the proposed allowable value change does not affect the existing licensing basis, and is, therefore, acceptable.

(8) ITS Table 3.3.6.1-1 (JFD DB11) - Function 2.f, Primary Containment Isolation, Main Steam Tunnel Radiation - High

The licensee proposed changes to the following Allowable Value, CTS Table 3.2-1 Item 7, Main Steam Line Tunnel High Radiation, from "≤ 3 x Normal Rated Full Power Background" to "≤ 3 times Normal Full Power Background."

The CTS includes the definition of Rated Thermal Power. The ITS includes the definitions of Rated Thermal Power and Thermal Power. The CTS definition of Rated Thermal Power refers to both a steady state nuclear steam supply output and reactor core thermal power. The ITS definition of Rated Thermal Power includes only reactor core thermal power, which is consistent with NUREG-1433, Rev 1. The ITS definition of Thermal Power is the same as the Rated Thermal Power without specifying the reactor power. There is no change to the reactor power of 2536 MWt. Therefore, this is an editorial change and is, therefore, acceptable.

(9) ITS Surveillance Requirement 3.3.7.3.1 (DOC M1), Emergency Service Water System Instrumentation Channel Calibration

The licensee proposed to set the Allowable Value, CTS Surveillance Requirement 4.11.D.1.e, Emergency Service Water Instrument Channel Calibration, as "≥ 40 psig and ≤ 50 psig."

The proposed allowable value has been established consistent with the JAFNPP Engineering Standards Manual, IES-3A, "Instrument Loop Accuracy and Setpoint Calculation Methodology." The methodology used to determine the allowable value is consistent with the methodology described in ISA-S67.04-1994, Part I, "Setpoints for Nuclear Safety-Related Instrumentation." The licensee has not committed to complying with RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," which endorses ISA-S67.04-1994, Part I with exceptions and clarifications. Nonetheless, the licensee has provided a comparison of IES-3A with RG 1.105, Revision 3. In this comparison the licensee adequately addressed each of the RG 1.105 exceptions and clarifications.

The proposed allowable value was calculated by applying calibration based errors to the trip setpoint; thereby establishing an operability limit associated with the entire loop of the instrumentation function. The licensee has stated that any changes to the safety analysis limits applied in the methodologies were evaluated, and that it has confirmed that applicable design requirements of the associated systems are maintained. The use of this methodology for establishing Allowable Values and Trip Setpoints ensures design or safety analysis limits are not exceeded in the event of transients or accidents and accounts for uncertainties and environmental conditions. Therefore, the proposed allowable value change does not affect the existing licensing basis, and is, therefore, acceptable.

(10) ITS Table 3.3.8.1-1 (DOC L1):

The licensee proposed changes to the following Allowable Values:

- Function 1.a, 4.16 kV Emergency Bus Undervoltage (Loss of Voltage) Bus Undervoltage

  Change Allowable Value, CTS Table 3.2-2, Item 22, 4kV Emergency Bus Undervoltage Relay (Loss of Voltage), from "85 ± 4.81 secondary volts" to "≥ 80.2 V and ≤ 89.8 V."

- Function 1.b, 4.16 kV Emergency Bus Undervoltage (Loss of Voltage) Time Delay

  Change Allowable Value, CTS Table 3.2-2, Item 23, 4kV Emergency Bus Undervoltage Timer (Loss of Voltage), from "2.50 ± 0.11 sec" to "≥ 2.4 seconds and ≤ 2.6 seconds."

- Function 2.a, 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) Bus Undervoltage

  Change Allowable Value, CTS Table 3.2-2, Item 19, 4kV Emergency Bus Undervoltage Relay (Degraded Voltage), from "110.6 ± 0.8 secondary volts" to "≥ 109.8 V and ≤ 111.4 V."

- Function 2.b, 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) Time Delay (LOCA)

  Change Allowable Value, CTS Table 3.2-2, Item 20, 4kV Emergency Bus Undervoltage Timer (Degraded Voltage LOCA), from "8.96 ± 0.55 sec" to "≥ 8.4 seconds and ≤ 9.5 seconds."

- Function 2.c, 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) Time Delay (non-LOCA)

  Change Allowable Value, CTS Table 3.2-2, Item 21, 4kV Emergency Bus Undervoltage Timer (Degraded Voltage non-LOCA), from "43.8 ± 2.8 sec" to "≥ 41.0 seconds and ≤ 46.6 seconds."

The above proposed allowable values have been established consistent with the JAFNPP Engineering Standards Manual, IES-3A, "Instrument Loop Accuracy and Setpoint Calculation Methodology." The methodology used to determine the allowable values is consistent with the methodology described in ISA-S67.04-1994, Part I, "Setpoints for Nuclear Safety-Related Instrumentation." The licensee has not committed to complying with RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," which endorses ISA-S67.04-1994, Part I with exceptions and clarifications. Nonetheless, the licensee has provided a comparison of IES-3A with RG 1.105, Revision 3. In this comparison, the licensee adequately addressed each of the RG 1.105 exceptions and clarifications.

The proposed allowable values were calculated by applying calibration based errors to the trip setpoints; thereby establishing any operability limit associated with the entire loop of each instrumentation function. The licensee has stated that any changes to the safety analysis limits applied in the methodologies were evaluated, and that it has confirmed that applicable design requirements of the associated systems are maintained. The use of this methodology for establishing Allowable Values and Trip Setpoints ensures design or safety analysis limits are not exceeded in the event of transients or accidents and accounts for uncertainties and environmental conditions. Therefore, the proposed allowable value changes do not affect the existing licensing basis, and are, therefore, acceptable.

(11) <u>ITS SR 3.3.8.2.3 (DOC M3) Change of Allowable value</u>

The proposed change involves a change to the allowable value of the Reactor Protection System (RPS) Electric Power Monitoring System. The RPS Electric Power Monitoring System is provided to isolate the RPS bus from the motor generator (MG) set or an alternate power supply in the event of overvoltage, undervoltage, or underfrequency. The RPS Electric Power Monitoring System is necessary to meet the assumptions of the FitzPatrick safety analyses by ensuring that the equipment powered from the RPS buses can perform its intended function.

CTS 4.9.G.2 list the undervoltage value of the RPS alternate power source monitors as $\geq$ 108 V. The undervoltage value listed in ITS SR 3.3.8.2.3 is $\geq$109.9 V. The licensee's justification provided with this change states that the methodology used to determine this allowable value is consistent with the methodology discussed in ISA-S67.04 -1994, Part II, "Methodologies for Determination of Setpoints for Nuclear Safety-Related Instrumentation." The licensee also states that the proposed value will ensure the most limiting voltage requirement is met. Based on the above justification and the fact that the change has been made in a conservative direction, the proposed change is acceptable.

(12) <u>ITS 3.4.9 (DOC L2) RCS P/T Limits changes</u>:

The CTS and STS both provide that the temperature differential between the reactor coolant system and the reactor vessel bottom drain line be less than or equal to 145 ^{o}F during a recirulation loop startup. This requirement ensures that the differential temperature between the bottom head drain line and the reactor coolant is within limits that have been established to avoid a thermal overstress condition to the Control Rod Drive (CRD) stub tubes and in-core housing welds which could result from sweeping hot water across these relatively cooler vessel structures and associated components. The temperature in the bottom head region is measured by monitoring the temperature of flow being drawn out from the bottom head drain line. JAFNPP also has, however, experienced plugging of the bottom head drain line. If the bottom head drain is plugged, the CTS would require the licensee to shut down the plant to restart an idle recirculation pump.

JAFNPP requests that an alternate method of verifying this temperature differential is met be allowed. The new method in ITS 3.4.9.4 would provide an option to verify the recirulation flow to assure proper thermal mixing and thereby avoid thermal overstress conditions. If the bottom head drain is plugged, the CTS would require the licensee to shut down the plant to restart an idle recirculation pump. The change would avoid unnecessary plant shutdowns when the bottom drain line is plugged or if the drain line flow is low. In addition, General Electric (GE) has determined by testing and experiments that the alternate method to verify the differential temperature between the bottom head coolant temperature and the reactor pressure vessel (RPV) temperature is valid.

GE has determined that two distinct operating conditions need to be considered to prevent excessive thermal stress of the lower head region of the RPV when a recirculation pump is started. In the first condition, when reactor coolant flow is greater than 40 percent of the rated value, there will be sufficient mixing and turbulence to prevent significant stratification of cooler water in the bottom head region. In the second case, when flow is less than 40 percent of rated flow (for the limited time of 30 minutes), stratification of cooler water in the lower head region will be insufficient to be of concern with respect to thermal stress. Based on the analysis by GE, the staff agrees that stratification in the lower head region will not be a problem for short periods of operation below 40 percent of rated flow, and thus the proposed change is acceptable.

(13) <u>ITS 3.5.1 (DOC L7) ECCS - Reduce HPCI and LPCI pump flow rates to SAFER/GESTER-LOCA flow rates</u>

The proposed changes involve reductions of the high-pressure coolant injection (HPCI) system and low-pressure coolant injection (LPCI) system flow rates from those specified in the approved CTS. The specific Technical Specification sections requested for NRC review and approval are: JAFNPP CTS 4.5.C.1, “High Pressure Coolant Injection (HPCI) system,” and CTS 4.5.A.3, “Low Pressure Coolant Injection (LPCI) system.” The licensee proposes to decrease the required HPCI flow rate from 4250 gpm to 3400 gpm and the required LPCI flow rate from 8910 gpm to 7700 gpm. The basis for the reductions of the HPCI and LPCI flow rates is the new approved SAFER/GESTR-LOCA methodology used to show compliance with the 10 CFR 50.46 criteria for the design basis loss-of-coolant accident (LOCA).

The licensee is requesting to use the HPCI and LPCI system flow rates that are inputs to the SAFER/GESTR-LOCA analysis to show compliance to the 10 CFR 50.46 criteria using the 10 CFR Part 50 Appendix K requirements. The plant-specific LOCA analysis using the accepted SAFER/GESTR-LOCA methodology is presented in NEDO-31317P, Revision 2, dated April 1993. This analysis was performed in conformance with NRC requirements and demonstrates conformance with the ECCS acceptance criteria of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The analysis evaluated a number of plant-specific break sizes to establish the behavior of the nominal and Appendix K peak cladding temperature (PCT) as a function of break size. Different single failures were also evaluated to identify the worst cases. The Licensing Basis PCT for JAFNPP is 1620 deg.F, which is below the PCT limit of 2200 deg.F. The calculated Upper Bound PCT for JAFNPP is 1510 deg.F, which is below the Upper Bound PCT limit of 1600 deg.F. With the verification that the Licensing Basis PCT is greater than the Upper Bound PCT for JAFNPP, the level of safety and conservatism of the plant-specific evaluation meets the NRC approved criteria. Therefore, the requirements of Appendix K are satisfied with the proposed lower flow rates, and the proposed changes are acceptable.

(14) <u>ITS SRs 3.5.3.3 and 3.5.3.4 (DOC M3) RCIC Flow Rate Test</u>

The licensee proposed to change the test criteria from a single periodic pump flow rate test (every 92 days) with a pressure range of 1195 psig - 150 psig in the CTS, to two pump flow rate tests (one at high pressure and one at low pressure), consistent with the NUREG. The high pressure test (ITS SR 3.5.3.3) would be performed in a pressure range of 1040 psig - 970 psig, which represents a nominal value within the normal operating reactor pressure range.

The low pressure test (ITS SR 3.5.3.4) would be performed at a pressure less than or equal to 165 psig (as opposed to a pressure greater than or equal to 150 psig in CTS). The test frequency at low pressure is changed from every 92 days to every 24 months. Reactor pressure of 165 psig or below is near the lower limit of operability/capability of the RCIC turbine, yet provides a small range to the lower limit to conduct the test. This pressure range represents conditions of lower driving pressure for the RCIC turbine, and thus, is a more restrictive condition under which to provide the required flow.

Accordingly, these changes will ensure that the RCIC system is tested at both the high and low pressure ranges of its operating capabilities at the proposed frequencies and is considered more restrictive on plant operation than the CTS. These tests are necessary to ensure that RCIC remains operable over its full pressure range. Therefore, this change is acceptable.

(15) <u>ITS SR 3.6.1.1.1 (DOC L3), Primary Containment Leakage Rate Testing Program</u>

The proposed change deletes the current JAFNPP CTS 4.7.A.1. CTS 4.7.A.1 requires inspection of the accessible interior surfaces of the drywell and above the water line of the torus (suppression chamber) once every 24 months.

The licensee stated that the visual examination required by CTS 4.7.A.2.a (ITS surveillance requirement (SR) 3.6.1.1.1) duplicates the visual examination required by CTS 4.7.A.1 except for the frequency of the required examinations. CTS 4.7.A.2.a (ITS SR 3.6.1.1.1) is required by the Primary Containment Leakage Rate Testing Program, which is based on Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, Option B. It requires visual inspection be performed prior to each Type A test and two additional times during each 10-year interval. Thus, the CTS 4.7.A.2.a (ITS SR 3.6.1.1.1) requires visual examination be performed at least 3 times in each 10-year period while the CTS 4.7.A.1 required visual examination be performed 5 times in a 10-year period. The licensee stated that additional examinations are performed as required by the inservice inspection (ISI) program and every 5 years as required by the maintenance rule. The licensee further stated that the results of examinations conducted over more than 20 years of plant operation and through 14 refueling outages has shown that no significant deterioration has taken place.

Based on the above, and the requirements of 10 CFR 50.55a(ix), “Examination of metal containments and the liners of concrete containments,” Section (E), which requires a licensee to perform visual examination, as required by Subsection IWE, 3 times in a 10-year interval, the staff finds the proposed change to delete CTS 4.7.A.1 is inconsequential as far as containment visual examination is concerned, and that the additional examinations required by CTS 4.7.A.2.a (ITS SR 3.6.1.1.1) and the maintenance rule, as stated by the licensee, provide an added enhancement. Therefore, the proposed change is acceptable.

(16) ITS 3.6.1.3 (DOC L13) Primary Containment Isolation Valves

The proposed change revises LPCI and CS testable check valve testing requirements to the frequency in the Primary Containment Leakage Rate Testing Program (PCLRT).

The staff reviewed the licensee's proposed change and finds that FitzPatrick’s LPCI and CS Systems containment isolation is accomplished by the use of two motor-operated valves outside the containment and one air operated check valve inside the containment. The motor operated isolation valves are tested per the IST program. The staff finds the small increase in the test interval (6 months, until test results indicate additional relaxation is acceptable) of the LPCI and CS system air operated check valves and the change to "In accordance with the Primary Containment Leakage Rate Testing Program" to be acceptable. The staff concurs with the licensee that testing over a long period has demonstrated reliability, and that other isolation methods exist to limit potential leakage. The staff also agrees that diverse instrumentation methods to detect potential leaks to the reactor building makes the proposed relaxation of the test frequency acceptable. The staff, therefore, finds the proposed revision of the SR for the testing frequency of LPCI and CS system air operated testable check valves from "once per 24 months" in CTS 4.7.D.2.c to ITS SR 3.6.1.3.11 "In accordance with the Primary Containment Leakage Rate Testing Program" acceptable.

(17) ITS SR 3.6.1.7.2, (DOC L4), Suppression Chamber-to-Drywell Vacuum Breakers

The proposed change revises the frequency of performing a functional test of each required vacuum breaker from 31 days to a new schedule in accordance with the IST Program which is 92 days.

The Frequency of CTS 4.7.A.5.a, which requires exercising each Suppression Chamber-to-Drywell vacuum breaker through an open-close cycle, is proposed to be extended from “monthly” to a frequency that is “In Accordance with the Inservice Testing

(IST) Program" in the proposed IST SR 3.6.1.7.2. The licensee stated that at FitzPatrick, the vacuum breakers are not located in the harsh environment of the suppression chamber. The valves are located in the reactor building (secondary containment) where the environment is similar to that which exists for many primary and secondary containment isolation valves which are subjected to tests on a frequency that is in accordance with the IST Program (92 days). In addition, similar SRs for the Reactor Building-to-Suppression Chamber vacuum breakers, which are of a similar design, have similar design functions, are also located in the reactor building, and are performed on a frequency that is in accordance with the IST Program as stated in CTS 4.7.A.4.a (ITS SR 3.6.1.6.2).

The licensee stated in DOC L4 that a historical review of Suppression Chamber-to-Drywell vacuum breaker surveillance data has been performed for the past 5 years and the data indicate there were no failures of the vacuum breakers to properly operate through a full open-close cycle operation. Therefore, based on (1) the valve reliability performance and (2) the longer test interval that has been approved for the similar Reactor Building-Suppression Chamber vacuum breakers and other valves located in areas with a similar environment (not a harsh environment), the staff finds the proposed extension of the SR in the proposed ITS SR 3.6.1.7.2 from the current 31 days to a frequency that is "In accordance with Inservice Testing Program" (92 days) to be acceptable.

(18) <u>ITS 3.6.2.3.2 (JFD PA4) - Residual Heat Removal (RHR) Suppression Pool Cooling</u>.

The proposed change revises ITS SR 3.6.2.3.2 by adding the word "required" to make it clearer that the SR is applicable to only the single required RHR pump in a subsystem rather than both pumps in a subsystem that are provided by design. The licensee also added "required" to the Bases of SR 3.6.2.3.2 to clarify that all RHR pumps need not be tested under this SR. In response to the staff RAI, the licensee stated that only one RHR pump is needed to satisfy the operability requirements of an RHR subsystem and that one RHR subsystem is capable of maintaining the primary containment peak pressure and temperature below design limits. The licensee has further stated that "The RHR pumps are required to be tested by the IST program every 92 days," and there is no change in the RHR pump testing frequency or testing requirements under ITS SR 3.5.1.7. This is consistent with the JAFNPP licensing basis as described in the UFSAR. Hence, the licensee's proposal to test only one RHR pump for suppression pool cooling as proposed in ITS SR 3.6.2.3.2 is acceptable.

(19) <u>ITS 3.8.4 (DOCs M2 and L4) DC Sources - Operating</u>

The JAFNPP CTS 3.9.E currently requires that during power operation, and if one battery becomes unavailable, repairs shall be made immediately, and continued reactor operation is permissible for a period not to exceed 7 days total per calender-month provided that:

- The other battery including its battery charger, and distribution systems is operable.
- Pilot cell voltage, specific gravity, and overall voltage and temperature is measured immediately and daily thereafter for the operable battery.
- The availability of the unaffected Emergency Diesel Generator System shall be demonstrated in accordance with Specification 4.9.B.5.

The CTS 3.9.E actions do not include a specific Action for an inoperable 125 VDC battery charger. Therefore, if a 125 VDC charger is inoperable, CTS 3.0.C must be entered and the plant must be in Cold Shutdown in 24 hours. The new ITS 3.8.4 (from previous CTS 3.9.E) combines the inoperability of a charger with its associated battery, referred to as a 125 VDC power subsystem.

The licensee has proposed to allow 8 hours to restore the inoperable 125 VDC power subsystem (i.e., inoperable battery, inoperable battery charger, or inoperable battery charger and the associated inoperable battery) to Operable status. The licensee stated that the 8-hour Completion Time has been selected because it allows sufficient time for operator assessment and action for restoring the division of 125 VDC electrical power and also minimizes the time operating without a full compliment of equipment.

The staff reviewed the information provided by the licensee in its submittal and concludes that the proposed change to allow 8 hours for the 125 VDC power subsystem rather than 7 days for an inoperable battery is more conservative and therefore acceptable.

4.0 COMMITMENTS RELIED UPON

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

5.0 LICENSE CONDITIONS

A license condition to define the schedule to begin performing the new and revised SRs after the implementation of the ITS is included in the license amendment issuing the ITS. This schedule is:

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

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

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

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

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

Also, a license condition is included that will enforce the relocation of requirements from the CTS to licensee-controlled documents. The relocations are provided in Table R, “Relocated Specifications,” and in Table LA, “Removal of Details Matrix.” The license condition states that the relocations will be completed upon implementation of the ITS. However, relocations to the UFSAR shall be reflected in updates completed in accordance with 10 CFR 50.71(e). This schedule is acceptable.

Finally, two license conditions are deleted because the first condition’s requirements have been incorporated into the ITS, and the second condition’s requirements need not be located in the ITS pursuant to 10 CFR 50.36; rather, they may be relocated to plant controlled documents. License Condition 2.C.(4) required a program to be implemented to reduce leakage from the systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This license condition has been incorporated into ITS 5.5.2. License Condition 2.C.(5) required a program to be implemented to ensure the capability to accurately determine the airborne iodine concentration in areas vital to the mitigation of or recovery from an accident. The requirement related to this license condition has been relocated to the UFSAR as discussed in the attached Table LA.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the ITS conversion amendment for JAFNPP. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on August 14, 2001 (66 FR 42683) for the proposed conversion of the CTS to ITS for JAFNPP. The licensee’s supplements dated February 6, March 27, April 26, June 11, and June 12, 2002 (two letters) provided clarifying information, and did not change the scope of the requested action. Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

8.0 CONCLUSION

The JAFNPP ITS provides clearer, more readily understandable requirements to ensure safe operation of the plant. The NRC staff concludes that the ITS for JAFNPP satisfy the guidance in the Final Policy Statement on TS improvements for nuclear power reactors with regard to the content of TS, and conform to the STS provided in NUREG-1433, Revision 1, or NUREG-1434, Revision 1, with appropriate modifications for plant-specific considerations. The NRC staff further concludes that the ITS satisfy Section 182a of the Atomic Energy Act, 10 CFR 50.36, and other applicable standards. On this basis, the NRC staff concludes that the proposed ITS for JAFNPP are acceptable.

The NRC staff has also reviewed the plant-specific changes to the CTS as described in this SE. On the basis of the evaluations described herein for each of the changes, the NRC staff also concludes that these changes are acceptable.

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security, or to the health and safety of the public.

Attachments:
1. List of Acronyms
2. Table A of Administrative Changes Matrix
3. Table M of More Restrictive Changes Matrix
4. Table L of Less Restrictive Changes Matrix
5. Table LA of Removal of Details Matrix
6. Table R of Relocated Specifications

Principal Contributors:	C. Schulten	A. Chu	R. Tjader
	N. Le	J. Foster	R. Giardina
	B. Marcus	G. Thomas	C. Harbuck
	G. Vissing	P. Gill	B. Thomas
	J. Lazevnick	H. Li	K. Kavanagh
	R. Goel	E. Tomlinson	

Date: July 3, 2002

List of Acronyms

AC	Air Conditioning or Alternating Current
ADS	Automatic Depressurization System
AOT	Allowed Outage Time
APLHGR	Average Planar Linear Heat Generation Rate
APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
ATWS-RPT	Anticipated Transient Without Scram - Recirculation Pump Trip
BPWS	Banked Position Withdrawal Sequence
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CFR	Code of Federal Regulations
CFT	Channel Functional Test
COLR	Core Operating Limits Report
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CREF	Control Room Envelope Filtration
CST	Condensate Storage Tank
CTS	Current Technical Specification
DBA	Design-Basis Accident
DC	Direct Current
DG	Diesel Generator
DOC	Discussion of Change (from the CTS)
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFCV	Excess Flow Check Valve
EOC-RPT	End of Cycle - Recirculation Pump Trip
EPA	Electrical Protection Assembly
ESF	Engineered Safeguard Feature
FR	Federal Register
FRTP	Fraction of Rated Thermal Power
GDC	General Design Criteria
GE	General Electric
HEPA	High Efficiency Particulate Air
HPCS	High Pressure Core Spray
Hz	Hertz
IRM	Intermediate Range Monitor
ISI	Inservice Inspection
ITS	Improved (converted) Technical Specifications
Kv	Kilovolt
kW	Kilowatt
LCO	Limiting Condition for Operation
LHGR	Linear Heat Generation Rate
LLS	Low-Low Set
LOCA	Loss of Coolant Accident

LOOP	Loss of Offsite Power
LOP	Loss of Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LPRM	Local Power Range Monitor
LSFT	Logic System Functional Test
MCPR	Minimum Critical Power Ratio
MFLPD	Maximum Fraction of Limiting Power Density
MG	Motor Generator
MSIV	Main Steam Isolation Valve
MWD/T	Megawatt Days/short Ton
NMP2	Nine Mile Point Unit 2
NUMAC	Nuclear Measurement Analysis and Control
OPDRV	Operation with a Potential for Draining the Reactor Vessel
PAM	Post-Accident Monitoring
P/T	Pressure/Temperature
QA	Quality Assurance
RAI	Request for Additional Information
RBM	Rod Block Monitor
RCS	Reactor Coolant System
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSCS	Rod Sequence Control System
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
RWM	Rod Worth Minimizer
SCIV	Secondary Containment Isolation Valve
SDC	Shutdown Cooling
SDM	Shutdown Margin
SDV	Scram Discharge Volume
SE	Safety Evaluation
SER	Safety Evaluation Report
SGT	Standby Gas Treatment
SLC	Standby Liquid Control
SR	Surveillance Requirement
SRM	Source Range Monitor
SRV	Safety/Relief Valve
SSER	Supplemental Safety Evaluation Report
STS	Improved Standard Technical Specification(s), NUREG-1433/4, Rev. 1
SW	Service Water
TRM	Technical Requirements Manual
TS	Technical Specifications

TSTF	Technical Specifications Task Force (re: generic changes to the STS)
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply
USAR	Updated Final Safety Analysis Report
V	Volt
VAC	Volts Alternating Current

cc:

Mr. Jerry Yelverton
Chief Executive Officer
Entergy Operations
1340 Echelon Parkway
Jackson, MS 39213

Mr. Theodore H. Sullivan
Vice President Operations
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Mr. Dan Pace
Vice President, Engineering
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. John Kelly
Director - Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. George Tasick
Licensing Manager
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Resident Inspector's Office
U. S. Nuclear Regulatory Commission
P.O. Box 136
Lycoming, NY 13093

Mr. Harry P. Salmon, Jr.
Director of Oversight
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Ms. Charlene D. Faison
Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Supervisor
Town of Scriba
Route 8, Box 382
Oswego, NY 13126

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

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

Oswego County Administrator
Jack Tierney
46 East Bridge Street
Oswego, New York 13126

Mr. William M. Flynn, President
New York State Energy, Research,
and Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

Mr. Arthur Zaremba, Licensing Manager
Director, Safety Assurance
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Mr. Paul Eddy
Electric Division
New York State Dept. of Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

FitzPatrick Nuclear Power Plant

cc:

Michael J. Colomb
General Manager
Entergy Nuclear Operations, Inc.
James A. FitzPatrick Nuclear Power Plant
P.O. Box 110
Lycoming, NY 13093

Mr. James Knubel
Vice President, Operations Support
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. John M. Fulton
Assistant General Counsel
Entergy Nuclear Generation Co.
Pilgrim Station
600 Rocky Hill Road
Plymouth, MA 02360

Mr. J. Spath, Program Director
New York State Energy, Research, and
Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

Mr. Ronald Schwartz
SRC Consultant
64 Walnut Drive
Spring Lake Heights, NJ 07762

Mr. Ronald J. Toole
SRC Consultant
Toole Insight
605 West Horner Street
Ebensburg, PA 15931

Mr. Charles W. Hehl
SRC Consultant
Charles Hehl, Inc.
1486 Matthew Lane
Pottstown, PA 19465

Mr. Tim Judson
Organizer
Citizens Awareness Network
140 Bassett St.
Syracuse, NY 13210

Deborah Katz
Executive Director
Citizens Awareness Network
P.O. Box 83
Shelburne Falls, MA 01370

Shawn McConnell
NYPIRG Project Coordinator
13 Hewitt Union
SUNY Oswego
Oswego, NY 13126

Tom Dellwo
Oswego NYPIRG Environmental
Project Leader
13 Hewitt Union
SUNY Owego
Oswego, NY 13126