July 20, 2001

Mr. L. W. Myers
Senior Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Post Office Box 4
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT RE: IMPLEMENTATION OF THE REVISED THERMAL DESIGN PROCEDURE, ETC. (TAC NOS. MB0848 AND MB0849)

Dear Mr. Myers:

The Commission has issued the enclosed Amendment No. 239 to Facility Operating License No. DPR-66 and Amendment No. 120 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 27, 2000, as supplemented by letters dated March 28, April 12, June 9, June 13, and June 29 (3), 2001.

These amendments approve: (1) implementation of the revised thermal design procedure (RTDP), (2) revisions to reactor trip system and engineered safety feature actuation system trip setpoints and allowable values, (3) the addition of a TS Bases control program, (4) relocation of certain TS requirements to the core operating limits report, the licensing requirements manual, or to the TS Bases, and (5) other changes including the deletion of a license condition.

Implementation of the RTDP facilitates a 1.4 percent power uprate at each unit, which the licensee has requested in a separate license amendment request, dated January 18, 2001.

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

Sincerely,

/RA/

Lawrence J. Burkhart, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures: 1. Amendment No. 239 to DPR-66
2. Amendment No. 120 to NPF-73
3. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

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PUBLIC	SLittle	ACRS	FAkstulewicz
PDI-1 R/F	LBurkhart	BPlatchek, RI	RDennig
EAdensam	OGC	EMarinos	RCorreia
GHill (4)	WBeckner		

ACCESSION NO. ML011910223 *See previous concurrence

OFFICE	PDI-1/PM	PDI-2/LA	SC/EEIB*	SC/SRXB*	SC/RTSB*	PDI-1/SC(A)	OGC*
NAME	LBurkhart	SLittle for MO'Brien	EMarinos	FAkstulewicz	RDennig	RCorriea	AHodgdon
DATE	7/18/01	7/18/01	7/11/01	7/11/01	7/12/01	7/18/01	7/16/01

OFFICIAL RECORD COPY

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PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 239
License No. DPR-66

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee) dated December 27, 2000, as supplemented on March 28, April 12, June 9, June 13, and June 29 (3), 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-66 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 239, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard Correia, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Page 4 of the License

Date of Issuance: July 20, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 239

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following page of the Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
Page 4	Page 4

Replace the following page of Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
II	II
III	III
X	X
XV	XV
XVI	XVI
XIX	XIX
1-8	1-8
Section 2.0 cover sheet	Section 2.0 cover sheet
2-1	2-1
2-2	-
2-5	-
2-6	-
2-7	-
2-7a	-
2-8	-
2-9	-
2-10	-
Bases for Safety Limits and Limiting Safety System Settings cover sheet	Bases for Safety Limits cover sheet
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	-
B 2-4	-
B 2-5	-
B 2-6	-
B 2-7	-
B 2-8	-
3/4 2-12	3/4 2-12
3/4 2-13	-
3/4 3-1	3/4 3-1
3/4 3-2	3/4 3-2

Remove	Insert
3/4 3-3	3/4 3-3
3/4 3-4	3/4 3-4
-	3/4 3-4a
3/4 3-5	3/4 3-5
-	3/4 3-5a
-	3/4 3-5b
3/4 3-14	3/4 3-14
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-19a	3/4 3-19a
3/4 3-19b	3/4 3-19b
3/4 3-21	3/4 3-21
3/4 3-22	-
3/4 3-22a	-
3/4 3-23	-
3/4 3-24	-
3/4 3-24a	-
3/4 3-24b	-
3/4 3-31	3/4 3-31
3/4 3-31a	3/4 3-31a
3/4 10-4	3/4 10-4
3/4 10-6	3/4 10-6
3/4 10-7	3/4 10-7
B 3/4 2-11	B 3/4 2-11
B 3/4 3-1	B 3/4 3-1
B 3/4 3-1a	B 3/4 3-1a
B 3/4 3-1b	B 3/4 3-1b
B 3/4 3-1c	B 3/4 3-1c
B 3/4 3-1d	B 3/4 3-1d
B 3/4 3-1e	B 3/4 3-1e
B 3/4 3-1f	B 3/4 3-1f
B 3/4 3-1g	B 3/4 3-1g
B 3/4 3-1h	B 3/4 3-1h
-	B 3/4 3-1i
-	B 3/4 3-1j
-	B 3/4 3-1k
-	B 3/4 3-1l
-	B 3/4 3-1m
-	B 3/4 3-1n
-	B 3/4 3-1o
-	B 3/4 3-1p
B 3/4 4-1	B 3/4 4-1
6-18	6-18
6-19	6-19
6-26	6-26

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. NPF-73

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

   B. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (the licensee) dated December 27, 2000, as supplemented on March 28, April 12, June 9, June 13, and June 29 (3), 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No.120, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard Correia, Acting Chief, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 20, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 120

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

Replace the following pages of Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
II	II
III	III
XIV	XIV
XV	XV
1-7	1-7
Section 2.0 cover sheet	Section 2.0 cover sheet
2-1	2-1
2-2	-
2-3	-
2-4	-
2-5	-
2-6	-
2-7	-
2-8	-
2-9	-
2-10	-
Bases for Safety Limits and Limiting Safety System Settings cover sheet	Bases for Safety Limits cover sheet
B 2-1	B 2-1
B 2-2	B 2-2
B 2-3	-
B 2-4	-
B 2-5	-
B 2-6	-
B 2-7	-
B 2-8	-
3/4 2-11	3/4 2-11
3/4 2-12	-
3/4 3-1	3/4 3-1
3/4 3-2	3/4 3-2
3/4 3-3	3/4 3-3
3/4 3-4	3/4 3-4
3/4 3-4a	3/4 3-4a
3/4 3-5	3/4 3-5
3/4 3-5a	3/4 3-5a
3/4 3-5b	3/4 3-5b
3/4 3-5c	3/4 3-5c
3/4 3-14	3/4 3-14

Remove	Insert
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
-	3/4 3-20a
3/4 3-22a	3/4 3-22a
3/4 3-23	-
3/4 3-24	-
3/4 3-25	-
3/4 3-26	-
3/4 3-27	-
3/4 3-28	-
3/4 10-3	3/4 10-3
B 3/4 2-11	B 3/4 2-11
B 3/4 3-1	B 3/4 3-1
B 3/4 3-1a	B 3/4 3-1a
-	B 3/4 3-1b
-	B 3/4 3-1c
-	B 3/4 3-1d
-	B 3/4 3-1e
-	B 3/4 3-1f
-	B 3/4 3-1g
B 3/4 3-2	B 3/4 3-2
B 3/4 3-3	B 3/4 3-3
B 3/4 3-5	B 3/4 3-5
6-18	6-18
6-19	6-19
-	6-20
6-26	6-26
-	6-27

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 239 AND 120 TO FACILITY OPERATING

LICENSE NOS. DPR-66 AND NPF-73

PENNSYLVANIA POWER COMPANY

OHIO EDISON COMPANY

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

THE TOLEDO EDISON COMPANY

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION

By letter dated December 27, 2000 (Ref. 1), as supplemented on March 28 (Ref. 2), April 12 (Ref. 3), June 9 (Ref. 4), June 13 (Ref. 5), and June 29 (Ref. 6, 7, and 8), 2001, the FirstEnergy Nuclear Operating Company, et al. (FENOC; the licensee) submitted a request for changes to the Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2), Technical Specifications (TSs) and operating licenses. The amendment request proposes (1) implementation of the revised thermal design procedure (RTDP) which would result in revisions to the core safety limits, the departure from nucleate boiling (DNB) parameters, and the overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT) trip setpoints, (2) revision of reactor trip system (RTS) and engineered safety feature actuation system (ESFAS) instrumentation trip setpoints and allowable values, (3) relocation of certain requirements from the TS to the core operating limits report (COLR) or to the licensing requirements manual (LRM), and (4) other changes including the deletion of license condition 2.C(3) regarding less than 3-loop operation. Editorial and administrative changes and TS Bases changes were proposed for consistency and clarity.

In support of the proposed RTS instrumentation and ESFAS instrumentation trip setpoints and allowable values changes and the implementation of the RTDP, the licensee submitted, as attachments to its December 27, 2000, request, the following proprietary plant-specific topical reports (TRs):

(1) “Westinghouse Setpoint Methodology for Protection Systems for Beaver Valley Power Station - Unit 1, WCAP-11419, Revision 2, December 2000" (WCAP-11419) (Ref. 9);

(2) “Westinghouse Setpoint Methodology for Protection Systems for Beaver Valley Power Station - Unit 2, WCAP-11366, Revision 4, December 2000" (WCAP-11366) (Ref. 10);

(3) “Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 1, WCAP-15264, Revision 3, December 2000" (WCAP-15264) (Ref. 11); and

(4) “Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 2, WCAP-15265, Revision 2, December 2000" (WCAP-15265) (Ref. 12).

The licensee also submitted the following non-proprietary versions of the TRs:

(1) “Westinghouse Setpoint Methodology for Protection Systems for Beaver Valley Power Station - Unit 1, WCAP-15407, December 2000" (WCAP-15407) (Ref. 13);

(2) “Westinghouse Setpoint Methodology for Protection Systems for Beaver Valley Power Station - Unit 2, WCAP-15408, December 2000" (WCAP-15408) (Ref. 14);

(3) “Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 1, WCAP-15336, Revision 2, December 2000" (WCAP-15336) (Ref. 15); and

(4) “Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 2, WCAP-15337, Revision 2, December 2000" (WCAP-15337) (Ref. 16).

By letter dated June 9, 2001, the licensee revised some of the uncertainty allowances associated with the use of the Caldon Leading Edge Flow Meter (LEFM) Checkplus™ system for BVPS-2 and submitted Revision 3 to WCAP-15265 (Ref. 17).

The March 28, April 12, June 9, June 13, and June 29 (3), 2001, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

During review of the December 27, 2000, amendment request, it became evident that adoption of a TS Bases control program would be necessary to justify some of the relocations requested. The licensee submitted an amendment dated March 28, 2001 (Ref. 18), that, in part, requested addition of a TS Bases control program. The June 13, 2001, letter requested that the TS Bases control portion of the March 28, 2001, amendment request be issued in a manner to support the December 27, 2000, amendment request. Consequently, the TS Bases control program portion of the March 28, 2000, amendment request is included in this amendment and is discussed in this safety evaluation (SE).

The NRC staff’s initial proposed no significant hazards consideration determination for the March 28, 2001, amendment request that included the addition of a TS Bases control program was published in the *Federal Register* on June 20, 2001 (66 FR 33111).

2.0 BACKGROUND

The licensee currently uses the mini-RTDP (Ref. 19) for safety analyses for BVPS-1 and 2. This amendment request proposes to replace the mini-RTDP methodology with the RTDP methodology. The proposed revisions to the core safety limits, DNB parameters, and the OTΔT and OPΔT trip setpoints are a result of implementation of the RTDP (Ref. 20). The licensee also intends to use the Caldon Leading Edge Flow Meter (LEFM)✓™ and LEFM CheckPlus™ systems for the feedwater flow measurements at BVPS-1 and 2, respectively, which reduces the total power calorimetric measurement uncertainty. With the RTDP methodology and the reduction of total power calorimetric measurement uncertainty, an increase in DNB margin is realized in the DNB safety analysis. This DNB margin gain facilitates the implementation of a 1.4 percent power uprate at each unit, which the licensee has requested in a separate license amendment request, dated January 18, 2001 (Ref. 21).

In the standard deterministic method of safety analyses, the uncertainties of important reactor system parameters are included in the analyses by conservatively assuming all adverse uncertainty conditions occurring simultaneously. A design limit for the DNB ratio (DNBR) is established such that there is at least a 95 percent probability with a 95 percent confidence level that DNB will not occur when the calculated DNBR is equal to or greater than the design limit. The design DNBR limit is generated from the critical heat flux (CHF) correlation that is used in the analyses, based on the capability and quality of the correlation to predict the CHF test data.

Alternatively, the uncertainties of certain reactor system parameters are accounted for statistically, and are combined with the uncertainty of the CHF correlation to establish the design DNBR limit. The nominal values of these parameters are then used in the safety analyses. Compared to the deterministic approach, the statistical treatment of uncertainties would increase the design DNBR limit; however, an overall DNB margin gain would be realized through the use of nominal values of the parameters whose uncertainties have been accounted for in the design DNBR limit.

Westinghouse developed several methodologies in the statistical treatment of instrumentation uncertainties for the DNBR analysis in Westinghouse-designed plants: improved thermal design procedure (ITDP) (Ref. 22), RTDP, and mini-RTDP. In the ITDP, the uncertainties of the system parameters are statistically combined. The system parameter uncertainty statistics and the CHF correlation statistics are then combined directly, rather than statistically, to establish the design DNBR limit. The RTDP is an extension of the ITDP, in that the uncertainties of the system parameters and the CHF correlation uncertainty are statistically, rather than deterministically, combined into the calculation of the design limit DNBR. In both the ITDP and RTDP methodologies, the system uncertainties considered include those associated with reactor system operating parameters, fabrication parameters, nuclear and thermal parameters. The mini-RTDP is a conservative application of the RTDP in that only the uncertainties in the nuclear peaking factors and fuel fabrication parameters are combined with the CHF correlation uncertainties to define the design DNBR limit. The uncertainties of the reactor system operating parameters (e.g., reactor power, flow, temperature, pressure) are excluded from the statistical combination process.

The ITDP, RTDP, and mini-RTDP methodologies have been accepted by NRC for use in licensing applications, and have been used widely in the Westinghouse-designed plants.

However, the uncertainty values of the reactor system parameters included in the statistical treatment are evaluated on a plant-specific basis.

In Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, the Commission established its regulatory requirements for TS content. In doing so, the Commission emphasized those matters related to preventing accidents and mitigating 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" (see Statement of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports," of December 17, 1968 (33 FR 18610)).

10 CFR 50.36 requires that TS include items in the following five specific categories: (1) safety limits, limiting safety system settings (LSSS) and limiting control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls.

However, the rule does not specify particular TS requirements. Therefore, NRC and industry representatives worked to develop guidelines for improving nuclear power plant TS content and quality. On February 6, 1987, the Commission issued their "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). In September 1992, the Commission issued the Westinghouse standard TS (STS) as NUREG-1431 (and also STS for the other Nuclear Steam System Supplier [NSSS] vendors), which were developed using the guidance and criteria contained in the Commission's Interim Policy Statement. The Westinghouse STS are a model for developing improved TS for Westinghouse plants. The results from applying the Interim Policy Statement criteria to generic system functions were published in a "Split Report" issued to the NSSS Owners Groups in May 1988. The Interim Policy Statement criteria along with the Writer's Guide ensured that the improved TS would consistently reflect system configurations and operating characteristics for all NSSS designs. In addition, the generic Bases provide considerable information about the basis for the STS requirements.

On July 22, 1993, the Commission issued its Final Policy Statement indicating 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 improved STS safety benefits and encouraged licensees to use the improved STS as the basis for plant-specific TS amendments and for complete conversions to the improved STS. Further, the Final Policy Statement gave guidance for evaluating the required scope of plant-specific TS and defined the guidance criteria for determining which of the LCOs and associated surveillances should remain in 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 Company's hearing (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed the following:

> [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.

Using this approach, licensees should keep in TS existing LCO requirements that fall within or satisfy any of the Final Policy Statement criteria. 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 36593, July 19, 1995).

During review of individual plant license amendments to adopt improved STS the NRC and industry representatives continued to make generic improvements in STS content through joint meetings with the staff and the Nuclear Energy Institute Technical Specifications Task Force Owners Group. A generic change process, Technical Specification Task Force Travelers, was developed to ensure equivalent treatment of technical issues and format changes common to one or more vendor STS, thus preserving the baseline standardization of the improved STS. As a result of this process, Revision 1 of the NUREG STS was issued in April 1995, followed by Revision 2 in April 2001.

3.0 EVALUATION

3.1 Changes Associated With Implementation of the RTDP

For this license amendment request, the licensee proposes to replace the mini-RTDP methodology with the RTDP methodology for safety analyses. The use of the RTDP methodology would include four operating parameters uncertainties. These parameters are pressurizer pressure, primary coolant temperature (T-avg), reactor power, and reactor coolant system (RCS) flow, which are frequently monitored and used for control purposes. In addition, the licensee will use the Caldon LEFM✓™ and LEFM CheckPlus™ systems for the feedwater flow measurement in BVPS-1 and 2, respectively, which reduces the feedwater flow measurement uncertainty and, consequently, the power measurement uncertainty. With the RTDP methodology and the reduction of measurement uncertainties, an increase in DNB margin is realized in the DNB safety analysis.

As a result of implementation of the RTDP methodology, the design DNBR limit is revised. The reactor core safety limits (RCSL) figure, which defines the acceptable operating regions in a reactor power-average coolant temperature-pressure map such that the design DNBR limit and boiling limit are not violated, is revised. The RTS OTΔT and OPΔT setpoints, which are based on the RCSLs, are also revised.

Though the RTDP methodology has been accepted by the NRC for licensing applications, each licensee is required to justify the plant-specific uncertainty values of the system parameters included in the RTDP. For this amendment request, the review will be limited to the four operating parameters, which are the difference between the RTDP and mini-RTDP currently used by BVPS-1 and 2.

3.1.1 Implementation of RTDP Methodology - Operating Parameters Measurement Uncertainties

In Attachment D to the December 27, 2000, letter, the licensee provided WCAP-15264, Revision 3, and WCAP-15265, Revision 2, which describes RTDP instrumentation uncertainty methodology for BVPS-1 and 2, respectively. These TRs document the detailed calculations of the measurement uncertainties of the pressurizer pressure, T-avg, RCS flow, and reactor power. Pressurizer pressure is a controlled parameter, and the uncertainty reflects the control system. Reactor coolant temperature is a controlled parameter via the temperature input to the rod control system, and the uncertainty reflects this control system. RCS flow is monitored using the RCS cold leg indicators, which are normalized by the performance of a calorimetric flow measurement at the beginning of each fuel cycle. Reactor power is monitored by the daily performance of a secondary side heat balance. The uncertainty calculations of these four parameters are documented in WCAP-15264 and WCAP-15265, Section III, with the final uncertainty value of each parameter summarized in Section IV. The calculations also include the instrumentation uncertainties for the daily power calorimetric measurement at the 1.4 percent power uprate conditions using the Caldon LEFM✓™ system in the feedwater header for BVPS-1, and using the Caldon LEFM CheckPlus™ system for BVPS-2. The licensee, by its letter of June 9, 2000, submitted WCAP-15265, Revision 3, which revised some of the uncertainty allowances associated with the use of the LEFM CheckPlus™ system for BVPS-2. Therefore, the NRC staff's review of BVPS-2 operating parameters uncertainties is based on WCAP-15265, Revision 3.

Section II of WCAP-15264 and WCAP-15265 describes the methodologies used to calculate measurement channel instrument error allowances for precision parameter indication using special test equipment of a digital voltmeter at the process rack, the plant process computer, and parameters of control systems. The methodologies used to combine the error components for a channel to obtain the channel statistical allowance (CSA) is the square-root-of-sum-of-the-squares (SRSS) of those groups of components that are statistically independent. Those uncertainties that are dependent are combined arithmetically into independent groups, which are then systematically combined. The uncertainties used are considered to be random, two-sided distributions. This methodology is a common industry standard, has been used extensively in many license amendment applications and is accepted by NRC. The relationships between the error components and the channel instrument error allowance are variations of the basic Westinghouse setpoint methodology and are based on BVPS-1 and 2 specific procedures and processes.

Section III of WCAP-15264 and WCAP-15265 provides uncertainty values of various components associated with the measurements of the pressurizer pressure, T-avg, RCS flow, and reactor power, as well as the overall uncertainty values of these operating parameters. In its letter dated March 28, 2001, the licensee described the process used to generate and verify the uncertainty numbers listed in the TRs, and the process used to update the setpoints when a plant protection system or RTDP instrumentation is modified. The uncertainties calculation by Westinghouse utilized information such as the following:

- identification of the equipment such as transmitters, process racks, control board meters, etc.,

- identification of measurement and test equipment used to calibrate the transmitters and process racks,
- plant calibration and functional test procedures,
- drift data or drift magnitudes for transmitters and process racks,
- plant conditions for which the equipment is scaled, and
- plant calorimetric measurement procedures and specifications of the equipment used to perform the measurements.

Westinghouse then develops an uncertainty model for each function and determines the uncertainty for the control, protection or indication functions. When equipment is replaced, the licensee must assure that the replacement equipment is, with respect to the assumed uncertainty, equivalent to or better than the installed equipment. Verification of equivalency must include the confirmation of the same or better statistical characteristics to assure that the original calculation is still bounding.

WCAP-15264 and WCAP-15265, Tables 1 and 2, provide the instrumentation uncertainty values of various components associated with the measurement and the total CSAs of the pressurizer pressure and T-avg, which are controlled by automatic control systems. The pressure control channel uncertainties include allowances for the pressure transmitters, the process racks/indicators and the control systems. Also included are a limit of error to account for the seismic effects associated with the Rosemont transmitter, an allowance for pressure overshoot or undershoot, and a bias for temperature compensation of Barton transmitters and long-term drift effects. The T-avg control channel uncertainties include allowances for hot leg and cold leg streaming and the uncertainties associated with the use of various equipment such as resistance temperature detectors (RTD), turbine pressure transmitter, process racks/indicators and controller. Also included is the automatic control deadband.

The RCS flow surveillance is periodically performed with the process computer/control board indicators from the cold leg elbow tap transmitters to ensure that the RCS flow is maintained above the assumed safety analysis value. The elbow tap RCS flow measurement instrument channels are normalized at the beginning of every fuel cycle by a secondary side power-based calorimetric flow measurement. The RCS flow measurement uncertainties consist of the uncertainties associated with the calorimetric flow measurement and the control board indicators for periodic surveillance.

The calorimetric flow measurement is based on the primary and secondary side thermal equilibrium such that the RCS flow is determined from the steam generator thermal output with corrections for the reactor coolant pump (RCP) heat input and primary system heat loss, and the enthalpy rise of the primary coolant. The calculation of the calorimetric flow measurement considers measurements of such parameters as feedwater flow, feedwater enthalpy, steam enthalpy, RCP heat addition, hot leg and cold leg enthalpy, and bias. Table 5 of WCAP-15264 and WCAP-15265 provides the uncertainties of various components associated with calorimetric flow measurement, as well as the overall uncertainty value. The control board indicator uncertainties include those associated with the elbow tap pressure drop measurement, and ΔP transmitters. Table 6 of WCAP-15264 and WCAP-15265 provides the uncertainty values of various components associated with the cold leg elbow tap RCS flow surveillance and control board indicator, as well as the overall RCS flow measurement uncertainty.

The daily reactor power measurement is based on the steam generator thermal output. Assuming primary and secondary side equilibrium, the core power is determined by the sum of thermal output of steam generators, the primary side heat loss, and the RCP heat addition. The steam generator thermal output is determined by the secondary side calorimetric measurement, which is determined by multiplying the feedwater flow and the difference in the steam and feedwater enthalpy, with the correction of blowdown. The feedwater flow is measured using inputs from flow venturi and ΔP transmitters placed in the feedwater lines. The secondary side power calorimetric measurement uncertainties calculation, provided in Table 9 of WCAP-15264 and WCAP-15265, considers uncertainties associated with the feedwater flow venturi, feedwater and steam enthalpy, pump heat addition, steam generator blowdown flow and enthalpy.

The licensee also calculates the daily power measurement from the feedwater flow measurement using the Caldon LEFM✓™ and the LEFM CheckPlus™ systems for BVPS-1 and 2, respectively. The results of these measurements are used in place of the feedwater flow venturi measurement in the plant process computer. Tables 10 and 12 of WCAP-15264 and WCAP-15265 provide the component uncertainty values associated with the LEFM✓™ and LEFM CheckPlus™ feedwater flow measurements for BVPS-1 and 2, respectively. Except for the uncertainties associated with the feedwater flow measurement, other uncertainty components are the same as Table 9 for venturi feedwater flow measurement. The feedwater flow measurement uncertainty using Caldon LEFM✓™ system has been reviewed and accepted previously. The measurement uncertainty for the LEFM CheckPlus™ system used in WCAP-15265, Revision 2, has not been reviewed before. In its letter dated April 12, 2001, the licensee states that the LEFM CheckPlus™ system is similar to the LEFM✓™ system, except that it has twice as many transducers compared to the LEFM✓™ system. The LEFM CheckPlus™ is essentially two LEFM✓™ systems combined, and therefore provides feedwater flow measurement that is at least as accurate as that provided by the LEFM✓™ system. In WCAP-15265, Revision 3, the instrument uncertainty value of the feedwater flow for the LEFM✓™ system is used for the LEFM CheckPlus™ system. The NRC staff finds that the licensee provided sufficient justification to allow the same feedwater flow measurement uncertainty value for the LEFM CheckPlus™ system as that of the LEFM✓™ system. This value of uncertainty is reflected in Revision 3 to WCAP-15265.

Based on its review the NRC staff finds that the licensee has sufficiently identified, adequately evaluated, and accurately combined the plant-specific measurement components, parameters and associated uncertainties involved in the total power calorimetric measurement. The NRC staff finds the arithmetic in the CSA calculations of the four operating parameters in Section III of WCAP-15264, Revision 3, and WCAP-15265, Revision 3, to be consistent with the accepted uncertainty calculational methodology. In addition, the licensee has stated that it has processes to ensure that these uncertainties are maintained. Consequently, the NRC staff finds the proposed implementation of the RTDP including the treatment of operating parameter uncertainties acceptable.

3.1.2 Revision of Design and Safety Analysis DNBR Limits Based on RTDP

The proposed TS changes revise the Bases for TS 2.1.1, “Safety Limit” to reflect the DNBR limit changes resulting from the utilization of the RTDP.

BVPS currently uses the mini-RTDP procedure and currently has a DNBR limit of 1.17. The mini-RTDP includes the uncertainties of the nuclear peaking factors, fuel fabrication parameters, the THINC-IV thermal hydraulic code, and the WRB-1 CHF correlation. For conservatism, the design DNBR limits using the mini-RTDP methodology is 1.21. To maintain a margin in the safety analyses, the licensee is conservatively using a safety analysis DNBR limit of 1.33.

With the use of the RTDP, the uncertainty values of pressurizer pressure, T-avg, RCS flow and reactor power are calculated in WCAP-15264, Revision 3, and WCAP-15265, Revision 3, respectively, for BVPS-1 and 2. With the addition of these uncertainties to the uncertainties treated in the mini-RTDP methodology, the RTDP design DNBR limits are calculated to be 1.24 for typical cells and 1.23 for thimble cells. These RTDP design DNBR limits are calculated based on the RTDP methodology described in WCAP-11397-P-A. The actual derivations of these design DNBR limits are documented in the licensee’s letter dated June 29, 2001. The NRC staff has reviewed these calculations and concluded that these design DNBR limits for the typical and thimble cells are based on implementation of the RTDP, are conservative as compared to the current design DNBR limits and are, therefore, acceptable.

To maintain margin in the design safety analysis, the revised TS Bases, Section 2.1.1, originally indicated a safety analysis DNBR limit of 1.36 for both typical and thimble cells. However, in subsequent safety analyses related to a power uprate of 1.4 percent, the results show a minimum DNBR of 1.335 for the complete loss of RCS flow event for BVPS-1 and 2. The licensee revised the safety analysis DNBR limit to 1.33 to bound the analyses at uprated conditions. The margins between the safety analysis DNBR limit of 1.33 and the design DNBR limits are sufficiently large to accommodate a generic rod bow penalty of 1.3 percent. The licensee in its letter of June 29, 2001, revised the safety analysis DNBR limit to 1.33. The NRC staff concludes that the revised design DNBR limits of 1.24 and 1.23, respectively, for the typical and thimble cells, and the safety analysis DNBR limit of 1.33 provided in the revised TS Bases 2.1.1 are accurately determined and result from the proper implementation of the RTDP and are acceptable.

3.1.3 Revision to Reactor Core Safety Limits

The proposed TS changes would revise TS Figure 2.1-1, “Reactor Core Safety Limit, Three Loop Operation.” The RCSL figure defines the acceptable operation regions with various RTS functions that the specified acceptable fuel design limits (SAFDL), i.e., the design DNBR limit and the centerline fuel melting limit, are satisfied during normal operation and anticipated operational occurrences (AOO). The RCSL figure shows the loci of points of RCS average temperature as a function of rated thermal power (RTP) and pressurizer pressure for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. Since the design DNBR limits, as well as the safety analysis DNBR limits, are revised as a result of using the RTDP, the reactor core safety limits figure is revised accordingly.

In Attachments A-1 and A-2, to the December 27, 2000, letter, the licensee provided proposed revisions to Figure 2.1-1, for BVPS-1 and 2, respectively. The licensee, in its letter dated June 29, 2001, stated that these revised RCSLs are established based on the safety analysis DNBR limit of 1.36 and the 1.4 percent uprated power conditions. This revised RCSL figure would be more restrictive than if it were derived from the safety analysis DNBR limit of 1.33 at the current rated power conditions. The NRC staff finds that the proposed revisions to Figure 2.1-1 reflect the proper implementation of the RTDP, and conservatively reflect the change to the DNBR limits and are, therefore, acceptable.

3.1.4 Revision of the DNB Parameters

TS Table 3.2-1, "DNB Parameters," which is referenced in LCO 3.2.5, specifies the limits on T-avg, pressurizer pressure, and RCS total flow rate, to assure that each of these parameters is maintained within the normal steady state envelope of operation assumed in the design-basis transient and accident analyses. This license amendment request proposed to revise the DNB parameters values as follows:

For BVPS-1, the limiting values of RCS T-avg, pressurizer pressure, and RCS total flow would be revised from 580.7 °F, 2220 pound per square inch absolute (psia), and 261,600 gallons per minute (gpm), to 580.0 °F, 2215 psia, and 267,400 gpm, respectively. For BVPS-2, the limit values of RCS T-avg, pressurizer pressure, and RCS total flow would be revised from 580.2 °F, 2220 psia, and 261,600 gpm, to 579.9 °F, 2214 psia, and 267,200 gpm, respectively. In addition, the term "indicated value" would be added to the DNB parameters table to indicate the limits are based on the indicated values, rather than the analytical values used in the safety analysis.

These revisions are made because the use of the RTDP methodology provides additional operating margin to the DNB parameters. The RTDP takes advantage of the statistical combination of the uncertainties of reactor power, RCS flow, temperature, and pressure to establish the design and safety analysis DNBR limits. The nominal values of these operating parameters are used as the initial conditions of the design-basis transients in the safety analysis. By letter dated June 29, 2001, the licensee provided a summary of non-loss-of-coolant accident analyses with the nominal values of the operating parameters at the 1.4 percent uprated power conditions as the initial conditions. The results show that the minimum DNBRs are greater than the safety analysis DNBR limit of 1.33. The licensee also clarified the "indicated values" of the RCS T-avg and the pressurizer pressure are based on the "nominal values" used in the RTDP safety analysis and the indication uncertainties which are slightly conservative with respect to the uncertainty values used in determining the RTDP design DNBR limits. The NRC staff finds the changes to the DNB parameters of RCS T-avg, pressurizer pressure, and RCS flow in TS Table 3.2-1 reflect the proper implementation of the RTDP and the revised DNBRs and are, therefore acceptable.

The proposed TS changes would also revise the footnotes of LCO 3.2.5. The footnote (1), which stated that the values presented in Table 3.2-1 correspond to analytical limits used in the safety analysis, is deleted and is replaced with footnote (1) under current TABLE 3.2-1, which states that the pressurizer pressure limit is not applicable during either a THERMAL POWER ramp increase in excess of 5 percent RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10 percent RATED THERMAL POWER. The deletion of footnote (1) is consistent with the RTDP methodology. The replacement of footnote (1) under current Table 3.2-1 to LCO 3.2.5 is to maintain the current basis, and is acceptable.

As discussed in Section 3.4.1.2 of this SE, Table 3.2-1 is being relocated to the COLR with the relocated values changed from analysis values to indicated values. LCO 3.2.5.c is revised to specify that the RCS total flow rate is $\geq$ 261,600 gpm and $\geq$ the limit specified in the COLR. A new footnote (3) is added to LCO 3.2.5 to indicate that the RCS total flow rate of 261,600 gpm is the analytical limit used in the safety analysis. This number is the minimum NRC-approved value for RCS total flow. The proposed changes assure that a lower flow than that reviewed by NRC will not be used. This proposed change follows the guidance contained in Technical Specification Task Force (TSTF) proposal TSTF-339, Revision 2, and is acceptable.

3.1.5 Revision of OTΔT and OPΔT Trip Functions:

The OTΔT and OPΔT trip functions are designed to provide primary protection against DNB and fuel centerline melt through excessive linear heat generation rates (LHGR) during transients. Since the coolant temperature difference (ΔT) between the reactor vessel outlet and inlet is nearly proportional to the core power, which is a dominant parameter for the DNBR and LHGR, the indicated ΔT serves as a primary parameter for these trip functions. The analytical methods used to derive the LSSSs for the OTΔT and OPΔT trips are described in the Westinghouse TR WCAP-8745 (Ref. 23), which has been accepted by NRC for referencing in license applications.

3.1.5.1 Changes of OTΔT and OPΔT Trip Equations K Constants

The licensee proposed to revise the values of the OTΔT setpoint constant K_1 and OPΔT setpoint constant K_4. These revisions are made to provide operational margin and to be consistent with the RTDP methodology used. The methodology used to determine the values of K_1 and K_4 is described in Appendix B of WCAP-8745. The K_4 value of the OPΔT trip equation is based on the limit of 118 percent of nominal power. The K_1 value of the OTΔT trip equation is derived based on the RCSLs and is chosen to ensure the OTΔT trip equation satisfies the limits imposed by the OPΔT protection line and the steam generator safety valve open line in accordance with the methodology described in WCAP-8745. As described in Section 3.1.3 of this report, the RCSLs are revised as a result of using the RTDP. The values of the constants K_1 and K_4 in the OTΔT and OPΔT trip function equations are revised accordingly.

The OTΔT is credited in the analyses of the uncontrolled rod cluster control assembly bank withdrawal at power, loss of external electrical load and/or turbine trip, and accidental depressurization of the RCS. The OPΔT is not credited in the design-basis transients and accidents analyses. The licensee has re-analyzed the design-basis transients with the revised OTΔT trip equation to demonstrate the safety analysis DNBR limit is not violated. The NRC staff finds that the revised values of the constants K_1 and K_4 in the OTΔT and OPΔT trip function are appropriate and are a result of the implementation of the RTDP and are, therefore, acceptable.

3.1.5.2 Deletion of BVPS-1 OTΔT and OPΔT Trip Equations Dynamic Time Constants

The proposed TS change would delete the lag compensators for the measured ΔT and T-avg, i.e., $1/(1+\tau_4 S)$ and $1/(1+\tau_5 S)$, respectively, from the OTΔT and OPΔT trip equations for BVPS-1. The deletion of these lag compensators implicitly sets the dynamic compensation time constants, τ_4 and τ_5, which are used in the lag compensators, to zero. The current TS requirement is to set the time constants to a value equal to or less than 2 seconds. The licensee states that these time constants have been removed from the analog channel hardware by plant hardware change Design Change Package (DCP) 698. These time constants are implicitly set to 0 to be consistent with the required setpoint value, and reflect the hardware configuration of the BVPS-1 RTS. In its letter dated March 28, 2001 (Ref. 9), the licensee explains the differences between the BVPS-1 and 2 RTSs. The BVPS-1 reactor protection system is an older design, where the OPΔT reactor trip did not utilize the lead/lag module on the ΔT side of the bistable and thus it was not included in the as-built configuration. The BVPS-1 RTS does not have lag filters on the ΔT side of the bistable and on the T-avg side of the bistable. Therefore, the time constants τ_4 and τ_5 for the filters are set to zero to represent the absence of the filter in the channel. The licensee states that Westinghouse has confirmed that the lead/lag module is not modeled in the safety analysis for BVPS-1. Since the deletion of the lag compensators on the measured ΔT and T-avg reflects the hardware configuration of the BVPS-1 RTS and the design-basis safety analysis, this change is acceptable.

3.1.5.3 Deletion of f(ΔI) from the OPΔT trip setpoint equation

The flux difference trip reset functions, f(ΔI), is designed to correct for the effect of axial neutron flux difference, ΔI, on the OTΔT and the OPΔT trip setpoints. In WCAP-8745-P-A, it is stated that no f(ΔI) function is required to preclude fuel centerline melting during overpower incidents in 16x16 and 17x17 fuel assembly plants. Therefore, the flux difference trip reset function, f(ΔI) for BVPS-1 and f_2 (ΔI) for BVPS-2, for the OPΔT trip function is set to 0 for all ΔI in current TS for both BVPS units. The proposed TS would eliminate the f(ΔI) reset function all together. Since the f(ΔI) reset function is not modeled in the safety analysis nor included in the OPΔT trip setpoint methodology calculation, its deletion from the OPΔT trip setpoint equation is acceptable.

3.1.5.4 Change in BVPS-1 RCS average Temperature

The proposed change would revise the T-avg at RTP in the OTΔT and OPΔT trip functions, T′ and T″, respectively, from 576.3 °F to 576.2 °F. This change is necessary to make the TS values consistent with the nominal RCS average temperature assumed in the safety analysis. The reduction of T′ and T″ values is conservative in nature because it results in reduced OTΔT and OPΔT setpoints. It is, therefore, acceptable.

3.1.5.5 Addition of Inequality in OTΔT and OPΔT trip equations Constants Values

The proposed changes would replace the equalities in the TS of the values of various constants in the OTΔT and OPΔT setpoints equations with inequalities, "≤" or "≥". The equalities of the following constants in the OTΔT and OPΔT trip equations are replaced with inequalities "≤":

K_1, K_4,
T' and T" (T-avg at the RTP in OTΔT and OPΔT equations, respectively),
τ_2, τ_5, and τ_6 (time constants for various dynamic compensators),
τ_3 associated with Unit 2 ΔT measurement.

The following constants are replaced with "≥":

K_2, K_3 , K_5 and K_6,
P' (nominal pressurizer pressure)
τ_1, τ_4, and τ_7 (Time constants for various compensators)
τ_3 associated with BVPS Unit 1 T-avg measurement.

The need to use inequalities for these constants is to ensure that the safety analysis values assumed for the setpoints would not violate the TS values. In order to assure that the settings selected are conservative with respect to the values assumed in the analyses, a directional conservatism is chosen for each constant associated with the OTΔT and OPΔT setpoint equations to ensure that the actual OTΔT and OPΔT setpoints are no greater than those assumed in the safety analyses.

The determination of "≥" or "≤" is based on the directional conservatism that would result in lower ΔT setpoints for the OTΔT and OPΔT trip functions. The NRC staff has performed an arithmetic evaluation to conclude that the inequality signs incorporated for these constants result in the decreasing ΔT in both the OTΔT and OPΔT trip functions. These changes are conservative and ensure that the setpoints are no greater than those assumed in the safety analyses and are, therefore, acceptable.

3.1.5.6 Revision of OTΔT and OPΔT Trip Setpoint Allowable Values

The allowable values for the OTΔT and OPΔT trip functions (Functional Units 7 and 8 of TS Table 2.2-1) setpoints, respectively, are specified in NOTES 3 and 4 for BVPS-1, and NOTES 2 and 4 for BVPS-2. Each of these NOTES specifies that the channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than the specified percent. The specified percentage for the OTΔT and OPΔT trip setpoints are 1.3 percent and 2.9 percent, respectively, for BVPS-1, and 1.6 percent and 2.5 percent, respectively, for BVPS-2. The TS changes would reduce these specified percentages to 0.5 percent of ΔT span.

The allowable value of a trip function is used to determine instrumentation channel operability. For the revisions to the RTS trip setpoint allowable values, the licensee states that the original calculations were completed using generic numbers for these uncertainties, whereas the latest revision of the WCAPs utilized plant-specific information for these variables.

WCAP-11419, Revision 2, and WCAP-11366, Revision 4, respectively, provide detailed calculations of the instrumentation uncertainties of the RTS for BVPS-1 and 2. Section 4.4 of WCAP-11419 and WCAP-11366 provides the determination of the TS allowable values for various trip functions. The allowable values for the TS are determined by adding (or subtracting) the calibration accuracy of the device tested during the Channel Operational Test to the nominal trip setpoint (NTS) in the non-conservative direction (i.e., toward or closer to the safety analysis limit) for the application. For those channels that provide trip actuation via a bistable in the process racks, the calibration accuracy is defined by the Rack Calibration

Accuracy term. The magnitude of the calibration accuracy term is as specified in the station procedures. The rack calibration accuracies for both OTΔT and OPΔT trip functions of both BVPS units are determined to be 0.5 percent of ΔT span, as shown in Tables 3-5 and 3-6, respectively, in the TRs. Therefore, the revised value of 0.5 percent of ΔT span specified for the OTΔT and OPΔT trip setpoints allowable values are consistent with those determined in the protection system instrument uncertainties and are, therefore, acceptable.

3.2 Revision to RTS and ESFAS Trip Setpoint and Allowable Values

Paragraph (c)(1)(ii)(A) of 10 CFR 50.36 requires, in part, that where a LSSS setting is specified for a variable on which a safety limit has been placed, the setting be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. Trip setpoint in nuclear safety-related instruments should be selected to provide sufficient allowance between the trip setpoint and the safety limit to account for uncertainties. The trip setpoint should be the value that the final setpoint device is set to actuate. The safety limit can be defined in terms of directly measured process variables such as pressure or temperature. Safety limit can also be defined in terms of a calculated variable involving two or more measured process variables. An example of a calculated variable is the DNBR.

The existing trip setpoint methodology utilized for BVPS-1 and 2 follows the guidance of Instrument Society of America (ISA) Standard S67.04-1982, “Setpoint for Nuclear Safety-Related Instrumentation.” The revised trip setpoint and allowable values are based on the setpoint methodology of ISA Standard S67.04-1994. ISA-S67.04-1994 provides some guidance on instrument drift evaluation and uncertainty term development for the evaluation of an instrument surveillance interval. Section 4.3 of ISA-S67.04-1994 states that the LSSS may be the trip setpoint, an allowable value, or both. The TSTF No. 355 to NUREG-1431, “Standard Technical Specifications Westinghouse Plants,” Revision 1, entitled “Westinghouse Standard Technical Specifications - Reactor Trip System and Engineered Safety Feature Actuation Instrumentation” recommended that the allowable value be designated as the LSSS. In association with the trip setpoint and LCOs, the LSSS establishes the threshold for protective system action to prevent acceptable limits from being exceeded during design-basis accidents. The LSSS ensures that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

ISA-S67.04-1994 provides a discussion on the purpose and application of an allowable value. The allowable value is the limiting value that the trip setpoint can have when tested periodically, beyond which the instrument channel is considered inoperable and corrective action must be taken in accordance with the TSs. The allowable value relationship to the setpoint methodology and testing requirements in the TS must be documented.

WCAP-11419 documents the instrument uncertainty calculations for the RTS and ESFAS trip functions for BVPS-1. WCAP-11366 documents the instrument uncertainty calculations for the RTS and ESFAS trip functions for BVPS-2. The approaches discussed in both WCAP-11419 and WCAP-11366 are consistent with ISA Standard S67.04, 1994. NRC Regulatory Guide (RG) 1.105, “Setpoint for Safety-Related Instrumentation,” Revision 3, 1999, endorses the 1994 version of ISA S67.04. The NRC staff finds that the methodology described in the WCAP-11419 and WCAP-11366 is in compliance with 10 CFR 50.36, Paragraph (c)(1)(ii)(A) and is, therefore, acceptable.

With respect to the instrument uncertainty, the basic methodology used by the licensee is the SRSS technique. The methodology used to combine the uncertainty components for a channel is an appropriate combination of those groups that are statistically and functionally independent. Those uncertainties which are not independent are conservatively treated by arithmetic summation and then combined with the independent terms.

By letter dated March 28, 2001, the licensee stated that, for any uncertainty calculation (protection function or RTDP) performed by licensee (supported by Westinghouse), the following information is utilized:

- Identification of the equipment - transmitters, process racks, control board meters, etc. This includes the specification sheet data, particularly for transmitters,

- Identification of measurement and test equipment used to calibrate the transmitters and process racks. This includes the specification data for the digital volt meters and precision gauges used by the plant,

- Plant calibration and functional test procedures,

- Drift data or drift magnitudes for transmitters and process racks,

- Plant conditions for which the equipment is scaled, and

- Plant calorimetric measurement procedures and specifications on the equipment used to perform the measurement.

The licensee then develops an uncertainty model for each function and determines the uncertainty for the control, protection or indication function. When equipment is replaced, the licensee will ensure that the replacement equipment is equivalent to or better than the installed equipment. Verification of equivalency must include the confirmation of the same or better statistical characteristics to assure that the original calculation is still bounding.

Due to demonstrated compliance with 10 CFR 50.36, adherence to the guidance contained in RG 1.105, ISA Standard S67.04, 1994, and appropriate modeling and maintenance of uncertainties, the NRC staff finds that the methodologies presented in WCAP-11419 and WCAP-11366 for establishing revised RTS and ESFAS trip setpoint and allowable values are acceptable.

Furthermore, the NRC staff has verified the proposed trip setpoint values and the allowable values with the setpoint methodology documents WCAP-11419 and WCAP-11366. The allowable values are being modified due to the plant-specific analysis which resulted in changes to the uncertainties used in the determination of the allowable values. The trip setpoint values and the allowable values listed in the proposed changes are consistent with the setpoint documentation. The proposed values for the RTS and ESFAS trip setpoints, in general, are more restrictive than the currently specified setpoints. The more restrictive trip setpoints are to ensure that the current safety analysis limits continue to be met. The NRC staff finds that the RTS and ESFAS trip setpoints and allowable values have been adequately determined in accordance with the acceptable methodologies described in WCAP-11419 and WCAP-11366 and, consequently, these revised trip setpoint and allowable values are acceptable. Therefore, the NRC staff finds the following TS changes acceptable:

BVPS-1:

- Revision of the values for “Trip Setpoint” contained in Table 2.2-1, “Reactor Trip System Instrumentation Trip Setpoint,” for Table Item Nos. 12, 13, and 15.

- Revision of the values for “Allowable Value” contained in Table 2.2-1 for Functional Units 2, 3, 4, 5, 6, 9, 10, 11, 12, 13, 14, 15, 17A, and 20.

- Revision of allowable value description for Functional Unit 20D contained in Table 2.2-1.

- Revision of the values for “Allowable Value” contained in Table 3.3-4, “Engineered Safety Feature Actuation Instrumentation Trip Setpoints,” for Functional Units 1.c, 1.d, 1.e, 1.1.c, 1.1.d, 2.c, 3.b.3, 4.c, 4.d, 4.e, 5.a, 6.a, 6.b, 6.c, 7.a, 7.b, 8.b, and 8.c.

- Revision of the values for “Trip Setpoint” contained in Table 3.3-4 for Table Item Nos. 5.a, 6.b, 6c, 7.a, and 7.b.

BVPS-2:

- Revision of the values for “Allowable Value” contained in Table 2.2-1, for Functional Units 2, 3, 4, 5, 6, 9, 10, 11, 12, 13, 15, 16, and 22.

- Revision of allowable value description for Functional Unit 22.d contained in Table 2.2-1.

- Revision of the value for “Trip Setpoint” contained in Table 2.2-1 for Table Item No. 13.

- Revision of the values for “Allowable Value” contained in Table 3.3-4, for Functional Units 1.c, 1.d, 1.e, 1.1.b, 2.c, 3.b.3, 4.c, 4.d, 4.e, 5.b, 6.a, 6.b, 6.c, 7.b, 7.c, 8.b, and 8.c.

- Revision of the values for “Trip Setpoint” contained in Table 3.3-4 for Table Item Nos. 1.d, 1.1.b, 5.b, 6.b, 6.c, and 7.b.

BVPS-1 and 2

- Deletion of the existing inequalities applied to the trip setpoint values (excluding the time constants) specified in the “Trip Setpoint” column of Table 2.2-1 with the exception of Table Item No. 17.B, “Turbine Stop Valve,” for BVPS-1 and Table Item No. 17.b, “Turbine Stop Valve Closure,” for BVPS-2.

- Deletion of the inequalities applied to the trip setpoint values (excluding the time constants) specified in the “Trip Setpoint” column of Table 3.3-4.

3.3 Addition of a TS Bases Control Program

In its letter dated March 28, 2001, the licensee proposed to revise existing TS Administrative Controls Programs by the addition of a new program, the TS Bases Control Program, TS 6.18.

This change provides guidance for making changes to the TS Bases and assures changes that may pose an unreviewed safety question or involve a TS change are not made to the Bases of the TS without prior NRC approval. Although this program provides a reasonable method to address changes to the TS Bases and is applicable to the licensee, the inclusion of this program represents an additional TS requirement not present in the current TS. Therefore, this change is considered more restrictive. This change is an additional restriction on plant operation that enhances safety and is acceptable.

3.4 Relocating Existing TS Requirements to Licensee-Controlled Documents

When TS requirements have been shown to give little or no safety benefit, their removal from the TS may be appropriate. In most cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the Owners Groups comments on STS. The NRC staff reviewed generic relaxations contained in the NUREG-1431 STS and found them acceptable because they are consistent with current licensing practices and the Commission's regulations. The NRC staff reviewed the BVPS-1 and 2 licensing bases to ensure a basis exists for adopting the proposed elements of model NUREG-1431 STS requirements for RTS and ESFAS instrumentation TS. Thus, a basis is established for proposed revisions to TS that delete details of system design and system description including design limits for meeting TS requirements.

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 plan as described in UFSAR Chapter 17. In 10 CFR 50.59, controls are specified for changing the facility as described in the UFSAR, and in 10 CFR 50.54(a) criteria are specified for changing the QA plan. In TS, the Bases also contain descriptions of system design. Proposed Specification 6.18 specifies controls for changing the Bases. Removing details of system design from the TS is acceptable because this information will be adequately controlled in the UFSAR, controlled design documents and drawings, or the TS Bases, as appropriate. Cycle-specific design limits are moved from the TS to the COLR. TS Administrative Controls are revised to include the programmatic requirements for the COLR.

3.4.1 Relocation of Cycle-Specific Values of Operating and Trip Parameters to COLR

In WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," the Westinghouse Owners Group (WOG) provides justification and methodologies for the relocation from the TS to the COLRs of the following cycle-specific operating parameters values: (1) the RCSL figure, (2) the DNB parameters of RCS T-avg, pressurizer pressure, and RCS flow rate, and (3) the OTΔT and OPΔT trip setpoint parameter values. The NRC staff has accepted WCAP-14483-A (Ref. 24) for license applications. Accordingly, the licensee proposed to relocate the RCSL figure, the DNB parameters values, and the OTΔT/OPΔT Trip Parameter Values to the COLR. Relocating the cycle-specific parameter limits to the COLR enables the plant to implement cycle-specific changes to the limit and setpoint values without having to submit a license amendment request for NRC approval.

For the implementation of the relocation of these cycle-specific values of the operating parameters and the OTΔT and OPΔT trip setpoint parameters from the TS to the COLR, the definition of the COLR specified in paragraph 1.37 of BVPS TS is revised to reflect the cycle-specific parameters values, and to be consistent with that defined in NUREG-1431, Rev. 1 (Ref. 25). The revision of the COLR definition is consistent with that contained in NUREG-1431 and dos not result in any substantive change in operating requirements. It is, therefore, acceptable.

In addition, the TS Administrative Control Section 6.9.5, “Core Operating Limits Report (COLR),” would be revised by adding (1) the following cycle-specific core operating limits, which are documented in the COLR:

2.1.1	Reactor Core Safety Limits
3.2.5	DNB Parameters
3.3.1.1	Reactor Trip System Instrumentation - OTΔT and OPΔT setpoint parameter values,

and (2) the analytical methods used to determine these cycle-specific OTΔT and OPΔT setpoint parameter values, i.e., WCAP-8745-P-A. This is consistent with the method described in WCAP-14483-A, and is, therefore, acceptable.

3.4.1.1 Relocation of Core Safety Limits

The proposed changes would revise TS 2.1.1 by (1) the relocation of Figure 2.1-1, “Reactor Core Safety Limits, Three Loop Operation,” to the COLR, (2) the addition of specifications 2.1.1.1 and 2.1.1.2, which specify that the DNBR limit of 1.17 for the WRB-1 DNB correlation, and the peak fuel centerline temperature limit of 4700 °F, respectively, shall not be exceeded, and (3) the replacement of the “ACTION” statement by stating that “If Safety Limit 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.”

Consistent with WCAP-14483-A, the first two revisions would specify in the TS the specified acceptable fuel design limits (SAFDLs) with respect to DNB and center fuel melt, rather than the derived relationship among the operating parameters specified in the RCSL figure. The revised Action statement reflects the changes in the TS from the RCSL figure to the SAFDLs. The actions for violation of a safety limit specified in TS 2.1.1 will include the restoration of safety limit compliance and entry into Hot Standby within 1 hour. This proposed change is consistent with the Standard TS of NUREG-1431, Rev. 1. The licensee also would revise the Bases for TS 2.1.1, “Reactor Core,” to reflect (1) the use of RTDP statistical uncertainty treatment methodology with the design DNBR limits of 1.24 and 1.23 for the typical and thimble cells, respectively, and the safety analysis limit of 1.33 for both typical and thimble cells, and (2) the relocation of the RCSL figure to the COLR, and (3) the identification that the RCSL figure is based on the enthalpy hot channel factor limits provided in the COLR.

With regard to item (3), the change in the Action statement is consistent with the safety limits violation action requirement 2.2.1 for Westinghouse STS. The change also reflects the relocation of the RCSL figure to the COLR and the addition of the SAFDLs in TS 2.1.1.

The NRC staff finds that the changes associated with the relocation of core safety limits follow the general guidance contained in WCAP-14483-A, and are acceptable.

3.4.1.2 Relocation of DNB Parameters

The proposed changes would relocate the DNB parameters values to the COLR. The NRC-approved methodology used to derive the parameters is contained in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology" (Ref. 26), which is referenced in TS 6.9.5.b as one approved analytical methodology. The Bases for LCO 3.2.5 are also revised to address the relocation of the DNB parameters to the COLR. Attachment A-1 and A-2 of the December 27, 2000, letter, specifies the limits of the DNB related parameters as follows:

a. RCS T-avg $\leq$ the limit specified in the COLR,
b. Pressurizer pressure $\geq$ the limit specified in the COLR, and
c. RCS total flow rate $\geq$ 261,600 gpm and $\geq$ the limit specified in the COLR.

A footnote is added to note that the RCS flow limit of 261,600 gpm is the analytical limit used in the safety analysis. This is the minimum NRC-approved value for RCS total flow rate. The RCS flow limits specified in the COLR are the indicated values of 267,400 gpm for BVPS-1, and 267,200 gpm for BVPS-2, which include the RCS instrument uncertainties. A footnote in the COLR indicated that the RCS flow limit includes allowance for normalization of the cold leg elbow taps with a beginning of cycle precision RCS flow calorimetric measurement and verification on a periodic basis via control board indication. Therefore, the NRC staff concludes that the implementation of the relocation of DNB parameters to the COLR is consistent with the approved method described in WCAP-14483-A and, is acceptable.

The proposed changes would revise the Bases for TS 3/4.2.5 by adding a statement that the (DNB parameter) variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, based on maximum analyzed steam generator tube plugging, is retained in the LCO.

The NRC staff finds that the changes associated with the relocation of DNB parameters follow the general guidance contained in WCAP-14483-A, and are acceptable.

3.4.1.3 Relocation of OTΔT and OPΔT Trip Setpoints Parameter Values

The proposed TS changes would revise Function Units 7 and 8, OTΔT and OPΔT, respectively. In the trip setpoint column, the wording "see Note 1" and "see Note 2" for the OTΔT and OPΔT trips functions, respectively, would be replaced with the wording "see Technical Specification table Notation (A) on Table 3.3-1" and "see Technical Specification table Notation (B) on Table 3.3-1." In the Allowable Value column in the revised Table 3.3-1, the wording "see Note 3" and "see Note 4," for the OTΔT and OPΔT trip functions, respectively, would be replaced with "see Table Notation (A)" and "see Table Notation (B)." Under the revised Table 3.3-1, Table Notations (A) and (B) are added, which, respectively, describes the OTΔT and OPΔT trip equations and defines the definition and units of measure of each term in these equations. In addition, Tables (A) and (B) also state that the values of the parameters in the OTΔT and OPΔT trip equations are specified in the COLR. BASES 2.2.1 and 3/4.3.1 and 3/4.3.2 are revised to indicate the OTΔT and OPΔT trip equations parameters values are specified in the COLR. In addition, TS 6.9.5 , "Core Operating Limits Report (COLR)," is also revised to include WCAP-8745-P-A as the acceptable analytical method used to determine the OTΔT and OPΔT trip equations. In the draft COLR for BVPS-1 and 2, respectively, specific values of these parameters are specified.

The implementation of the relocation of the OTΔT and OPΔT trip functions to the COLR is consistent with the approved methodology of WCAP-14483-P-A and is, therefore, acceptable.

3.4.2 Relocation of Trip Setpoint Requirements to the LRM

TS 2.2.1, "Reactor Trip System Instrumentation Setpoints," and TS 3.3.2.1, "Engineered Safety Feature Actuation System Instrumentation," require the trip setpoints for required functions to be set consistent with the values shown in the Trip Setpoint column of Table 2.2-1 and Table 3.3-4, respectively. TS 2.2.1 and TS 3.3.2.1 Actions require inoperable channels to be restored to OPERABLE and the trip setpoints adjusted to the correct "trip setpoint" values. Trip setpoints are operational details of instrumentation operability. The instrumentation setpoint allowable value, however, is a required limit for the associated Function and this value is retained in the TSs. The relocated trip setpoints are not required to be in the TSs to provide adequate protection of the public health and safety. Therefore, these details are to be relocated to the LRM and the references to these setpoints in Table 2.2-1 and Table 3.3-4 are deleted. The LRM will be incorporated into the BVPS-1 and 2 UFSAR at implementation. Any changes to the relocated trip setpoints in the LRM will be controlled by the provisions of 10 CFR 50.59. Therefore, the NRC staff finds these changes acceptable.

3.4.3 Relocation of Existing TS Design Information to the TS Bases

The TS Table 2.2-1 Function 12 (Loss of RCS Flow) trip setpoint and allowable value are revised to be consistent with the assumptions of the applicable safety analyses. The TS specifies these setpoints as a percent of design flow per loop and a footnote to Table 2.2-1 specifies that the design flow is 87,200 gpm per loop. The licensee proposes to revise the loss of flow setpoints to "percent of indicated flow" from "percent of design flow." The associated footnote specifying design flow per loop is moved to the applicable section of the RTS Bases.

The licensee states that the low flow reactor trip function is modeled in the non-LOCA safety analyses as a percent of the assumed RCS loop flow. Although the design flow is explicitly modeled in the Partial and Complete Loss of Flow and Locked Rotor analyses, the assumed trip setpoint, in percent, is relative to the initial loop flow and not the design loop flow. Thus, the reactor trip function setpoint is not intended to be based on a specific loop flow rate. The BVPS-1 and 2 TS 3.2.5, RCS Pressure, Temperature, and Flow Departure from DNB, includes a requirement that the actual measured flow is greater than or equal to the required design flow. Thus, for a symmetric transient with all RCPs coasting down, a trip setpoint based on percent of the normal flow (not design flow) is conservative since the flow rate at the time of reactor trip will be greater than or equal to the flow rate assumed in the UFSAR analysis. This is also true for an asymmetric transient (i.e., Locked Rotor event). For the Locked Rotor event, the flow reduction in the affected loop is very fast such that if the setpoint is based on normal flow, and not explicitly design flow, there will be no significant change in the time of reactor trip. Also, a Partial Loss of Flow event (single loop coastdown) is bounded by the analysis of the Complete Loss of Flow event; therefore, a postulated asymmetric flow condition will not result in this analysis violating the DNB design basis. Therefore, deleting the design flow value from TS Table 2.2-1 and revising the setpoints to reflect the normal flow instead of design flow will not significantly impact the non-LOCA safety analyses and the conclusions in the FSAR for these analyses remain valid. In addition, BVPS-1 and 2 TS 3.2.5.c continues to require the design flow be maintained above the limit specified in that TS. The descriptive information regarding these low flow setpoints, including specifying the design flow value and the requirement for the setpoints to be greater than or equal to the design flow, are being moved to the Bases for the low flow trip function.

3.4.4 Conclusion Regarding Relocating Existing TS Requirements

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

- TS Bases controlled by TS 6.18, "Technical Specifications Bases Control Program."
- FSAR (includes the Licensing Requirements Manual (LRM) by reference) controlled by 10 CFR 50.59.
- Core Operating Limits Report controlled by TS 5.6.5, “Core Operating Limits Report (COLR).”

To the extent that requirements and information have been relocated to licensee-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 current TS, the NRC staff has concluded that they do not fall within any of the four criteria in the Final Policy Statement (discussed in Section 2.0 of this SE). Accordingly, existing detailed information and specific requirements, such as generally described above, may be deleted from the current TS.

3.5 Less Restrictive Technical Changes to Existing TS Requirements

Changes to the TS involving deleting portions of TS requirements were evaluated as relaxation of required actions and relaxation of LCO requirements. The following discussions address why various TS relaxations are not required to be included in TS.

3.5.1 Relaxation of Required Actions

Existing TS require that, in the event specified LCOs are not met, penalty factors to reactor operation, such as resetting setpoints and power reductions, shall be initiated as the method to reestablish the appropriate limits. The NUREG-1431 STS are also constructed to specify Actions for conditions of required features made inoperable. Adopting TS actions that are like NUREG-1431 STS Action requirements is acceptable because the plant remains within analyzed parameters by performance of required Actions, or the Actions are constructed to minimize risks associated with continued operation while providing time to repair inoperable features. Such Actions retain the margin to safety thereby providing assurance that operations that could result in a challenge to safety systems are exited in a time period that is commensurate with the safety importance of the system. The following changes are generally consistent with NUREG-1431 STS, and changes characterized as relaxation of required actions are acceptable.

Proposed changes, for BVPS-1 only, include deletion of Action a. contained in LCO 3.3.2.1 and modification of Action b. by deleting the letter “b” designation. The existing TS 3.3.2.1 Action includes requirements for an ESFAS setpoint less conservative than the required allowable value. The applicable Action requirement must be applied until the channel is restored to operable status with the trip setpoint readjusted to within TS limits. The replacement action (TS 3.3.2.1 Action A) requires the Action shown in Table 3.3-3 for one or more inoperable instrument channel(s). In the context of the revised TS format and considering that the TS Bases contain the operability details for trip setpoint setting requirements, the deletion of the TS 3.3.2.1 Action requirements is made to conform with the presentation and format of this information in the NUREG-1431 STS and effectively provide the same requirements.

For BVPS-2 only, the licensee proposes to delete Actions a., b.1, and b.2 contained in LCO 3.3.2.1. Existing TS 3.3.2.1 Action a. requires the ESFAS instrument or interlock trip setpoint to be adjusted if the trip setpoint is less conservative than the required trip setpoint but more conservative than the allowable value. Existing TS 3.3.2.1 Actions b.1 and b.2 require readjustment of ESFAS instrument or interlock trip setpoints that are less conservative than the allowable value and within 12 hours determine that the specified TS 3.3.2.1 equation is consistent with the established TS setpoint equation or apply remedial action requirements. In the context of the revised setpoint methodology, and TS rules for constructing Actions, proposed TS LCO 3.3.2.1 Action a. replaces the existing TS 3.3.2.1 Action a. and b. statements without changing the technical intent of the action. The reorganization of TS 3.3.2.1 Action requirements is made to generally conform with the presentation and format of this information in the STS.

3.5.2 <u>Relaxation of LCOs</u>

The current TS are generally constructed to provide LCO requirements that specify the protective limit that is required to meet safety analysis assumptions for required features. When conducting routine but infrequent testing that is otherwise prohibited by TS, it is advantageous to include special exceptions in plant TS that suspend specification limits provided deliberate remedial operational limits are established. The newly established protective limits replace the protective limits previously found to be acceptable to the NRC staff for meeting the LCO. The proposed TS changes provide the same degree of protection required by the safety analysis and provide flexibility for meeting limits without adversely affecting operations since equivalent features are required to be operable. These changes are consistent with STS, and changes specified as relaxation of LCO requirements are acceptable.

For BVPS-1 only, TS 3/4.10.3, “Special Test Exceptions Pressure/Temperature Limitation - Reactor Criticality,” 3/4.10.4, “Physics Test,” and 3/4.10.5, “Special Test Exception No Flow Tests,” would be revised by deleting the trip setpoint requirement for the intermediate and power range instrumentation specified in the LCO.

For BVPS-2 only, TS 3/4.10.3, “Special Test Exceptions Physics Tests,” would be revised by deleting the trip setpoint requirement for the intermediate and power range instrumentation specified in the LCO.

3.5.3 Removal of Details of System Design, Design Limits and System Description

Existing TS Section 2.2, "Limiting Safety System Settings - Reactor Trip Instrumentation Setpoints," is deleted to reflect the removal of setpoint information from this TS. The TS 2.2.1 Safety Limit statement for the RTS instrumentation is effectively replaced by the LCO 3.3.1 statement and associated Bases. Since the 2.2 safety limit requirements are moved into the RTS Instrumentation, the LCO 3.3.1 statement requiring the RTS instrumentation to be operable is applicable. Consistent with the format and presentation of STS LCO statements, the operability details (interlock and setpoint setting requirements) are discussed in the associated TS Bases. The RTS Bases contain extensive discussions pertaining to the required trip setpoints and allowable values. The Bases information effectively includes the requirement of the TS 2.2.1 safety limit statement. Reliance on the information contained in the Bases for system operability requirements and design information is acceptable since changes to the information in the Bases is controlled by the Bases Control Program specified in the administrative controls section of the TS.

Proposed changes include revision of ESFAS TS 3.3.2.1 LCO and Action to reflect the relocation of ESFAS Trip Setpoints to the LRM and the combining of ESFAS Tables 3.3-4 and 3.3-3. The LCO statement for the ESFAS instrumentation is moved to the associated Bases consistent with the format and presentation of STS LCO statements. The operability details (trip setpoint setting requirements) are discussed in the associated Bases of the LCO, which are controlled by Specification 6.18 - Bases Control Program.

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

TS Bases changes reflect the deletion of TS Section 2.2 by the movement of the Bases for TS Section 2.2.1 to the Bases for Sections 3/4.3.1 and 3/4.3.2. These changes provide consistency and clarity with the TS changes and are acceptable.

3.6 Other Miscellaneous Changes

- For BVPS-1, revision of the following items to delete references corresponding to 2-loop operation, or loop stop valves being open: facility operating license condition 2.C.(3), Table 2.2-1 Note 1 for the OTΔT "K" coefficients that apply to less than 3-loop operation in Table 2.2-1, Bases for TS Section 2.2 for OTΔT and Loss of Flow descriptions, and Bases for TS Section 3/4.4 for Reactor Coolant Loops.

The BVPS-1 Facility Operating License contains a condition in paragraph 2.C.(3), which states that:

> FENOC shall not operate the reactor at power levels above P-7 (as defined in Table 3.3-1 of Specification 3.3.1.1 of the Technical Specifications, Appendix A) with less than three (3) reactor coolant loops in operation until safety analyses and approval for less than three loop operation at power levels above P-7 has been granted by the Commission by amendment of this license.

TS LCO 3.4.1.1 specifies that all reactor coolant loops shall be in operation in Modes 1 and 2. Since interlock P-7 is approximately 10 percent full power, operation at power level above P-7 would be MODE 1 operation, and the operation with less than three reactor coolant loops would not be allowed per LCO 3.4.1.1. Therefore, the license condition stated above is not necessary and its removal is acceptable.

In the current BVPS-1 TS, the OTΔT trip function in Table 2.2-1 includes three sets of values for constants K_1, K_2, and K_3 for 3-loop operation, 2-loop operation with no loops isolated, and 2-loop operation with 1 loop isolated. The proposed change would delete the references to the two sets of values for 2-loop operations.

As discussed above, LCO 3.4.1.1 does not permit plant operation in Modes 1 and 2 with less than three RCS loops in operation. Also, LCO 3.4.1.4.1 requires all RC loop isolation valves to be open during plant operation in Modes 1 through 4. Since the OTΔT Trip function is applicable only for Modes 1 and 2 operation during which the operation with two reactor coolant loops is prohibited, the references to the values of OTΔT trip function constants for 2-loop operations are not needed. Therefore, the proposed deletion of references to 2-loop operations for BVPS-1 is correct and appropriate and is acceptable.

- For BVPS-1, revision of Table 3.3-3, Table 3.3-4 and Table 4.3-2 to delete Item 7.d regarding Auxiliary Feedwater Emergency Bus Undervoltage.

This change is necessary to correct a discrepancy between the TS description and the protection system design. The undervoltage relay does not directly start the motor-driven auxiliary feedwater (AFW) pumps. The start of the motor-driven AFW pumps is accomplished indirectly via a combination of (1) emergency bus feed breaker opening, (2) logic signals from steam generator low-low level, safety injection, or trip of all main feedwater pumps, and (3) diesel generator sequencer actuation. The ESFAS related TS Table 3.3-3 items 7.a, 7.c, 7.e, and emergency diesel generator TS 4.8.1.1.2.b.3(b) cover the above inputs that cause the start of the motor-driven AFW pumps. The NRC staff finds that Item 7.d is not necessary as a requirement for the starting of the motor-driven AFW pump and that its deletion is appropriate and acceptable.

The NRC staff finds that the following proposed changes are administrative or editorial in nature, are reasonably compatible with improved STS, do not result in any substantive change in operating requirements, and are, therefore, acceptable:

- The definition that "ΔT is measured RCS ΔT, °F" is added to the OTΔT and OPΔT trip functions equations contained in Notes 1 and 2 of Table 2.2-1 (BVPS-1).

- The term "or interlock" is added to the current Action b of LCO 3.3.2.1. The first letter in certain words of LCO 3.3.2.1 and SR 4.3.2.1.1 would be capitalized (BVPS-1).

- Revision of column heading titled "Allowable Values" to "Allowable Value" for Table 2.2-1 and Table 3.3-4 (BVPS-1).

- Deletion of the reference to Table 3.2-1 in the table index (BVPS-1).

- Deletion of the reference to Figure 2.1-1 in the figure index (BVPS-1)

- For Table 2.2-1, correction of a typographical error in Note 1 for the OTΔT trip setpoint equation. The superscript in the term τ^1 should be a subscript in the following expression: $(1+\tau^1 s)/(1+\tau_2 s)$ (BVPS-2).

- The addition of a new footnote to Table 2.2-1 which defines the term RTP (BVPS-2).

- Reformat of Table 3.3-1 header to make it more consistent with other instrument tables in the TSs (BVPS-2).

- Revision of index pages to reflect numbering changes in Section 6.0 (BVPS-2)

- Revision of certain RTS and ESFAS instrumentation Function descriptions to make the nomenclature used in the TS Tables consistent from Table to Table for each affected instrument function (BVPS-1 and 2).

- Simplification and reformat of the OTΔT and OPΔT equations to be consistent with the improved STS (BVPS-1 and 2).

- Revision of TS 3.2.5, DNB Parameters, LCO, Actions, and SRs to improve consistency with the improved STS (BVPS-1 and 2).

- Clarification of the RTS and ESFAS Actions with the addition of note, similar to that contained in the improved STS, allowing separate Action statement entry for each instrument function (BVPS-1 and 2).

- Revision of instrumentation Bases text to eliminate the repetition of specific setpoint values in the Bases discussions (BVPS-1 and 2).

- Grammar and punctuation changes to the TS and Bases (BVPS-1 and 2).

- Reformat of pages, including repagination, due to the deletion and addition of text as discussed elsewhere in this SE (BVPS-1 and 2).

- Rotation of the page footer on Table pages, as needed, to be consistent with the text format (BVPS-1 and 2).

- Replacement of equal signs with the words “is” or “are” for the OTΔT and OPΔT equation parameters contained in Table 2.2-1 (BVPS-1 and 2).

- Addition of parameter units of measure for certain parameters of OTΔT and OPΔT equations contained in Table 2.2-1 (BVPS-1 and 2)

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (66 FR 20002 for the changes associated with the December 27, 2000, amendment request and 66 FR 33111 for the changes associated with the March 28, 2001, amendment request that included the addition of a TS Bases control program). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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Date: July 20, 2001

7.0 REFERENCES

1. Letter from L. W. Myers, FirstEnergy Nuclear Operating Company, to USNRC, "Beaver Valley Power Station, Unit No. 1 and Unit No. 2, BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73, License Amendment Request Nos. 286 and 158," L-00-143, December 27, 2000, Agencywide Documents Access and Management System (ADAMS) Accession No. ML003782095.

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