**FENOC**

FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
P.O. Box 4
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**Lew W. Myers**
Senior Vice President

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December 27, 2000
L-00-143

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

**Subject: Beaver Valley Power Station, Unit No. 1 and 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**

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the above licenses in the form of changes to the technical specifications and a Unit 1 operating license condition. The proposed amendment revises Reactor Trip System Instrumentation and Engineered Safety Feature Actuation System Instrumentation trip setpoints and allowable values; utilizes the Revised Thermal Design Procedure (RTDP) to generate additional Departure from Nucleate Boiling (DNB) margin to allow a revision to the core safety limits, DNB parameters and Overtemperature and Overpower ΔT trip setpoints; relocates certain requirements from the Technical Specifications to either the Core Operating Limits Report (COLR) or Licensing Requirements Manual (LRM); revises the associated Bases sections to reflect the proposed changes; and also includes miscellaneous changes to the Technical Specifications and Bases sections.

Proposed revision to a license condition and technical specification changes for Unit No. 1 are presented in Attachment A-1. Proposed technical specification changes for Unit No. 2 are presented in Attachment A-2. The safety analysis (including the no significant hazards evaluation) is presented in Attachment B.

The Proprietary Information Notice, Copyright Notice, a Westinghouse application for withholding proprietary information (CAW-00-1425), applicable to the WCAPs listed below, are presented in Attachment C-1. WCAP-11419 Rev. 2, "Setpoint Methodology for Protection Systems for Beaver Valley Power Station - Unit 1" (Proprietary Class 2C) is presented in Attachment C-2. WCAP-15407, "Setpoint Methodology for Protection Systems for Beaver Valley Power Station -

AP01

Unit 1" (Non-Proprietary Class 3) is presented in Attachment C-3. WCAP-11366 Rev. 4, "Setpoint Methodology for Protection Systems for Beaver Valley Power Station - Unit 2" (Proprietary Class 2C) is presented in Attachment C-4. WCAP-15408, "Setpoint Methodology for Protection Systems for Beaver Valley Power Station - Unit 2" (Non-Proprietary Class 3) is presented in Attachment C-5.

The Proprietary Information Notice, Copyright Notice, a Westinghouse application for withholding proprietary information (CAW-00-1431), applicable to the WCAPs listed below, are presented in Attachment D-1. WCAP-15264 Rev. 3, "Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 1" (Proprietary Class 2) is presented in Attachment D-2. WCAP-15336 Rev. 2, "Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 1" (Non-Proprietary Class 3) is presented in Attachment D-3. WCAP-15265 Rev. 2, "Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 2" (Proprietary Class 2) are presented in Attachment D-4. WCAP-15337 Rev. 2, "Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 2" (Non-Proprietary Class 3) is presented in Attachment D-5.

As the Class 2 or Class 2C proprietary reports contain information proprietary to Westinghouse, they are supported by affidavits signed by Westinghouse, the owner of the information. These affidavits set forth the basis on which the information may be withheld from public disclosure by the Commission and address with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit(s) should reference CAW-00-1425 or CAW-00-1431 (as applicable) and should be addressed to H. A. Sepp, Manager, Regulatory and Licensing Engineering, Westinghouse Electric Company, LLC, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Draft changes to the Core Operating Limits Report are contained in Attachment E. Draft changes to the Licensing Requirements Manual are contained in Attachment F.

This proposed amendment is a follow up to License Amendment Request (LAR) No. 220 (BVPS Unit No. 1) and LAR No. 88 (BVPS Unit No. 2) previously submitted by letter L-99-006 dated January 18, 1999 (TAC No. MA4616 and MA4617) which proposed technical specification changes to revise the non-conservative trip setpoints and allowable values. LAR Nos. 220 and 88 were withdrawn by FENOC in letter L-00-100 dated July 29, 2000. In letter L-00-100, FENOC stated that our plan was to submit a new LAR to address the non-conservative trip setpoints and allowable values by December 31, 2000. This LAR fulfills the commitment stated above.

These changes have been reviewed by the Beaver Valley review committees. These changes were determined to be safe and do not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the evaluation presented in Attachment B.

An implementation period of up to 120 days is requested following the effective date of this amendment. A license condition is also requested that will permit revised technical specification requirements not to be met until the next scheduled performance of the applicable surveillance test(s) following amendment implementation. This license condition is necessary since the proposed change includes revision of the requirements for numerous trip setpoints and allowable values. This license condition will allow readjustment of the instrumentation setpoints to meet the revised requirements during future performances of normally scheduled surveillances. This plan for implementation will avoid the need to perform numerous surveillance tests in a short period of time. This plan for implementation will also avoid potential compliance issues that may arise when transitioning from the current technical specifications requirements to the revised technical specification requirements. This request for a license condition is similar to those granted by the NRC for licensee's converting to the improved standard technical specifications.

This change is requested to be approved in a timeframe that will support implementation of license amendment requests numbers 289 (Unit 1) and 161 (Unit 2) that will allow a 1.4 % power uprate at each unit. This LAR will be submitted to the NRC in the near future.

If there are any questions concerning this matter, please contact Mr. Thomas S. Cosgrove, Manager, Licensing at 724-682-5203.

Sincerely,

Marc T. Pearson for
Lew W. Myers

c: Mr. L. J. Burkhart, Project Manager
Mr. D. M. Kern, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

**Subject:** **Beaver Valley Power Station, Unit No. 1 and 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**

I, Marc P. Pearson, being duly sworn, state that I am Director, Plant Services of FirstEnergy Nuclear Operating Company (FENOC), that I am authorized to sign and file this submittal with the Nuclear Regulatory Commission on behalf of FENOC, and that the statements made and the matters set forth herein pertaining to FENOC are true and correct to the best of my knowledge and belief.

FirstEnergy Nuclear Operating Company

Marc P. Pearson
Director, Plant Services - FENOC

COMMONWEALTH OF PENNSYLVANIA

COUNTY OF BEAVER

Subscribed and sworn to me, a Notary Public, in and for the County and State above named, this 27 th day of December, 2000.

My Commission Expires:

Notarial Seal
Sheila M. Fattore, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Sept. 30, 2002
Member, Pennsylvania Association of Notaries

ATTACHMENT A-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 286

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(next page is 6)

(3) Less Than Three Loop Operation

~~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 for less than three loop operation have been submitted by the licensees 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.~~

(4) Steam Generator Water Rise Rate

Deleted per License Amendment No. 24.

(5) Fire Protection Program

FENOC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report (UFSAR) for the facility, subject to the following provision: FENOC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Deleted per License Amendment No. ____.

(Proposed Wording)

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SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~ ← DELETE


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| ~~2.2 LIMITING SAFETY SYSTEM SETTINGS~~ | |
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(Proposed Wording)

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(Proposed Wording)

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(Proposed Wording)

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(Proposed Wording)

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(Proposed Wording)

2) Major changes in the design of radwaste treatment systems (liquid, gaseous and solid) that could significantly increase the quantities or activity of effluents released or volumes of solid waste stored or shipped offsite from those previously considered in the FSAR and SER (e.g., use of asphalt system in place of cement);

3) Changes in system design which may invalidate the accident analysis as described in the SER (e.g., changes in tank capacity that would alter the curies released); and

4) Changes in system design that could potentially result in a significant increase in occupational exposure of operating personnel (e.g., use of temporary equipment without adequate shielding provisions.)

MEMBER(S) OF THE PUBLIC

1.36 MEMBER(S) OF THE PUBLIC means any individual except when that individual is receiving an occupational dose.

CORE OPERATING LIMITS REPORT

cycle specific parameter

DELETE

1.37 The CORE OPERATING LIMITS REPORT (COLR) is the unit~~-~~specific document that provides ~~core operating~~ limits for the current ~~operating~~ reload cycle. These cycle~~-~~specific ~~core operating~~ limits shall be determined for each reload cycle in accordance with Specification ~~6.9.1.12~~. Plant operation within these ~~operating~~ limits is addressed in individual specifications.

DELETE

DELETE

6.9.5

DELETE

parameter



(Proposed Wording)

SECTION 2.0

SAFETY LIMITS

~~AND~~

~~LIMITING SAFETY SYSTEM SETTINGS~~ ← DELETE

(Proposed Wording)

## 2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~ DELETE

### 2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature ($T_{avg}$) shall not exceed the limits ~~shown~~ specified in ~~Figure 2.1-1 for 3 loop operation.~~

REPLACE WITH INSERT "1"

APPLICABILITY: MODES 1 and 2.

ACTION:

~~Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.~~

REPLACE WITH INSERT "2"

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.



(Proposed Wording)

Insert 1

the COLR; and the following Safety Limits shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for WRB-1 DNB correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained ≤ 4700°F.

Insert 2

If Safety Limit 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

REPLACE WITH INSERT "3" AND RELOCATE TO COLR

DELETE This page

~~DPR-66~~

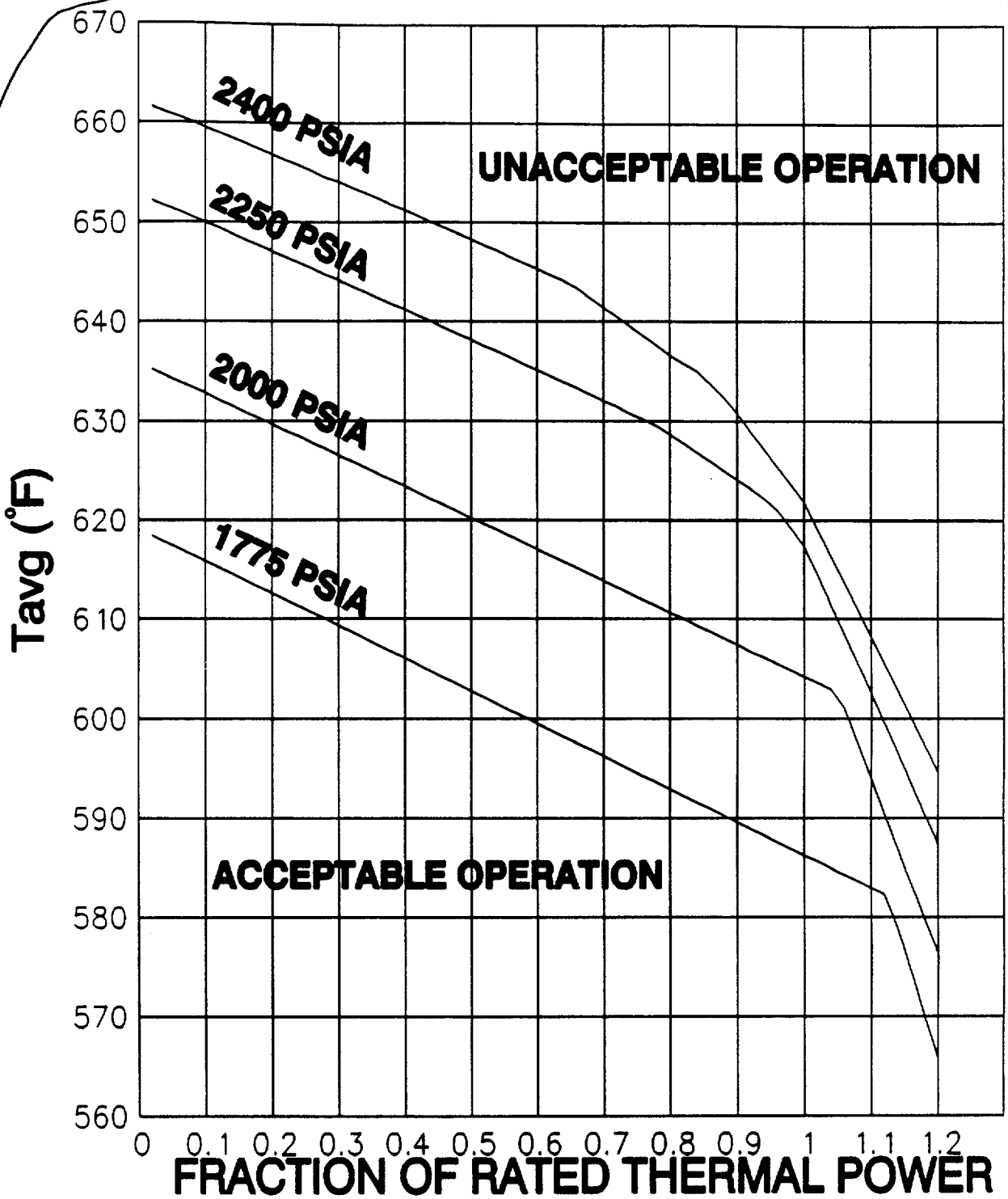


**FIGURE 2.1-1**

**REACTOR CORE SAFETY LIMIT**
**THREE LOOP OPERATION**



(Proposed Wording)

Insert 3

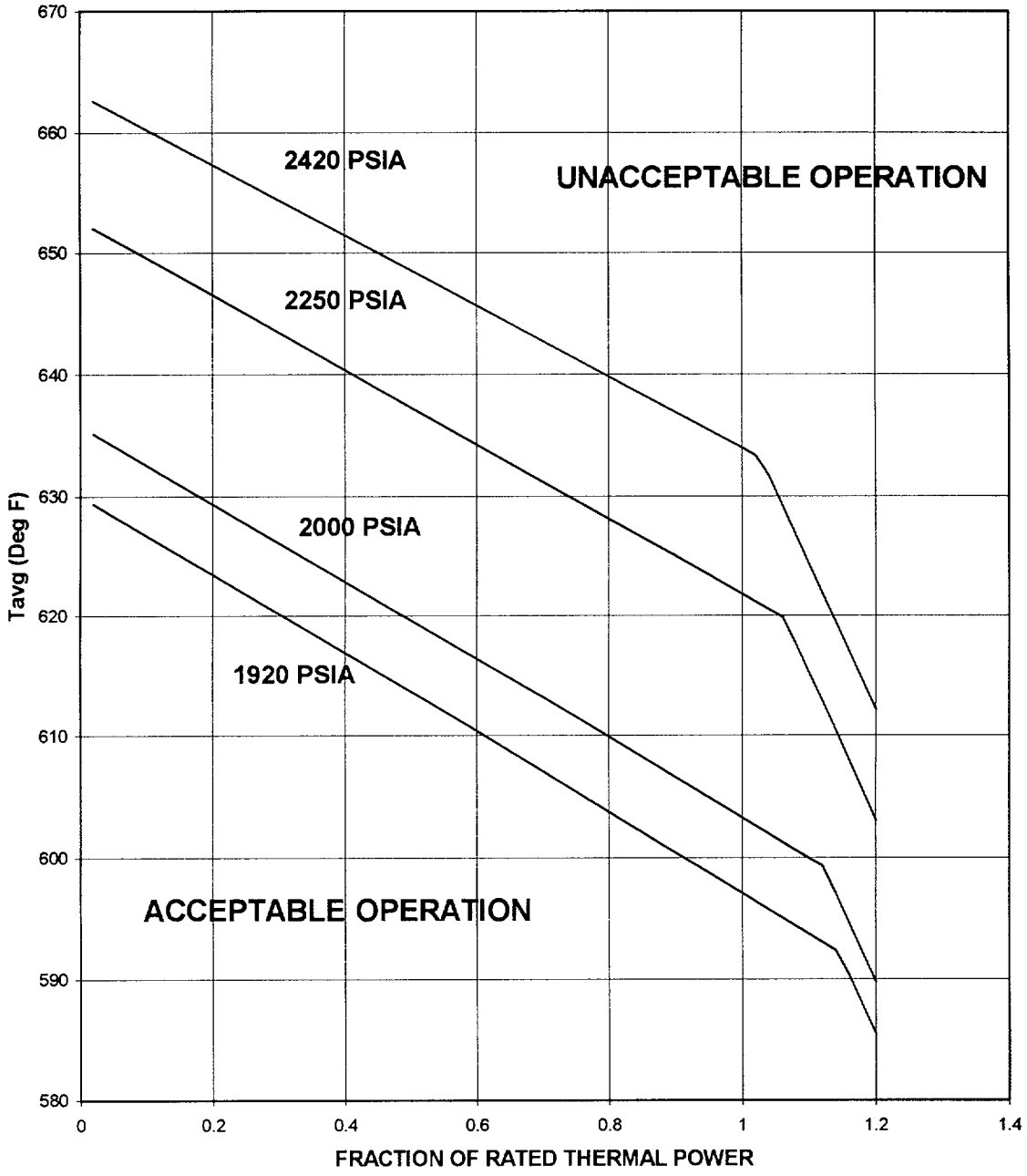


**Figure 2.1-1**
**REACTOR CORE SAFETY LIMIT**
**THREE LOOP OPERATION**

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## ~~2.2 LIMITING SAFETY SYSTEM SETTINGS~~

### ~~REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS~~

~~2.2.1 The reactor trip system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.~~

~~APPLICABILITY: AS SHOWN FOR EACH CHANNEL IN TABLE 3.3-1.~~

~~ACTION:~~

~~With a reactor trip system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.~~

RELOCATE TO LRM

DELETE THIS PAGE

MOVE TO TABLE 3.3-1

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION ~~TRIP SETPOINTS~~

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 1. Manual Reactor Trip | Not Applicable | Not Applicable |
| 2. Power Range, Neutron Flux | ~~Low Setpoint~~ ≤ 25% of RATED THERMAL POWER | ~~Low Setpoint~~ ≤ ~~27.3~~ 25.5% of RATED THERMAL POWER |
| b. Low Setpoint ← ADD | | DELETE |
| a. High Setpoint | ~~High Setpoint~~ ≤ 109% of RATED THERMAL POWER | ~~High Setpoint~~ ≤ ~~111.3~~ 109.5% of RATED THERMAL POWER |
| 3. Power Range, Neutron Flux, High Positive Rate | ≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds | ≤ ~~6.3~~ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds |
| 4. Power Range, Neutron Flux, High Negative Rate | ≤ 5% of RATED THERMAL POWER with a time constant ≥ 2 seconds | ≤ ~~6.3~~ 5.5% of RATED THERMAL POWER with a time constant ≥ 2 seconds |
| 5. Intermediate Range, Neutron Flux | ≤ 25% of RATED THERMAL POWER | ≤ ~~31.1~~ 27.9% of RATED THERMAL POWER |
| 6. Source Range, Neutron Flux | ≤ $10^5$ counts per second | ≤ ~~1.4~~ 1.3 x $10^5$ counts per second |
| 7. Overtemperature ΔT | See ~~Note 1~~ Technical Specification Table Notation (A) on Table 3.3-1 | See ~~Note 3~~ Table Notation (A) |
| 8. Overpower ΔT | See ~~Note 2~~ Technical Specification Table Notation (B) on Table 3.3-1 | See ~~Note 4~~ Table Notation (B) |
| 9. Pressurizer Pressure--Low | ≥ 1945 psig | ≥ ~~1934~~ 1941 psig |
| 10. Pressurizer Pressure--High | ≤ 2385 psig | ≤ ~~2394~~ 2389 psig |
| 11. Pressurizer Water Level--High | ≤ 92% of instrument span | ≤ ~~93.9~~ 92.5% of instrument span |
| 12. Loss of Flow | ≥ ~~90~~ 90.2% of ~~design flow* per loop~~ indicated loop flow | ≥ ~~89.0~~ 89.8% of ~~design flow* per~~ indicated loop flow |

~~*Design flow is 87,200 gpm per loop~~ ← DELETE



(Proposed Wording)

RELOCATE TO LRM

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MOVE TO TABLE 3.3-1

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION ~~TRIP SETPOINTS~~

| FUNCTIONAL UNIT | NOMINAL* TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 13. Steam Generator Water Level-Low-Low | ≥ ~~17~~ 15.1% of narrow range instrument span-each steam generator | ≥ ~~10.7~~ 14.6% of narrow range instrument span-each steam generator |
| 14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level | ≤ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level | ≤ ~~43.4~~ 40.6% of full steam flow at RATED THERMAL POWER coincident with steam generator water level |
| | ≥ 25% of narrow range instrument span-each steam generator | ≥ ~~23.1~~ 24.5% of narrow range instrument span-each steam generator |
| 15. Undervoltage-Reactor Coolant Pumps | ≥ ~~2750 volts~~ 75% of nominal bus voltage-each bus | ≥ ~~2687 volts~~ 71.2% of nominal bus voltage-each bus |
| 16. Underfrequency-Reactor Coolant Pumps | ≥ 57.5 Hz - each bus | ≥ 57.4 Hz - each bus |
| 17. Turbine Trip | | |
| A. Auto stop oil pressure | 45 psig | ≥ ~~± 5~~ 42.9 psig |
| B. Turbine Stop Valve Closure | ≥ 1% open | ≥ 1% open |
| 18. Safety Injection Input from ESF | Not Applicable | Not Applicable |
| 19. Reactor Coolant Pump Breaker Position Trip | Not Applcable | Not Applicable |
| 20. Reactor Trip System Interlocks | | |
| A. Intermediate Range Neutron Flux, P-6 | ≥ $1 \times 10^{-10}$ Amps | ≥ ~~6~~ 9.0 $\times 10^{-11}$ Amps |

ADD

* With the exception of Functional Unit number 17.B.

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

20. (Continued)

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| B. Power Range Neutron Flux, P-8 | ≤ 30% RATED THERMAL POWER | ≤ ~~22.3~~ 30.5% RATED THERMAL POWER |
| C. Power Range Neutron Flux, P-9 | ≤ 49% RATED THERMAL POWER | ≤ ~~51.3~~ 49.5% RATED THERMAL POWER |
| D. Power Range Neutron Flux, P-10 (Input to P-7) | 10% RATED THERMAL POWER | ≥ ~~7.7~~ 9.5% and ≤ ~~12.3~~ 10.5% RATED THERMAL POWER on decreasing power; ≤ ~~12.3~~ 10.5% RATED THERMAL POWER on increasing power |
| E. Turbine Impulse Chamber Pressure, P-13 (Input to P-7) | ≤ 10% of RTP Turbine Impulse Pressure Equivalent | ≤ ~~12.3~~ 10.5% of RTP Turbine Impulse Pressure Equivalent |

RELOCATE TO LRM

MOVE TO TABLE 3.3-1



(Proposed Wording)

DELETE THIS PAGE

A: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel, 0.5% ΔT span for the Tavg channel, 0.5% ΔT span for the Pressurizer Pressure channel and 0.5% ΔT span for the f(ΔI) channel.

ΔT is measured RCS ΔT, °F

**TABLE 2.2-1 (Continued)**

**REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS**

**NOTATION**

**NOTE 1:** Overtemperature $\Delta T \left(\frac{1}{1+\tau_4 S}\right) \leq \Delta T_o \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)\left[T\left(\frac{1}{1+\tau_5 S}\right) - T'\right] + K_3 (P-P') - f(\Delta I)\right]$

where:

$\Delta T_o$ = Loop specific indicated ΔT at RATED THERMAL POWER, °F

$T$ = IS = Average temperature, °F, measured RCS

$T'$ = IS = 576.3°F indicated $T_{avg}$ at RATED THERMAL POWER ≤576.2 COLR — specified in the COLR.

$P$ = IS = Pressurizer pressure, psig, measured

$P'$ = IS = 2235 psig indicated RCS nominal operating pressure ≥2250 psia → COLR — specified in the COLR.

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for $T_{avg}$ dynamic compensation.

$\tau_1$ & $\tau_2$ are the = Time constants utilized in the lead-lag controller for $T_{avg}$ — specified in the COLR.
$\tau_1 = 30$ secs, $\tau_2 = 4$ secs. → $\tau_1 \geq 30$ sec, $\tau_2 \leq 4$ secs → COLR

$\frac{1}{1+\tau_4 S}$ = Lag compensator on measured ΔT

$\tau_4$ = Time constant utilized in the lag compensator for ΔT, ≤ 2 secs.

$\frac{1}{1+\tau_5 S}$ = Lag compensator on measured Tavg

$\tau_5$ = Time constant utilized in the lag compensator for Tavg ≤ 2 secs.

$S$ = IS = the Laplace transform operator, $sec^{-1}$

MOVE TO TABLE 3.3-1 Table Notation

(Proposed Wording)

DELETE This Page

$K_1$ is specified in the COLR.
$K_2$ is specified in the COLR.
$K_3$ is specified in the COLR.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

| Operation with 3 Loops | Operation with 2 Loops (no loops isolated) | Operation with 2 Loops (1 loop isolated) |
|---|---|---|
| $K_1$ = 1.18 ≤ 1.259 → COLR | $K_1$ = 0.99 | $K_1$ = 1.1 |
| $K_2$ ≥ 0.01655 /°F | $K_2$ = 0.01655 | $K_2$ = 0.01655 |
| $K_3$ ≥ 0.000801 /psig | $K_3$ = 0.000801 | $K_3$ = 0.000801 |

→ COLR

as specified in the COLR.

and f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between -23 percent and +11 percent, f (ΔI) = 0 (where $q_t$ and $q_b$ are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of $(q_t - q_b)$ exceeds -23 percent, the ΔT trip setpoint shall be automatically reduced by 1.54 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $(q_t - q_b)$ exceeds +11 percent, the ΔT trip setpoint shall be automatically reduced by 1.91 percent of its value at RATED THERMAL POWER.

COLR

MOVE TO Table 3.3-1 Table Notation

(Proposed Wording)

DELETE THIS PAGE

B: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel and 0.5% ΔT span for the Tavg channel.

ΔT is measured RCS ΔT, °F.

MOVE TO TABLE 3.3-1 TABLE Notation

**TABLE 2.2-1 (CONTINUED)**

**REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS**

**NOTATION**

**NOTE 2:** Overpower $\Delta T \left(\frac{1}{1+\tau_4 S}\right) \leq \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right)\left(\frac{1}{1+\tau_5 S}\right) T - K_6 \left[T\left(\frac{1}{1+\tau_5 S}\right) - T''\right] - f(\Delta I)\right]$

is loop specific

where:

$\Delta T_o$ = Indicated ΔT at RATED THERMAL POWER, °F.

measured RCS

$T$ = Average temperature, °F.

$T''$ = Indicated $T_{avg}$ at RATED THERMAL POWER specified in the COLR. ≤ 576.3°F → ≤576.2°F → COLR

$K_4$ = 1.07 → ≤1.0916 → COLR

$K_5$ = ≥0.02/°F for increasing average temperature → COLR

$K_6$ = ≥0.00128/°F for $T > T''$; $K_6 = (0)$ for $T \leq T''$ (COLR)

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for $T_{avg}$ dynamic compensation.

$\tau_3$ = Time constant utilized in the rate lag controller for $T_{avg}$ specified in the COLR. $\tau_3$ = 10 secs

$\frac{1}{1+\tau_4 S}$ = Lag compensator on measured ΔT

$\tau_4$ = Time constant utilized in the lag compensator for ΔT, ≤ 2 secs.

$\frac{1}{1+\tau_5 S}$ = Lag compensator on measured $T_{avg}$

$\tau_5$ = Time constant utilized in the lag compensator for Tavg ≤ 2 secs.

$S$ = Laplace transform operator, $sec^{-1}$.

$f(\Delta I)$ = 0 for all ΔI

REVISED TO new notes A and B

**NOTE 3:** The channel's maximum trip point shall not exceed its computed trip point by more than 1.3 percent. → 0.5%

**NOTE 4:** The channel's maximum trip point shall not exceed its computed trip point by more than 2.9 percent.



(Proposed Wording)

BASES

FOR

SAFETY LIMITS

~~AND~~

~~LIMITING SAFETY SYSTEM SETTINGS~~ ← DELETE

(Proposed Wording)

## 2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. REPLACE WITH INSERT "4"

~~The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting fuel rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for the WRB-1 correlation).~~

REPLACE WITH INSERT "5"

~~In meeting this design basis, uncertainties in nuclear and thermal parameters, and fuel fabrication parameters were combined statistically with the DNB correlation uncertainties to determine the plant DNBR uncertainty and establish the design DNBR limit such that there is at least a 95% probability with 95% confidence level that the minimum DNBR for the limiting fuel rod is greater than or equal to the DNBR limit. For this application, the design DNBR limit is 1.21. This DNBR value must be met in plant safety analyses using nominal values of the input parameters that were included in the DNBR uncertainty evaluation.~~ In addition, margin has been maintained in the design by meeting a safety analysis DNBR limit of ~~1.33~~ 1.36 for both typical and thimble cells in performing safety analyses.

The ~~curves of Figure 2.1-1, show~~ figure provided in the COLR shows the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature 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.

ADD INSERT "6"

Insert 4

The DNB thermal design criterion is that the probability of DNB not occurring on the most limiting rod is at least 95 percent at a 95 percent confidence level for any Condition I or II event.

Insert 5

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes have been statistically combined with the DNB correlation uncertainties to determine the DNBR Design Limits which are 1.24 for typical and 1.23 for thimble cell.

Insert 6

The figure is based on enthalpy hot channel factor limits provided in the COLR.

REPLACE WITH INSERT "7"

~~The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.62 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:~~

~~$$F^N_{\Delta H} \leq 1.62\ [1 + 0.3\ (1-P)]$$~~

~~where P is the fraction of RATED THERMAL POWER~~

~~These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trip will reduce the setpoint to provide protection consistent with core safety limits.~~

## 2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping and fittings are designed to ANSI B 31.1 and the valves are designed to ASA 16.5 which permit a maximum transient pressure of 120% (2985) psig of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig to demonstrate integrity prior to initial operation.



(Proposed Wording)

Insert 7

The reactor core Safety Limits are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core Safety Limits are used to define the various Reactor Protection System (RPS) functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower $\Delta T$ reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and $\Delta I$ that the reactor core Safety Limits will be satisfied during steady state operation, normal operational transients, and AOOs.

DELETE THIS PAGE

BASES

~~2.2.1 REACTOR TRIP SET POINTS~~

REPLACE WITH INSERT "8"

~~The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a Trip Setpoint less conservative than its Setpoint Limit but within its specified Allowable Value is acceptable on the basis that each Allowable Value is equal to or less than the drift allowance assumed to occur for each trip used in the accident analyses.~~

~~For the Overtemperature ΔT, and Overpower ΔT trips, Table 2.2-1 specifies several time constants (τ's) in terms of equalities, such as τ=x seconds. These time constants represent nominal values and are periodically adjusted to within their specified calibration accuracy. With the time constants within the specified accuracy the accident analysis produces acceptable results.~~

DELETE

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low set point provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above ~~approximately 9 per cent of RATED THERMAL POWER~~) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below ~~approximately 9 percent of RATED THERMAL POWER~~). DELETE

the P-10 setpoint

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2



(Proposed Wording)

Insert 8

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

Technical specifications are required by 10 CFR 50.36 to contain Limiting Safety System Settings (LSSS) defined by the regulation as "...settings for automatic protective devices...so chosen that automatic protective action will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytic Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytic Limit therefore ensures that the SL is not exceeded.

The Allowable Values (Nominal Trip Setpoints ± the calibration tolerance) specified in Table 3.3.1 are considered the LSSS as identified in 10 CFR 50.36 and have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents.

The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides Reactor Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System.

The Reactor Trip System initiates a turbine trip signal whenever reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

The difference between T' (Overtemperature $\Delta T$) or T" (Overpower $\Delta T$) and the loop specific, indicated, full power $T_{avg}$ shall be less than or equal to the $T_{avg}$ allowances for such differences in the uncertainty calculations for these functions. In addition, T' and T" shall be less than or equal to the full power $T_{avg}$ modeled in the safety analyses as an initial condition assumption; i.e., the numerical value specified in the COLR. In the event that the difference between a T' or T" set to the numerical value specified in the COLR and a loop specific, indicated, full power $T_{avg}$ is greater than the $T_{avg}$ allowances for such differences in the uncertainty calculations, T' or T" shall be reduced until the difference allowances in the uncertainty calculations are satisfied; i.e., T' or T" are set to a loop specific, full power value less than the numerical value specified in the COLR. These reductions in the values of T' and T" are consistent with the recommendations of Westinghouse Technical Bulletin ESBU-TB-96-07-RO, "Temperature Related Functions," 11/5/96.

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MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2

BASES

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the design DNBR limit for control rod drop accidents. At high power a single or multiple rod drop accident could cause flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. For those transients on which reactor trip on power range negative rate trip is not postulated, it is shown that the minimum DNBR is greater than the design DNBR limit.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor start-up. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range Channels will initiate a reactor trip at ~~about $10^{+5}$ counts per second~~ the trip setpoint unless manually blocked when P-6 becomes active. The Intermediate Range Channels will initiate a reactor trip at a current level proportional to ~~approximately 25 percent of RATED THERMAL POWER~~ the trip setpoint unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown ~~on Figure 2.1-1~~ in the COLR. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table ~~B.2-1~~ 3.3-1.



(Proposed Wording)

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BASES

~~Operation with a reactor coolant loop out of service below the 3 loop P-8 set point does not require reactor protection system set point modification because the P-8 set point and associated trip will prevent DNB during 2, loop operation exclusive of the Overtemperature ΔT set point. 2 loop operation above the 3 loop P-8 set point is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature ΔT channels and raising the P-8 set point to its 2 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.~~

DELETE

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip provides protection by tripping the reactor in the event of a loss of reactor coolant pressure.

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief

MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2



(Proposed Wording)

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BASES

through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

the trip setpoint

The loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above ~~11 percent of RATED THERMAL POWER~~ (P-7), an automatic the trip setpoint reactor trip will occur if the flow in any two loops drop below ~~90% of nominal full loop flow~~. DELETE Above ~~31%~~ (P-8) ~~of RATED THERMAL POWER~~, an automatic reactor trip will occur if the flow in any single loop drops below ~~90% of nominal full loop flow~~. ~~This latter trip will prevent the minimum value of the DNBR from going below the design DNBR limit during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature $\Delta$T trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature $\Delta$T trip setpoint adjusted to the value specified for 2 loop operation, the P-8 trip at 66% RATED THERMAL POWER with loop stop valves open and at 71% RATED THERMAL POWER with a loop stop valve closed will prevent the minimum value of the DNBR from going below the design DNBR limit during normal operational transients and anticipated transients with 2 loops in operation.~~

DELETE

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table ~~2.2-1~~ 3.3-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall

MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2

BASES

reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow ~~exceeds the feedwater flow by > 1.55 x 10⁶ lbs/hour~~. The Steam Generator Low Water level portion of the trip is activated when the water level drops below ~~25 percent~~ drops below the trip setpoint, as indicated by the narrow range instrument. These trip ~~values~~ Setpoints include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The trip ~~specified~~ setpoints assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.9 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.3 seconds.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-9. Each of the turbine trips ~~provide~~ provides turbine protection and ~~reduce~~ reduces the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2

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(Proposed Wording)

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB resulting from the opening of two or more pump breakers above P-7. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip set point is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System.

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

P-6 Above the setpoint P-6 allows the manual block of the Source Range reactor trip and de-energizing of the high voltage to the detectors. Below the setpoint Source Range level trips are automatically reactivated and high voltage restored.

P-7 Above the setpoint P-7 automatically enables reactor trips on low flow or coolant pump breaker open in more than one primary coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. Below the setpoint the above listed trips are automatically blocked.

P-8 Above the setpoint P-8 automatically enables reactor trip on low flow in one or more primary coolant loops. Below the setpoint P-8 automatically blocks the above listed trip.

P-9 Above the setpoint P-9 automatically enables a reactor trip on turbine trip. Below the setpoint P-9 automatically blocks a reactor trip on turbine trip.

P-10 Above the setpoint P-10 allows the manual block of the Intermediate Range reactor trip and the low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip and de-energizes the Source Range high voltage power. Below the setpoint the Intermediate Range reactor trip and the low setpoint Power Range reactor trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2



(Proposed Wording)

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## 3/4.2.5 DNB PARAMETERS

### LIMITING CONDITION FOR OPERATION

3.2.5 The following ~~DNB related~~ parameters shall be maintained within the limits ~~shown on Table 3.2-1(1):~~ specified below:

a. Reactor Coolant System $T_{avg}$ is less than or equal to the limit specified in the COLR,

b. Pressurizer Pressure is greater than or equal to the limit(1) specified in the COLR, and

c. Reactor Coolant System Total Flow Rate ≥ 261,600 gpm(3) and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1(2).

ACTION:

With ~~any of~~ one or more of the above DNB related parameters ~~exceeding its limit~~ not within limits, restore the DNB related parameter to within ~~its~~ limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours. (S) DELETE

### SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the above DNB related parameters ~~of Table 3.2-1~~ DELETE shall be verified to be ~~indicating~~ DELETE within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit (S) by measurement at least once per 18 months.

REPLACE WITH FOOTNOTE (1) ON PAGE 3/4 2-13

~~(1) The values presented in Table 3.2-1 correspond to analytical limits used in the safety analyses.~~

(2) The provisions of Specification 4.0.4 are not applicable for Reactor Coolant System total flow rate to allow a calorimetric flow measurement and the calibration of the Reactor Coolant System total flow rate indicators.

ADD → (3) Analytical limit used in the safety analysis.



(Proposed Wording)

DELETE THIS PAGE

~~TABLE 3.2-1~~

DNB PARAMETERS

(Technical Specification 3.2.5)

| PARAMETER | | ~~3 Loops In Operation~~ | Indicated Value |
|---|---|---|---|
| Reactor Coolant System $T_{avg}$ | 580.0 | ≤ ~~580.7~~ °F | (1) |
| Pressurizer Pressure | 2215 | ≥ ~~2220~~ psia | (2) |
| Reactor Coolant System Total Flow Rate | 267,400 | ≥ ~~261,600~~ gpm | (3) |

RELOCATE TO COLR

(1) Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of 10% RATED THERMAL POWER.

MOVE TO PAGE 3/4 2-12

(1) The Reactor Coolant System (RCS) $T_{avg}$ value includes allowances for rod control operation and verification via control board indication.

(2) The pressurizer pressure value includes allowances for pressurizer pressure control operation and verification via control board indication.

(3) The RCS total flow rate includes allowances 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.

ADD

RELOCATE TO COLR

## 3/4.3 INSTRUMENTATION

### 3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1. (1) ← ADD

#### SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by interlock operation. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

---

(1) Separate ACTION statement entry is allowed for each Function.

ADD



(Proposed Wording)

ADD ALLOWABLE VALUE COLUMN FROM TABLE 2.2-1

## TABLE 3.3-1

## REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 1. Manual Reactor Trip | 2 | 1 | 2 | 1, 2, 3(3), 4(3) and 5(3) | 12 |
| 2. Power Range, Neutron Flux | | | | | |
| a. High Setpoint | 4 | 2 | 3 | 1, 2 | 2 |
| b. Low Setpoint | 4 | 2 | 3 | 1(1), 2 | 2 |
| 3. Power Range, Neutron Flux High Positive Rate | 4 | 2 | 3 | 1, 2 | 2 |
| 4. Power Range, Neutron Flux High Negative Rate | 4 | 2 | 3 | 1, 2 | 2 |
| 5. Intermediate Range, Neutron Flux | 2 | 1 | 2 | 1(1), 2, 3(3), 4(3) and 5(3) | 3 |
| 6. Source Range, Neutron Flux | | | Not Applicable | (8) | |
| a. With Rod Withdrawal Capability | 2 | 1 | 2 | 2(2), 3(3), 4(3) and 5(3) | 4 |
| b. With All Rods Fully Inserted and Without Rod Withdrawal Capability | 2 | 0 | 1 | 3, 4, and 5 | 5 |
| 7. Overtemperature ΔT | 3 | 2 | 2 | 1, 2 | 7 |
| 8. Overpower ΔT | 3 | 2 | 2 | 1, 2 | 7 |
| 9. Pressurizer Pressure-Low (Above P-7) | 3 | 2 | 2 | 1, 2 | 7 |



(Proposed Wording)

ADD ALLOWABLE VALUE COLUMN FROM TABLE 2.2-1

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

| | FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 10. | Pressurizer Pressure-High | 3 | 2 | 2 | 1, 2 | 7 |
| 11. | Pressurizer Water Level-High (Above P-7) | 3 | 2 | 2 | 1, 2 | 7 |
| 12. | Loss of Flow - Single Loop (Above P-8) | 3/loop | 2/loop in any operating loop | 2/loop in each operating loop | 1 | 7 |
| 13. | Loss of Flow - Two Loops (Above P-7 and below P-8) | 3/loop | 2/loop in two operating loops | 2/loop each operating loop | 1 | 7 |
| 14. | Steam Generator Water Level-Low-Low (Loop Stop Valves Open) | 3/loop | 2/loop | 2/loop | 1, 2 | 7 |
| 15. | Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level | 2/loop-level and 2/loop-flow mismatch | 1/loop-level coincident with 1/loop-flow mismatch in same loop | 1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch | 1, 2 | 7 |
| 16. | Undervoltage-Reactor Coolant Pumps (Above P-7) | 3-1/bus | 2 | 2 | 1 | 7 |
| 17. | Underfrequency-Reactor Coolant Pumps (Above P-7) | 3-1/bus | 2 | 2 | 1 | 7 |



(Proposed Wording)

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 2.2-1

| | FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 18. | Turbine Trip (Above P-9) | | | | | |
| | A. Auto Stop Oil Pressure | 3 | 2 | 2 | 1 | 7 |
| | B. Turbine Stop Valve Closure | 4 | 4 | 4 | 1 | 8 |
| 19. | Safety Injection Input from ESF | 2 | 1 | 2 | 1, 2 | 1 |
| 20. | Reactor Coolant Pump Breaker Position Trip (Above P-7) | 1/breaker | 2 | 1/breaker per operating loop | 1 | 11 |
| 21. | Reactor Trip Breakers | 2 | 1 | 2 | 1, 2 | 40 |
| | | 2 | 1 | 2 | 3(3), 4(3), 5(3) | 39 |
| 22. | Automatic Trip Logic | 2 | 1 | 2 | 1, 2 | 1 |
| | | 2 | 1 | 2 | 3(3), 4(3), 5(3) | 39 |
| 23. | Reactor Trip System Interlocks | | | | | |
| | A. Intermediate Range Neutron Flux, P-6 | 2 | 1 | 1 | 2 | 3 |
| | B. Power Range Neutron Flux, P-8 | 4 | 2 | 3 | 1 | 12 |
| | C. Power Range Neutron Flux, P-9 | 4 | 2 | 3 | 1 | 12 |
| | D. Power Range Neutron Flux, P-10 | 4 | 2 | 3 | 1 | 12 |
| | E. Turbine Impulse Chamber Pressure, P-13 | 2 | 1 | 1 | 1 | 12 |



(Proposed Wording)

TABLE 3.3-1 (Continued)

TABLE NOTATION

(1) Trip function may be manually bypassed in this Mode above P-10.

(2) Trip function may be manually bypassed in this Mode above P-6.

(3) With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1, provided the other channel is OPERABLE.

ACTION 2 - With one power range neutron flux channel inoperable,[4] perform one of the following, as applicable:

a. Power Range High Neutron Flux Channel

1. Place the inoperable channel in trip within 6 hours and reduce THERMAL POWER to less than or equal to 75 percent RATED THERMAL POWER within the next 6 hours and perform SR 4.2.4,[5] or

2. Place the inoperable channel in trip within 6 hours and perform SR 4.2.4,[5] or

3. Be in MODE 3 within 12 hours.

b. All other channels

1. Place the inoperable channel in trip within 6 hours, or

2. Be in MODE 3 within 12 hours.

ADD TABLE NOTATION (A) FROM TABLE 2.2-1

ADD TABLE NOTATION (B) FROM TABLE 2.2-1

(8) In this condition, source range Function does not provide reactor trip but does provide indication.

---

(4) The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels.

(5) Only required to be performed when the power range high neutron flux channel input to QPTR is inoperable.



(Proposed Wording)

INSTRUMENTATION

## 3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.2.1 The engineered safety feature actuation system instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE ~~with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4~~. DELETE

APPLICABILITY: As shown in Table 3.3-3.

ACTION: DELETE

~~a. With an engineered safety feature actuation system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint Value.~~

~~b.~~ With an engineered safety feature actuation system instrumentation channel or interlock inoperable, take the ~~action~~ ACTION shown in Table 3.3-3. (1)

### SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each engineered safety feature actuation system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by interlock operation. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESF function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESF function as shown in the "Total No. of Channels" Column of Table 3.3-3.

(1) Separate ACTION statement entry is allowed for each Function. ← ADD



(Proposed Wording)

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

| | FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 1. | SAFETY INJECTION AND FEEDWATER ISOLATION | | | | | |
| | a. Manual Initiation | 2 | 1 | 2 | 1, 2, 3, 4 | 18 |
| | b. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 13, 36 |
| | c. Containment Pressure-High | 3 | 2 | 2 | 1, 2, 3 | 14 |
| | d. Pressurizer Pressure-Low | 3 | 2 | 2 | 1, 2, 3(1) | 14 |
| | e. ~~Low~~ Steamline Pressure — Low (DELETE) | 3/loop | 2/loop any loop | 2/loop any loop | 1, 2, 3(1) | 14 |



(Proposed Wording)

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 1.1 SAFETY INJECTION-TRANSFER FROM INJECTION TO THE RE-CIRCULATION MODE | | | | | |
| a. Manual Initiation | 2 sets<br>2 switches/ set | 1 set | 2 sets | 1, 2, 3, 4 | 18 |
| b. Automatic Actuation Logic Coincident with Safety Injection Signal | 2 | 1 | 2 | 1, 2, 3 | 18 |
| c. Refueling Water Storage Tank Level-Low | 4 | 2 | 3 | 1, 2, 3 | 16 |
| d. Refueling Water Storage Tank Level - Auto QS Flow Reduction | 1 per train | 1 per train | 1 per train | 1, 2, 3 | 18 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 2. CONTAINMENT SPRAY | | | | | |
| a. Manual | 2 sets | 1 set<br>2 switches | 2 sets | 1, 2, 3, 4 | 18 |
| b. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 13 |
| c. Containment Pressure-- High-High | 4 | 2 | 3 | 1, 2, 3 | 16 |
| 3. CONTAINMENT ISOLATION | | | | | |
| a. Phase "A" Isolation | | | | | |
| 1) Manual | 2 | 1 | 2 | 1, 2, 3, 4 | 18 |
| 2) From Safety Injection Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 13 |
| b. Phase "B" Isolation | | | | | |
| 1) Manual | 2 sets<br>(2 switches/set) | 1 set | 2 sets | 1, 2, 3, 4 | 18 |
| 2) Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3, 4 | 13 |
| 3) Containment Pressure-- High-High | 4 | 2 | 3 | 1, 2, 3 | 16 |



(Proposed Wording)

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 4. STEAM LINE ISOLATION | | | | | |
| a. Manual | 2/steam line | 1/steam line | 2/operating steam line | 1, 2, 3 | 18 |
| b. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3 | 13 |
| c. Containment Pressure Intermediate-High-High | 3 | 2 | 2 | 1, 2, 3 | 14 |
| d. ~~Low~~ (DELETE) Steamline Pressure - Low | 3/loop | 2/loop any loop | 2/loop any loop | 1, 2, 3(1) | 14 |
| e. ~~High~~ (DELETE) ~~Steam~~ Steamline Pressure Rate - High Negative | 3/loop | 2/loop any loop | 2/operating loop | 3(2) | 14 |



(Proposed Wording)

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 5. TURBINE TRIP & FEEDWATER ISOLATION | | | | | |
| a. Steam Generator Water Level-- High-High, P-14 | 3/loop | 2 loop in any operating loop | 2/loop in each operating loop | 1, 2, 3 | 14 |
| 6. LOSS OF POWER (Emergency Bus Undervoltage) | | | | | |
| a. 4.16kv ~~Bus~~ | 1/4.16kv Bus | | | | |
| 1. Loss of Voltage (trip feeder) | | 1/4.16kv Bus | 1/4kv Bus | 1, 2, 3, 4 | 33 |
| 2. Loss of Voltage (start diesel) | 1/4.16kv Bus | 1/4.16kv Bus | 1/4kv Bus | 1, 2, 3, 4 | 33 |
| b. ~~Grid Degraded Voltage (4.16kv Bus)~~ | 2/4.16kv Bus | 2/Bus | 2/Bus | 1, 2, 3, 4 | 34 |
| c. ~~Grid Degraded Voltage (480v Bus)~~ | 2/480v Bus | 2/Bus | 2/Bus | 1, 2, 3, 4 | 34 |

4.16 Kv Emergency Bus Undervoltage (Degraded Voltage)

480 volt Emergency Bus Undervoltage (Degraded Voltage)

(Proposed Wording)

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 7. AUXILIARY FEEDWATER | | | | | |
| a. Steam Gen. Water Level-Low-Low (Loop Stop valves open) | | | | | |
| i. Start Turbine Driven Pump | 3/stm. gen. | 2/stm. gen. any stm. gen. | 2/stm. gen. | 1, 2, 3 | 14 |
| ii. Start Motor Driven Pumps | 3/stm. gen. any 2 stm. gen. | 2/stm. gen. any 2 stm. gen. | 2/stm. gen. | 1, 2, 3 | 14 |
| b. Undervoltage-RCP (Start Turbine Driven Pump) | (3)-1/bus | 2 | 2 | 1 | 14 |
| c. S.I. (Start All Auxiliary Feedwater Pumps) | See 1 above (all S.I. initiating functions and requirements) | | | | |
| d. ~~Emergency Bus Undervoltage (Start Motor Driven Pumps)~~ | ~~1/bus~~ | ~~1~~ | ~~1~~ | ~~1, 2, 3~~ | ~~18~~ |
| e. Trip of Main Feedwater Pumps (Start Motor Driven Pumps) | 1/pump | 1 | 1 | 1, 2, 3 | 18 |

(Deleted)



(Proposed Wording)

(Proposed Wording)

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 8. ESF INTERLOCKS | | | | | |
| a. Reactor Trip, P-4 | 2 | 1 | 2 | 1, 2, 3 | 38 |
| b. Pressurizer Pressure, P-11 | 3 | 2 | 2 | 1, 2, 3 | 38 |
| c. Low-Low Tavg, P-12 | 3 | 2 | 2 | 1, 2, 3 | 38 |

TABLE 3.3-3 (Continued)

ACTION STATEMENTS

ACTION 33 - With the number of OPERABLE Channels one less than the Total Number of Channels, the Emergency Diesel Generator associated with the 4kv Bus shall be declared inoperable and the ACTION Statements for Specification 3.8.1.1 or Specification 3.8.1.2, as appropriate, shall apply.

ACTION 34 -

a. With the number of OPERABLE channels one less than the Minimum Number of Channels, place the inoperable channel in the tripped condition within 1 hour; otherwise, immediately enter the applicable ACTION statement(s) for the associated Emergency Diesel Generator made inoperable by the degraded voltage start instrumentation.

b. With the number of OPERABLE channels two less than the Minimum Number of Channels, restore at least one of the two channels to OPERABLE status and place the other in the tripped condition within 1 hour; otherwise, immediately enter the applicable ACTION statement(s) for the associated Emergency Diesel Generator made inoperable by the degraded voltage start instrumentation.

ACTION 36 - The block of the automatic actuation logic introduced by a reset of safety injection shall be removed by resetting (closure) of the reactor trip breakers within one hour of an inadvertent initiation of safety injection providing that all trip input signals have reset due to stable plant conditions. Otherwise, the requirements of ACTION statement 13 shall have been met.

ACTION 37 - Not applicable.

ACTION 38 - With less than the Minimum Number of Channels OPERABLE, within one hour determine by observation of the associated permissive annunciator window(s) (bistable status lights or computer checks) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

(Next page is 3/4 3-29)



(Proposed Wording)

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TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 1. SAFETY INJECTION, ~~TURBINE TRIP~~ (DELETE) AND FEEDWATER ISOLATION | | |
| a. Manual Initiation | Not Applicable | Not Applicable |
| b. Automatic Actuation Logic | Not Applicable | Not Applicable |
| c. Containment Pressure--High | ≤ 1.5 psig | ≤ ~~2.4~~ 1.8 psig |
| d. Pressurizer Pressure-Low | ≥ 1845 psig | ≥ ~~1830~~ 1841 psig |
| e. Steamline Pressure-Low | ≥ 500 psig steam line pressure | ≥ ~~488~~ 495.8 psig steam line pressure |

RELOCATE TO LRM

MOVE TO TABLE 3.3-3



(Proposed Wording)

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## TABLE 3.3-4 (Continued)

## ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 1.1 SAFETY INJECTION-TRANSFER FROM INJECTION TO THE RECIRCULATION MODE | | |
| a. Manual Initiation | Not Applicable | Not Applicable |
| b. Automatic Actuation Logic Coincident with Safety Injection Signal | Not Applicable | Not Applicable |
| c. Refueling Water Storage Tank Level-Low | 18'8-1/2" | 18'5" ≥ ~~18'2-1/2"~~ and 19'0" ≤ ~~19'2-1/2"~~ |
| d. Refueling Water Storage Tank Level - Auto QS Flow Reduction | 8'6" | ≥ ~~8'0"~~ 8'2.5" and ≤ ~~9'0"~~ 8'9.5" |

RELOCATE TO LRM

MOVE TO TABLE 3.3-3



(Proposed wording)

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 2. CONTAINMENT SPRAY | | |
| a. Manual Initiation | Not Applicable | Not Applicable |
| b. Automatic Actuation Logic | Not Applicable | Not Applicable |
| c. Containment Pressure--High-High | ≤ 8.0 psig | ≤ ~~8.9~~ 8.3 psig |
| 3. CONTAINMENT ISOLATION | | |
| a. Phase "A" Isolation | | |
| 1. Manual | Not Applicable | Not Applicable |
| 2. From Safety Injection Automatic Actuation Logic | Not Applicable | Not Applicable |
| b. Phase "B" Isolation | | |
| 1. Manual | Not Applicable | Not Applicable |
| 2. Automatic Actuation Logic | Not Applicable | Not Applicable |
| 3. Containment Pressure--High-High | ≤ 8.0 psig | ≤ ~~8.9~~ 8.3 psig |

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MOVE TO TABLE 3.3-3



(Proposed Wording)

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~~DPR-66~~ MOVE TO TABLE 3.3-3

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT (NOMINAL) | ALLOWABLE VALUES[a] |
|---|---|---|
| 4. STEAM LINE ISOLATION | | |
| a. Manual | Not Applicable | Not Applicable |
| b. Automatic Actuation Logic | Not Applicable | Not Applicable |
| c. Containment Pressure--Intermediate-High-High | ≤ 3.0 psig | ≤ ~~3.9~~ 3.3 psig |
| d. Steamline Pressure-Low | ≥ 500 psig steam line pressure | ≥ ~~488~~ 495.8 psig steam line pressure |
| e. High Negative Steam Pressure Rate | ≤ 100 psi with a time constant ≥ 50 seconds | ≤ ~~127~~ 104.2 psi with a time constant ≥ 50 seconds |
| 5. TURBINE TRIP AND FEEDWATER ISOLATION | | |
| a. Steam Generator Water Level High-High | ≤ ~~73~~ 81.2% of narrow range instrument span each steam generator | ≤ ~~76.9~~ 81.7% of narrow range instrument span each steam generator |
| 6. LOSS OF POWER | | |
| a. 1. 4.16kv Emergency Bus Undervoltage (Loss of Voltage) (Trip Feed) | ≥ 75% of nominal bus voltage with a 1 ± 0.1 second time delay | ≥ ~~73~~ 71.2% of nominal bus voltage with a 1 ± 0.1 second time delay |
| 2. 4.16kv Emergency Bus (Start Diesel) | ≥ 75% of nominal bus voltage with a < 0.9 second time delay (includes auxiliary relay times) | ≥ ~~74~~% of nominal bus voltage with a < 0.9 second time delay (includes auxiliary relay times) |
| b. 4.16kv Emergency Bus Undervoltage (Degraded Voltage) | ≥ ~~90~~ 93.7% of nominal bus voltage with a 90 ± 5 second time delay | ≥ ~~88~~ 93.4% of nominal bus voltage with a 90 ± 5 second time delay |
| c. 480v Emergency Bus Undervoltage (Degraded Voltage) | ≥ ~~90~~% of nominal bus voltage with a 90 ± 5 second time delay | ≥ ~~88~~% of nominal bus voltage with a 90 ± 5 second time delay |

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 7. AUXILIARY FEEDWATER | | |
| a. Steam Generator Water Level-low-low | 15.1 ≥ ~~13~~% of narrow range instrument span each steam generator | 14.6 ≥ ~~10.7~~% of narrow range instrument span each steam generator |
| b. Undervoltage - RCP | 75% nominal ≥ ~~2750 volts~~ RCP bus voltage | 71.2% nominal ≥ ~~2687 volts~~ RCP bus voltage |
| c. S.I. | See 1 above (all SI Setpoints) | |
| d. ~~Emergency Bus Undervoltage~~ (Deleted) | DELETE ~~≤ 3350 volts~~ | DELETE ~~≤ 3325 volts~~ |
| e. Trip of Main Feedwater Pumps | Not Applicable | Not Applicable |

RELOCATE TO LRM

MOVE TO TABLE 3.3-3



(Proposed wording)

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | NOMINAL TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 8. ESF INTERLOCKS | | |
| a. P-4 | Not Applicable | Not Applicable |
| b. P-11 | ≤ 2000 psig | ≤ ~~2010~~ 2004 psig |
| c. P-12 | ≥ 541°F | ≥ ~~539~~ 540.5°F |

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MOVE TO TABLE 3.3-3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 4. STEAM LINE ISOLATION | | | | |
| a. Manual | N.A. | N.A. | R | 1, 2, 3 |
| b. Automatic Actuation Logic | N.A. | N.A. | M(1) | 1, 2, 3 |
| c. Containment Pressure--Intermediate-High-High | S | R | Q | 1, 2, 3 |
| d. Steamline Pressure--Low | S | R | Q | 1, 2, 3 |
| e. Steamline Pressure Rate-High | S | R | Q | 1, 2, 3 |
| 5. TURBINE TRIP & FEEDWATER ISOLATION | | | | |
| a. Steam Generator Water Level--High-High | S | R | Q | 1, 2, 3 |
| 6. LOSS OF POWER | | | | |
| a. 4.16kv Emergency Bus Undervoltage (Loss of Voltage) Trip Feed & Start Diesel | N.A. | R | M | 1, 2, 3, 4 |
| b. 4.16kv and 480v Emergency Bus Undervoltage (Degraded Voltage) | N.A. | R | M | 1, 2, 3, 4 |

Negative



(Proposed Wording)

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

| FUNCTIONAL UNIT | CHANNEL CHECK | CHANNEL CALIBRATION | CHANNEL FUNCTIONAL TEST | MODES IN WHICH SURVEILLANCE REQUIRED |
|---|---|---|---|---|
| 7. AUXILIARY FEEDWATER | | | | |
| a. Steam Generator Water Level-Low-Low | S | R | Q | 1, 2, 3 |
| b. Undervoltage - RCP | S | R | M | 1, 2 |
| c. S.I. | See 1 above (all SI surveillance requirements) | | | |
| d. ~~Emergency Bus Undervoltage~~ (Deleted) | ~~N.A.~~ | ~~R~~ | ~~R~~ | ~~1, 2, 3~~ |
| e. Trip of Main Feedwater Pumps | N.A. | N.A. | R | 1, 2, 3 |
| 8. ESF INTERLOCKS | | | | |
| a. P-4 | N.A. | N.A. | R | 1, 2, 3 |
| b. P-11 | N.A. | R | Q | 1, 2, 3 |
| c. P-12 | N.A. | R | Q | 1, 2, 3 |

(Proposed Wording)

## PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

### LIMITING CONDITION FOR OPERATION

3.10.3 The minimum temperature and pressure conditions for reactor criticality of Specifications 3.1.1.5 and 3.4.9.1 may be suspended during low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5 percent of RATED THERMAL POWER,

Deleted, and

b. ~~The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at ≤ 25% of RATED THERMAL POWER, and~~

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation shown on Figures 3.4-2 and 3.4-3.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER > 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the unacceptable region of operation on Figures 3.4-2 and 3.4-3, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the analysis required by Specification 3.4.9.1 prior to the next reactor criticality.

### SURVEILLANCE REQUIREMENTS

4.10.3.1 The Reactor Coolant System shall be verified to be within the acceptable region for operation of Figures 3.4-2 and 3.4-3 at least once per hour.

4.10.3.2 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMA POWER at least once per hour.



(Proposed Wording)

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided;

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER and

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at ≤25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER >5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be ≤5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.



(Proposed Wording)

SPECIAL TEST EXCEPTIONS

NO FLOW TESTS

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set ≤ 25% of RATED THERMAL POWER

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The THERMAL POWER shall be determined to be less than the P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.5.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

3/4.2.4 QUADRANT POWER TILT RATIO (QPTR) (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, the symmetric thimble flux map can be used to confirm that QPTR is within limits.

With one excore detector inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore results may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the surveillance should be within 2 percent of the tilt shown by the most recent flux map data.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than or equal to the design DNBR limit throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

These 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.

REPLACE WITH INSERT "9"

BASES

3/4.3.1 AND 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation, and 3) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

OPERABILITY of the following trips in Table 3.3-1 provides additional diverse or anticipatory protection features and is not credited in the accident analyses:

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level; Undervoltage - Reactor Coolant Pumps (Above P-7); Underfrequency Reactor Coolant Pumps (Above P-7); Turbine Trip (Above P-9); Reactor Coolant Pump Breaker Position Trip (Above P-7); Turbine Impulse Chamber Pressure, P-13.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report as approved by the NRC and documented in the SER (letter to J. J. Sheppard from Cecil O. Thomas dated February 21, 1985). Jumpers and lifted leads are not an acceptable method for placing equipment in bypass as documented in the NRC safety evaluation report for this WCAP.

ADD BASES FROM SECTION 2.2.1

Insert 9

## 3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated action and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The Allowable Values specified in Table 3.3-1 and Table 3.3-3 are the initial values for consideration of channel operability. A channel is OPERABLE with a nominal trip setpoint value outside its calibration tolerance band provided the trip setpoint "as found" value does not exceed its associated Allowable Value.

Additional administratively controlled limits for operability of a device are determined by device drift being less than the value required for the surveillance interval. In the event the device exceeds the administratively controlled limit, operability of the device may be evaluated by other device performance characteristics, e.g., comparison to historical device drift data, calibration characteristics, response characteristics and short term drift characteristics. A device (relay, transmitter, process rack module, etc.), whose "as found" value is in excess of the calibration tolerance, but within the additional operability criteria (administratively controlled limit), is considered operable but must be recalibrated such that the "as left" value is within the two sided (±) calibration tolerance. Plant procedures set administrative limits ("as left" and "as found" criteria) to control the determination of operability by setting minimum standards based on the setpoint methodology and the uncertainty values included in the determination of the Nominal Trip Setpoint, and allow the use of other device characteristics to evaluate operability.

Insert 9 (Continued)

The Engineered Safety Features Actuation System and Reactor Trip System Nominal Trip Setpoints specified in the Licensing Requirements Manual (LRM) are the nominal values* at which the instrumentation is set for each functional unit. A Setpoint is considered to be consistent with the nominal value when the measured "as left" Setpoint is within the administratively controlled (±) calibration tolerance identified in plant procedures (which specifies the difference between the Allowable Value and Nominal Trip Setpoint). Additionally, a trip setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions provided that the ± calibration tolerance band remains the same and the allowable value is also adjusted accordingly in the conservative direction to meet the assumptions of the setpoint methodology. The conservative direction is established by the direction of the inequality applied to the Allowable Value.

The setpoint methodology, used to derive the Nominal Trip Setpoints, is based upon combining all of the uncertainties in the channels. Inherent in the determination of the Nominal Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels should be capable of operating within the allowances of these uncertainty magnitudes. Occasional drift in excess of the allowance may be determined to be acceptable based on the other device performance characteristics. Device drift in excess of the allowance that is more than occasional, may be indicative of more serious problems and would warrant further investigation.

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* With the exception of the Reactor Trip System Functional Unit number 17.B for the Turbine Stop Valve Position trip. The trip setpoint specified in the LRM for Functional Unit number 17.B is not a nominal value. The trip setpoint for this Functional Unit is adjusted to be consistent with the trip setpoint value specified in the LRM in lieu of adjusting the setpoint within an established calibration tolerance band.

REPLACE WITH INSERT "10"

BASES

~~3/4.3.1 AND 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION (Continued)~~

The surveillance requirements for the Manual Trip Function, Reactor Trip Breakers and Reactor Trip Bypass Breakers are provided to reduce the possibility of an Anticipated Transient Without Scram (ATWS) event by ensuring OPERABILITY of the diverse trip features (Reference: Generic Letter 85-09).

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

ESF response times which include sequential operation of the RWST and VCT valves are based on values assumed in the Non-LOCA safety analyses and are provided in Section 3 of the Licensing Requirements Manual. These analyses take credit for injection of borated water. Initial borated water is supplied by the BIT, however, injection of borated water from the RWST is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When sequential operation of the RWST and VCT valves is not included in the response times, the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times will assure that the assumptions used for the LOCA and Non-LOCA analyses with respect to operation of the VCT and RWST valves are valid.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

The Engineered Safety Feature Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates turbine trip, closes main feedwater valves on Tavg below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows safety injection block so that components can be reset or tripped. Reactor not tripped - prevents manual block of safety injection.

REPLACE WITH INSERT "10"

BASES

~~3/4.3.1 AND 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION (Continued)~~

P-11 Above the setpoint P-11 automatically reinstates safety injection actuation on low pressurizer pressure, automatically blocks steamline isolation on high steam pressure rate, enables safety injection and steamline isolation on low steamline pressure (with Loop Stop Valves Open), and enables auto actuation of the pressurizer PORVs.

Below the setpoint P-11 allows the manual block of safety injection actuation on low pressurizer pressure, allows manual block of safety injection and steamline isolation on low steamline pressure (with Loop Stop Valves Open) and enabling steamline isolation on high steam pressure rate, automatically disables auto actuation of the pressurizer PORVs unless the Reactor Vessel Over Pressure Protection System is in service.

P-12 Above the setpoint P-12 automatically reinstates an arming signal to the steam dump system. Below the setpoint P-12 blocks steam dump and allows manual bypass of the steam dump block to cooldown condenser dump valves.

Table 3.3-1 Action 2 has been modified by two notes. Note (4) allows placing the inoperable channel in the bypass condition for up to 4 hours while performing: a) routine surveillance testing of other channels, and b) setpoint adjustments of other channels when required to reduce the setpoint in accordance with other technical specifications. The 4 hour time limit is justified in accordance with WCAP-10271-P-A, Supplement 2, Revision 1, June 1990. Note (5) only requires SR 4.2.4 to be performed if a Power Range High Neutron Flux channel input to QPTR becomes inoperable. Failure of a component in the Power Range High Neutron Flux channel which renders the High Neutron Flux trip function inoperable may not affect the capability to monitor QPTR. As such, determining QPTR using the movable incore detectors once per 12 hours may not be necessary.

The following discussion pertains to Table 3.3-3, Functional Units 6.b and 6.c and the associated ACTION 34. The degraded voltage protection instrumentation system will automatically initiate the separation of the offsite power sources from the emergency buses. This action results in an automatic diesel generator start signal being generated as a direct result of the supply breakers opening between the normal and emergency buses. The failure of the degraded voltage protection system results in a loss of one of the automatic start signals for the diesel generator. Therefore, the ACTION statement requires the affected diesel generator to be declared inoperable if the required actions cannot be met within the specified time period.



(Proposed Wording)

REPLACE WITH INSERT "10"

BASES

~~3/4.3.1 AND 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION (Continued)~~

The instrumentation functions that receive input from neutron detectors are modified by a note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RATED THERMAL POWER. The power range neutron detector CHANNEL CALIBRATION is performed every 18 months but is not required for entry into MODE 2 or 1 on unit startup because the unit must be in at least MODE 1 to perform the test. The neutron detector CHANNEL CALIBRATION for the source range and intermediate range detectors consists of obtaining detector characteristics and performing an engineering evaluation of those characteristics. The intermediate range neutron detector CHANNEL CALIBRATION is performed every 18 months but is not required for entry into MODE 2 on unit startup because the unit must be in at least MODE 2 to perform the test. The source range neutron detector CHANNEL CALIBRATION is performed every 18 months but is not required for entry into MODE 2 or 3 on unit shutdown because the unit must be in at least MODE 3 to perform the test. The P-6 permissive neutron detector CHANNEL CALIBRATION is performed in conjunction with the intermediate range neutron detectors. The overtemperature $\Delta T$, P-8, P-9 and P-10 permissive neutron detector CHANNEL CALIBRATIONS are performed in conjunction with the power range neutron detectors.

Source Range Neutron Flux

DELETE

The limiting condition for operation (LCO) requirement for the source range neutron flux trip function ensures that protection is provided against an uncontrolled rod cluster control assembly (RCCA) bank rod withdrawal accident from a subcritical condition during startup with the reactor trip breakers (RTBs) closed. This trip function provides redundant protection to the Power Range Neutron Flux-Low Setpoint and Intermediate Range Neutron Flux trip functions (See UFSAR Section 14.1.1 ~~and Specification 2.2.1 Bases~~). In MODES 3, 4, and 5, with the RTBs closed, administrative controls also prevent the uncontrolled withdrawal of rods. The nuclear instrumentation system (NIS) source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. In Modes 3, 4, and 5, with the reactor trip breakers closed, the source range detectors provide an automatic trip function with a setpoint in the shutdown range and the intermediate range detectors provide an automatic trip function with a setpoint in the power range. Therefore, the functional capability at the specified trip setpoint is assumed to be available.



(Proposed Wording)

REPLACE WITH INSERT "10"

BASES

~~3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION (Continued)~~

The LCO requires two channels of source range neutron flux to be OPERABLE when the RTBs are closed. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip function. The LCO also requires one channel of the source range neutron flux to be OPERABLE in MODE 3, 4, or 5 with RTBs open. In this case, the source range function is to provide control room indication and the high flux at shutdown alarm. The outputs of the function to RTS logic are not required OPERABLE when the RTBs are open.

The source range neutron flux function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The function also provides visual neutron flux indication in the control room.

In MODE 2 when below the P-6 setpoint during a reactor startup, the source range neutron flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and Power Range Neutron Flux-Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range detectors are de-energized and not functional.

In MODE 3, 4, or 5 with the reactor shut down and with the control rod drive (CRD) system capable of rod withdrawal, the source range neutron flux trip function must also be OPERABLE. If the CRD system is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. However, their monitoring function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution.

Suitable detectors used in place of primary source range neutron flux monitors are recognized as alternate detectors. Alternate detectors may be used in place of primary source range neutron flux monitors as long as the required neutron flux indication, high flux at shutdown alarm, and source range high neutron flux trip functions are provided.



(Proposed Wording)

REPLACE WITH INSERT "10"

BASES

~~3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION (Continued)~~

ACTION 4

Item (a) applies to one inoperable source range neutron flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

Item (b) applies to one inoperable source range neutron flux trip channel when in MODE 3, 4, or 5, with the RTBs closed and the CRD system capable of rod withdrawal. With the unit in this condition, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to OPERABLE status. If the channel cannot be returned to an OPERABLE status, 1 additional hour is allowed to open the RTBs. Once the RTBs are open, rod withdrawal is not possible and the unit enters ACTION 5. The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour to open the RTBs, are justified in WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

Item (c) applies to two inoperable source range neutron flux trip channels when in MODE 2, below the P-6 setpoint, and performing a reactor startup, or in MODE 3, 4, or 5 with the RTBs closed and the CRD system capable of rod withdrawal. With the unit in this condition, below P-6, the NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, rod withdrawal is not possible and the unit enters ACTION 5.



(Proposed Wording)

REPLACE WITH INSERT "10"

BASES

~~3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION (Continued)~~

ACTION 5

This ACTION applies when the required number of OPERABLE source range neutron flux channels is not met in MODE 3, 4, or 5 with the RTBs open. With the unit in this condition, the NIS source range performs the monitoring function. With less than the required number of source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation. However, a note applicable to this ACTION allows plant cooldown as long as the shutdown margin is adequate to account for the positive reactivity addition resulting from the temperature change. This ensures the core is controlled and the shutdown margin requirements are satisfied for all applicable events. In addition to suspension of positive reactivity additions, the valve(s) that controls the addition of unborated water to the RCS must be closed within 1 hour. The isolation of unborated water sources will preclude a boron dilution accident.

Also, the shutdown margin (SDM) must be verified within 1 hour and once every 12 hours thereafter as per SR 4.1.1.1.1 or 4.1.1.2, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within 1 hour allows sufficient time to perform the calculations and determine that the SDM requirements are met. The SDM must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Item (a) precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour frequency is adequate. This does not include xenon decay which is accounted for in the shutdown margin surveillance. The completion times of within 1 hour and once per 12 hours are based on operating experience in performing the ACTIONS and the knowledge that unit conditions will change slowly.

SOURCE RANGE NEUTRON FLUX

SURVEILLANCE REQUIREMENTS (SR)

CHANNEL CHECK

The alternate source range detectors are modified by a note to indicate they are not subject to the source range detector surveillance requirements until they have been connected to the applicable circuits and are required to be OPERABLE. This complies with the testing requirements for components that are required to be OPERABLE.



(Proposed Wording)

REPLACE WITH INSERT "10"

BASES

~~3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION (Continued)~~

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK for a single channel involves a qualitative assessment of the channel indication to verify the channel is operating in the approximate range for the expected plant conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The frequency is based on operating experience that demonstrates channel failure is rare. Thus, performance of the CHANNEL CHECK ensures that undetected overt channel failure is limited to 12 hours. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

When the control rods are fully inserted and are not capable of withdrawal, inadvertent control rod withdrawal is not a concern and one source range detector can adequately monitor the core.

CHANNEL FUNCTIONAL TEST

The alternate source range detectors are modified by a note to indicate they are not subject to the source range detector surveillance requirements until they have been connected to the applicable circuits and are required to be OPERABLE. This complies with the testing requirements for components that are required to be OPERABLE.



(Proposed Wording)

REPLACE WITH INSERT "10"

BASES

~~3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION (Continued)~~

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure the entire channel will perform the intended function. Setpoints must be within the Allowable Values. The frequency of 92 days is justified for certain channels in WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

This surveillance is modified by a Note that specifies testing when below P-6 and is clarified to address the transition from MODE 2 to MODE 3. A transition into MODE 3 with the reactor trip breakers closed is often made for a short period of time during plant shutdown. During a normal shutdown, the reactor trip breakers are opened shortly after entering MODE 3. The transition time in MODE 3 from when the reactor trip breakers are closed to when they are opened is less than the time required to perform the CHANNEL FUNCTIONAL TEST prior to entering MODE 3. Therefore, an allowance to enter MODE 3 without first performing the source range CHANNEL FUNCTIONAL TEST is warranted.

When performing the CHANNEL FUNCTIONAL TEST for manual initiation functions, the injection of a simulated signal into the channel as close to the primary sensor as practicable is accomplished by manually operating the function's manual switch(es).

CHANNEL CALIBRATION

The alternate source range detectors are modified by a note to indicate they are not subject to the source range detector surveillance requirements until they have been connected to the applicable circuits and are required to be OPERABLE. This complies with the testing requirements for components that are required to be OPERABLE.

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. The CHANNEL CALIBRATION for the source range neutron detectors consists of obtaining the detector plateau and preamp discriminator curves, evaluating those curves, and establishing detector operating conditions as directed by the detector manufacturer. The 18 month frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage since performance at power is not possible. The protection and monitoring functions are also calibrated at an 18 month frequency as is normal for reactor protection instrument channels. Operating experience has shown these components usually pass the surveillance when performed on the 18 month frequency.



(Proposed Wording)

Insert 10

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

## 3/4.4.1.1, 2, 3 REACTOR COOLANT LOOPS

DELETE

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR limit during all normal operations and anticipated transients. In Modes 1 and 2, ~~with one reactor coolant loop not in operation, THERMAL POWER is restricted to less than or equal to 31 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above the design DNBR limit.~~ A loss of flow in two loops will cause a reactor trip if operating above P-7 DELETE ~~(11 percent of RATED THERMAL POWER)~~ while a loss of flow in one loop will cause a reactor trip if operating above P-8 ~~(31 percent of RATED THERMAL POWER)~~. DELETE

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, due to the initial conditions assumed in the analysis for the control rod bank withdrawal from a subcritical condition, two operating coolant loops are required to meet the DNB design basis for this Condition II event.

In MODES 4 and 5, a single reactor coolant loop or RHR subsystem provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to the enable temperature set forth in Specification 3.4.9.3 are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water level in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 25°F above each of the RCS cold leg temperatures.



(Proposed Wording)

6.9.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

ADD INSERT "11" →

3.1.3.5 Shutdown Rod Insertion Limits
3.1.3.6 Control Rod Insertion Limits
3.2.1 Axial Flux Difference-Constant Axial Offset Control
3.2.2 Heat Flux Hot Channel Factor-$F_Q(Z)$
3.2.3 Nuclear Enthalpy Rise Hot Channel Factor-$F^N_{\Delta H}$

ADD INSERT "12" →

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (Westinghouse Proprietary).

ADD INSERT "13" →

WCAP-10266-P-A Rev. 2/WCAP-11524-NP-A Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," Kabadi, J. N., March 1987; including Addendum 1-A "Power Shape Sensitivity Studies" 12/87 and Addendum 2-A "BASH Methodology Improvements and Reliability Enhancements" 5/88.

WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT." September 1974 (Westinghouse Proprietary).

T. M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 -- Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

Insert 11

2.1.1 Reactor Core Safety Limits

Insert 12

3.2.5 DNB Parameters

3.3.1.1 Reactor Trip System Instrumentation - Overtemperature and Overpower $\Delta T$ Setpoint Parameter Values

Insert 13

WCAP-8745-P-A, Design Bases for the Thermal Overtemperature $\Delta T$ and Thermal Overpower $\Delta T$ trip functions, September 1986.

ATTACHMENT A-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 158

The following is a list of the affected pages:

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DELETE



(Proposed Wording)

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(Proposed Wording)

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(Proposed Wording)

MAJOR CHANGES

1.35 MAJOR CHANGES to radioactive waste systems (liquid, gaseous and solid), as addressed in the PROCESS CONTROL PROGRAM, shall include the following:

1) MAJOR CHANGES in process equipment, components, structures, and effluent monitoring instrumentation from those described in the Final Safety Analysis Report (FSAR) or the Hazards Summary Report and evaluated in the staff's Safety Evaluation Report (SER) (e.g., deletion of evaporators and installation of demineralizers; use of fluidized bed calciner/incineration in place of cement solidification systems);

2) MAJOR CHANGES in the design of radwaste treatment systems (liquid, gaseous, and solid) that could significantly increase the quantities or activity of effluents released or volumes of solid waste stored or shipped offsite from those previously considered in the FSAR and SER (e.g., use of asphalt system in place of cement);

3) Changes in system design which may invalidate the accident analysis as described in the SER (e.g., changes in tank capacity that would alter the curies released); and

4) Changes in system design that could potentially result in a significant increase in occupational exposure of operating personnel (e.g., use of temporary equipment without adequate shielding provisions).

MEMBER(S) OF THE PUBLIC

1.36 MEMBER(S) OF THE PUBLIC means any individual except when that individual is receiving an occupational dose.

CORE OPERATING LIMITS REPORT

Cycle specific parameter

DELETE

1.37 The CORE OPERATING LIMITS REPORT (COLR) is the unit specific document that provides ~~core operating~~ limits for the current ~~operating~~ reload cycle. These cycle specific ~~core operating~~ limits shall be determined for each reload cycle in accordance with Specification 6.9.5. Plant operation within these ~~operating~~ limits is addressed in individual specifications.

DELETE

DELETE

parameter

DELETE



(Proposed Wording)

SECTION 2.0

SAFETY LIMITS

~~AND~~

~~LIMITING SAFETY SYSTEM SETTINGS~~

DELETE

(Proposed Wording)

## 2.0 SAFETY LIMITS ~~AND LIMITING SAFETY SYSTEM SETTINGS~~ DELETE

### 2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature ($T_{avg}$) shall not exceed the limits ~~shown~~ Specified in ~~Figure 2.1-1 for 3-loop operation.~~ REPLACE WITH INSERT "1"

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour. REPLACE WITH INSERT "2"

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4, and 5

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.



(Proposed Wording)

Insert 1

the COLR; and the following Safety Limits shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.17 for WRB-1 DNB correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained ≤ 4700°F.

Insert 2

If Safety Limit 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

REPLACE WITH INSERT "3" AND RELOCATE TO COLR

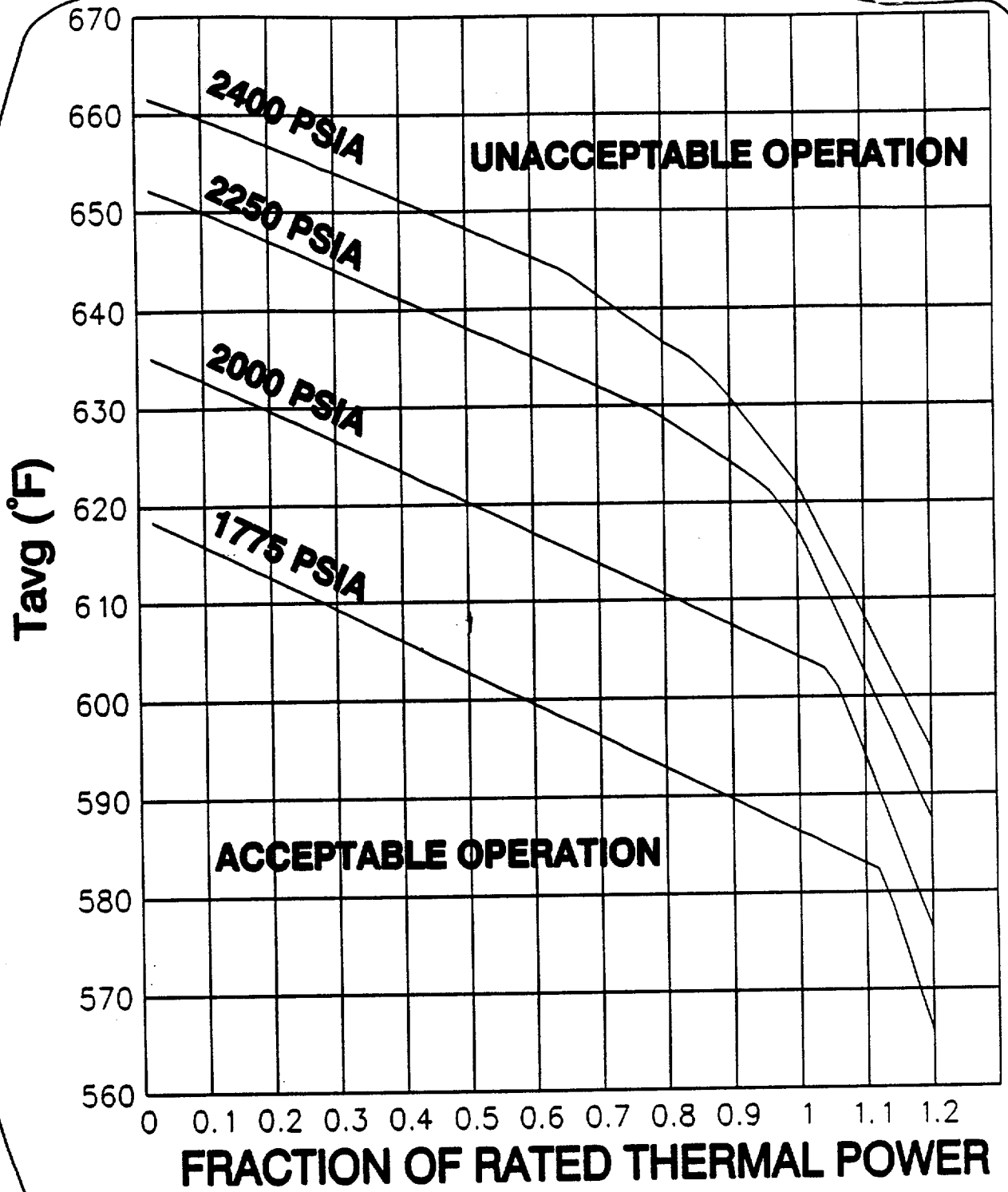


**FIGURE 2.1-1**

**REACTOR CORE SAFETY LIMIT**
**THREE LOOP OPERATION**



DELETE THIS PAGE

(Proposed Wording)

Insert 3

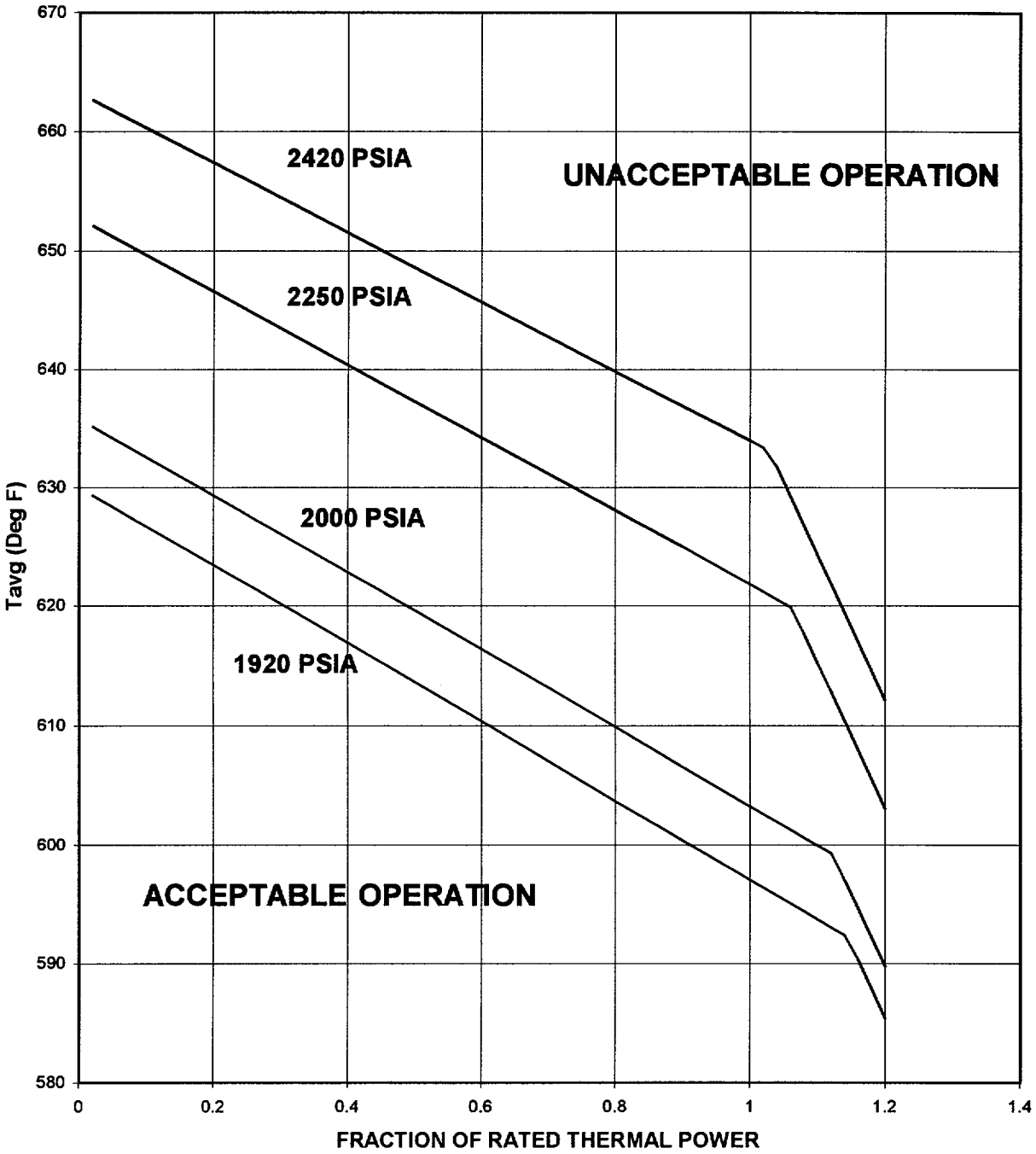


**Figure 2.1-1**
**REACTOR CORE SAFETY LIMIT**
**THREE LOOP OPERATION**

DELETE THIS PAGE

~~2.2 LIMITING SAFETY SYSTEM SETTINGS~~

~~REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS~~

~~2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be consistent with the Trip Setpoint values shown in Table 2.2-1.~~

~~APPLICABILITY: As shown for each channel in Table 3.3-1.~~

~~ACTION:~~

~~a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1 adjust the Setpoint consistent with the Trip Setpoint value.~~

~~b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Value column of Table 2.2-1, either:~~

~~1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel or~~

~~2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.~~

~~EQUATION 2.2-1~~ $Z + R + S \leq TA$

~~where:~~

~~Z = The value for column Z of Table 2.2-1 for the affected channel,~~

~~R = The "as measured" value (in percent span) of rack error for the affected channel,~~

~~S = Either the "as measured" value (in percent span) of the sensor error, or the value of Column S (Sensor Error) of Table 2.2-1 for the affected channel, and~~

~~TA = The value from Column TA (Total Allowance in % of span) of Table 2.2-1 for the affected channel.~~

~~NPF-73~~

RELOCATE TO LRM

Technical Specification Table Notation (A) on Table 3.3-1

Technical Specification Table Notation (B) on Table 3.3-1

DELETE THIS PAGE

MOVE TO TABLE 3.3-1

## TABLE 2.2-1

## REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

DELETE

| FUNCTIONAL UNIT | ~~ALLOWANCE (TA)~~ | ~~Z~~ | ~~S~~ | NOMINAL TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. Manual Reactor Trip | ~~N/A~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| 2. Power Range, Neutron Flux, | | | | | |
| a. High Setpoint | ~~7.5~~ | ~~4.56~~ | ~~0~~ | ≤ 109% of RTP* | ≤ ~~111.1~~ 109.5% of RTP* |
| b. Low Setpoint | ~~8.3~~ | ~~4.56~~ | ~~0~~ | ≤ 25% RTP* | ≤ ~~27.1~~ 25.5% of RTP* |
| 3. Power Range, Neutron Flux, High Positive Rate | ~~1.6~~ | ~~0.50~~ | ~~0~~ | ≤ 5% of RTP* with a time constant ≥ 2 seconds | ≤ ~~6.3~~ 5.5% of RTP* with a time constant ≥ 2 seconds |
| 4. Power Range, Neutron Flux High Negative Rate | ~~1.6~~ | ~~0.50~~ | ~~0~~ | ≤ 5% of RTP* with a time constant ≥ 2 seconds | ≤ ~~6.3~~ 5.5% of RTP* with a time constant ≥ 2 seconds |
| 5. Intermediate Range, Neutron Flux | ~~17.0~~ | ~~8.41~~ | ~~0~~ | ≤ 25% RTP* | ≤ ~~30.9~~ 27.9% of RTP* |
| 6. Source Range, Neutron Flux | ~~17.0~~ | ~~10.01~~ | ~~0~~ | ≤ $10^5$ cps | ≤ ~~1.4~~ 1.3 x $10^5$ cps |
| 7. Overtemperature ΔT | ~~7.0~~ | ~~5.10~~ | ~~See Note 5~~ | See ~~Note 1~~ | See ~~Note 2~~ Table Notation (A) |
| 8. Overpower ΔT | ~~4.9~~ | ~~1.71~~ | ~~1.49~~ | See ~~Note 3~~ | See ~~Note 4~~ Table Notation (B) |
| 9. Pressurizer Pressure-Low | ~~3.1~~ | ~~0.71~~ | ~~1.67~~ | ≥ 1945 psig** | ≥ ~~1935~~ 1941 psig** |
| 10. Pressurizer Pressure-High | ~~6.2~~ | ~~4.96~~ | ~~0.67~~ | ≤ 2375 psig | ≤ ~~2383~~ 2379 psig |
| 11. Pressurizer Water Level-High | ~~8.0~~ | ~~2.18~~ | ~~1.67~~ | ≤ 92% of instrument span | ≤ ~~93.0~~ 92.5% of instrument span |
| 12. Loss of Flow | ~~2.5~~ | ~~1.39~~ | ~~0.60~~ | ≥ 90% of indicated loop ~~design~~ flow** | ≥ ~~88.9~~ 89.6% of indicated loop ~~design~~ flow* |

DELETE

* = RATED THERMAL POWER

~~** Loop design flow = 87,200 gpm.~~ DELETE

** Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 seconds for lag. Channel calibration shall ensure that these time constants are adjusted to those values.



MOVE TO TABLE 3.3-1 AND ALSO RELOCATE TO LRM

(Proposed Wording)

DELETE THIS PAGE



TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | ~~ALLOWANCE (TA)~~ (DELETE) | ~~Z~~ | ~~S~~ | NOMINAL ★★★ TRIP SETPOINT | MOVE TO TABLE 3.3-1 ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 13. Steam Generator Water Level-Low-Low | ~~11.5~~ | ~~10.18~~ | ~~1.67~~ | ≥ ~~11.5~~ 16.5% of narrow range instrument span-each steam generator | ≥ ~~10~~ 16% of narrow range instrument span-each steam generator |
| 14. DELETED. | | | | | |
| 15. Undervoltage - Reactor Coolant Pumps | ~~27.7~~ | ~~1.39~~ | ~~0~~ | ≥ 75% of ~~BUS~~ nominal bus voltage-each bus | ≥ ~~72~~ 71.2% of ~~BUS~~ nominal bus voltage - each bus |
| 16. Underfrequency-Reactor Coolant Pumps | ~~10.0~~ | ~~0.20~~ | ~~0~~ | ≥ 57.5 Hz-each bus | ≥ ~~57.4~~ 57.45 Hz-each bus |
| 17. Turbine Trip | | | | | |
| a. Emergency Trip Header Low Pressure | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | ≥ 1000 psig | ≥ 958 psig |
| b. Turbine Stop Valve Closure | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | ≥ 1% open | ≥ 1% open |
| 18. Safety Injection Input from ESF | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. ADD | N.A. |
| 19. Reactor Coolant Pump Breaker Position Trip | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| 20. Reactor Trip Breakers | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |

ADD

★★★ with the exception of Functional Unit number 17.b.

RELOCATE TO LRM

DELETE THIS PAGE

RELOCATE TO LRM

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

MOVE TO TABLE 3.3-1

| FUNCTIONAL UNIT | ~~ALLOWANCE (TA)~~ | ~~Z~~ | ~~S~~ | NOMINAL TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 21. Automatic Trip ~~and Interlock~~ Logic | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| 22. Reactor Trip System Interlocks | | | | | |
| a. Intermediate Range Neutron Flux, P-6 | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | ≥1 x $10^{-10}$ amps | ≥6 x $10^{-11}$ amps |
| b. Power Range Neutron Flux, P-8 | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | ≤30% of RTP* | ≤32.1% of RTP* |
| c. Power Range Neutron Flux, P-9 | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | ≤49% of RTP* | ≤51% of RTP* |
| d. Power Range Neutron Flux, P-10 (Input to P-7) | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | 10% of RTP* | ~~≥7.9% and ≤12.1% of RTP*~~ ≥ 9.5 % RTP* on increasing power and ≤ 10.5 % RTP* on decreasing power |
| e. Turbine Impulse Chamber Pressure, P-13 (Input to P-7) | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | ≤10% of RTP* turbine impulse pressure equivalent | ≤12.1% of RTP* turbine impulse pressure equivalent |

DELETE

DELETE

★ = RATED THERMAL POWER

ADD

★ = RATED THERMAL POWER

ADD

The overtemperature ΔT Function Allowable Value shall not exceed the following nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel, 0.5% ΔT span for the $T_{avg}$ channel, 0.5% ΔT span for the Pressurizer Pressure channel and 0.5% ΔT span for the f(ΔI) channel.

DELETE THIS PAGE

~~TABLE 2.2-1 (Continued)~~

~~REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS~~

~~NOTATION~~

(A):

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)}\left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_o \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)}\left[T\left(\frac{1}{1+\tau_6 S}\right) - T'\right] + K_3 (P - P') - f_1(\Delta I)\right\}$$

Where:

$\Delta T$ = Measured ΔT (RCS), °F

$\frac{1+\tau S}{1+\tau_2 S}$ = (is) the function generated by the lead-lag compensator on measured ΔT

$\tau_1$, $\tau_2$ = (are the) time constants utilized in (the) lead-lag compensator for ΔT; $\tau_2 = 3$ s; ≤ 3 sec. → COLR

$T_1$ ≥ 8 s (sec.) → COLR (specified in the COLR.)

$\frac{1}{1+\tau_3 S}$ = (is the) lag compensator on measured ΔT

$\tau_3$ = (is the) time constants utilized in the lag compensator for ΔI; $\tau_3$ ≤ 0 s (sec.) → COLR (specified in the COLR.)

$\Delta T_o$ = (is the loop specific) indicated ΔT at RATED THERMAL POWER, °F

$K_1$ = 1.28; ≤ 1.311 → COLR (is specified in the COLR.)

$K_2$ = ≥ 0.0183/°F; → COLR

$\frac{1+\tau_4 S}{1+\tau_5 S}$ = (is) the function generated by the lead-lag compensator for $T_{avg}$ dynamic compensation;

$\tau_4$, $\tau_5$ = (are the) time constants utilized in lead-lag compensator for $T_{avg}$; $T_4$ ≥ 30 s, $T_5$ ≤ 4 s (sec.) → COLR (specified in the COLR.)

→ MOVE to Table 3.3-1 Table Notation (Proposed Wording)

DELETE THIS PAGE



## TABLE 2.2-1 (Continued)

## REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS NOTATION (Continued)

| | |
|---|---|
| T | = Average temperature, °F |
| $\frac{1}{1 + \tau_6 S}$ | = Lag compensator on measured $T_{avg}$ |
| $\tau_6$ | = Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6 \geq 0$ s |
| T' | = ≤ 576.2°F (Nominal) $T_{avg}$ at RATED THERMAL POWER |
| $K_3$ | = ≥ 0.00082 |
| P | = Pressurizer Pressure, psig |
| P' | = 2235 psig (Nominal RCS operating pressure) |
| S | = Laplace transform operator, $s^{-1}$ |

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -33% and +9%, $f_1(\Delta I) = 0$, where $q_t$ and $q_b$ are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -33%, the ΔT Trip Setpoint shall be automatically reduced by 2.52% of its value at RATED THERMAL POWER; and

(iii) For each percent that the magnitude $q_t - q_b$ exceeds +9%, the ΔT Trip Setpoint shall be automatically reduced by 1.75% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.6% of ΔT span.





MOVE TO TABLE 3.3-1 TABLE NOTATION (Proposed Wording)

The Overpower ΔT Function Allowable Value shall not exceed the following nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel and 0.5% ΔT span for the Tavg channel.

DELETE THIS PAGE



TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

(B):

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \frac{(1)}{(1+\tau_3 S)} \le \Delta T_o \{K_4 - K_5 \frac{\tau_7 S}{(1+\tau_7 S)} \frac{(1)}{(1+\tau_6 S)} T - K_6 [T \frac{(1)}{(1+\tau_6 S)} - T''] - f_2(\Delta I)\}$$

Where:

$\Delta T$ = Measured ΔT (RCS, °F)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = is the function generated by the lead-lag compensator on measured ΔT

$\tau_1, \tau_2$ = time constants utilized in the lead-lag compensator for ΔT (specified in the COLR) — $\tau_1 \ge 8$ s, $\tau_2 \le 3$ s → COLR

$\frac{1}{1+\tau_3 S}$ = Lag compensator on measured ΔT

$\tau_3$ = is the time constant utilized in the lag compensator for ΔT (specified in the COLR) — $\tau_3 \le 0$ s → COLR

$\Delta T_o$ = indicated ΔT at RATED THERMAL POWER (°F)

$K_4$ = 1.0781; ≤ 1.094 → COLR

$K_5$ = ≥ 0.02/°F for increasing average temperature and = 0 for decreasing average temperature; → COLR

$\frac{\tau_7 S}{1+\tau_7 S}$ = the function generated by the rate-lag compensator for $T_{avg}$ dynamic compensation;

$\tau_7$ = time constant utilized in the rate-lag compensator for $T_{avg}$ (specified in the COLR) — $\tau_7 \ge 10$ s → COLR

is the loop specific

is specified in the COLR

is the

MOVE TO TABLE 3.3-1
TABLE NOTATION

(proposed Wording)

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION (Continued)

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured $T_{avg}$

$\tau_6$ = Time constant utilized in the measured $T_{avg}$ lag compensator, $\tau_6 \geq 0$ s.

$K_6$ = 0.0012/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$; (1°F)

T = Average temperature, °F

T'' = Indicated $T_{avg}$ at RATED THERMAL POWER (Calibration temperature for ΔI instrumentation, ≤576.2°F);

S = Laplace transform operator, $s^{-1}$; and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.5% of ΔT span.

NOTE 5: The sensor error for temperature is 1.49% and 0.73% of span for pressure.

Is the function generated by the

Is the

Is specified in the COLR.

Is measured RCS

is

Is the

0.5%

For Tavg specified in the COLR.

≤ sec.

COLR

COLR

Specified in the COLR.

COLR

sec$^{-1}$.

Revised as now Note B on Page 2-9

MOVE TO TABLE 3.3-1 TABLE NOTATION



(Proposed Wording)

BASES

FOR

SAFETY LIMITS

~~AND~~

~~LIMITING SAFETY SYSTEM SETTINGS~~ ← DELETE

(Proposed Wording)

BASES

## 2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the WRB-1 correlation. The WRB-1 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

REPLACE WITH INSERT "4"

~~The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting fuel rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation in this application). The correlation DNBR limit is based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit (1.17 for the WRB-1 correlation).~~

REPLACE WITH INSERT "5"

~~Incorporating the peaking factor uncertainties in the correlation limit results in a DNBR design limit value of 1.21. This DNBR value must be met in plant safety analyses using nominal values of the input parameters that were included in the DNBR uncertainty evaluation.~~ In addition, margin has been maintained in the design by meeting a safety analysis DNBR limit of ~~1.33~~ 1.36 for both typical and thimble cells in performing safety analyses.

The ~~curve of Figure 2.1-1~~ figure provided in the COLR shows the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature 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.

The ~~curves are~~ figure is based on enthalpy hot channel factor limits provided in the ~~Core Operating Limits Report~~ (COLR).

DELETE

## Insert 4

The DNB thermal design criterion is that the probability of DNB not occurring on the most limiting rod is at least 95 percent at a 95 percent confidence level for any Condition I or II event.

## Insert 5

In meeting the DNB design criterion, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters and computer codes have been statistically combined with the DNB correlation uncertainties to determine the DNBR Design Limits which are 1.24 for typical and 1.23 for thimble cell.

REPLACE WITH INSERT "6"

BASES

2.1.1 REACTOR CORE, continued

~~These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f(ΔI) function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trip will reduce the setpoint to provide protection consistent with core safety limits.~~

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this safety limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel, pressurizer, and the RCS piping, valves and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig to demonstrate integrity prior to initial operation.

~~2.2.1~~ REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

~~The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The setpoint for a reactor trip system or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.~~

~~To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the reactor trip setpoints have been specified in Table 2.2-1. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is~~

REPLACE WITH INSERT "7"



MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2

(Proposed Wording)

Insert 6

The reactor core Safety Limits are established to preclude violation of the following fuel design criteria:

a. There must be at least a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core Safety Limits are used to define the various Reactor Protection System (RPS) functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower $\Delta T$ reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and $\Delta I$ that the reactor core Safety Limits will be satisfied during steady state operation, normal operational transients, and AOOs.

Insert 7

Technical specifications are required by 10 CFR 50.36 to contain Limiting Safety System Settings (LSSS) defined by the regulation as "...settings for automatic protective devices...so chosen that automatic protective action will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytic Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytic Limit therefore ensures that the SL is not exceeded.

The Allowable Values (Nominal Trip Setpoints ± the calibration tolerance) specified in Table 3.3.1 are considered the LSSS as identified in 10 CFR 50.36 and have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents.

The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides Reactor Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System.

The Reactor Trip System initiates a turbine trip signal whenever reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

The difference between T' (Overtemperature $\Delta T$) or T" (Overpower $\Delta T$) and the loop specific, indicated, full power $T_{avg}$ shall be less than or equal to the $T_{avg}$ allowances for such differences in the uncertainty calculations for these functions. In addition, T' and T" shall be less than or equal to the full power $T_{avg}$ modeled in the safety analyses as an initial condition assumption; i.e., the numerical value specified in the COLR. In the event that the difference between a T' or T" set to the numerical value specified in the COLR and a loop specific, indicated, full power $T_{avg}$ is greater than the $T_{avg}$ allowances for such differences in the uncertainty calculations, T' or T" shall be reduced until the difference allowances in the uncertainty calculations are satisfied; i.e., T' or T" are set to a loop specific, full power value less than the numerical value specified in the COLR. These reductions in the values of T' and T" are consistent with the recommendations of Westinghouse Technical Bulletin ESBU-TB-96-07-RO, "Temperature Related Functions," 11/5/96.

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BASES

~~2.2.1~~ REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS, continued

acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its trip setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified trip setpoint. S or Sensor Drift is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

For the Overtemperature $\Delta T$, Overpower $\Delta T$, and Low Pressurizer Pressure trips, Table 2.2-1 specifies several time constants ($\tau$'s) in terms of equalities, such as $\tau$=x seconds. These time constants represent nominal values and are periodically adjusted to within their specified calibration accuracy. With the time constants within the specified accuracy the accident analysis produces acceptable results. For time lags in the Tavg and Delta T circuits, these values are set as close to "zero" as practical to provide minimum filter action.

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(Proposed Wording)

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BASES

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range, Neutron Flux

The Power Range, Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry. The low setpoint provides redundant protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed when P-10 is active (two of the four power range channels indicate a power level of above ~~approximately 10 percent of RATED THERMAL POWER~~) and is automatically reinstated when P-10 becomes inactive (three of the four channels indicate a power level below ~~approximately 10 percent of RATED THERMAL POWER~~). DELETE the P-10 setpoint

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of rod ejection events from any power level. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from partial power.

The Power Range Negative Rate trip provides protection to ensure that the minimum DNBR is maintained above the design DNBR limit for control rod drop accidents. At high power a multiple rod drop accident could cause local flux peaking which, when in conjunction with nuclear power being maintained equivalent to turbine power by action of the automatic rod control system, could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than the design DNBR limit.

Intermediate and Source Range, Nuclear Flux

The Intermediate and Source Range, Nuclear Flux trips provide reactor core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The

MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2

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BASES

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Source Range Channels will initiate a reactor trip at ~~about $10^{+5}$~~ the trip setpoint ~~counts per second~~ unless manually blocked when P-6 becomes active. The intermediate range channels will initiate a reactor trip at a current level proportional to ~~approximately 25 percent of RATED THERMAL POWER~~ unless manually blocked when P-10 becomes active. Although no explicit credit was taken for operation of the Source Range Channels in the accident analyses, operability requirements in the Technical Specifications will ensure that the Source Range Channels are available to mitigate the consequences of an inadvertent control bank withdrawal in MODES 3, 4 and 5.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit, thermowell, and RTD response time delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low pressure reactor trips. This setpoint includes corrections for changes in density and heat capacity of water with temperature and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown ~~on Figure 2.1-3~~ in the COLR. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notation in Table ~~2.2-1~~ 3.3-1.

Overpower ΔT

The Overpower ΔT reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux Trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for transport, thermowell, and RTD response time delays from the core to RTD output indication. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam line breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Release."

MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2

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MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2

BASES

Pressurizer Pressure

The Pressurizer High and Low Pressure trips are provided to limit the pressure range in which reactor operation is permitted. The High Pressure trip is backed up by the pressurizer code safety valves for RCS overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The Low Pressure trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Pressure trip is automatically blocked by P-7 ~~(a power level of approximately 10% of RATED THERMAL POWER or turbine impulse chamber pressure at approximately 10% of full power equivalent)~~; and on increasing power, automatically reinstated by P-7. DELETE

Pressurizer Water Level

The Pressurizer High Water Level trip ensures protection against Reactor Coolant System overpressurization by limiting the water level to a volume sufficient to retain a steam bubble and prevent water relief through the pressurizer safety valves. On decreasing power, the pressurizer high water level trip is automatically blocked by P-7 DELETE ~~(a power level of approximately 10 percent of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10 percent of full power equivalent)~~; and on increasing power, automatically reinstated by P-7. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above P-7 ~~10 percent of RATED THERMAL POWER~~, an automatic the trip setpoint reactor trip will occur if the flow in any two loops drop below ~~90 percent of nominal full loop flow~~. DELETE Above ~~30 percent~~ P-8 ~~of RATED THERMAL POWER~~, automatic reactor trip will occur if the flow in any single loop an drops below ~~90 percent of nominal full loop flow~~. the trip setpoint DELETE

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The

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BASES

specified setpoint provides allowance that ~~here~~ there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The ~~specified~~ trip setpoints assure a reactor trip signal is generated before the low flow trip setpoint is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip setpoint is reached shall not exceed 0.6 seconds.

On decreasing power, the Undervoltage and Underfrequency Reactor Coolant Pump bus trips are automatically blocked by P-7 ~~(a power level of approximately 10 percent of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10 percent of full power equivalent)~~; and on increasing power, reinstated automatically by P-7.

DELETE

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-9. Each of the turbine trips ~~provide~~ provides turbine protection and ~~reduce~~ reduces the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in TABLE 3.3-3.

MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2

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BASES

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB resulting from the opening of two or more pump breakers above P-7. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip setpoint is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 Above the setpoint P-6 allows the manual block of the Source Range reactor trip and de-energizing of the high voltage to the detectors. Below the setpoint source range level trips are automatically reactivitated and high voltage restored.

P-7 Above the setpoint P-7 automatically enables reactor trips on low flow or coolant pump breaker open in more than one primary coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure and pressurizer high level. Below the setpoint the above listed trips are automatically blocked.

P-8 Above the setpoint P-8 automatically enables reactor trip on low flow in one or more primary coolant loops. Below the setpoint P-8 automatically blocks the above listed trip.

P-9 Above the setpoint P-9 automatically enables a reactor trip on turbine trip. Below the setpoint P-9 automatically blocks a reactor trip on turbine trip.

P-10 Above the setpoint P-10 allows the manual block of the Intermediate Range reactor trip and the low setpoint Power Range reactor trip; and automatically blocks the Source Range reactor trip and de-energizes the Source Range high voltage power. Below the setpoint the Intermediate Range reactor trip is automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

MOVE TO BASES SECTION 3/4.3.1 and 3/4.3.2

POWER DISTRIBUTION LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits ~~shown on Table 3.2-1(1)~~: specified below:

a. Reactor Coolant System $T_{avg}$ is less than or equal to the limit specified in the COLR,

b. Pressurizer Pressure is greater than or equal to the limit(1) specified in the COLR, and

c. Reactor Coolant System Total Flow Rate ≥ 261,600(3) and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1(2).

ACTION:

With ~~any of~~ one or more of the above DNB related parameters ~~exceeding its~~ not within limit(s), restore the parameter DNB related to within ~~its~~ DELETE limit within 2 hours or reduce THERMAL POWER to less than 5 percent of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the above DNB related parameters ~~of Table 3.2-1~~ DELETE shall be verified to be ~~indicating~~ DELETE within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit(s) by measurement at least once per 18 months.

REPLACE WITH FOOTNOTE (1) ON PAGE 3/4 2-12

~~(1) The values presented in Table 3.2-1 correspond to analytical limits used in the safety analyses.~~

(2) The provisions of Specification 4.0.4 are not applicable for Reactor Coolant System total flow rate to allow a calorimetric flow measurement and the calibration of the Reactor Coolant System total flow rate indicators.

(3) Analytical limit used in the safety analysis. ADD



(Proposed Wording)

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~~TABLE 3.2-1~~

DNB PARAMETERS

(Technical Specification 3.2.5)

RELOCATE TO COLR

| PARAMETER | Indicated Value | ~~3 Loops In Operation~~ |
|---|---|---|
| Reactor Coolant System $T_{avg}$ | 579.9 | ≤ ~~580.2~~ °F (1) |
| Pressurizer Pressure | 2214 | ≥ ~~2220~~ psia (2) |
| Reactor Coolant System Total Flow Rate | 267,200 | ≥ ~~261,600~~ gpm (3) |

(1) Limit 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% RATED THERMAL POWER.

MOVE TO PAGE 3/4 2-11

(1) The Reactor Coolant System (RCS) $T_{avg}$ value includes allowances for rod control operation and verification via control board indication.

(2) The pressurizer pressure value includes allowances for pressurizer pressure control operation and verification via control board indication.

(3) The RCS total flow rate includes allowances 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 plant process computer.

ADD

RELOCATE TO COLR



(Proposed Wording)

## 3/4.3 INSTRUMENTATION

### 3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION: As shown in Table 3.3-1. (2) ← ADD

#### SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor trip system instrumentation channel and interlock and automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements[1] during the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the interlocks shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by interlock operation. The total interlock function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by interlock operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

---

(1) For the automatic trip logic, the surveillance requirements shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output including, as a minimum, a continuity check of output devices.

(2) Separate ACTION statement entry is allowed for each Function.

ADD



(Proposed Wording)

ADD ALLOWABLE VALUE COLUMN FROM TABLE 2.2-1

## TABLE 3.3-1

## REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 1. Manual Reactor Trip | 2 | 1 | 2 | 1, 2, 3(3), 4(3) and 5(3) | 12 |
| 2. Power Range, Neutron Flux | | | | | |
| a. High Setpoint | 4 | 2 | 3 | 1, 2 | 2 |
| b. Low Setpoint | 4 | 2 | 3 | 1(1), 2 | 2 |
| 3. Power Range, Neutron Flux High Positive Rate | 4 | 2 | 3 | 1, 2 | 2 |
| 4. Power Range, Neutron Flux High Negative Rate | 4 | 2 | 3 | 1, 2 | 2 |
| 5. Intermediate Range, Neutron Flux | 2 | 1 | 2 | 1(1), 2, 3(3), 4(3) and 5(3) | 3 |
| 6. Source Range(8), Neutron Flux | | | | | |
| a. With Rod Withdrawal Capability | 2 | 1 | 2 | 2(2), 3(3), 4(3) and 5(3) | 4 |
| b. With All Rods Fully Inserted and Without Rod Withdrawal Capability | 2 | 0 | 1 | 3, 4, and 5 | 5 |
| 7. Overtemperature ΔT | 3 | 2 | 2 | 1, 2 | 7 |
| 8. Overpower ΔT | 3 | 2 | 2 | 1, 2 | 7 |
| 9. Pressurizer Pressure-Low (Above P-7) | 3 | 2 | 2 | 1, 2 | 7 |
| 10. Pressurizer Pressure-High | 3 | 2 | 2 | 1, 2 | 7 |

N.A.

(9)

(8) Alternate detectors may only be used for monitoring purposes Without Rod Withdrawal Capability until detector functions are modified to permit equivalent alarm and trip functions.

ADD ALLOWABLE VALUE COLUMN FROM TABLE 2.2-1

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 11. Pressurizer Water Level-High (Above P-7) | 3 | 2 | 2 | 1, 2 | 7 |
| 12. Loss of Flow - Single Loop (Above P-8) | 3/loop | 2/loop in any operating loop | 2/loop in each operating loop | 1 | 7 |
| 13. Loss of Flow - Two Loop (Above P-7 and below P-8) | 3/loop | 2/loop in two operating loops | 2/loop in each operating loop | 1 | 7 |
| 14. Steam Generator Water Level-Low-Low | 3/loop | 2/loop | 2/loop | 1, 2 | 7 |
| 15. DELETED. | | | | | |
| 16. Undervoltage-Reactor Coolant Pumps (Above P-7) | 3-1/bus | 2 | 2 | 1 | 7 |
| 17. Underfrequency-Reactor Coolant Pumps (Above P-7) | 3-1/bus | 2 | 2 | 1 | 7 |
| 18. Turbine Trip (Above P-9) | | | | | |
| a. Emergency Trip Header Low Pressure | 3 | 2 | 2 | 1 | 7 |
| b. Turbine Stop Valve Closure | 4 | 4 | 4 | 1 | 8 |



(Proposed wording)

ADD ALLOWABLE VALUE COLUMN FROM TABLE 2.2-1

TABLE 3.3-1 (Continued

REACTOR TRIP SYSTEM INSTRUMENTATION

| | FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 19. | Safety Injection Input from ESF | 2 | 1 | 2 | 1, 2 | 1 |
| 20. | Reactor Coolant Pump Breaker Position Trip (Above P-7) | 1/breaker | 2 | 1/breaker per operating loop | 1 | 11 |
| 21. | Reactor Trip Breakers | 2 | 1 | 2 | 1, 2 | 40 |
| | | 2 | 1 | 2 | 3(3), 4(3), 5(3) | 39 |
| 22. | Automatic Trip Logic | 2 | 1 | 2 | 1, 2 | 1 |
| | | 2 | 1 | 2 | 3(3), 4(3), 5(3) | 39 |
| 23. | Reactor Trip System Interlocks | | | | | |
| | a. Intermediate Range Neutron Flux, P-6 | 2 | 1 | 2 | 2 | 44 |
| | b. Power Range Neutron Flux, P-8 | 4 | 2 | 3 | 1 | 44 |
| | c. Power Range Neutron Flux, P-9 | 4 | 2 | 3 | 1 | 44 |
| | d. Power Range Neutron Flux, P-10 | 4 | 2 | 3 | 1, 2 | 44 |
| | e. Turbine Impulse Chamber Pressure, P-13 | 2 | 1 | 2 | 1 | 44 |



(Proposed Wording)

TABLE 3.3-1 (Continued)

TABLE NOTATION

(1) Trip function may be manually bypassed in this MODE above P-10.

(2) Trip function may be manually bypassed in this MODE above P-6.

(3) With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1, provided the other channel is OPERABLE.

ACTION 2 - With one power range neutron flux channel inoperable,[4] perform one of the following, as applicable:

a. Power Range High Neutron Flux Channel

1. Place the inoperable channel in trip within 6 hours and reduce THERMAL POWER to less than or equal to 75 percent RATED THERMAL POWER within the next 6 hours and perform SR 4.2.4,[5] or

ADD TABLE NOTATION (A) FROM TABLE 2.2-1

ADD TABLE NOTATION (B) FROM TABLE 2.2-1

2. Place the inoperable channel in trip within 6 hours and perform SR 4.2.4,[5] or

3. Be in MODE 3 within 12 hours.

b. All other channels

1. Place the inoperable channel in trip within 6 hours, or

2. Be in MODE 3 within 12 hours.

(9) In this condition, source range Function does not provide reactor trip but does provide indication.

(4) The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels.

(5) Only required to be performed when the power range high neutron flux channel input to QPTR is inoperable.

## 3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

### LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE ~~with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4~~. DELETE

**APPLICABILITY:** As shown in Table 3.3-3.

**ACTION:** DELETE

~~a. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4 adjust the Setpoint consistent with the Trip Setpoint value.~~

~~b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-4, either:~~

~~1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or~~

~~2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.~~

~~EQUATION 2.2-1~~ $Z + R + S \leq TA$

~~where:~~

~~Z = The value for Column Z of Table 3.3-4 for the affected channel,~~

~~R = The "as measured" value (in percent span) of rack error for the affected channel,~~

~~S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Drift) of Table 3.3-4 for the affected channel, and~~

~~TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.~~

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3. (2)



(2) Separate ACTION statement entry is allowed for each Function.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

| | FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 1. | SAFETY INJECTION AND FEEDWATER ISOLATION | | | | | |
| | a. Manual Initiation | 2 | 1 | 2 | 1, 2, 3, 4 | 18 |
| | b. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 13, 36 |
| | c. Containment Pressure-High | 3 | 2 | 2 | 1, 2, 3 | 14 |
| | d. Pressurizer Pressure-Low | 3 | 2 | 2 | 1, 2, 3(1) | 14 |
| | e. Steamline Pressure-Low | 3/loop | 2/loop any loop | 2/operating loop | 1, 2, 3(1) | 14 |
| 1.1. | SAFETY INJECTION-TRANSFER FROM INJECTION TO THE RECIRCULATION MODE | | | | | |
| | a. Automatic Actuation Logic Coincident with Safety Injection Signal | 2 | 1 | 2 | 1, 2, 3, 4 | 18 |
| | b. Refueling Water Storage Tank Level-Extreme Low | 4 | 2 | 3 | 1, 2, 3, 4 | 16 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 2. CONTAINMENT SPRAY | | | | | |
| a. Manual Initiation | 2 sets (2 switches/set) | 1 set | 2 sets | 1, 2, 3, 4 | 18 |
| b. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 13 |
| c. Containment Pressure--High-High | 4 | 2 | 3 | 1, 2, 3 | 16 |
| 3. CONTAINMENT ISOLATION | | | | | |
| a. Phase "A" Isolation | | | | | |
| 1) Manual Initiation | 2 | 1 | 2 | 1, 2, 3, 4 | 18 |
| 2) Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 13 |
| 3) Safety Injection | See Item 1. above for all Safety Injection initiating functions and requirements. | | | | |
| b. Phase "B" Isolation | | | | | |
| 1) Manual Intiation | 2 sets (2 switches/set) | 1 set | 2 sets | 1, 2, 3, 4 | 18 |
| 2) Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3, 4 | 13 |
| 3) Containment Pressure--High-High | 4 | 2 | 3 | 1, 2, 3, 4 | 16 |

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 4. STEAM LINE ISOLATION | | | | | |
| a. Manual Initiation | | | | | |
| 1. Individual | 1/steam line | 1/steam line | 1/operating steam line | 1, 2, 3 | 41 |
| 2. System | 2 sets (2 switches/set) | 1 set | 2 sets | 1, 2, 3 | 18 |
| b. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3 | 13 |
| c. Containment Pressure Intermediate-High-High | 3 | 2 | 2 | 1, 2, 3 | 14 |
| d. Steamline Pressure-Low | 3/loop | 2/loop any loop | 2/operating loop | 1, 2, 3(1) | 14 |
| e. Steamline Pressure Rate--High Negative | 3/loop | 2/loop any loop | 2/operating loop | 3(2) | 14 |
| 5. TURBINE TRIP & FEEDWATER ISOLATION | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2 | 42 |
| b. Steam Generator Water Level -- High-High, P-14 | 3/loop | 2/loop in any operating loop | 2/loop in each operating loop | 1, 2, 3 | 14 |



(Proposed Wording)

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 5. TURBINE TRIP & FEEDWATER ISOLATION (Continued) | | | | | |
| c. Safety Injection | See Item 1 above for all Safety Injection initiating functions and requirements. | | | | |
| 6. LOSS OF POWER | | | | | |
| a. 4.16kv Emergency Bus | | | | | |
| 1. Undervoltage (Trip Feed) | 2/4.16kv Bus | 2/4.16kv Bus | 2/4.16kv Bus | 1, 2, 3, 4 | 33 |
| 2. Undervoltage (Start Diesel) | 1/4.16kv Bus | 1/4.16kv Bus | 1/4kv Bus | 1, 2, 3, 4 | 33 |
| b. 4.16kv Emergency Bus (Degraded Voltage) | 2/4.16kv Bus | 2/Bus | 2/Bus | 1, 2, 3, 4 | 34 |
| c. 480 Volt Emergency Bus (Degraded Voltage) | 2/480v Bus | 2/Bus | 2/Bus | 1, 2, 3, 4 | 34 |
| 7. AUXILIARY FEEDWATER(3) | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | 2 | 1 | 2 | 1, 2, 3 | 42 |

(3) Manual initiation is included in Specification 3.7.1.2.



(Proposed Wording)

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

ADD ALLOWABLE VALUE COLUMN FROM TABLE 3.3-4

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 7. AUXILIARY FEEDWATER (Continued) | | | | | |
| b. Steam Gen. Water Level--Low-Low | | | | | |
| 1. Start Turbine Driven Pump | 3/stm. gen. | 2/stm. gen. any stm. gen. | 2/stm. gen. | 1, 2, 3 | 14 |
| 2. Start Motor Driven Pumps | 3/stm. gen. | 2/stm. gen. any 2 stm. gen. | 2/stm. gen. | 1, 2, 3 | 14 |
| c. Undervoltage-RCP (Start Turbine Driven Pump) | (3)-1/bus | 2 | 2 | 1, 2 | 14 |
| d. Safety Injection (Start All Auxiliary Feedwater Pumps) | See 1 above (all SI initiating functions and requirements) | | | | |
| e. Trip of Main Feedwater Pumps (Start Motor Driven Pumps) | 1/pump | 2 | 2 | 1, 2, 3 | 18 |
| 8. ENGINEERED SAFETY FEATURE INTERLOCKS | | | | | |
| a. Reactor Trip, P-4 | 2 | 1 | 2 | 1, 2, 3 | 45 |
| b. Pressurizer Pressure, P-11 | 3 | 2 | 2 | 1, 2, 3 | 38 |
| c. Low-Low $T_{avg}$, P-12 | 3 | 2 | 2 | 1, 2, 3 | 38 |

TABLE 3.3-3 (Continued)

ACTION STATEMENTS

ACTION 42 - With the number of OPERABLE Channels one less than the minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.1 provided the other Channel is OPERABLE.

ACTION 45 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

(Next page is 3/4 3-33)



(Proposed Wording)

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TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | ~~TOTAL ALLOWANCE (TA)~~ | ~~Z~~ | ~~SENSOR DRIFT (S)~~ | TRIP SETPOINT (NOMINAL) | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 1. SAFETY INJECTION AND FEEDWATER ISOLATION | | | | | |
| a. Manual Initiation | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| b. Automatic Actuation Logic and Actuation Relays | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| c. Containment Pressure - High | ~~2.6~~ | ~~0.71~~ | ~~1.67~~ | ≤ 1.5 psig | ≤ ~~2.0~~ 1.8 psig |
| d. Pressurizer Pressure - Low | ~~12.5~~ | ~~10.71~~ | ~~1.67~~ | ≥ ~~1845~~ 1856 psig | ≥ ~~1840~~ 1852 psig |
| e. Steamline Pressure - Low | ~~14.2~~ | ~~1.21~~ | ~~1.67~~ | ≥ 500 psig* | ≥ ~~479~~ 494 psig* |
| 1.1 SAFETY INJECTION TRANSFER FROM INJECTION TO THE RECIRCULATION MODE | | | | | |
| a. Automatic Actuation Logic, Coincident with Safety Injection Signal | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| b. Refueling Water Storage Tank Level - Extreme Low | ~~2.8~~ | ~~0.64~~ | ~~1.30~~ | ≥ ~~37~~ 38 feet ~~6~~ 0 inches | ≥ 37 feet ~~4~~ 11 inches |
| 2. CONTAINMENT SPRAY | | | | | |
| a. Manual Initiation | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |

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*Time constants utilized in the lead-lag controllers for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| FUNCTIONAL UNIT | ~~TOTAL ALLOWANCE (TA)~~ | ~~Z~~ | ~~SENSOR DRIFT (S)~~ | TRIP SETPOINT NOMINAL | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 2. CONTAINMENT SPRAY (Continued) | | | | | |
| b. Automatic Actuation Logic and Actuation Relays | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| c. Containment Pressure -- High-High | ~~2.9~~ | ~~0.71~~ | ~~1.67~~ | ≤ 8.0 psig | ≤ ~~8.7~~ 8.3 psig |
| 3. CONTAINMENT ISOLATION | | | | | |
| a. Phase "A" Isolation | | | | | |
| 1. Manual Initiation | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| 2. Automatic Actuation Logic and Actuation Relays | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| 3. Safety Injection | See Functional Unit 1. above for all Safety Injection Trip Setpoints ~~and Allowable Values~~. | | | | |
| b. Phase "B" Isolation | | | | | |
| 1. Manual Initiation | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| 2. Automatic Actuation Logic and Actuation Relays | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| 3. Containment Pressure-- High-High | ~~2.9~~ | ~~0.71~~ | ~~1.67~~ | ≤ 8.0 psig | ≤ ~~8.7~~ 8.3 psig |

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~~3/4 3-25~~ (Proposed wording)

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

MOVE TO TABLE 3.3-3

| FUNCTIONAL UNIT | ~~TOTAL ALLOWANCE (TA)~~ | ~~Z~~ | ~~SENSOR DRIFT (S)~~ | TRIP SETPOINT (NOMINAL) | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 4. STEAM LINE ISOLATION | | | | | |
| a. Manual Initiation | | | | | |
| 1. Individual | ~~N.A~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| 2. System | ~~N.A~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| b. Automatic Actuation Logic and Actuation Relays | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| c. Containment Pressure--Intermediate-High-High | ~~2.9~~ | ~~0.71~~ | ~~1.67~~ | ≤ 3.0 psig | ≤ ~~3.7~~ 3.3 psig |
| d. Steam Line Pressure - Low | ~~14.2~~ | ~~1.21~~ | ~~1.67~~ | ≥ 500 psig* | ≥ ~~479~~ 494 psig* |
| e. Steamline Pressure Rate - High Negative | ~~3.0~~ | ~~0.50~~ | ~~0~~ | ≤ 100 psi with a time constant ≥ 50 seconds | ≤ ~~121~~ 103.6 psi with a time constant ≥ 50 seconds |
| 5. TURBINE TRIP AND FEEDWATER ISOLATION | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| b. Steam Generator Water Level -- High-High, P-14 | ~~5.0~~ | ~~2.18~~ | ~~1.67~~ | ≤ ~~75~~ 80.6% of narrow range instrument span | ≤ ~~76.9~~ 81.1% of narrow range instrument span |
| c. Safety Injection | See Functional Unit 1. above for all Safety Injection Trip Setpoints ~~and Allowable Values~~. | | | | |

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*Time constants utilized in the lead-lag controllers for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

MOVE TO TABLE 3.3-3

| FUNCTIONAL UNIT | ~~TOTAL ALLOWANCE (TA)~~ | ~~Z~~ | ~~SENSOR DRIFT (S)~~ | TRIP SETPOINT (NOMINAL) | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 6. LOSS OF POWER | | | | | |
| a. 4.16 kV Emergency Bus<br>1. Undervoltage (Trip Feed) | ~~15.0~~ | ~~1.39~~ | ~~0.0~~ | ≥ 75% of nominal Bus Voltage with a 1 ± 0.1 second time delay | ≥ ~~73~~ 71.2% of nominal Bus Voltage with a 1 ± 0.1 second time delay |
| 2. Undervoltage (Start Diesel) | ~~15.0~~ | ~~1.39~~ | ~~0.0~~ | ≥ 75% of nominal Bus Voltage, 20 cycles ± 2 cycles | ≥ ~~73~~ 71.2% of nominal Bus Voltage, 20 cycles ± 2 cycles |
| b. 4.16 kV Emergency Bus (Degraded Voltage) | ~~15.0~~ | ~~1.39~~ | ~~0.0~~ | ≥ ~~90~~ 93.4% of nominal Bus Voltage with a 90 ± 5 second time delay | ≥ ~~88~~ 93.1% of nominal Bus Voltage with a 90 ± 5 second time delay |
| c. 480 Volt Emergency Bus (Degraded Voltage) | ~~15.0~~ | ~~1.39~~ | ~~0.0~~ | ≥ ~~90~~ 93.4% of nominal Bus Voltage with a 90 ± 5 second time delay | ≥ ~~88~~ 93.1% of nominal Bus Voltage with a 90 ± 5 second time delay |
| 7. AUXILIARY FEEDWATER* | | | | | |
| a. Automatic Actuation Logic and Actuation Relays | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |

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*Manual initiation is included in Specification 3.7.1.2

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

MOVE TO TABLE 3.3-3

| FUNCTIONAL UNIT | ~~TOTAL ALLOWANCE (TA)~~ | ~~Z~~ | ~~SENSOR DRIFT (S)~~ | TRIP SETPOINT (NOMINAL) | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 7. AUXILIARY FEEDWATER (Continued) | | | | | |
| b. Steam Generator Water Level--Low-Low | | | | | |
| 1. Start Turbine Driven Pump | ~~11.5~~ | ~~10.18~~ | ~~1.67~~ | ≥ ~~11.5~~ 16.5% of narrow range instrument span | ≥ ~~10.7~~ 16% of narrow range instrument span |
| 2. Start Motor Driven Pumps | ~~11.5~~ | ~~10.18~~ | ~~1.67~~ | ≥ ~~11.5~~ 16.5% of narrow range instrument span | ≥ ~~10.7~~ 16% of narrow range instrument span |
| c. Undervoltage - RCP (Start Turbine Driven Pump) | ~~27.7~~ | ~~1.39~~ | ~~0.0~~ | ≥ ~~75~~% of nominal bus voltage | ≥ ~~73~~ 71.2% of nominal bus voltage |
| d. Safety Injection (Start All Auxiliary Feedwater Pumps) | See Item 1. above for all Safety Injection Trip Setpoints ~~and Allowable Values~~. | | | | |
| e. Trip of Main Feedwater Pumps (Start Motor-Driven Pumps) | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

MOVE TO TABLE 3.3-3

| FUNCTIONAL UNIT | ~~TOTAL ALLOWANCE (TA)~~ | ~~Z~~ | ~~SENSOR DRIFT (S)~~ | TRIP SETPOINT NOMINAL | ALLOWABLE VALUE |
|---|---|---|---|---|---|
| 8. ENGINEERED SAFETY FEATURE INTERLOCKS | | | | | |
| a. Reactor Trip, P-4 | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | N.A. | N.A. |
| b. Pressurizer Pressure, P-11 | ~~N.A.~~ | ~~N.A.~~ | ~~N.A.~~ | ≤ 2000 psig | ≤ ~~2010~~ 2004 psig |
| c. Low-Low $T_{avg}$, P-12 | ~~4.0~~ | ~~0.82~~ | ~~0.87~~ | ≥ 541°F | ≥ ~~538.5~~ 540.5°F |

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SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.1.5, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

Deleted, and

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. ~~The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at $\leq$ 25% of RATED THERMAL POWER, and~~

c. The Reactor Coolant System lowest operating loop temperature ($T_{avg}$) is $\geq$ 531°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature ($T_{avg}$) < 531°F, restore $T_{avg}$ to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature ($T_{avg}$) shall be determined to be $\geq$ 531°F at least once per 30 minutes during PHYSICS TESTS.

(Proposed Wording)

## 3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than or equal to the design DNBR limit throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

These 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.

BASES

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

REPLACE WITH INSERT "8"

~~The OPERABILITY of the Reactor Protection System and Engineered Safety Feature Actuation System Instrumentation and interlocks ensure that 1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation, and 3) sufficient system functional capability is available from diverse parameters.~~

~~The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.~~

OPERABILITY of the following trips in Table 3.3-1 provides additional diverse or anticipatory protection features and is not credited in the accident analyses:

Undervoltage - Reactor Coolant Pumps (Above P-7); Underfrequency Reactor Coolant Pumps (Above P-7); Turbine Trip (Above P-9); Reactor Coolant Pump Breaker Position Trip (Above P-7); Turbine Impulse Chamber Pressure, P-13.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report as approved by the NRC and documented in the SER (letter to J. J. Sheppard from Cecil O. Thomas dated February 21, 1985). Jumpers and lifted leads are not an acceptable method for placing equipment in bypass as documented in the NRC safety evaluation report for this WCAP.

ADD BASES FROM SECTION 2.2.1

Insert 8

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated action and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The Allowable Values specified in Table 3.3-1 and Table 3.3-3 are the initial values for consideration of channel operability. A channel is OPERABLE with a nominal trip setpoint value outside its calibration tolerance band provided the trip setpoint "as found" value does not exceed its associated Allowable Value.

Additional administratively controlled limits for operability of a device are determined by device drift being less than the value required for the surveillance interval. In the event the device exceeds the administratively controlled limit, operability of the device may be evaluated by other device performance characteristics, e.g., comparison to historical device drift data, calibration characteristics, response characteristics and short term drift characteristics. A device (relay, transmitter, process rack module, etc.), whose "as found" value is in excess of the calibration tolerance, but within the additional operability criteria (administratively controlled limit), is considered operable but must be recalibrated such that the "as left" value is within the two sided (±) calibration tolerance. Plant procedures set administrative limits ("as left" and "as found" criteria) to control the determination of operability by setting minimum standards based on the setpoint methodology and the uncertainty values included in the determination of the Nominal Trip Setpoint, and allow the use of other device characteristics to evaluate operability.

Insert 8 (Continued)

The Engineered Safety Features Actuation System and Reactor Trip System Nominal Trip Setpoints specified in the Licensing Requirements Manual (LRM) are the nominal values* at which the instrumentation is set for each functional unit. A setpoint is considered to be consistent with the nominal value when the measured "as left" setpoint is within the administratively controlled (±) calibration tolerance identified in plant procedures (which specifies the difference between the Allowable Value and Nominal Trip Setpoint). Additionally, a trip setpoint may be set more conservative than the Nominal Trip Setpoint as necessary in response to plant conditions provided that the ± calibration tolerance band remain the same and the Allowable Value is also adjusted accordingly in the conservative direction to meet the assumptions of the setpoint methodology. The conservative direction is established by the direction of the inequality applied to the Allowable Value.

The setpoint methodology, used to derive the Nominal Trip Setpoints, is based upon combining all of the uncertainties in the channels. Inherent in the determination of the Nominal Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels should be capable of operating within the allowances of these uncertainty magnitudes. Occasional drift in excess of the allowance may be determined to be acceptable based on the other device performance characteristics. Device drift in excess of the allowance that is more than occasional, may be indicative of more serious problems and would warrant further investigation.

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* With the exception of the Reactor Trip System Functional Unit number 17.b for the Turbine Stop Valve Position trip. The trip setpoint specified in the LRM for Functional Unit number 17.b is not a nominal value. The trip setpoint for this Functional Unit is adjusted to be consistent with the trip setpoint value specified in the LRM in lieu of adjusting the setpoint within an established calibration tolerance band.

BASES

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

~~The Engineered Safety Feature Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.~~

~~To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the setpoints have been specified in Table 3.3-4. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its trip setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, Z + R + S ≤ TA, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the trip setpoint and the value used in the analysis for the actuation. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified trip setpoint. S or Sensor Drift is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.~~

DELETE



(Proposed Wording)

BASES

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

~~The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.~~ DELETE

The surveillance requirements for the Manual Trip Function, Reactor Trip Breakers, and Reactor Trip Bypass Breakers are provided to reduce the possibility of an Anticipated Transient Without Scram (ATWS) event by ensuring OPERABILITY of the diverse trip features (Reference: Generic Letter 85-09).

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

ESF response times which include sequential operation of the RWST and VCT valves are based on values assumed in the non-LOCA safety analyses and are provided in Section 3 of the Licensing Requirements Manual. These analyses take credit for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction valves are closed following opening of the RWST charging pump suction valves. When sequential operation of the RWST and VCT valves is not included in the response times, the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times will assure that the assumptions used for the LOCA and Non-LOCA analyses with respect to operation of the VCT and RWST valves are valid.

The maximum response time for control room isolation on high radiation is based on ensuring that the control room remains habitable following a small line break outside the containment. From a control room habitability aspect, the worst case accident that does not initiate a Containment Isolation - Phase B signal is

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

shutdown because the unit must be in at least MODE 3 to perform the test. The P-6 permissive neutron detector CHANNEL CALIBRATION is performed in conjunction with the intermediate range neutron detectors. The overtemperature ΔT, P-8, P-9 and P-10 permissive neutron detector CHANNEL CALIBRATIONS are performed in conjunction with the power range neutron detectors.

Source Range Neutron Flux

The limiting condition for operation (LCO) requirement for the source range neutron flux trip function ensures that protection is provided against an uncontrolled rod cluster control assembly (RCCA) bank rod withdrawal accident from a subcritical condition during startup with the reactor trip breakers (RTBs) closed. This trip function provides redundant protection to the Power Range Neutron Flux-Low Setpoint and Intermediate Range Neutron Flux trip functions (see UFSAR Section 15.4.1 ~~and Specification 2.2.1 Bases~~). In MODES 3, 4, and 5, with the RTBs closed, administrative controls also prevent the uncontrolled withdrawal of rods. The nuclear instrumentation system (NIS) source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. In Modes 3, 4, and 5, with the reactor trip breakers closed, the source range detectors provide an automatic trip function with a setpoint in the shutdown range and the intermediate range detectors provide an automatic trip function with a setpoint in the power range. Therefore, the functional capability at the specified trip setpoint is assumed to be available.

DELETE

The LCO requires two channels of source range neutron flux to be OPERABLE when the RTBs are closed. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip function. The LCO also requires one channel of the source range neutron flux to be OPERABLE in MODE 3, 4, or 5 with RTBs open. In this case, the source range function is to provide control room indication and the high flux at shutdown alarm. The outputs of the function to RTS logic are not required OPERABLE when the RTBs are open.

The source range neutron flux function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The function also provides visual neutron flux indication in the control room.

REPORTING REQUIREMENTS (Continued)

6.9.3 ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

- - - - - - - - - - - - - - - - - - NOTE - - - - - - - - - - - - - - - - - -
A single submittal may be made for a multi-unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.
- - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -

The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous 12 months of operation shall be submitted before April 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program (PCP) and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I Section IV.B.1.

6.9.4 MONTHLY OPERATING REPORT

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

6.9.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

ADD INSERT "9" →

3.1.3.5 Shutdown Rod Insertion Limits
3.1.3.6 Control Rod Insertion Limits
3.2.1 Axial Flux Difference-Constant Axial Offset Control
3.2.2 Heat Flux Hot Channel Factor-$F_Q(Z)$
3.2.3 Nuclear Enthalpy Rise Hot Channel Factor-$F^N_{\Delta H}$

ADD INSERT "10" →

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (Westinghouse Proprietary).

ADD INSERT "11" →



(Proposed Wording)

Insert 9

2.1.1 Reactor Core Safety Limits

Insert 10

3.2.5 DNB Parameters

3.3.1.1 Reactor Trip System Instrumentation - Overtemperature and Overpower $\Delta T$ setpoint parameter values

Insert 11

WCAP-8745-P-A, "Design Bases for the Thermal Overtemperature $\Delta T$ and Thermal Overpower $\Delta T$ Trip Functions," September 1986.

ATTACHMENT B

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 286 and 158
REVISION OF REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION REQUIREMENTS AND DNB PARAMETER REQUIREMENTS
AND ASSOCIATED BASES SECTIONS

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A. DESCRIPTION OF AMENDMENT REQUEST

I. GENERAL DESCRIPTION OF AMENDMENT REQUEST

The proposed License Amendment Request (LAR) revises the Beaver Valley Power Station (BVPS) Unit No. 1 operating license and technical specifications and the BVPS Unit No. 2 technical specifications (TS). The proposed amendment has been divided into five distinct categories. These five categories have been carried throughout Sections A, C, D and E of this LAR.

**1. REVISION TO SETPOINT AND ALLOWABLE VALUES**

The LAR revises non-conservative trip setpoints and allowable values contained in Limiting Safety System Settings (LSSS) 2.2.1 titled "Reactor Trip System Instrumentation Setpoints" and Limiting Condition For Operation (LCO) 3.3.2.1 titled "Engineered Safety Feature Actuation System Instrumentation." The non-conservative trip setpoints and allowable values currently contained in the BVPS Unit No. 1 and No. 2 technical specifications have been revised under a Basis For Continued Operation (BCO) and are under administrative control to ensure plant operation remains within the design analysis limits. This proposed amendment is a follow up to LAR No. 220 (BVPS Unit No. 1) and LAR No. 88 (BVPS Unit No. 2) previously submitted by letter L-99-006 dated January 18, 1999 (TAC No. MA4616 and MA4617) which proposed TS changes to revise the non-conservative trip setpoints and allowable values. LAR Nos. 220 and 88 were withdrawn by letter L-00-100 dated July 29, 2000 due to the inability to make a Bases wording change that would have defined the LSSS as the allowable values.

This proposed LAR also revises the allowable values contained in LSSS 2.2.1 so that the LSSS can now be defined as the allowable values. Additionally, the Nuclear Regulatory Commission (NRC) staff verbally requested that BVPS follow the guidance contained in the NRC approved Technical Specification Traveler Form (TSTF) Number 355 to NUREG 1431 Revision 1 titled "Standard Technical

Specification – Westinghouse Plants" (ISTS) when preparing a subsequent LAR pertaining to the Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) setpoints. This LAR utilizes the guidance contained in TSTF Number 355.

The Reactor Protection Setpoint Program being implemented at both Beaver Valley Units 1 and 2, [References 3 and 4], has been revised to incorporate the latest Westinghouse Methodology. The proposed changes to the RTS and ESFAS trip setpoints in the technical specifications will eliminate the need for the current administrative controls to ensure that non-conservative trip setpoints are maintained consistent with safety analysis assumptions.

The applicable Bases sections will be revised to address the calibration tolerance band. This proposed change provides a clear delineation of the design analysis requirements as they are translated into the plant technical specifications. The proposed amendment will facilitate a two-sided calibration tolerance around the trip setpoint. The uncertainty calculations, utilized in the development of the required trip setpoint, assume that the "as left" tolerance (both conservative and non-conservative direction) is satisfied on a reasonable, statistical basis. It is acceptable for the "as left" condition, immediately after calibration of the process rack modules or the bistable, to be in the non-conservative direction provided that the magnitude is within the calibration tolerance band specified in plant procedure(s). These terms are defined in the setpoint WCAP, [References 3 and 4], for each Unit.

## 2. REVISED THERMAL DESIGN PROCEDURE (RTDP)

The Revised Thermal Design Procedure (RTDP), [Reference 1], was utilized to generate revised Core Safety Limits (Figure 2.1-1) as well as the resulting Overtemperature and Overpower ΔT setpoints and associated uncertainties. RTDP was also utilized to generate additional Departure from Nucleate Boiling (DNB) margin. The additional margin is utilized in determining additional operational margin in the DNB parameters identified in Technical Specification 3/4.2.5. The WCAPS for RTDP Instrument Uncertainty Methodology [References 13 and 14] for BVPS are contained in Attachment D of this LAR.

## 3. RELOCATION TO CORE OPERATING LIMITS REPORT (COLR)

This amendment request proposes to relocate the reactor core safety limits in Section 2.1, parameters and coefficients used in the Overtemperature and Overpower Δ T reactor trip setpoint algorithms in Table 2.2-1, and the DNB parameters in 3/4.2.5 to the COLR. This amendment also expands the Administrative Controls that govern the COLR to include the above reactor core safety limits, DNB parameters and Overtemperature and Overpower ΔT trip setpoint parameter values.

In 1988, the NRC issued Generic Letter (GL) 88-16, entitled "Removal of Cycle Specific Parameter Limits from Technical Specifications." The GL proposed the relocation of selected technical specification limits to a "Core Operating Limits Report" that would be maintained by the licensee and submitted to the NRC for information. The primary motivation for relocating cycle-specific limits was to avoid the need to submit a license amendment request each time the cycle-specific limit changed. Initially the core limits proposed for relocation included the moderator temperature coefficient, bank insertion limits, power distribution limits and boron concentration in Mode 6. A requirement for relocating a specific parameter is that the parameter must be generated with an NRC-approved methodology which must be referenced in the COLR Administrative Controls section of the plant technical specifications.

Recently, the Westinghouse Owner's Group submitted WCAP 14483 entitled "Generic Methodology for Expanded Core Operating Limits Report" to the NRC for approval. In addition to the technical specifications requirements proposed for relocation in GL 88-16, the WCAP proposed relocating the Safety Limits Figure, the Overtemperature and Overpower Δ T trip setpoint parameters and coefficients and the DNB limits to the COLR. The NRC Safety Evaluation Report (SER) approved the WCAP recommendations but required the DNB limit on reactor coolant system (RCS) flow be retained in the technical specifications. The NRC SER approving the WCAP was issued on January 19, 1999.

This LAR follows the guidance provided by the Westinghouse Owner's Group in WCAP 14483-A, of January 19, 1999.

## 4. RELOCATION TO LICENSING REQUIREMENTS MANUAL (LRM)

The proposed LAR relocates the trip setpoints contained in Table 2.2-1 titled "Reactor Trip System Instrumentation Trip Setpoints" and Table 3.3-4 titled "Engineered Safety Feature Actuation System Instrumentation Trip Setpoints" to the applicable Unit's BVPS LRM. The allowable values that are also contained in Table 2.2-1 and Table 3.3-4 will be relocated to Tables 3.3-1 and 3.3-3, respectively, consistent with the single table format of the ISTS. Technical Specification 2.2 titled "LSSS – Reactor Trip System Instrumentation Setpoints" and Table 3.3-4 titled "ESFAS Instrumentation Trip Setpoints" will be deleted consistent with the format of the ISTS. This change also includes simplification of the associated instrumentation tables.

## 5. MISCELLANEOUS CHANGES

In the course of incorporating the other more significant changes included in this LAR, several additional changes are proposed. The additional changes are not directly related to the other changes proposed in this LAR, however, they do affect the same TS material and were not considered substantial enough to warrant a separate LAR. These changes are also not technical in nature. The additional changes are proposed to improve the internal consistency of the BVPS TS, simplify the presentation of requirements, provide clarifications, and improve consistency with the ISTS. This category of change also includes revising the BVPS Unit No. 1's Facility Operating License and technical specifications requirements relating to two reactor coolant loop operation and deleting the BVPS Unit No. 1 technical specification requirements pertaining to Auxiliary Feedwater Emergency Bus Undervoltage.

# II. DETAILED DESCRIPTION OF AMENDMENT REQUEST

## 1. REVISION TO SETPOINT AND ALLOWABLE VALUES

The proposed changes include the following:

For BVPS Unit No. 1 only:

- Revision of the values for "Trip Setpoint" contained in Table 2.2-1 titled "Reactor Trip System Instrumentation Trip Setpoints," for Table item numbers 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, 17.A, and 20,

- Revision of the trip setpoint and allowable value descriptions for Functional Units 2 and 12 contained in Table 2.2-1. Also, for Functional Unit 12, the associated footnote designated by a single asterisk would be deleted,

- Revision of the trip setpoint and allowable value description for Functional Unit 15 contained in Table 2.2-1,

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

- Revision of Table 2.2-1 Note 1 for Functional Unit 7 and incorporation into Table Notation (A). Table 2.2-1 Note 2 for Functional Unit 8 would also be revised and incorporated into Table Notation (B),

- Revision of the value for T′ specified in Note 1 of Table 2.2-1 for the Overtemperature ΔT equation. Additionally, the value for T" specified in Note 2 of Table 2.2-1 for the Overpower ΔT equation would be revised,

- Deletion of time constants τ4 and τ5 and associated Tavg and ΔT lag compensator functions that are specified in Notes 1 and 2 of Table 2.2-1 for the Overtemperature and Overpower ΔT setpoint,

- Revision of Table 2.2-1 Notes 3 and 4 and incorporation into the new Table Notations (A) and (B) respectively,

- An editorial change would be made to the τ3 value specified in Note 2 of Table 2.2-1,

- Deletion of the term f(ΔI) from the Overpower ΔT Trip Setpoint equation ,

- Revision of the Bases section titled "2.2.1 Reactor Trip Setpoints" to be consistent with the proposed changes,

- Revision of the values for "Allowable Value" contained in Table 3.3-4 titled "Engineered Safety Feature Actuation System 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 numbers 5.a, 6.b, 6.c, 7.a and 7.b,
- Revision of the trip setpoint and allowable value description for Functional Unit 7.b contained in Table 3.3-4,
- Revision of the Bases section titled "3/4.3.1 and 3/4.3.2 Protective and Engineered Safety Features (ESF) Instrumentation" to be consistent with the proposed changes,
- Editorial changes to the Index and Bases sections have also been included in this LAR.

For BVPS Unit No. 2 only:

- Revision of the values for "Allowable Value" contained in Table 2.2-1 titled "Reactor Trip System Instrumentation Trip Setpoints" for Functional Units 2, 3, 4, 5, 6, 9, 10, 11, 12, 13, 15, 16, and 22,
- Revision of the allowable value and trip setpoint descriptions for Functional Unit 12 contained in Table 2.2-1. The associated footnote designated by a double asterisk would be deleted,
- Revision of the trip setpoint and allowable value description for Functional Unit 15 contained in Table 2.2-1,
- Revision of the 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 number 13,
- Revision of Table 2.2-1 Note 1 for Functional Unit 7 and incorporation into Table Notation (A). Table 2.2-1 Note 3 for Functional Unit 8 would also be revised and incorporated into Table Notation (B),

- Revision of Table 2.2-1 Notes 2 and 4 and incorporation into the new Table Notations (A) and (B) respectively,

- Deletion of the term f2(ΔI) from the Overpower ΔT Trip Setpoint equation ,

- The footnote on Table 2.2-1, page 2-4, currently designated by three asterisks, would be designated by two asterisks. This footnote would also be modified by the addition of "≥" to the stated lead time constant and the addition of "≤" to the stated lag time constant,

- Revision of the values for "Allowable Value" contained in Table 3.3-4 titled "Engineered Safety Features Actuation System Instrumentation Trip Setpoints," 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 numbers 1.d, 1.1.b, 5.b, 6.b, 6.c, 7.b,

- Deletion of the values and column headings for "Total Allowance (TA)", "Z", and "S", specified in Tables 2.2-1 and 3.3-4. This deletion includes Equation 2.2-1 in LSSS 2.2.1 and Note 5 specified in Table 2.2-1,

- Revision of the Bases section titled "2.2.1 Reactor Trip System Instrumentation Setpoints" to be consistent with the proposed changes,

- Revision of Bases 3/4.3.1 and 3/4.3.2 "Reactor Trip System and Engineered Safety Features Actuation System Instrumentation" to reflect the proposed changes including the deletion of the Equation 2.2-1 specified in LCO 3.3.2.1.

<u>For BVPS Unit Nos. 1 and 2</u>:

- 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 17.B "Turbine Stop Valve" for BVPS Unit No. 1 and item 17.b "Turbine Stop Valve Closure" for BVPS Unit No. 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,

- The addition of the term "nominal" to the Trip Setpoint column in Tables 2.2-1 (includes equations contained in Table Notes 1 and 2 for BVPS Unit No. 1 and equations contained in Table Notes 1 and 3 for BVPS Unit No. 2), and Table 3.3-4. A new footnote would be added to Table 2.2-1 which denotes an exception to the nominal classification for item 17.B (BVPS Unit No. 1)/17.b (BVPS Unit No. 2),

- Revision of various terms in the Overtemperature and Overpower ΔT setpoint equations specified in Table 2.2-1. These changes include the addition of inequalities, the addition of units of measure, and the revision of term descriptions,

- Modification of Table 3.3-1, Functional Unit 6.b, Applicable Modes column, by a new table notation (8) for BVPS Unit No. 1 and new table notation (9) for BVPS Unit No. 2. This proposed change includes the addition of "Not Applicable" for BVPS Unit No. 1 and "N.A." for BVPS Unit No. 2 to the "Allowable Value" column of revised Table 3.3-1.

**2. RTDP**

The proposed changes include the following:

- Revision of Figure 2.1-1 "Reactor Core Safety Limits Three Loop Operation" contained in Safety Limit 2.1.1,

- Revision of the Bases section for Safety Limit 2.1.1 to reflect changes due to the utilization of RTDP,

- Revision of the K1 and K4 values contained in the Overtemperature and Overpower ΔT equations specified in Table 2.2-1,

- Revision of DNB parameter values specified in Table 3.2-1 titled "DNB Parameters" contained in Technical Specification 3/4.2.5, titled "Power Distribution Limits-DNB Parameters." This change includes the modification of the existing footnotes and the addition of new footnotes. The DNB parameters in Table 3.2-1 would be specified as "indicated values" consistent with the wording contained in the modifying footnotes,

- Revision and renumbering the current footnote (1) as new footnote (3) in Technical Specification 3/4.2.5. The new footnote (3) will modify the RCS total flow rate specified in the proposed wording of LCO 3.2.5.c.

## 3. RELOCATION TO COLR

The proposed changes include the following:

- Revision of index pages to reflect page numbering changes in Section 6.0 (BVPS Unit No.2 only),
- Deletion of the reference to Table 3.2-1 in the table index (BVPS Unit No.1 only),
- Deletion of the reference to Figure 2.1-1 in the figure index (BVPS Unit No.1 only),
- Revision of the definition of the Core Operating Limits Report (1.37) specified in Section 1.0 of the BVPS technical specifications,
- Modification of Safety Limit 2.1.1 by replacing the reference to Figure 2.1-1 with a reference to the COLR limits and the addition of limits for departure from nucleate boiling ratio and peak fuel centerline temperature,
- Revision of the Safety Limit 2.1.1 action requirement,
- Relocation of Safety Limit Figure 2.1-1 to the COLR,
- Relocation of Overtemperature and Overpower ΔT trip setpoint parameters and coefficients in Table 2.2-1 to the COLR. The words "specified in the COLR" would also be added to the equation terms with relocated values,
- Replacement of certain equality symbols with the words "is" or "are", as applicable, for the Overtemperature and Overpower ΔT equation parameters contained in Table 2.2-1,
- Revision of the safety limit bases section 2.1.1 titled "reactor core" to reflect the revisions to Safety Limit 2.1.1,

- Revision of Bases section titled "2.2.1 Reactor Trip Setpoints" to reflect the relocation of Figure 2.2-1 to the COLR,

- Relocation of the DNB parameters for Reactor Coolant System average temperature and pressurizer pressure, and RCS total flow rate, specified in Table 3.2-1 of TS 3.2.5, to the COLR. The minimum limit for RCS total flow rate of ≥ 261,600 gpm will be retained in LCO 3.2.5.c,

- Movement of the current footnote (1) specified in Table 3.2-1 of Technical Specification 3/4.2.5 to modify LCO 3.2.5.b pertaining to pressurizer pressure,

- Revision of Bases section for 3/4.2.5 titled "DNB Parameters,"

- Revision of the Administrative Control Section 6.9.5 for the Core Operating Limits Report to include reactor core safety limits, Overtemperature and Overpower ΔT parameter values, and the DNB parameters,

- Revision of the Administrative Control Section 6.9.5 for the Core Operating Limits Report to include WCAP-8745-P-A, Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT trip functions, September 1986.

## 4. RELOCATION TO LRM

The proposed changes include the following:

- Relocation of the RTS Trip Setpoints specified in TS Table 2.2-1 and the ESFAS Trip Setpoints specified in TS Table 3.3-4 to the Licensing Requirements Manual (LRM),

- Movement of the remaining RTS Allowable Values (LSSS) specified in Table 2.2-1 to the RTS TS Table 3.3-1,

- Movement of the remaining ESFAS Allowable Values from TS Table 3.3-4 to TS Table 3.3-3,

- Deletion of Tables 2.2-1 and 3.3-4 from the TS.

As a result of the re-organization of the instrument TS Tables described above the following changes are also being proposed:

- Movement of Bases section 2.2.2 to the Bases section 3/ 4 3.1 and 3/ 4.3.2,
- Deletion of Section 2.2 titled "Limiting Safety System Settings –Reactor Trip Instrumentation Setpoints" to reflect the removal of setpoint information from this TS,
- Movement of the Overtemperature and Overpower ΔT setpoint equations to Table Notation section of Table 3.3-1,
- Renumbering of the Overtemperature and Overpower ΔT setpoint equation notes specified in Table 2.2-1 and combining the Allowable Value and Trip Setpoint notes into a single note to accommodate moving this information as described above,
- For BVPS Unit No. 1 only, deletion of Action a. contained in LCO 3.3.2.1 and modification of Action b. by deleting the letter "b" designation,
- For BVPS Unit No. 2 only, deletion of Actions a., b.1, and b.2 contained in LCO 3.3.2.1 and modification of Action c. by deleting the letter "c" designation,
- Revision of ESFAS TS 3.3.2.1 LCO and Actions to reflect the relocation of ESFAS Trip Setpoints to the LRM and combining of ESFAS Tables 3.3-4 and 3.3-3,
- Administrative changes that include changes to the TS Index, Table Index (BVPS Unit No. 1 only), Title pages, format of the TS RTS and ESFAS instrumentation Tables, and Bases as well as pagination changes and changes to page headers and footers to support the deletions and revisions listed above.
- For BVPS Unit No. 1 only, TS 3/4.10.3 Special Test Exceptions Pressure/Temperature Limitation – Reactor Criticality, 3/4.10.4 Physics Tests, and 3/4.10.5 Special Test Exceptions No Flow Tests would be revised by deleting the trip setpoint requirement for the intermediate and power range instrumentation specified in the LCO. In addition, the page format will be updated to delete double lines on the left side of the page.

- For BVPS Unit No. 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.

## 5. MISCELLANEOUS

The proposed changes include the following:

For BVPS Unit No. 1 only:

- The term ΔT in the Overtemperature and Overpower ΔT setpoint equations, contained in Notes 1 and 2 of Table 2.2-1, would be defined,

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

- Revision of the following items to delete references corresponding to two loop operation, or loop stop valves being open, Facility Operating License Condition 2.C.(3), Table 2.2-1 Note 1 for the Overtemperature ΔT "K" coefficients that apply to less than 3 loop operation in Table 2.2-1, Bases section for "Overtemperature ΔT" on page B 2-5, Bases Section for "Loss of Flow" on page B 2-6, and Bases section for "Reactor Coolant Loops" on page B 3/4 4-1,

- Revision of the column heading title "Allowable Values" to "Allowable Value" for Table 2.2-1 and Table 3.3-4,

- Revision of Table 3.3-3 titled "Engineered Safety Feature Actuation System Instrumentation," Table 3.3-4 and Table 4.3-2 titled "Engineered Safety Feature Actuation System Instrumentation Surveillance Requirements," to delete Item 7.d, pertaining to Auxiliary Feedwater Emergency Bus Undervoltage.

For BVPS Unit No. 2 only:

- For Table 2.2-1, correction of a typographical error in Note 1 for the Overtemperature ΔT trip setpoint equation. In the term:

$$\frac{\left(1 + \tau^1 s\right)}{\left(1 + \tau_2 s\right)}$$

  the number one designation, for the equations specified in Notes 1 and 3 of Table 2.2-1, should be a subscript instead of a superscript,

- The addition of a new footnote to Table 2.2-1 on page 2-6 which defines the term "RTP". This footnote is consistent with a similar footnote for Table 2.2-1 on page 2-4,

- Reformat of BVPS Unit No. 2 RTS Instrument Table 3.3.1 header to make the header more consistent with other instrument tables in the TS.

For BVPS Unit Nos. 1 and 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,

- Simplification and reformat of the Overtemperature and Overpower ΔT equations consistent with the ISTS,

- Revision of TS 3.2.5, DNB Parameters, LCO, Actions, and Surveillance Requirements to improve consistency with the ISTS,

- Clarification of the RTS and ESFAS Actions with ISTS note allowing separate Action statement entry for each instrument function,

- Revision of instrumentation Bases text to eliminate the repetition of specific setpoint values in the Bases discussions,

- Grammatical and punctuation changes to the TS and Bases sections,

- Reformat of pages to permit incorporation of additional text and updating to current page format. The format changes include the addition and deletion of technical specification pages and page renumbering (includes the addition of wording designating the next page number),

- Rotation of the page footer on table pages (as applicable) to be consistent with the text format.

B. DESIGN BASES

**Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS)**

The purpose of the RTS and ESFAS is to automatically initiate protective actions to aid in the mitigation of certain design basis accidents and to provide a backup to operator actions in the form of permissive interlocks that ensure bypassable functions are in operation under the conditions assumed in the safety analyses. In order for the RTS and ESFAS to function correctly, the trip setpoints must be adjusted to values that ensure that the trip device will actuate at or before the value assumed in the safety analysis when all known uncertainties in the channel are accounted for. A methodology has been developed that properly accounts for channel uncertainties and was used to validate the trip setpoints and allowable values which are proposed in this license amendment request. The allowable values for BVPS are determined by adding (or) subtracting the calibration accuracy of the device tested during the channel operational test to the trip setpoint in the non-conservative direction (i.e., toward or closer to the safety analysis limit) for functional unit. This results in a smaller delta between the trip setpoint and the allowable value (i.e., more conservative). For those channels that provide trip actuation via a bistable in the process rack, the calibration accuracy is defined by the rack calibration term. For a limited number of channels that provide trip actuation without being processed via the process rack (e.g., Auto Stop Oil Pressure) the allowable value is defined by device drift or repeatability. The magnitude of the calibration accuracy term is as specified in the station procedures. The methodology is documented in the following reports listed below.

Westinghouse reports WCAP-11419 Rev. 2, "Westinghouse Setpoint Methodology for Protection Systems Beaver Valley Unit 1," December 2000, [Reference 3], and WCAP-11366, Rev. 4, "Westinghouse Setpoint Methodology for Protection

Systems Beaver Valley Unit 2," December 2000, [Reference 4], (contained in Attachment C of this LAR), provide revised Reactor Trip and Engineered Safety Feature Actuation System Trip Setpoints and Allowable Values applicable to the current BVPS Units No. 1 and 2 protection systems.

The methodology used is the statistical "square root sum of the squares" which has been utilized in other Westinghouse reports and is consistent with the current licensing basis methodology. The methodology enacted explicitly accounts for the calibration tolerance band of the final proposed nominal trip setpoint value.

## THERMAL AND HYDRAULIC DESIGN OF THE REACTOR CORE

The overall objective of the thermal and hydraulic design of the reactor core is to provide adequate heat transfer which is compatible with the heat generation distribution in the core such that the heat removal by the RCS or the Emergency Core Cooling System (ECCS) assures that the following performances and safety criteria requirements are met:

- Fuel damage is not expected during normal operation and operational transients (ANS Condition I) or any transient conditions arising from faults of moderate frequency (ANS Condition II).
- The reactor can be brought to a safe state following a Condition III event with only a small fraction of the fuel rods damaged. This fuel damage might preclude resumption of operation without considerable outage time.
- The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events.

In order to satisfy these requirements, the following design bases have been established for the thermal hydraulic design of the reactor core: Departure from Nucleate Boiling Design Basis, Fuel Temperature Design Basis, Core Flow Design Basis, and Hydrodynamic Stability Design Basis.

The existing thermal-hydraulic analysis utilized for BVPS is based on the Mini-Revised Thermal Design Procedure (Mini-RTDP) and the WRB-1 DNB correlation as described in the Beaver Valley UFSAR, [Reference 5]. The Mini-RTDP is a

conservative application of the Revised Thermal Design Procedure (RTDP). In the Mini-RTDP, uncertainties in the nuclear peaking factors, fuel fabrication parameters, and THINC-IV transient codes are combined statistically with the DNB correlation uncertainties to define the DNBR design limit such that there is at least a 95 percent probability (with 95 percent confidence) that DNB will not occur when the calculated minimum DNBR is equal to or greater than the design limit. The uncertainties included in the Mini-RTDP method are for the nuclear enthalpy hot-channel factor, $F^N_{\Delta H}$; the enthalpy rise engineering hot-channel factor, $F^E_{\Delta H,1}$; and the THINC-IV transient codes. Since the uncertainties in these parameters are considered in determining the design DNBR value, the plant safety analyses are performed using input values without uncertainties for these parameters. The DNB analysis is being modified to incorporate the RTDP as licensed by Westinghouse. The W-3 correlation and Standard Design Procedure (STDP) are still used when conditions are outside the range of the WRB-1 correlation, and of the RTDP.

Historically, the DNBR limit has been 1.30 for Westinghouse application. In this application, the WRB-1 correlation is used. The WRB-1 DNB correlation is based entirely on rod bundle data and takes credit for the significant improvements in the accuracy of the critical heat flux predictions over previous DNB correlations. The approval by the NRC that a 95/95 limit DNBR of 1.17 is appropriate has been documented in Reference 6.

With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor which is used to define the design limit DNBR that satisfies the DNB design criterion. The criterion is that the probability that DNB will not occur on the most limiting fuel rod is at least 95% (at a 95% confidence level) for any Condition I and II event. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values. The uncertainties included are the nuclear enthalpy hot-channel factor, $F^N_{\Delta H}$; the enthalpy rise engineering hot-channel factor, $F^E_{\Delta H,1}$; and the THINC-IV transient codes; uncertainties, based on surveillance data, associated with vessel coolant flow, core power, coolant temperature, system pressure and effective flow fraction (i.e., bypass

flow). The increase in DNB margin is realized when nominal values of the preceding factors are used in the DNB safety analysis.

The improved THINC-IV PWR design modeling method as described in Reference 7 was used for the DNB analysis. This modeling scheme improves the accuracy by minimizing the inaccuracies, which result in the use of the perturbation technique in the solution of the governing equations. No changes to the basic THINC-IV models and correlations were made for the improved modeling scheme.

Fuel rod thermal evaluations are performed at rated power, maximum overpower and during transients at various burnups, to assure the fuel temperature design basis as well as the fuel integrity design bases are met. During modes of operation associated with ANS Condition I and II events, there is at least a 95 percent probability at the 95 percent confidence level that the peak kW/ft fuel rod will not exceed the uranium dioxide (UO2) melting temperature. If UO2 melting is precluded, preservation of fuel geometry is assured, and the potentially adverse effects of molten fuel-cladding interactions are avoided. To preclude center melting and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of 4,700 °F has been selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluations.

A minimum of 93.5 percent of the thermal flow rate passes through the fuel rod region of the core and is effective for fuel rod cooling and to assure that the core flow design basis is met. Core cooling evaluations are based on the thermal flow rate entering the reactor vessel. A maximum of 6.5 percent of this value is allotted as bypass flow.

The hydrodynamic stability design basis is met by assuring that the modes of operation associated with ANS Condition I and II events will not lead to hydrodynamic instability. Boiling flow may be susceptible to thermohydraulic instabilities. These instabilities are undesirable in the reactor core because they may cause a change in thermohydraulic conditions that may lead to a reduction in the DNB heat flux relative to that observed during a steady flow condition or to undesired forced vibrations of core components.

C. JUSTIFICATION

## 1. REVISION TO SETPOINT AND ALLOWABLE VALUES

All of the RTS and ESFAS uncertainties have been recalculated as provided in WCAP-11419, Rev 2 and WCAP-11366, Rev 4, for Beaver Valley Units 1 and 2, respectively. The methodology used is the "square root of the sum of the squares" which has been utilized in other Westinghouse reports; i.e., WCAP-10395, "Statistical Evaluation of LOCA Heat Source Uncertainty" [Reference 8], and WCAP-8567, "Improved Thermal Design Procedure" (ITDP) [Reference 9]. RTDP uncertainties are calculated using the same methodology employed for ITDP. The NRC, noting acceptability of statistical techniques for the application requested, approved WCAP-8567. Also, various ANSI, American Nuclear Society, and Instrument Society of America standards approve the use of probabilistic and statistical techniques in determining safety-related setpoints (specifically ANSI/ANS Standard 58.4-1979, "Criteria for Technical Specifications for Nuclear Power Stations," and ISA Standard S67.04, 2000, "Setpoints for Nuclear Safety-Related Instrumentation"), [Reference 10]. The methodology used in this license amendment is essentially the same as that used for V. C. Summer in August, 1982: approved via NUREG-0717, Supplement No. 4 "Safety Evaluation Report related to the Operation of Virgil C. Summer Nuclear Station, Unit No. 1," Docket No. 50-395, August 1982.

The proposed values for the RTS and ESFAS trip setpoints, with the exception of RTS Functional Units 7 and 8 and ESFAS Functional Units (5.a for BVPS Unit No. 1 only) and (5.b for BVPS Unit No. 2 only), are more restrictive than the currently specified setpoints. The more restrictive trip setpoints are being proposed to ensure that the current safety analysis limits continue to be met. Administrative controls are currently in place to ensure that RTS and ESFAS trip setpoints are maintained at the values required to meet the current safety analysis limits. Typically, the setpoint changes were due to the previously referenced plant-specific analysis. The amount of measurement error calculated was increased which results in the trip setpoint being required to be moved closer to the nominal plant operating values.

The allowable values typically 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 net results of these analyses were changes to allowable values which are more restrictive and less restrictive. The majority of the proposed

allowable values are more restrictive with the exception of RTS Functional Unit 15 (for BVPS Unit No. 2 only) and ESFAS Functional Units (5a, 6a.1, 6a.2 for BVPS Unit No. 1 only) and (5.b, 6a.1, 6a.2, 7.c for BVPS Unit No. 2 only) which are less restrictive. Administrative controls are currently in place to ensure that RTS and ESFAS allowable values are consistent with the current safety analysis limits. The allowable value is used to determine instrumentation channel operability. The difference between the bistable trip setpoint and the associated allowable value is compiled by the addition of several uncertainties associated with the process racks or the final output device, in the cases of a remote actuation device. The original calculations were completed using generic numbers for these uncertainties. The latest revision of the WCAPs utilized plant-specific information for these variables. In several cases, discrepancies were noted and centered around instrumentation uncertainties associated with the plant process racks. These items included rack comparator settings accuracy and rack measurement and test equipment accuracy.

The proposed changes to the RTS and ESFAS trip setpoints and allowable values will eliminate the need to administratively control these setpoints and allowable values.

The RTS and ESFAS Bases will be revised to address the calibration tolerance band. This proposed change will provide a clear delineation of the design analysis requirements as they are translated into the plant TS.

The trip setpoint and allowable value description for the RTS Functional Unit 12, Loss of Flow, is being changed from "design flow per loop" to "indicated loop flow" to be consistent with the revised analysis assumptions and NSAL-00-088, [Reference 12]. This change protects against loop-to-loop RCS flow asymmetries. Due to loop flow asymmetry one of the three loops could have an indicated flow greater than the two remaining loops and be the faulted (tripping) loop. In that case, if the low flow trip is set to 90 percent of design flow for that loop, the indicated total flow could drop such that the trip would occur when the total flow is less than that assumed in the safety analysis. The allowable value was changed to reflect the method described above. Loss of Flow is credited in the Complete Loss of Forced Reactor Coolant Flow, Partial Loss of Forced Reactor Coolant Flow, and the Single Reactor Coolant Pump Locked Rotor events. The proposed changes to the trip setpoint and allowable values, including the associated description of these values,

does not have an adverse impact on the conclusions of the events described above and documented in the UFSAR.

The proposed values for the BVPS Unit No. 2 RTS Functional Unit 13 and ESFAS Functional Unit 7.b, Steam Generator Water Level Low-Low, were revised to provide more plant operational margin (based on the current administratively controlled limits) by allocating the excess margin to the trip setpoint. The current administratively controlled trip setpoint value is 18.2%. The current administratively controlled allowable value is 17.7%. The Steam Generator Water Level Low-Low is credited in the Loss of Normal Feedwater, Major Rupture of a Main Feedwater Pipe, and Loss of Offsite Power to the Station Auxiliaries (Station Blackout) events. The changes to the trip setpoint and allowable value do not adversely impact the conclusions of these events as documented in the UFSAR.

The proposed trip setpoints associated with the BVPS Unit No. 1 ESFAS Functional Unit 5.a and BVPS Unit No. 2 ESFAS Functional Unit 5.b, Turbine Trip and Feedwater Isolation on Steam Generator Water Level High-High, were established as part of the recently completed RTS/ESFAS program as documented in References 3 and 4. The safety analysis limit associated with Steam Generator High-High was also redefined as part of this program which, in turn, provided additional margin. The excess margin was re-allocated to the trip setpoint for operational purposes. The allowable value, which is based on the calibration tolerance band, was also increased accordingly. The Feedwater Isolation on Steam Generator Water Level High-High Trip is credited in the "Excessive Heat Removal Due to Feedwater Systems Malfunctions" event. The changes to the trip setpoint and allowable value do not adversely impact the conclusions of this event as documented in the UFSAR.

The allowable values provided in this LAR are based on calibration tolerance. However, for BVPS Unit Nos. 1 and 2 ESFAS Functional Units 6.a.1 and 6.a.2, Loss of Power on 4.16 kv Emergency Bus Undervoltage, the calibration tolerance was much smaller than the drift term; therefore, the allowable value is based on the equipment performance specifications. Equipment performance specifications were chosen to calculate the allowable value rather than the calibration tolerance in order to provide plant operational margin. As with Functional Units 6.a.1 and 6.a.2 above, the allowable values for Auxiliary Feedwater on Undervoltage RCP, Functional Unit 7.b for BVPS Unit No. 1 and Functional Unit 7.c for BVPS Unit

No. 2, are also based on equipment performance specification and not calibration tolerance. This calculation process was also used for the following functional units: Auxiliary Feedwater on Undervoltage – RCP, Functional Unit 7.b for BVPS Unit No. 1 and Functional Unit 7.c for BVPS Unit No. 2; Undervoltage-Reactor Coolant pump, Functional Units 15 for BVPS Units 1 and 2; Turbine Trip on Auto stop oil pressure, Functional Unit 17.A for BVPS Unit No. 1; and Underfrequency-Reactor Coolant Pumps, Functional Units 16 for BVPS Unit No. 2.

The deletion of the inequalities from the trip setpoints stated in Table 2.2-1 (with the exception of the turbine stop valve trip setpoint) and Table 3.3-4 along with the change in terminology from "Trip Setpoint" to "Nominal Trip Setpoint" is consistent with the setpoint methodology used to develop the required trip setpoints. The proposed amendment will facilitate a two-sided calibration tolerance around the trip setpoints as provided in References 3 and 4. The uncertainty calculations, utilized in the validation of the nominal trip setpoint, assume that the "as left" tolerance (both in the conservative and non-conservative direction) is satisfied on a reasonable, statistical basis. It is acceptable for the "as left" condition, immediately after calibration of the process rack modules or the bistable, to be in the non-conservative direction provided that the magnitude is within the calibration tolerance band specified in plant procedure(s). The specified calibration tolerance band will consist of rack calibration accuracy in conjunction with the rack comparator setting accuracy as required. These terms are defined in the setpoint WCAPs [References 3 and 4] for each unit.

Currently the technical specifications provide the same values for the dynamic compensation terms, the lead/lag on Overtemperature $\Delta T$ and the rate lag on Overpower $\Delta T$, that are input in the safety analyses. BVPS Units Nos. 1 and 2 do not include inequalities for these terms. Additionally, the Overtemperature and Overpower $\Delta T$ hardware for BVPS Unit No. 1 does not allow the technical specification value to be entered as provided. According to the functional requirements, the hardware for the time constants is adjustable in increments such that any setpoint can be obtained within ±10%. Therefore, the setting for BVPS Unit No. 1 has been set as close to the required values as possible and within ±10%. When these time constants are assumed in the safety analysis, the values modeled are the values provided in the technical specification. Note that this analysis approach is not consistent with that used for setpoints. The safety analysis values assumed for the setpoint are different from the technical specification value by some

amount sufficient to cover the calculated static instrument uncertainties and bounding time response. Note also that when the hardware allows the settings to be within ±10% of the desired value, the setting may not be in the conservative direction with respect to the analysis. In order to assure that the settings selected are conservative with respect to the values assumed in the analyses, a direction of conservatism can be chosen for each constant associated with the Overtemperature and Overpower ΔT setpoint equations. This can be accomplished by replacing the equalities with inequalities to specifically allow these settings to be set in a conservative fashion. The inequalities for the "τ values" are documented in References 3 and 4. Inequalities for T′, T″, and P′ are also documented in Reference 3 and 4. The equalities for T′, T″, $\tau_2$, $\tau_5$ and $\tau_6$ are changed to "≤". The equalities for P′, $\tau_1$, $\tau_4$ and $\tau_7$ are changed to "≥". The equality for $\tau_3$ when associated with $T_{avg}$ (BVPS Unit No. 1) is changed to "≥" and when associated with ΔT (BVPS Unit No. 2) is changed to "≤". Since the "τ values" are no longer set within some tolerance of a specific value but set "≤" or "≥" of a given value, the 2.2.1 Bases discussion on nominal time constants has been removed. The inequalities for the "K-values" are documented in the ISTS and clarified in Reference 11. The equalities for K1 and K4 are changed to "≤". The equalities for K2, K3, K5, and K6 are changed to "≥".

Additional changes, which are categorized as editorial, have been made to the notes for the Overtemperature ΔT Trip Setpoint and Overpower ΔT Trip Setpoint to improve readability, remove excessive detail, and provide clarity. The description of $\Delta T_0$ is being changed from "Indicated ΔT at RATED THERMAL POWER" to "loop specific indicated ΔT at RATED THERMAL POWER". For BVPS Unit No. 2 only, the wording "Calibration temperature for ΔT instrumentation" would be deleted from the T″ term description. To assure that the calculations of these two setpoints are performed in accordance with the setpoint methodology, loop specific has been added to the description of $\Delta T_0$. Additional changes were made to assure that the calculations were performed consistent with the methodology. The description of "average temperature", T, is being changed to "measured RCS average temperature". The "indicated $T_{avg}$ at RATED THERMAL POWER" or "nominal $T_{avg}$ at RATED THERMAL POWER", T′ and T″, are being changed to "$T_{avg}$ at RATED THERMAL POWER". The "Pressurizer Pressure", P, is being changed to "measured pressurizer pressure". The pressure value for P′ is changed from "2235 psig" to "≥ 2250 psia". The temperature value for T′ (BVPS Unit No. 1

only) will be revised by changing the equality to an inequality. For BVPS Unit No. 1 only, the unit of "/°F" was added to K2, and K6. The unit of "/psia" was added to K3. Units for pressure are changed from "psig" to "psia". For BVPS Unit No. 1 only, the unit, $sec^{-1}$, has been added to "Laplace transform operator." For BVPS Unit No. 1 only, the terms "controller for $T_{avg}$ dynamic compensation" and "controller for $T_{avg}$" have been changed to "compensator for $T_{avg}$. For BVPS Unit No. 2 only, the terms "compensator for $T_{avg}$ dynamic compensation" have been changed to "compensator for $T_{avg}$". For BVPS Unit No. 1 only, the P′ description is being changed from "indicated RCS nominal operating pressure" to "nominal pressurizer pressure". For BVPS Unit No. 2 only, the P′ description is being changed from "nominal RCS operating pressure" to "nominal pressurizer pressure". For BVPS Unit No. 2 only, the abbreviation of 's' for seconds has been changed to 'sec'. For BVPS Unit No. 2 only, the words "function generated by" will be added to certain lead and/or lag compensator terms. For BVPS Unit No. 2 only, the description for $\tau_6$ is being changed by deleting the words "measured $T_{avg}$."

The f(ΔI) (BVPS Unit No. 1) and f2(ΔI) (BVPS Unit No. 2) will be deleted from the equation of the Overpower ΔT Trip Setpoint since it is not modeled in the safety analysis nor included in the setpoint methodology calculation. Defeating this function, rather than leaving it in the equation with a setting of zero, eliminates the possibility that it will adversely contribute to the Overpower ΔT Trip due to the limitations of the hardware and possible variations in the setpoint. The circuit, associated with the generation of f(ΔI) (BVPS Unit No. 1) and f2(ΔI) (BVPS Unit No. 2), will be defeated as part of the amendment implementation process.

The maximum amount by which the channel's Overtemperature and Overpower ΔT trip setpoint can exceed the computed trip setpoint for each of its inputs has been reduced to 0.5 percent ΔT span. These changes have been made to meet the revised analysis assumptions.

The Reactor Trip Setpoint Bases, 2.2.1, and the Protective and Engineered Safety Features Instrumentation Bases, 3/4.3.1 and 3/4.3.2, changes are made to ensure consistency between the proposed TS and the bases. The changes reflect the use of the Westinghouse Setpoint Methodology for Protection Systems, [References 3 and 4].

The Bases section for the RTS and ESFAS instrumentation will be revised by the addition of wording which clarifies that a setpoint can be set more conservatively than the nominal value as necessary in response to plant conditions. The proposed addition of this Bases wording is consistent with the existing technical specification requirements. The current technical specifications, with the inequalities in place, permit the trip setpoint to be set in the conservative direction of the specified trip setpoint. It is consistent with the setpoint methodology for the trip setpoint to be set more conservative than the Nominal trip setpoint provided that the calibration tolerance band remains the same and the allowable value is also adjusted accordingly in the conservative direction. The conservative direction is established by the direction of the inequality applied to the allowable value. The proposed Bases wording change utilizes the guidance provided in TSTF-355. The changes to the Bases are necessary to preserve consistency with the changes to the technical specifications and to provide supporting information to clarify the intent of the proposed technical specifications.

The format for defining the setpoint as a nominal trip setpoint is consistent with the proposed changes to the ISTS Revision 1 approved in Technical Specification Task Force (TSTF) Change Number 355. This format change for defining the setpoint as a nominal trip setpoint has been submitted and approved by the NRC for another Westinghouse plant (Vogtle). Vogtle Technical Specification Amendments 101 (Unit No. 1 TAC No. MA0194) and 79 (Unit No. 2 TAC No. MA0195) implemented this format change for defining the setpoint as a nominal trip setpoint.

The proposed change (for BVPS Unit No. 2 only) to delete the values and column headings for "Total Allowance (TA)", "Z", and "S", specified in Tables 2.2-1 and 3.3-4, Equation 2.2-1 in LSSS 2.2.1 and Note 5 specified in Table 2.2-1 is made because the flexibility allowed by the 5 column format, in making an operability determination to serve as the basis for making a determination relative to potential reportable events, is no longer necessary. In 1983, 10 CFR 50.73 (as documented in NUREG 1022) changed the reporting requirements from a single inoperable channel to loss of functionality for a trip; i.e., reportability occurs when two or more channels are inoperable for a trip function assuming compliance with other TS requirements. Therefore, the advantages of the 5 column format are no longer present. This proposed change is consistent with the current format contained in BVPS Unit No. 1 technical specification, and in the ISTS.

The values of T′ in the Overtemperature ΔT equation and the value of T" in the Overpower ΔT equation would be revised for BVPS Unit No. 1 only. The proposed change for T′ and T" from 576.3°F to 576.2°F is conservative in nature and results in a reduced overtemperature and overpower setpoint. This change is necessary to make the values for T′ and T" stated in the plant's technical specifications consistent with the nominal Reactor Coolant System average temperature assumed in the safety analysis.

The deletion of time constants $\tau_4$ and $\tau_5$ in BVPS Unit No. 1 Table 2.2-1 Notes 1 and 2 is being proposed so that the table correctly reflects the hardware configuration of the BVPS Unit No. 1 Trip System. The deleted time constants have been removed from the analog channel hardware by plant hardware change Design Change Package (DCP) 698. The current requirement is to set the time constants to a value equal to or less than 2 seconds. A deleted time constant is implicitly set to zero, which is consistent with the required setpoint value. The proposed change is made to remove the implication that the time constants exist in the hardware.

For the turbine stop valve position trip channel, the current setpoint and allowable value settings and inequalities are retained. This channel is an exception to the other channels currently listed in Table 2.2-1 because the sensing element is a limit switch. This limit switch is only subject to relatively course adjustment. This device is not subject to electronic drift between functional tests like true analog channels which include electronic signal conditioning which must be periodically recalibrated on line. Also, the safety analysis does not require a specific setting for this trip function. Therefore, the current setpoint is retained and the nominal value as defined for the other channels does not apply.

The proposed addition of a new table notation (8) for BVPS Unit No. 1 and new table notation (9) for BVPS Unit No. 2 to the "Applicable Modes" column for Functional Unit 6.b contained in Table 3.3-1 is a clarification of the current requirements. This Source Range function is not required to initiate a reactor trip when the control rods are not capable of being withdrawn. The column for "channels to trip" in Table 3.3-1 states "zero" for this function. Consistent with the associated Bases section, this function is only required to provide indication. The proposed addition this new table notation for this function is consistent with the ISTS. The addition of "Not Applicable" for BVPS Unit No. 1 and "N.A." for BVPS

Unit No. 2 to the "Allowable Value" column is consistent with a Function that is not required to initiate a reactor trip. This change is also consistent with the ISTS.

**2. RTDP**

The purpose for implementing the RTDP methodology is primarily to gain DNB margin. The margin recovery is gained by the combination of additional uncertainties over those used to calculate the DNBR limits with the current licensed DNB methodology which is mini-RTDP. This treatment of uncertainties provides a more realistic prediction of the DNBR limit which satisfies the design criterion. In 1998 a BCO was developed to support the RPS/ESFAS setpoints to justify continued operation with the current setpoints. In order to demonstrate an acceptable conclusion, Westinghouse utilized the existing DNB margin available at the time to address industry issues and still operate within the current limits. The proposed amendment would remove the current BCO in place for this technical area.

In order to increase the margins associated with the DNB limits, the changes to the technical specifications shown in Attachment A are being proposed. The proposed revisions involve the following:

a) Use of upgraded Reactor Trip System uncertainty calculations (See section 1 titled "REVISION TO SETPOINT AND ALLOWABLE VALUES"), and

b) Use of Revised Thermal Design Procedure (RTDP) Methodology

The current licensing basis utilizes the Mini-RTDP and the WRB-1 DNB correlation to generate the Core Thermal Limits. The Mini-RTDP is a conservative application of the Revised Thermal Design Procedure (RTDP). In the Mini-RTDP, uncertainties in the nuclear peaking factors, fuel fabrication parameters, and THINC-IV transient codes are combined statistically with the DNB correlation uncertainties to define the DNBR design limit such that there is at least a 95 percent probability (with 95 percent confidence) that DNB will not occur when the calculated minimum DNBR is equal to or greater than the design limit. The uncertainties included in the Mini-RTDP method are for the nuclear enthalpy hot-channel factor, $F^N_{\Delta H}$; the enthalpy rise engineering hot-channel factor, $F^E_{\Delta H,1}$; and the THINC-IV transient codes. Since the uncertainties in these parameters are

considered in determining the design DNBR value, the plant safety analyses are performed using input values without uncertainties for these parameters. The DNB analysis is being modified to incorporate the RTDP as licensed by Westinghouse. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are combined statistically to obtain the overall DNB uncertainty factor which is used to define the design limit DNBR that satisfies the DNB design criterion. The criterion is that the probability that DNB will not occur on the most limiting fuel rod is at least 95% (at a 95% confidence level) for any Condition I and II event. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the plant safety analyses are performed using input parameters at their nominal values. The uncertainties included are the nuclear enthalpy hot-channel factor, $F^N_{\Delta H}$; the enthalpy rise engineering hot-channel factor, $F^E_{\Delta H,1}$; and the THINC-IV transient codes; uncertainties, based on surveillance data, associated with vessel coolant flow, core power, coolant temperature, system pressure and effective flow fraction (i.e., bypass flow). The increase in DNB margin is realized when nominal values of the preceding factors are used in the DNB safety analysis. The Reactor Core Safety Limits form the basis for the reactor protection system Overtemperature ΔT and Overpower ΔT setpoints. Therefore, the justification for changing TS Figure 2.1-1 is to ensure that the revised basis for the Overtemperature ΔT and Overpower ΔT setpoints is presented.

The Safety Limits Bases changes are made to ensure consistency between the proposed TS and the bases. The changes reflect the use of the RTDP methodology. Also an editorial change (for BVPS Unit No. 1 only) is made by replacing discussions concerning the bases of Figure 2.1-1 with verbiage to indicate that the curves for Figure 2.1-1 are based on enthalpy hot channel factor limits provided in the Core Operating Limits Report (COLR).

The use of the RTDP Methodology provides additional operating margin to the BVPS Unit Nos. 1 and 2 Technical Specification 3/4.2.5, DNB Parameters. The RTDP takes advantage of the conservative use of statistical combination of the values for reactor power, RCS flow, temperature, and pressure. Therefore, due to the methodology, the TS values for the DNB parameters of reactor coolant system $T_{avg}$, pressurizer pressure, and reactor coolant system total flow rate are revised.

However, some of the margin that was gained was used to provide TS indicated values that include allowances for control board indication or process computer indication, whichever is appropriate per plant procedure. Note that the plant procedures use these parameters to monitor plant conditions to assure safe operation. For BVPS Unit No. 1, superscripts 1, 2, and 3 were added to the pressure, temperature, and flow parameter values to indicate that these values include uncertainty allowance for control board indication. For BVPS Unit No. 2, superscripts 1 and 2 were added to the pressure and temperature parameter values respectively, to indicate that these values include uncertainty allowance for control board indication. Superscript 3 was added to the flow parameter value to indicate that this value includes uncertainty allowance for process computer indication.

The K1 specified in Functional Unit 7, Overtemperature ΔT, was revised to provide operational margin and to be consistent with the RTDP method used. The K4 value specified in Functional Unit 8, Overpower ΔT, was also revised to provide operational margin and to be consistent with the RTDP method used. The Overtemperature ΔT is credited in the Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power, Loss of External Electrical Load and/or Turbine Trip, and Accidental Depressurization of the Reactor Coolant System. The Overpower ΔT is not credited in the Chapter 14 (BVPS Unit No. 1) or Chapter 15 (BVPS Unit No. 2) UFSAR events. The changes to the K-values do not adversely impact the conclusions of the Overtemperature ΔT credited events as documented in the UFSAR.

## 3. RELOCATION TO COLR

The methodology for relocating cycle-specific parameters has been approved by the NRC (Safety Evaluation relating to Topical Report WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," approved January 19, 1999). This proposed change would control the cycle-specific parameters within the acceptance criteria and assure conformance to 10 CFR50.36 by using the approved methodology instead of specifying technical specification values. Draft COLR changes that reflect the proposed relocation are provided in Attachment E of this LAR.

With the approval of this proposed LAR, future changes to the core safety limits will be able to be made without a license amendment. Since the safety limits

depend on power distribution parameters that have already been relocated to the Core Operating Limits Report, future changes to these parameters can be made without requiring a license amendment due to the safety limit change. Replacing the core safety limit figure with limits for departure from nucleate boiling ratio and peak fuel centerline temperature will have no impact on plant operation or safety. This change will not result in any change in the performance of safety related equipment or any change in plant operational procedures. Core safety limit violations are normally prevented by the automatic actions of the reactor trip system and the steam generator safety valves and do not require any specific operator actions.

The action requirement for Safety Limit 2.1.1 will be revised to reflect the relocation of the Safety limits curve and the addition of the two new limits specified in 2.1.1.1 and 2.1.1.2. The current action is not appropriate for the revised Safety limit 2.1.1 since it is based on only the limits specified in Figure 2.1-1. The actions for violation of a Safety Limit specified in 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 ISTS.

The proposed addition of Safety Limits 2.1.1.1 and 2.1.1.2 will provide more specific requirements regarding the safety limits (i.e., the fuel DNB design basis and the fuel centerline melt design basis), conforming with WCAP –14483-A. This proposed change follows the NRC approved guidance contained in TSTF- 339, Revision 2 to the ISTS.

Relocating the Overtemperature ΔT and Overpressure ΔT setpoint parameters to the COLR will enable the plant to implement cycle-specific changes to the setpoints without having to submit a license amendment. With this added flexibility the plant will be able to more efficiently take advantage of cycle-specific margins to optimize trip setpoints. Consistent with similar wording approved for Seabrook Unit No. 1 in amendment number 76 (TAC No. MA8764), the relocated Overtemperature ΔT and Overpressure ΔT setpoint parameters values (including inequalities/equalities and associated Units of measure) will be replaced with the wording "as specified in the COLR." Relocating the entire term (value, inequalities/equalities, and Units of measure) will eliminate unnecessary repetition of this terms in both the COLR and in the technical specifications. Relocating the DNB parameter limits to the Core Operating Limits Report will give Beaver Valley the flexibility to optimize margins between the cycle-specific analysis limits and plant operational limits.

The DNB parameters for pressurizer pressure and RCS average temperature are being relocated to the COLR. These relocated values have been changed from analysis values to indicated values. RCS total flow rate is also being changed from the analysis value to an indicated value. The indicated RCS total flow rate will be contained in the COLR. However, the current analysis value of 261,600 gpm will be retained in LCO 3.2.5. This number is the minimum NRC approved value for RCS total flow rate. Retaining this minimum analysis value will assure that a lower flow rate than reviewed by the NRC will not be used. This proposed change follows the NRC approved guidance contained in TSTF- 339, Revision 2 to the ISTS.

Consistent with the relocation of the reactor core safety limit figure, the DNB parameter limits and the Overtemperature and Overpower $\Delta T$ values, TS Section 6.9.5, Core Operating Limits Report, is revised to indicate that these values are relocated and updated to provide the approved analytical methods that were used to generate the relocated parameters. This change is consistent with Westinghouse WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," which was accepted for referencing in licensing application by the NRC staff on January 19, 1999.

The definition of the Core Operating Limits Report specified in Section 1.0 of the BVPS technical specifications is being revised to be consistent with the ISTS. This proposed change is necessary since the required scope of the COLR will be expanded by this proposed LAR. The COLR will now contain plant parameters which are not solely core operating limits. Therefore, the COLR definition is being revised to encompass the additional COLR scope by specifying cycle specific parameter limits instead of core operating limits. For BVPS Unit No. 1 only, the reference to Specification 6.9.1.12 is being changed to Specification 6.9.5 to accurately reflect the Administrative Control section that contains the COLR. The remaining changes are necessary to revise the current definition to be consistent with the ISTS. The proposed changes to the COLR definition are administrative or editorial in nature and do not affect plant safety.

## 4. RELOCATION TO LRM

The RTS and ESFAS TS in NUREG-1431, Rev. 1 (ISTS) including the NRC approved changes in TSTF Change Number 355 contain a reviewer's note that states, "Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit". The ISTS note recognizes that the LSSS are the only setpoints required in the TS by 10 CFR 50.36. TSTF-355 introduces changes to the RTS and ESFAS TS to more clearly and accurately describe the relation between the nominal trip setpoint and Allowable Value. When supported by the setpoint methodology, the LSSS (Allowable Value) may be the only setpoint limit specified in the TS to define instrument operability. This is consistent with the requirements of 10 CFR 50.36. In this case, the nominal trip setpoint and associated calibration tolerances are not required in the TS. The TSTF further recognizes that the reviewer's note described above allows the relocation of the trip setpoint information from the TS to an owner-controlled document.

Based on the regulations and other reference documents described above and the new BVPS setpoint methodology that establishes a nominal trip setpoint and an Allowable Value that corresponds to the required LSSS, BVPS proposes to relocate the nominal trip setpoint information for the RTS and ESFAS instrumentation in Tables 2.2-1 and 3.3-4 to the LRM. The BVPS LRM, for relocated TS requirements, is incorporated by reference in the UFSAR. The BVPS LRM was previously determined to be acceptable by the NRC for relocated TS requirements in BVPS License Amendment Numbers 233/115 (SER dated 9/7/00). Therefore, adequate regulatory control of the LRM exists to ensure that future changes to the trip setpoints are controlled by the provisions of 10 CFR 50.59 and that prior NRC review and approval of changes will be requested when required by 10 CFR 50.59. The provisions of 10 CFR 50.59 establish adequate controls for material relocated from the TS, including record retention and reporting requirements.

Additionally, the ISTS presentation of the RTS setpoints does not include a separate TS section for the LSSS as does the previous standard TS and the BVPS TS. The ISTS incorporates the LSSS into the RTS TS instrumentation table that contains all the requirements applicable to the RTS instrument functions. The ISTS also does not contain a separate setpoint table in the ESFAS TS as does the previous standard TS and the BVPS TS. The ESFAS setpoints are also integrated into the ESFAS TS Table that contains all the other requirements applicable to the ESFAS instrument

functions. The ISTS presentation of this information simplifies the TS eliminating redundancy and the need to refer between different instrumentation tables for setpoint information and instrument operability requirements. The human factored ISTS presentation of this information provides a substantial benefit to the end user and to the general clarity of the affected TS requirements.

In order to be more consistent with the ISTS and derive the benefits discussed above, BVPS proposes to move the RTS and ESFAS Allowable Values from Tables 2.2-1 and 3.3-4 to Tables 3.3-1 and 3.3-3. The end result will be a single more comprehensive instrumentation function table in each TS similar to the ISTS presentation of this information. The combining of instrument requirements in the TS tables is not intended to introduce a technical change to the information being moved. As the trip setpoints are being relocated to the LRM, the removal of the Allowable Values result in the elimination of the setpoint Tables (2.2-1 and 3.3-4) from the TS. However, the required setpoint information, LSSS for the RTS and allowable value for ESFAS, is retained in the TS. Therefore, the proposed change to move the Allowable Values from one TS table to another is considered administrative and is made to enhance the clarity and usability of the BVPS TS and improve consistency with the ISTS. The technical changes made to the Trip Setpoints and Allowable Values in this LAR are related to other changes discussed in Section 1 of this LAR. Draft LRM changes that reflect the proposed relocation are provided in Attachment F of this LAR.

Based on the discussions above, the proposed changes to relocate the Trip Setpoints from the TS and move the Allowable Values within the TS improve the presentation and clarity of the BVPS TS consistent with the ISTS. In addition, the proposed changes continue to maintain the LSSS within the TS per 10 CFR 50.36 and provide an adequate level of control for the relocated Trip Setpoints. Therefore, the proposed changes are acceptable.

As a result of combining the instrumentation Tables and relocating the Trip Setpoints and focusing instrumentation TS operability on the Allowable Value as described above, the following additional changes are proposed for the instrumentation TS:

1. Once the LSSS have been removed from TS section 2.2.1 and combined with the RTS TS, TS section 2.2.1 becomes redundant and unnecessary. TS section 2.2.1 contains the following requirements:

   a) That the instrumentation setpoints be maintained and adjusted consistent with the Trip Setpoint values listed in Table 2.2-1,

   b) That RTS TS Actions be entered when the setpoint is found to be less conservative than the Allowable Value listed on Table 2.2-1,

   c) An Applicability that simply references the RTS TS Applicability, and

   d) In the case of BVPS Unit No. 2, an equation that provides an alternate Action to immediately declaring a channel inoperable when the Allowable Value is exceeded.

   As previously explained, the current standard TS in the ISTS do not contain a specific TS section 2.2.1 for the LSSS. The setpoints associated with the RTS instrumentation are included in the RTS TS. Consistent with the guidance contained in the ISTS and TSTF-355, the TS 2.2.1 requirement to maintain and adjust the instrumentation setpoints consistent with the specified Trip Setpoints is explained in the standard TS Bases. The TS operability requirements focus on the Allowable Value or LSSS as required by 10 CFR 50.36. The TSTF and ISTS consider the maintenance of the Trip Setpoints within tolerance as operability related descriptive detail that is normally included in the ISTS bases. BVPS has proposed TS Bases additions that are consistent with the guidance of the ISTS and TSTF-355 to address the requirements to maintain and adjust the RTS instrumentation setpoints consistent with the specified Trip Setpoints.

   Once the Allowable Values are integrated into the RTS instrumentation operability requirements specified on Table 3.3-1, a separate TS 2.2.1 Action for exceeding the Allowable Value is no longer required. The LCO statement for the RTS TS refers to the requirements of Table 3.3-1 for the operability of the RTS instrumentation. If a requirement of Table 3.3-1 (including the Allowable Value) is not met for a specific instrument function a TS Action is applicable. The organization of all RTS instrumentation operability requirements (excluding surveillance requirements) on a single Table with a

single LCO statement reference to the Table is consistent with the format and function of the corresponding standard TS in the ISTS.

The TS 2.2.1 Applicability simply references the RTS Table 3.3-1. The inclusion of the Allowable Values from Table 2.2-1 in Table 3.3-1 provides all the operability requirements for the RTS instrumentation in one Table and eliminates the need to reference Table 3.3-1 for Applicability requirements. The Applicability of the RTS instrumentation is directly determined from Table 3.3-1 within the RTS TS. As such, the deletion of the reference to Table 3.3-1 in TS 2.2.1 has no impact on the RTS instrument requirements.

Based on the discussions above, the proposed changes serve to simplify the BVPS TS and make them more consistent with the ISTS without reducing the technical requirements important to the safe operation of the plant. The proposed changes incorporate features from the ISTS that result in a substantial improvement in the clarity and internal consistency of the TS. Therefore, these changes are acceptable.

The BVPS Unit No. 2 TS 2.2.1 contains an equation that provides an alternate Action for inoperable RTS instrumentation. This equation was intended to provide an alternate Action that could reduce the number of reports required for inoperable instrument channels. However, the applicable reporting requirements (10 CFR 50.73) were revised and the need for the alternate Action provided by this equation no longer exists. This equation is not part of the current standard TS and no longer has any impact on the operability of the RTS instrumentation. The elimination of this equation and all the associated terms from the BVPS TS is discussed in Section 1 "Revision to Setpoint and Allowable Values" of this LAR.

2. As part of moving the Allowable Values for the RTS instrumentation from Table 2.2-1 to Table 3.3-1, the Overtemperature and Overpower ΔT equation notes and terms are also moved. As the Allowable Values for the Overtemperature and Overpower ΔT instrument functions reference the Trip Setpoint equation for these functions, the trip setpoint equation is being retained within the TS and moved to Table 3.3-1 along with the Allowable Value. The existing BVPS TS contain four equation notes to address the Overtemperature and Overpower ΔT instrument functions. A separate note is provided for the Trip Setpoint and the Allowable Value for each function. As

the Allowable Value (LSSS) is being retained in Table 3.3-1, and for the Overtemperature and Overpower ΔT instrument functions the Trip Setpoint is integral to the Allowable Value, the separate equation notes for Trip Setpoint and Allowable Value are being combined in Table 3.3-1. BVPS is proposing only two notes be referenced in Table 3.3-1 instead of four. Each note will state the Allowable Value for one of the functions and reference the included Trip Setpoint equation. Additionally, in order to better integrate the new Overtemperature and Overpower ΔT equation notes into the Table 3.3-1 Table Notation, the note numbers are revised to an alpha designation "A" and "B".

The proposed changes to these equation notes (combination and renumbering) discussed above are changes to the format and presentation of the Overtemperature and Overpower ΔT instrument setpoint information. The changes are not technical in nature and are considered administrative changes necessary to conform to the single LSSS instrument value being retained in TS Table 3.3-1. Therefore, these changes are acceptable. In addition to the changes described above, this LAR contains several other changes to these equations (both in format and technical) that are related to other major changes and are discussed in Section 1 titled "REVISION TO SETPOINT AND ALLOWABLE VALUES."

3. Changes are proposed to the ESFAS TS LCO statement and Actions to simplify the TS consistent with the relocation of the Trip Setpoints from the TS and with the combined instrument Table containing all operability requirements (excluding surveillance requirements) for the ESFAS instrumentation.

   The current LCO refers to two Tables for the ESFAS instrument operability requirements (Table 3.3-3 and Table 3.3-4). Consistent with the elimination of Table 3.3-4 from the TS and the inclusion of the Allowable Values from Table 3.3-4 in Table 3.3-3, the reference to Table 3.3-4 in the ESFAS LCO is no longer required. The changes described earlier eliminated the need for a separate instrument table for setpoints and combined the ESFAS setpoint information with the other operability requirements in one table (ESFAS Table 3.3-3). Therefore, the LCO statement referring to Table 3.3-4 is no longer applicable and is deleted.

Similarly, the ESFAS Actions are also simplified. The current ESFAS Actions contain statements that reference the Allowable Values in Table 3.3-4 and another Action that references Table 3.3-3. A separate Action referring to the Allowable Value in Table 3.3-4 is no longer required. The proposed change described earlier introduces a single Table 3.3-3 containing all the operability requirements for the ESFAS Instrumentation (including the Allowable Values) consistent with how the ISTS is organized. The proposed change would make the Action and LCO statements consistent with reference to a single table for ESFAS instrumentation operability. Therefore, the current multiple Action statements are reduced to one Action referencing Table 3.3-3 consistent with the LCO requirements. This change makes the BVPS TS more consistent with the format of the ISTS, and eliminates unnecessary text and Table references from the TS.

The current ESFAS Actions also contain statements that require the adjustment of instrument setpoints consistent with the Trip Setpoints specified in the TS. Consistent with the guidance contained in ISTS and TSTF-355, the ESFAS requirement to adjust the instrumentation setpoints consistent with the specified Trip Setpoints is explained in the Bases. The TS operability requirements focus on the Allowable Value or LSSS as required by 10 CFR 50.36. The TSTF and ISTS consider the maintenance of the Trip Setpoints within tolerance as operability related descriptive detail that is normally included in the ISTS bases. BVPS has proposed TS Bases additions that are consistent with the guidance of the ISTS and TSTF-355 to address the requirements to adjust the ESFAS instrumentation setpoints consistent with the required Trip Setpoints.

The BVPS Unit No. 2 ESFAS Actions contain an option not present in the BVPS Unit No. 1 ESFAS Actions. The BVPS Unit No. 2 TS contains an equation that provides an alternate Action for inoperable ESFAS instrumentation. This equation was intended to provide an alternate Action that could reduce the number of reports required for inoperable instrument channels. However, the applicable reporting requirements (10 CFR 50.73) were revised and the need for the alternate Action provided by this equation no longer exists. This equation is not part of the current standard TS and no longer has any impact on the operability of the ESFAS instrumentation. The elimination of this equation and all the associated terms from the BVPS TS is

discussed in Justification Section 1 titled "REVISION TO SETPOINT AND ALLOWABLE VALUES."

The changes to the ESFAS LCO and Actions described above are considered administrative changes needed to support the relocation of the Trip Setpoints and the combination of the instrument tables discussed previously. The changes also serve to simplify the ESFAS TS and improve consistency with the ISTS. Therefore, these changes are acceptable.

4. The trip setpoints required in the technical specifications 3/4.10.3, 3/4.10.4, and 3/4.10.5 (for BVPS Unit No. 1) and 3/4.10.3 (for BVPS Unit No. 2) pertain to the normal operability requirements of instrumentation specified in the RTS technical specifications and bases. The RTS Instrumentation technical specification 3/4.3.1.1 specifies the operability requirements for the Intermediate and Power Range Instrumentation. The repetition of the trip setpoint requirements in the specifications listed above does not add new or different operability requirements. Therefore, the additional references to specific RTS instrument setpoints are redundant and may be removed from the technical specifications listed above without decreasing the requirements applicable to the affected instruments. Surveillances are retained in the technical specifications listed above to ensure the RTS instrumentation is operable when required. The removal of the specific trip setpoint values from these specifications is consistent with the content of similar Test Exceptions in the ISTS.

   The removal of this information from the technical specifications listed above facilitates the re-organization of the RTS technical specification requirements previously described in the LAR. Specifically, the RTS instrument trip setpoint values are being moved to the LRM. The RTS technical specification bases will continue to describe the operability requirements for the RTS instrumentation including requirements specific to the trip setpoint of each instrument function. The change described above simplifies the BVPS technical specifications by keeping the RTS instrumentation operability requirements within the RTS technical specification and bases.

5. In order to facilitate the relocation of the Trip Setpoints to the LRM and movement of Allowable Values to a combined instrumentation Table in the

RTS and ESFAS TS, the following additional and less significant administrative changes are proposed:

a) Changes to the TS index pages are required to reflect the proposed elimination of the LSSS section of the TS and the elimination of the TS Tables 2.2-1 and 3.3-4.

b) Certain TS title pages affected by the elimination of the separate LSSS TS section are revised or eliminated to reflect this change.

c) The format of retained RTS and ESFAS instrumentation Tables (3.3-1 and 3.3-3) is substantially revised to accommodate the addition of an Allowable Value column to these Tables. Additionally, the more comprehensive list of instrument functions on these Tables requires that the term "N.A." be added to the Allowable Value column where no setpoint is associated with the instrument function. For example, the current BVPS Unit No. 1 setpoint Tables (2.2-1 and 3.3-4) do not always include all functions such as manual switches which do not have a setpoint associated with them. When the BVPS Unit No. 1 setpoint information is transferred to the more comprehensive instrument function list on Tables 3.3-1 and 3.3-3 a line item is created in the Allowable Value column for all functions. In cases where no setpoint is associated with the function (e.g., manual switches) the term "N.A." or "Not Applicable" has been added to the Allowable Value column. This convention is consistent with the ISTS and BVPS Unit No. 2 presentation of this information. In order to keep the marked-up TS in Attachment A of this LAR clear and readable this level of detail is not specifically shown. These changes make both units more consistent with each other and with the ISTS.

d) The proposed changes include revisions to the TS bases that result from relocating the Trip Setpoints, moving the Allowable Values, and eliminating the LSSS section of the TS. The bases changes include moving the instrumentation bases from Section 2.2.1 (which was deleted) to the RTS TS bases in Section 3. In addition, the changes include new bases text applicable to the adjustment and maintenance of

the Trip Setpoints that is consistent with the guidance in the ISTS and TSTF-355 for instrumentation operability details.

e) Numerous changes to page headers and footers are proposed to address the deletions and revisions previously discussed. These changes are shown in the marked-up pages included in Attachment A of this LAR and include all the pagination changes necessary to ensure the revised pages are properly integrated into the TS document.

The administrative changes described above are made to support the relocation of the Trip Setpoints and the movement of the Allowable Values into comprehensive instrument operability Tables. The proposed revisions do not introduce technical changes to the TS and only affect the format and presentation of the BVPS TS. In addition, the proposed revisions serve to increase consistency between the BVPS units and with the ISTS. Therefore, these changes are acceptable.

## 5. MISCELLANEOUS

The proposed deletion of the various references to two loop operation and the loop stop valves being open for BVPS Unit No. 1 is administrative in nature. Plant operation in Modes 1 and 2 with less than three reactor coolant system loops in service is not permitted by LCO 3.4.1.1. In addition, LCO 3.4.1.4.1 requires all reactor coolant loop isolation valves to be open during plant operation in Modes 1 through 4. Based on the requirements for loop operation and loop isolation valve position contained in LCOs denoted above, the license condition which prohibits two loop operation along with the other references to two loop operation and loop isolation valve position are no longer necessary and can be deleted.

In the BVPS Unit No. 1 ESFAS tables associated with Technical Specification 3.3.2.1, the function for Auxiliary Feedwater (AFW) Pump Start on emergency bus undervoltage is deleted because the undervoltage relay does not directly start the motor-driven AFW pumps. The undervoltage start of the motor-driven AFW pumps is accomplished indirectly via a combination of 1) Emergency Bus feed breaker opening 2) valid technical specification required start signal from (Steam Generator low-low level, Safety Injection (SI) or trip of all operating main feedwater pumps), and 3) diesel generator sequencer actuation. During the above occurrences, electrical power to the motor-driven AFW pumps is shed. The emergency diesel

generator is started. The diesel generator (EDG) sequencer is then activated to add predetermined safety loads which include the automatic start of the motor-driven AFW pumps. Diesel start and sequencer operation on loss of emergency bus voltage and auxiliary feedwater pump start on the three actuation signals denoted above are included in the ESFAS related technical specifications, Table 3.3-3 and 3.3-4 items 7.a, 7.c, 7.e, and EDG related TS 4.8.1.1.2.b.3 (b).

NUREG 0800, "Standard Review Plan" (SRP), Appendix 8A, BTP PSB-1, "Adequacy of Station Electric Distribution System Voltages," requires that when an undervoltage condition has been detected, Class 1E loads are to be shed from the bus, and at BVPS Unit No. 1, the diesel generators are started. Loading of the emergency bus commences after the correct logic has been sent to the sequencer.

Editorial changes including typographical corrections and format changes were made as required. These type of changes do not affect plant safety.

The bases changes are made to ensure consistency between the proposed technical specifications and the bases. The additional information is intended to make the technical specifications less ambiguous.

The additional miscellaneous changes proposed in this LAR are intended to improve the internal consistency of the BVPS TS, simplify the presentation of requirements, provide clarifications, and improve consistency with the ISTS. A more detailed discussion of each additional change follows:

1. The current BVPS RTS and ESFAS instrumentation TS contain tables that list the instrument functions and specify various requirements applicable to each function. Each table contains a list of the functions addressed by the TS but does not always describe each function exactly the same. Slight variations in wording or presentation exist between table descriptions for the same instrument function. This LAR contains changes to the instrument function descriptions that are intended to make the instrument descriptions consistent between table lists. The specific changes made to the instrument function descriptions are identified in the marked-up table pages in Attachment A of the LAR. The proposed changes do not revise the technical requirements applicable to the affected functions. The changes are considered minor and editorial in nature. The proposed changes are intended to improve the clarity

and presentation of the TS information in the instrument tables. Therefore, these changes are acceptable.

2. The RTS Overtemperature and Overpressure ΔT Instrument functions consist of equations with a list of the terms used in the equations. The additional miscellaneous changes in this LAR include the reformat and simplification of these equations consistent with the ISTS. Other changes in this LAR move the equations and propose technical revisions to the terms used in these equations and are discussed in Section 1 "Revision to Setpoint and Allowable Values" of this LAR. The proposed format change for these equations will simplify the presentation of the technical information and improve consistency with the corresponding requirements in the ISTS. The specific changes to the presentation of the equations are shown in the marked-up pages in Attachment A of this LAR. As the reformat of these equations follows the guidance of the NRC approved ISTS, this change is acceptable.

3. The values for the DNB parameters specified in TS 3.2.5 (with the exception of minimum total RCS flow) are being relocated by a change discussed in Section 3 of this LAR. In addition to relocating the parameter values, several administrative changes are proposed to TS 3.2.5 to improve consistency with the ISTS. The parameter Table (3.2-1) would be deleted and the parameters addressed by the TS would be listed in the LCO statement. Existing Note 1 for Pressurizer Pressure in Table 3.2-1 is moved directly into the TS and is referenced from the LCO requirement for pressurizer pressure. As the parameter values are being relocated, the elimination of the parameter table is a simplification of the TS format that is reasonable and consistent with the ISTS presentation of this information. Additionally, changes are made to the Action statement applicable to the DNB parameters. The Action statement is revised to apply to "one or more" DNB parameters instead of "with any" when referring to DNB parameters not within limits. The required completion time for the Action (2 hours) remains unchanged and applies to all DNB parameters not within the specified limit. This change utilizes the more precise wording of the corresponding ISTS DNB requirements to clarify the application of the Action and the 2-hour completion time in the TS. The changes proposed to the Action statement are made solely to improve the clarity of the Action statement and to make the BVPS TS more consistent with the ISTS. The DNB surveillance requirements are revised consistent with the other changes made to

the LCO and the elimination of the DNB parameter table. The proposed revisions to the surveillances also improve consistency with the ISTS. As the proposed changes to the DNB TS follow the guidance of the NRC approved ISTS, these changes are acceptable.

4. The Action statements for the RTS and ESFAS TS are revised by the addition of Note 1 (for BVPS Unit No. 1) and Note 2 (for BVPS Unit No. 2) that states "Separate Action statement entry is allowed for each Function." The proposed note provides guidance applicable to the multiple Actions contained within the RTS and ESFAS TS and is consistent with the guidance provided in the ISTS for these TS. The ISTS note provides more specific guidance for the application of the Actions when more than one instrument function is inoperable. The note clarifies that each instrument function is treated independently and the Actions are applied separately for each inoperable instrument function. The clarification provided by the ISTS note is applicable to the format of the current BVPS RTS and ESFAS TS. Similar to the ISTS, the current BVPS TS provide Actions that are applied separately to each inoperable function. The clarification provided by the ISTS note is applicable to the BVPS TS Action statements as well as the ISTS Action Conditions. As the proposed change improves clarity and follows the guidance of the NRC approved ISTS, the change is acceptable.

5. The BVPS Unit No. 2 RTS Table 3.3-1 column header format is revised to be more consistent with the column headers in other tables. The header is revised to contain underlines for the column titles only instead of a continuous underline of the entire header. This change simply makes the BVPS Unit No. 2 table look the same as the other instrumentation tables and does not affect the technical content of the table. Therefore, this change is acceptable.

6. Changes to the RTS and ESFAS bases are proposed to remove setpoint values from the bases text and replace them with references to the setpoint. The Allowable Values continue to be specified in the TS and the Trip Setpoints will be specified in the LRM as previously discussed. The removal of specific values from the bases discussions reduces the effort required to revise the TS when the setpoint values are changed and results in a TS bases more consistent with the ISTS bases. This change simply reduces the repetition of values within the TS bases that may be subject to future changes. The proposed

change will reduce the potential for errors or inconsistencies in the bases if the specified values are revised. As this change does not alter the values specified in the TS and simply limits repetition of the specified values, it is acceptable.

D. SAFETY ANALYSIS

**1. REVISION TO SETPOINT AND ALLOWABLE VALUES**

By incorporating the setpoints in WCAPs 11366, Rev. 4 and 11419, Rev. 2, into the technical specifications, the assumptions in the safety analysis relative to initiation of RTS and ESFAS are preserved. The changes in format relative to the presentation of the nominal trip setpoints and allowable values and the proposed revisions to LSSS 2.2.1 and LCO 3.3.2.1 have no impact on the safety analysis. These changes are consistent with the assumptions in the setpoint methodology WCAPs and the assumptions for time constants modeled in dynamic transfer functions that are credited in the safety analysis.

The retention of the inequality in the setpoint for the turbine stop valve position trip channel has no safety analysis impact because a trip setpoint which is anywhere in the range permitted by the inequality will be conservative. Furthermore, the trip function is not credited as a primary trip in the safety analysis.

The revision to the Steam Generator Water Level High-High and Steam Generator Water Level Low-Low trip setpoints were made to provide operational margin. Explicit values were calculated for the revised allowable values using either calibration tolerance bands or equipment performance specification. The proposed changes to the RTS and ESFAS trip setpoints and allowable values in the technical specifications do not adversely impact the conclusions for credited events as documented in the UFSAR.

The proposed deletion of the lag compensation for $\Delta T$ and Tavg in the Overtemperature and Overpower trip setpoint equations and associated time constants (for BVPS Unit No. 1 only) reflects the current hardware configuration of the BVPS Unit No. 1 plant protection system hardware and does not have any impact on any safety analysis result.

The proposed amendment does involve a hardware change. The hardware change involves the deletion of $f(\Delta I)$ (BVPS Unit No. 1) and $f2(\Delta I)$ (BVPS Unit No. 2) for the Overpower $\Delta T$ Trip Setpoint. This function is not modeled in the safety analysis

nor included in the setpoint methodology calculation. Defeating this function, rather than leaving it in the equation with a setting of zero, eliminates the possibility that it will adversely contribute to the Overpower ΔT Trip due to the limitations of the hardware and possible variations in the setpoint.

For BVPS Unit No. 1 only, the proposed changes for T′ and T" from 576.3°F to 576.2°F results in a reduced Overtemperature and Overpower ΔT trip setpoint. At the current value of 576.3°F for T′ and T", adequate margin still exists such that the plant's safety analyses are not affected. However, these values are being revised to maintain the setpoint channel margins specified in WCAP-11419 Revision 2.

The proposed change (for BVPS Unit No. 2 only), to delete the values and column headings for "Total Allowance (TA)", "Z", and "S", specified in Tables 2.2-1 and 3.3-4, Equation 2.2-1 in LSSS 2.2.1 and Note 5 specified in Table 2.2-1, has no safety analysis impact. This change is consistent with the assumptions in the methodology used to calculate the trip setpoints.

The proposed editorial, administrative and format changes do not affect plant safety.

Therefore, based on the above, the proposed amendment does not adversely affect the safe operation of the plant.

**2. RTDP**

The Revised Thermal Design Procedure (RTDP) includes uncertainties based on surveillance data associated with core power, RCS temperature, flow and pressure. This methodology was approved by the NRC and uses the same methodology as defined in WCAP-8567, "Improved Thermal Design Procedure" [Reference 9]. With the core limit change, the values of the Overpower and Overtemperature ΔT reactor trip setpoints are changed in the analysis. The pressure, temperature and flow DNB parameters were also revised. A review of the BVPS Unit Nos. 1 and 2 UFSAR was performed to determine those events sensitive to changes in the Overtemperature and Overpower ΔT setpoints. Each of the events has been evaluated to determine if the various acceptance criteria were met. The limits on the DNB parameters assure that the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the UFSAR assumptions and have been demonstrated adequate for maintaining the required minimum DNBR above the applicable design

limits throughout each analyzed transient. In all cases, the acceptance criteria were met, and therefore the margin of safety is maintained.

The proposed editorial and format changes do not introduce any technical changes and therefore have no impact on plant safety.

Therefore, based on the above, the proposed amendment does not adversely affect the safe operation of the plant.

## 3. RELOCATION TO COLR

Replacing the core safety limit figure with a statement of the design criteria that is used to generate the safety limit has no effect on the safety analysis. Cycle-specific core limits will continue to be generated in the same way as in the past and will be accessible in the Core Operating Limits Report. Specific action to be taken in the event of a core safety limit violation continues to be included in the technical specifications.

The relocation of the specific parameters that define the Overtemperature and Overpower ΔT trip setpoints have no impact on the safety analysis because the resulting trip setpoints remain unchanged. All of the accident transients that rely on these reactor trips will continue to be protected by the Overtemperature and Overpower ΔT trip functions. The specific trip function surveillance requirements and action restrictions will continue to be specified in the technical specifications for operable channels as well as action restrictions for inoperable channels.

Relocation of the DNB parameter limits will have no impact on the safety analysis because the calculation of the limits will continue to use the same NRC-approved methodology as was used to calculate the previous limits. The specific surveillances and action restrictions will continue to be specified in the technical specifications for these parameter limits.

Therefore, based on the above, the proposed amendment does not adversely affect the safe operation of the plant.

## 4. RELOCATION TO LRM

BVPS proposes to relocate the nominal trip setpoint information for the RTS and ESFAS instrumentation in Tables 2.2-1 and 3.3-4 to the LRM. As previously explained, this change is based on the requirements of 10 CFR 50.36 and the guidance contained in the ISTS and TSTF-355. In addition, BVPS proposes to move the remaining RTS and ESFAS Allowable Values from Tables 2.2-1 and 3.3-4 to Tables 3.3-1 and 3.3-3. The end result of this move will be a single more comprehensive instrumentation function table in the RTS and ESFAS TS similar to the ISTS presentation of this information. As a result of the reorganization of the TS instrumentation tables described above, a number of administrative changes were also proposed and described in detail in the Justification section of this LAR.

The proposed relocation of the TS Trip Setpoints does not reduce the effectiveness of the TS requirements being relocated. Rather, the relocation of the requirements result in a change in the regulatory control required for future changes made to the setpoints. The Trip Setpoint requirements will continue to be implemented by the appropriate plant procedures (e.g., operating and maintenance procedures) in the same manner as before. The revised setpoint methodology, described elsewhere in this LAR, establishes the Allowable Value as the required LSSS and TS instrument setpoint operability criterion to support the existing plant safety analyses. The Allowable Value or LSSS is retained within the TS consistent with the requirements of 10 CFR 50.36. As such, instrumentation operability continues to be determined in accordance with the applicable setpoint methodology and the proposed change does not adversely impact any plant safety analysis described in the UFSAR. Future changes to the relocated requirements will be controlled in accordance with 10 CFR 50.59 instead of requiring a license amendment per 10 CFR 50.90. The relocated TS requirements in the LRM are incorporated by reference in the UFSAR and the LRM has been previously determined by the NRC to be acceptable for relocated TS in BVPS License Amendment Numbers 233/115 (SER dated 9/7/00). As such, adequate regulatory control of the destination documents exists to ensure that future changes to these requirements are controlled by the provisions of 10 CFR 50.59 and that prior NRC review and approval of changes will be requested when required per 10 CFR 50.59. The provisions of 10 CFR 50.59 establish adequate controls for material relocated from the TS, including record retention and reporting requirements. The provisions of 10 CFR 50.59 assure future changes to the relocated material will be consistent with safe plant operation.

The remainder of the changes associated with the reorganization of the instrument tables have all been evaluated as administrative changes that do not have a technical impact on the TS requirements. These additional changes include the following:

1. Movement of the Allowable Values within the TS Tables,
2. Elimination of TS 2.2.1,
3. The re-numbering of the Overtemperature and Overpower ΔT equation notes,
4. The revision of the ESFAS LCO and Actions,
5. Deleting the trip setpoint requirement for the intermediate and power range instrumentation specified in the LCO for the Special Test Exceptions, and
6. Other miscellaneous index or title page, table format, pagination, and bases changes made in support of the changes described above.

These administrative changes have been determined not to impact the effectiveness of the TS requirements and were made to enhance the usability and clarity of the TS consistent with the guidance in the ISTS. As such, these changes do not adversely affect the conclusions or description of any safety analysis described in the UFSAR.

Therefore, based on the above, the proposed amendment does not adversely affect the safe operation of the plant.

## 5. MISCELLANEOUS

For BVPS Unit No. 1 only, the proposed deletion of the references to two loop operation and the loop stop valves being open will not affect plant safety. LCO 3.4.1.1 and associated surveillance requirement ensure that the plant is not operated with less than three loops in operation in Modes 1 and 2. LCO 3.4.1.4.1 and associated surveillance requirement ensure that the loop stop valves remain open.

The deletion of the trip function for undervoltage start of the motor-driven AFW pumps (for BVPS Unit No. 1 only) does not have any impact on any safety analysis result.

In the course of incorporating the other more significant changes included in this LAR, several additional changes were proposed. These additional changes are not

technical but affect the same TS as the other more technical changes addressed by this LAR. These additional changes are proposed to improve the internal consistency of the BVPS TS, simplify the presentation of requirements, provide clarifications, and improve consistency with the ISTS. The additional proposed changes include the following:

1. 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,

2. Simplification and reformat of the Overtemperature and Overpower $\Delta T$ equations consistent with the ISTS,

3. Revision of TS 3.2.5, DNB Parameters, LCO, Actions, and Surveillance Requirements to improve consistency with the ISTS,

4. Clarification of the RTS and ESFAS Actions with ISTS note allowing separate Action statement entry for each instrument function,

5. Reformat of BVPS Unit No. 2 RTS Instrument Table 3.3.1 header to make the header more consistent with other instrument tables in the TS, and

6. Revision of instrumentation Bases text to eliminate the repetition of specific setpoint values in the Bases discussions.

The proposed changes listed above are administrative in nature and have been determined not to impact the effectiveness of the TS requirements. As previously stated, the proposed changes were made to improve internal consistency, enhance usability and clarity and improve consistency with the ISTS. The proposed changes do not revise any technical requirements or values specified in the TS. As such, these changes do not adversely affect the conclusions or description of any safety analysis described in the UFSAR.

Therefore, based on the above, the proposed amendment does not adversely affect the safe operation of the plant.

## E. NO SIGNIFICANT HAZARDS EVALUATION

The no significant hazard considerations involved with the proposed amendment have been evaluated. The evaluation focusing on the three standards set forth in 10 CFR 50.92(c) are as quoted below:

> The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:
>
> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> (3) Involve a significant reduction in a margin of safety.

The following evaluations are provided for the no significant hazards consideration standards.

### I. REVISION TO SETPOINT AND ALLOWABLE VALUES

The proposed amendment for the Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) trip setpoints and allowable values will eliminate the need for the current administrative controls to ensure that the trip setpoints are maintained consistent with safety analysis assumptions. The proposed amendment includes changes in the trip setpoint values and allowable values which are more restrictive and less restrictive than the current technical specification values.

For Beaver Valley Power Station (BVPS) Unit No. 1 only, time constants $\tau_4$ and $\tau_5$ and associated Tavg and $\Delta T$ lag compensator functions would be deleted that are specified in Notes 1 and 2 of Table 2.2-1 for the Overtemperature and Overpower $\Delta T$ trip setpoint.

The f(ΔI) (BVPS Unit No. 1) and f2(ΔI) (BVPS Unit No. 2) would be deleted from the equation of the Overpower ΔT Trip Setpoint.

The changes to the bases are necessary to preserve consistency with the changes to the technical specifications and to provide supporting information to clarify the intent of the proposed technical specifications.

The Bases for Limiting Condition For Operation (LCO) 3.3.1.1 and LCO 3.3.2.1 will be revised to address the calibration tolerance band. This proposed change provides a clear delineation of the design analysis requirements as they are translated into the plant technical specifications. The proposed amendment will facilitate a two-sided calibration tolerance around the trip setpoint. The uncertainty calculations, utilized in the development of the required trip setpoint, assume that the "as left" tolerance (both conservative and non-conservative direction) is satisfied on a reasonable, statistical basis. It is acceptable for the "as left" condition, immediately after calibration of the process rack modules or the bistable, to be in the non-conservative direction provided that the magnitude is within the calibration tolerance band specified in plant procedure(s).

Editorial, administrative and format changes have also been included in this proposed amendment.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The RPS and ESFAS trip functions are part of the accident mitigation response and are not themselves an initiator for any transient. Therefore, the probability of an accident previously evaluated is not significantly affected.

   This proposed amendment includes changes to RTS and ESFAS trip setpoints and allowable values that have been determined with the use of an approved methodology. The new values ensure that all automatic protective actions will be initiated at or before the condition assumed in the safety analysis. This change, which includes modification of the applicable Bases section(s), will allow the nominal trip setpoints to be adjusted within the calibration tolerance band allowed by the setpoint methodology. Plant operation with these revised values will not cause any design or analysis acceptance criteria to be exceeded. The structural and functional integrity of plant systems is

unaffected. There will be no adverse effect on the ability of the channels to perform their safety functions as assumed in the safety analyses. Since there will be no adverse effect on the trip setpoints or the instrumentation associated with the trip setpoints, there will be no significant increase in the consequences of any accident previously evaluated.

The proposed amendment does involve a hardware change. The hardware change involves the deletion of f(ΔI) (BVPS Unit No. 1) and f2(ΔI) (BVPS Unit No. 2) for the Overpower ΔT Trip Setpoint. This function is not modeled in the safety analysis nor included in the setpoint methodology calculation. Defeating this function, rather than leaving it in the equation with a setting of zero, eliminates the possibility that it will adversely contribute to the Overpower ΔT Trip due to the limitations of the hardware and possible variations in the setpoint.

Other changes in trip system function, content and format are proposed based on the current configuration of the trip system hardware at BVPS Unit No. 1. Similarly, since the ability of the instrumentation to perform its safety function is not adversely affected, there will be no significant increase in the consequences of any accident previously evaluated.

The proposed editorial, administrative and format changes do not affect plant safety.

Therefore, this change does not involve any significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed amendment includes changes to the format and magnitudes of nominal trip setpoints and allowable values that preserve all safety analysis assumptions related to accident mitigation. The protection system will continue to initiate the protective actions as assumed in the safety analysis. The proposed changes to Limiting Safety System Settings (LSSS) 2.2.1 and LCO 3.3.2.1 will continue to ensure that the trip setpoints are maintained consistent with the setpoint methodology and the plant safety analysis. The proposed amendment does involve a hardware change. The hardware change

involves the deletion of f(ΔI) (BVPS Unit No. 1) and f2(ΔI) (BVPS Unit No. 2) for the Overpower ΔT Trip Setpoint. This function is not modeled in the safety analysis nor included in the setpoint methodology calculation. Defeating this function, rather than leaving it in the equation with a setting of zero, eliminates the possibility that it will adversely contribute to the Overpower ΔT Trip due to the limitations of the hardware and possible variations in the setpoint. Therefore, this hardware change does not create the possibility of a new or different kind of accident from any accident previously evaluated. Plant operation will not be changed.

Other proposed changes are made so that the technical specifications more accurately reflect the plant-specific trip system hardware in BVPS Unit No. 1.

Furthermore, the proposed changes do not alter the functioning of the RTS and ESFAS.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The proposed RTS and ESFAS trip setpoints are calculated with an approved methodology. The proposed changes to LSSS 2.2.1 and LCO 3.3.2.1 will continue to ensure that the trip setpoints are maintained consistent with the setpoint methodology and the plant safety analysis. Therefore, the response of the RTS and ESFAS to accident transients reported in the Updated Final Safety Analysis Report (UFSAR) is unaffected by this change. This proposed amendment does involve a hardware change. The hardware change involves the deletion of f(ΔI) (BVPS Unit No. 1) and f2(ΔI) (BVPS Unit No. 2) for the Overpower ΔT Trip Setpoint. This function is not modeled in the safety analysis nor included in the setpoint methodology calculation. Defeating this function, rather than leaving it in the equation with a setting of zero, eliminates the possibility that it will adversely contribute to the Overpower ΔT Trip due to the limitations of the hardware and possible variations in the setpoint. Therefore, accident analysis acceptance criteria are not affected. Other proposed changes are made so that the protection system technical specifications more accurately reflect the plant-specific trip system hardware in BVPS Unit No. 1.

The proposed editorial, administrative, and format changes do not affect plant safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

## II.A REVISED THERMAL DESIGN PROCEDURE (RTDP) – OVERTEMPERATURE ΔT AND OVERPOWER ΔT

Revised core thermal limits reflected in Figure 2.1-1 for Units 1 and 2 were generated employing the RTDP methodology. Overtemperature and Overpower ΔT setpoints and associated uncertainties were calculated based on the new core thermal limits. A review of the Beaver Valley's UFSAR was performed to determine those events sensitive to changes in the Overtemperature and Overpower ΔT setpoints. Each of the events has been analyzed to determine if the various acceptance criteria were met. In all cases, the acceptance criteria were met.

The proposed amendment includes the following revisions:

- Revision of Figure 2.1-1 contained in Technical Specification 2.1.1, "Reactor Core Safety Limits Three Loops Operation" and associated Bases section;
- Revision of the K1 and K4 values contained in the Overtemperature ΔT and Overpower ΔT trip setpoint equations contained in Technical Specification Table 2.2-1, "Reactor Trip System Instrumentation Setpoints."

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The revised Figure 2.1-1 and the Overtemperature and Overpower ΔT reactor trip functions do not involve a significant increase in the probability or consequences of an accident previously evaluated because operation with these revised values will not cause any design or analysis acceptance criteria to be exceeded. The structural and functional integrity of all plant systems is unaffected. The Overtemperature and Overpower ΔT reactor trip functions are part of the accident mitigation response and are not themselves an initiator for any transient. Therefore, the probability of occurrence previously evaluated is not significantly affected.

The changes to Figure 2.1-1 and to the Overtemperature and Overpower ΔT reactor trip functions do not affect the integrity of the fission product barriers utilized for mitigation of radiological dose consequences as a result of an accident. Figure 2.1-1 provides restrictions to prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. It does not provide an automatic protective function but does provide the basis for the Overtemperature and Overpower ΔT reactor trip functions. These trip functions ensure that automatic protective actions will be initiated at or before the condition assumed in the safety analyses. These changes produce no adverse effect on the ability of these functions to perform their safety functions assumed in the safety analyses. In addition, the off-site mass releases used as input to the dose calculations are unchanged from those previously assumed. Therefore, the off-site dose predictions remain within the acceptance criteria of 10 CFR 100 limits for each of the transients affected. Since it has been concluded that the transient analyses results are unaffected by the parameter modifications, it is concluded that the probability or consequences of an accident previously evaluated are not significantly increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The revised Figure 2.1-1 and Overtemperature and Overpower ΔT reactor trip functions do not create the possibility of a new or different kind of accident from any accident previously evaluated because the setpoint adjustments do not affect accident initiation sequences. No new operating configuration is being imposed by the setpoint adjustments that would create a new failure scenario. In addition, no new failure modes or limiting single failures have been identified. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The changes to Figure 2.1-1 and to the Overtemperature and Overpower ΔT reactor trip functions do not involve a significant reduction in a margin of safety because the margin of safety associated with the Overtemperature and Overpower ΔT reactor trip functions, as verified by the results of the accident analyses, are within acceptable limits. All transients impacted by implementation of the RTDP methodology have been analyzed and have met the applicable accident analyses acceptance criteria. The margin of safety required for each affected safety analysis is maintained. This conclusion is not changed by the Overtemperature and Overpower ΔT setpoint modifications. The adequacy of the revised technical specification values to maintain the plant in a safe operating condition has been confirmed.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

## II.B RTDP - DEPARTURE FROM NUCLEATE BOILING (DNB) PARAMETER SURVEILLANCE REQUIREMENTS

The proposed amendment includes the following revisions:

- Revision of DNB parameter values specified in Table 3.2-1 titled "DNB Parameters" contained in Technical Specification (TS) 3/4.2.5, titled "Power Distribution Limits-DNB Parameters";
- Revision of footnote (1) and the addition of a new footnote (3) in Technical Specification 3/4.2.5.

The revised DNB Parameters provide additional operational margin to the limits associated with indicated Reactor Coolant $T_{avg}$, the indicated pressurizer pressure, and the measurement and indication of Reactor Coolant System (RCS) Flow by the use of indicated limits instead of analytical limits.

The limits on the DNB parameters assure that the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limit is consistent with the UFSAR assumptions and has been demonstrated to be adequate to maintain the required minimum Departure from

Nucleate Boiling Ratio (DNBR) above the applicable design limits throughout each analyzed transient. In all cases, the acceptance criteria were met, and therefore the margin of safety is maintained.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The intent of the change is to preserve the Safety Analyses Limits for DNB (TS 3/4.2.5). There is no increase in the probability or consequences of an accident previously evaluated because there is no change to any design or analysis acceptance criteria. The structural and functional integrity of any plant system is unaffected. The proposed license amendment revises the surveillance requirement acceptance criteria for the DNB parameters. The indicated DNB parameters preserve the assumptions used in the accident analysis and, therefore, there is no significant increase in probability or consequences previously evaluated.

   The changes to the DNB parameters do not affect the integrity of the fission product barriers utilized for mitigation of radiological dose consequences as a result of an accident. In addition, the off-site mass releases used as input to the dose calculations are unchanged from those previously assumed. Therefore, the off-site dose predictions remain within the limits of the 10 CFR 100 for each of the transients affected. Since it has been determined that the transient results are unaffected by these parameter modifications, it is concluded that the consequences of an accident previously evaluated are not significantly increased.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The revised DNB parameter values do not create the possibility of a new or different kind of accident from any accident previously evaluated. The setpoint values do not affect the assumed accident initiation sequences. No new operating configuration is being imposed by changing these parameters that would create a new failure scenario. In addition, no new failure modes

or single failures have been identified for any plant equipment. Therefore, the types of accidents defined in the UFSAR continue to represent the credible spectrum of events to be analyzed which determine safe plant operation.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The DNB parameters are consistent with the UFSAR assumptions and maintain the required minimum DNBR above the design limits throughout each analyzed transient. Thereby, the adequacy of the revised DNB parameter values to maintain the plant in a safe operating condition is confirmed.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

## III. RELOCATION TO COLR

This portion of the amendment request proposes that the cycle-specific Reactor Coolant System related technical specification parameter limits be relocated to the Core Operating Limits Report (COLR). This change will allow the Beaver Valley Power Station (BVPS) the flexibility to enhance plant operating margin and/or core design margins without the need for cycle-specific license amendment requests. The technical specification changes include: Relocation of core safety limits, Overtemperature ΔT, Overpressure ΔT, and DNB parameter values from the technical specifications to the Core Operating Limits Report. Revising Technical Specification 6.9.5, "Core Operating Limits Report," to reflect the relocations to the COLR and to add appropriate references.

The proposed change is consistent with the Nuclear Regulatory Commission (NRC) guidance for control of the Core Operating Limits Report as required in Generic Letter 88-16 and the NRC Safety Evaluation Report that approved WCAP 14483. The methodologies for calculating these parameters have been approved by the NRC (Safety Evaluation relating to Topical Report WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," approved January 19, 1999) and are consistent with the applicable limits in the Updated Final Safety

Analysis Report (UFSAR). The current technical specification method for controlling reactor physics parameters to assure conformance to 10 CFR 50.36 (which requires the lowest functional levels acceptable for continued safe operation) is to specify the values determined to be within the acceptance criteria using an NRC approved calculation methodology. If approved, the proposed amendment will not alter the input parameters or the methodologies for calculating these limits.

This proposed change would control the cycle-specific parameters within the acceptance criteria and assure conformance to 10 CFR50.36 by using the approved methodology instead of specifying technical specification values. The COLR will document the specific parameter limits resulting from NRC-approved calculations, including mid-cycle or other revisions to parameter values. Therefore, the proposed change is in conformance with the requirements of 10 CFR 50.36.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed amendment is a programmatic and administrative change that does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions. Because the design of the facility and system operating parameters are not being changed, the proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated.

   The cycle-specific values relocated into the COLR will continue to be controlled by the BVPS programs and procedures. Each accident analysis addressed in the UFSAR will be examined with respect to changes in the cycle-dependent parameters, which are obtained from the use of NRC approved reload design methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination, which will be conducted per the requirements of 10 CFR 50.59, will ensure that future reloads will not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed amendment is a programmatic and administrative change and does not result in any change in the manner in which the plant is operated or the way in which the Reactor Protection System provides plant protection. All of the accident transients analyzed in the UFSAR will continue to be protected by the same trip functions with the required trip setpoints. Removal of the cycle specific variables has no influence or impact on, nor does it contribute in any way to the probability or consequences of an accident. No safety-related equipment, safety function, or plant operation will be altered as a result of this proposed change. The cycle specific variables are calculated using the NRC approved methods, and submitted to the NRC to allow the staff to continue to review the values of these limits. The technical specifications will continue to require operation within the core operating limits, and appropriate actions will be required if these limits are exceeded.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

   The proposed change to relocate core safety limits, trip setpoint parameter values, and DNB parameter values to the Core Operating Limits Report represents an administrative change and no hardware changes are involved; therefore, no accident analysis acceptance criteria are affected.

   The margin of safety is not affected by the removal of cycle-specific core operating limits from the technical specifications. The margin of safety presently provided by current technical specifications remains unchanged. Appropriate measures exist to control the values of these cycle specific limits. The proposed amendment continues to require operation within the core limits as obtained from NRC-approved methodologies, and the actions to be taken if a limit is exceeded will continue to require that the plant be placed in Hot Standby within one hour.

   The development of the limits for future reloads will continue to conform to those methods described in NRC approved documentation. In addition, each

future reload will involve a 10 CFR 50.59 safety review to assure that operation of the unit within the cycle-specific limits will not involve a significant reduction in the margin of safety.

The proposed amendment is a programmatic and administrative change that provides assurance that plant operations continue to be conducted in a safe manner. The proposed amendment does not result in any change in the manner in which the plant is operated or the way in which the Reactor Protection System (RPS) provides plant protection. The proposed relocation does not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operation are determined. Therefore, the response of the RPS to accident transients described in the UFSAR is unaffected by this change.

As stated previously, this portion of the proposed amendment does not physically alter safety-related systems, nor does it affect the way in which safety-related systems perform their functions.

The accident transients are unaffected and the safety analysis acceptance limits are unaffected. The design of the facility and system operating parameters are not being changed.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

## IV. RELOCATION TO LICENSING REQUIREMENTS MANUAL (LRM)

The proposed change relocates the nominal trip setpoint information for the RTS and ESFAS instrumentation in Tables 2.2-1 and 3.3-4 to the LRM. This change is based on the requirements of 10 CFR 50.36 and the guidance contained in the Improved Standard Technical Specifications for Westinghouse Plants (ISTS) including NRC approved change Technical Specification Traveler Form (TSTF) - 355. In addition, the proposed change includes moving the remaining RTS and ESFAS Allowable Values from Tables 2.2-1 and 3.3-4 to Tables 3.3-1 and 3.3-3. The end result of this move will be a single more comprehensive instrumentation function table in the affected technical specifications similar to the ISTS presentation of this information. As a result of the reorganization of the TS instrumentation tables described above, the proposed changes include a number of

revisions that are administrative in nature and that have no technical impact on the technical specifications. These changes include the revision of certain Special Test Exceptions contained in technical specification Section 3/4.10.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change does not involve a significant increase in the probability of an accident previously evaluated because no changes are being made to any event initiator. Nor is any analyzed accident scenario being revised. The initiating conditions and assumptions for accidents described in the UFSAR remain as previously analyzed.

   The proposed change also does not involve a significant increase in the consequences of an accident previously evaluated. The change does not reduce the operability requirements for the affected instrumentation. The proposed relocation of TS requirements only affects the level of regulatory control involved in future changes to the requirements. The instrument setpoints will continue to be maintained in a similar manner as before. The conclusions and descriptions of the safety analyses described in the UFSAR remain unchanged.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve any physical changes to the plant or the modes of plant operation defined in the technical specifications. The proposed amendment does not involve the addition or modification of plant equipment nor does it alter the design or operation of any plant systems. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes.

   There are no changes in this amendment which would cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new mode of failure has been created and no new equipment performance

requirements are imposed. The proposed amendment has no effect on any previously evaluated accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The margin of safety depends on the maintenance of specific operating parameters and systems within design requirements and safety analysis assumptions.

The proposed change does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed change does not alter the functional capabilities assumed in a safety analysis for any system, structure, or component important to the mitigation and control of design bases accident conditions within the facility. Nor does this change revise any parameters or operating restrictions that are assumptions of a design basis accident. In addition, the proposed change does not affect the ability of safety systems to ensure that the facility can be placed and maintained in a shutdown condition for extended periods of time.

The relocation of TS requirements does not reduce the effectiveness of the requirements being relocated. Rather, the relocation of the TS requirements results in a change in the regulatory control required for future changes made to the requirements. The relocated requirements will continue to be implemented by the appropriate plant procedures (e.g., operating and maintenance procedures) in the same manner as before. However, future changes to the relocated requirements will be controlled in accordance with 10 CFR 50.59 instead of 10 CFR 50.90. The provisions of 10 CFR 50.59 establish adequate controls over requirements removed from the TS and assure future changes to these requirements will be consistent with safe plant operation.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

## V. MISCELLANEOUS

Description of amendment request: The proposed amendment would:

(1) Delete license condition 2.C. from the Beaver Valley Power Station (BVPS) Unit No. 1 operating license and delete unnecessary references to two-loop operation from BVPS Unit No. 1 technical specifications (TSs);

(2) Delete the TS requirement pertaining to Auxiliary Feedwater (AFW) Pump Auto-start on Emergency Bus Undervoltage, for BVPS Unit No. 1 only, to reflect the current configuration of plant hardware;

(3) The additional proposed changes consist of administrative, editorial, format, and bases revisions. The proposed changes are not technical in nature and are intended to improve the internal consistency of the BVPS TS, simplify the presentation of requirements, provide clarifications, and improve consistency with the ISTS. The proposed changes include the following:

a. 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,

b. Simplification and reformat of the Overtemperature and Overpower $\Delta T$ equations consistent with the ISTS,

c. Revision of TS 3.2.5, DNB Parameters, LCO, Actions, and Surveillance Requirements to improve consistency with the ISTS,

d. Clarification of the RTS and ESFAS Actions with ISTS note allowing separate Action statement entry for each instrument function,

e. Reformat of BVPS Unit No. 2 RTS Instrument Table 3.3.1 header to make the header more consistent with other instrument tables in the TS, and

f. Revision of instrumentation Bases text to eliminate the repetition of specific setpoint values in the Bases discussions.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The administrative change, for BVPS Unit No. 1 only, pertaining to two loop operation and Reactor Coolant System isolation valve position, does not affect plant safety. The technical specification requirements in LCOs 3.4.1.1 and 3.4.1.4.1 will continue to prohibit two loop operation and ensure safe plant operation by properly controlling the operation and position of the reactor coolant loops and Reactor Coolant System isolation valves.

The administrative change to delete line item 7.d, pertaining to Auxiliary Feedwater (AFW) Pump Auto-start on Emergency Bus Undervoltage, for BVPS Unit No. 1 only, from TS Tables 3.3-3, 3.3-4, and 4.3-2 will not affect plant safety because this function is not directly initiated by bus undervoltage. Rather, the automatic start of the motor-driven AFW pumps is accomplished by the combination of 1) Emergency Bus feed breaker opening 2) valid start signal from ESFAS, and 3) Emergency Diesel Generator (EDG) sequencer actuation. Requirements for these items are included in the ESFAS related TS, Table 3.3-3 and 3.3-4 items 7.a, 7.c, 7.e, and EDG related TS 4.8.1.1.2.b.3 (b). Therefore, since there is no change made to the plant hardware or its operation and requirements related to the AFW pump auto-start function are maintained elsewhere in the BVPS Unit No. 1 TS, deleting line item 7.d from BVPS Unit No. 1 TS Tables 3.3-3, 3.3-4, and 4.3-2 will not significantly change the probability or consequences of any accident previously evaluated.

The proposed change does not involve a significant increase in the probability of an accident previously evaluated because no changes are being made to any event initiator. Nor is any analyzed accident scenario being revised. The initiating conditions and assumptions for accidents described in the UFSAR remain as previously analyzed.

The proposed change also does not involve a significant increase in the consequences of an accident previously evaluated. The change does not reduce the effectiveness or scope of the affected TS. The proposed changes are administrative in nature and do not affect any technical or equipment operability requirements. The conclusions and descriptions of the safety analyses described in the UFSAR remain unchanged.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve any physical changes to the plant or the modes of plant operation defined in the TS. The proposed amendment does not involve the addition or modification of plant equipment nor does it alter the design or operation of any plant systems. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes.

   There are no changes in this amendment which would cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new mode of failure has been created and no new equipment performance requirements are imposed. The proposed amendment has no effect on any previously evaluated accident.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

   The margin of safety depends on the maintenance of specific operating parameters and systems within design requirements and safety analysis assumptions.

   The proposed change does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed change does not alter the functional capabilities assumed in a safety analysis for any system, structure, or component important to the mitigation and control of design bases accident conditions within the facility. Nor does this change revise any parameters or operating restrictions that are assumptions of a design basis accident. In addition, the proposed change does not affect the ability of safety systems to ensure that the facility can be placed and maintained in a shutdown condition for extended periods of time.

   The administrative change to delete line item 7.d, pertaining to AFW Pump Auto-start on Emergency Bus Undervoltage, BVPS Unit No. 1 only, from TS Tables 3.3-3, 3.3-4, and 4.3-2 will not affect plant safety because this function is not directly initiated by bus undervoltage. Rather, the automatic

start of the motor-driven AFW pumps is accomplished by the combination of (1) Emergency Bus feed breaker opening, (2) valid start signal from ESFAS, and (3) EDG sequencer actuation. Requirements for these items are included in the ESFAS related TS, Table 3.3-3 and 3.3-4 items 7.a, 7.c, 7.e, and EDG related TS 4.8.1.1.2.b.3 (b). Therefore, since there is no change made to the plant hardware or its operation and requirements related to the AFW pump auto-start function are maintained elsewhere in the BVPS Unit No.1 TS, deleting line item 7.d from BVPS Unit No. 1 TS Tables 3.3-3, 3.3-4, and 4.3-2 will not involve a significant reduction in a margin of safety.

The administrative change, for BVPS Unit No. 1 only, pertaining to two loop operation and Reactor Coolant System isolation valve position, does not affect plant safety. The technical specification requirements in LCOs 3.4.1.1 and 3.4.1.4.1 will continue to prohibit two-loop operation and ensure safe plant operation by properly controlling the operation and position of the reactor coolant loops and Reactor Coolant System isolation valve.

The other proposed changes are also administrative in nature and only affect the format or presentation of information in the TS. The proposed changes have no affect on the conclusions or descriptions of the safety analyses described in the UFSAR.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

## F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the considerations expressed above, it is concluded that the activities associated with this license amendment request satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

## G. ENVIRONMENTAL CONSIDERATION

The proposed amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. It has been determined that the proposed amendment involves 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. Accordingly, the proposed amendment meets 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 this proposed amendment.

## H. UFSAR CHANGES

Draft UFSAR changes have been prepared based on the proposed LAR.

## I. REFERENCES

1. Westinghouse "Revised Thermal Design Procedure" WCAP-11397-P-A.

2. Westinghouse "Generic Methodology for Expanded Core Operating Limits Report" WCAP-14483-A.

3. Westinghouse "Setpoint Methodology for Protection Systems for Beaver Valley Power Station - Unit 1," WCAP-11419, Rev. 2, December 2000

4. Westinghouse "Setpoint Methodology for Protection Systems for Beaver Valley Power Station - Unit 2," WCAP-11366, Rev. 4, December 2000.

5. Beaver Valley Units 1 and 2 "Updated Final Safety Analysis Report"

6. Westinghouse "Reference Report Vantage 5 Fuel Assembly," WCAP-10444-P-A, September 1985.

7. Westinghouse "Improved THINC-IV Modeling for PWR Core Design," WCAP-12330-A, September 1991.

8. Westinghouse "Statistical Evaluation of LOCA Heat Source Uncertainty," WCAP-10395, November 1983.

9. Westinghouse "Improved Thermal Design Procedure" WCAP-8567, July 1975.

10. ANSI/ANS Standard 58.4-1979, "Criteria for Technical Specifications for Nuclear Power Stations," and ISA Standard S67.04, 2000, "Setpoints for Nuclear Safety-Related Instrumentation."

11. NTD-RROI-SSO-95-430/NTD-NSA-TA-95-370, "Identification of Conservative Directions for Constants in OTΔT and OPΔT Reactor Trip Functions," September 1, 1995.

12. NSAL-00-008, "Reactor Coolant Loop Flow Asymmetry," May 22, 2000.

13. "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 1" WCAP-15264, Rev. 3, December 2000.

14. "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology for FirstEnergy Nuclear Operating Company Beaver Valley Unit 2" WCAP-15265, Rev. 2, December 2000.

ATTACHMENT C-1

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 286 and 158

Attached are the following items:

1. Proprietary Information Notice

2. Copyright Notice

3. Application for Withholding Proprietary Information from Public Disclosure

## PROPRIETARY INFORMATION NOTICE

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ATTACHMENT D-1

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License Amendment Request Nos. 286 and 158

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