ATTACHMENT A-1

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Change No. 208

Revise the Technical Specification as follows:

Remove Pages	Insert Pages
XXV	XXV
2-1	2-1
2-2	2-2
2-3	---
2-4	---
2-6	2-6
B 2-4	B 2-4
3/4 2-12	3/4 2-12
3/4 2-13	3/4 2-13

Figure Index

DELETE

(Proposed Wording)

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (Tavg) shall not exceed the limits shown in Figure 2.1-1 for 3 loop operation. and Figure 2.1-2 and Figure 2.1-3 for 2 loop operation.

DELETE

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.

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)

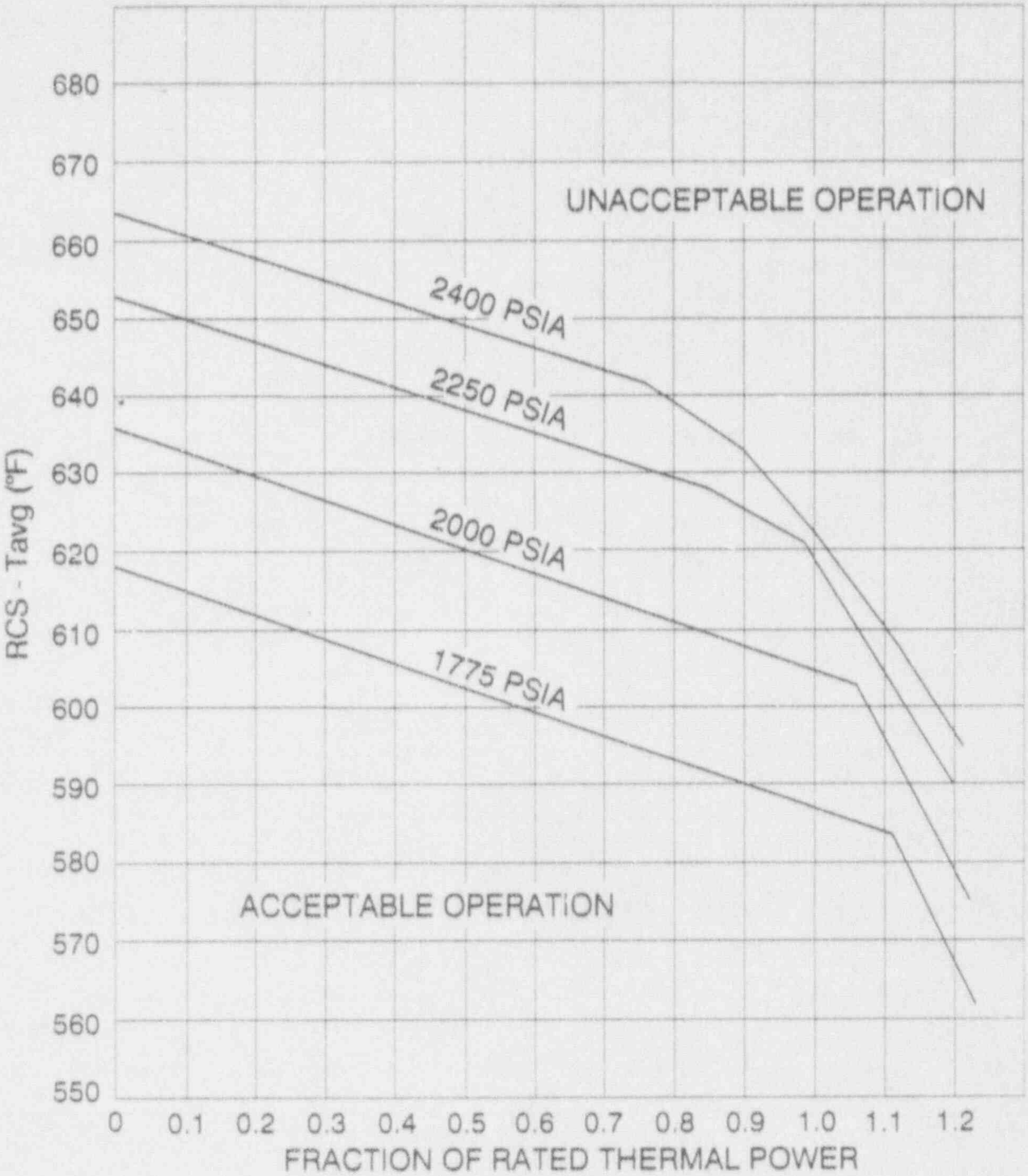

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT THREE LOOPS IN OPERATION

REPLACE WITH INSERT "A"

(Proposed Wording)

DPR-66

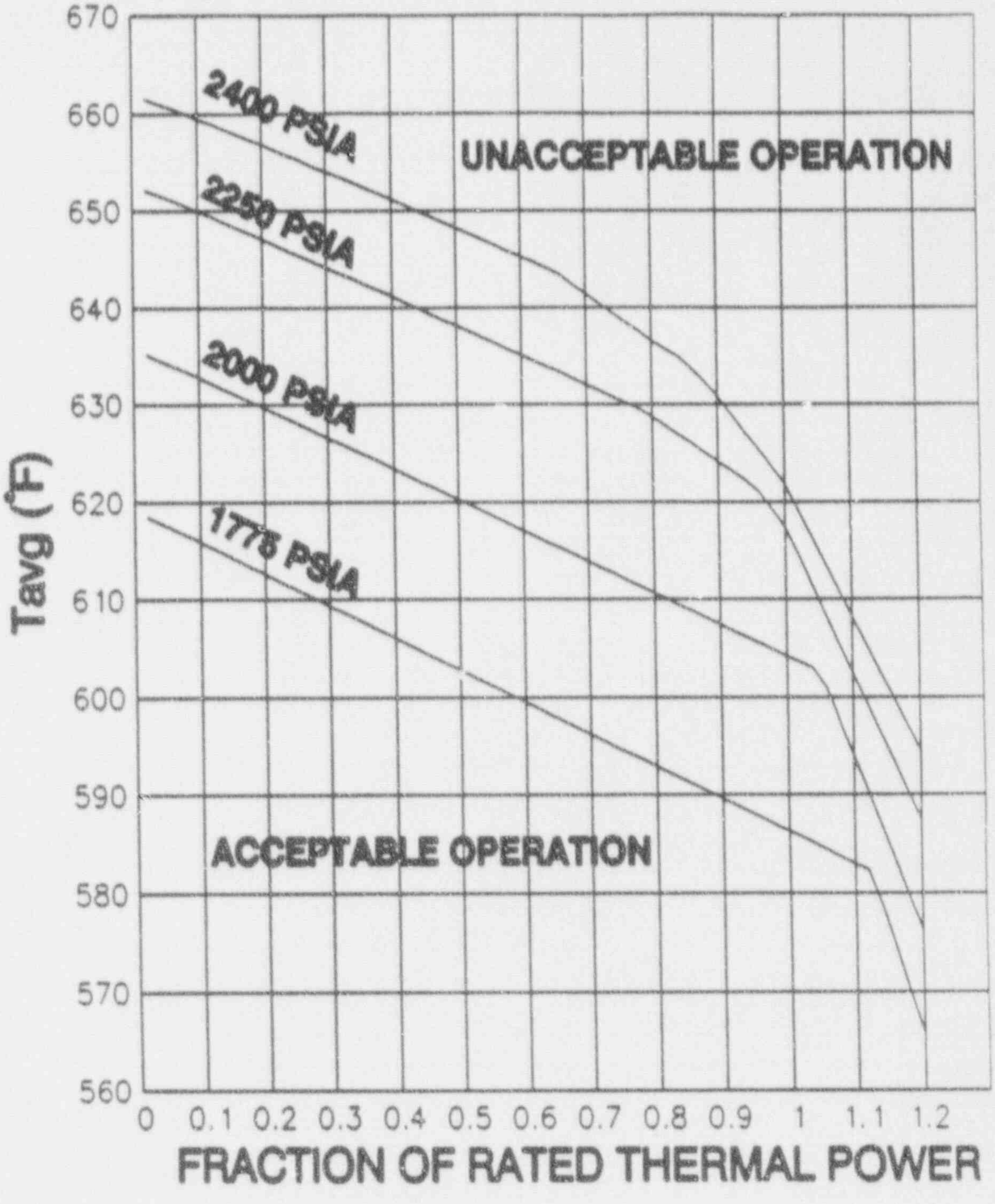

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT
THREE LOOP OPERATION

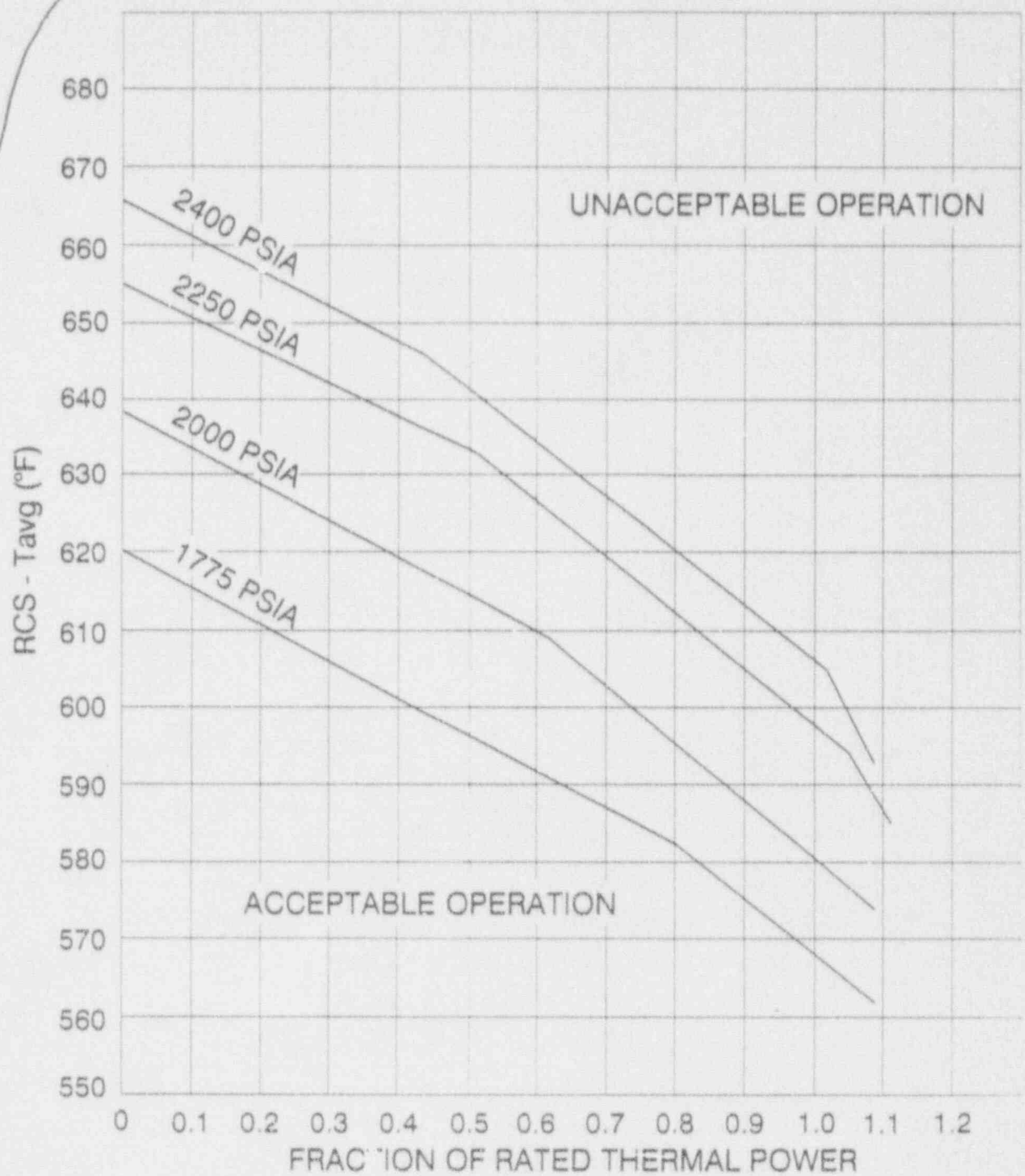

FIGURE 2.1-2

REACTOR CORE SAFETY LIMIT
TWO LOOPS IN OPERATION (ONE LOOP ISOLATED)

DELETE

(Proposed Wording)

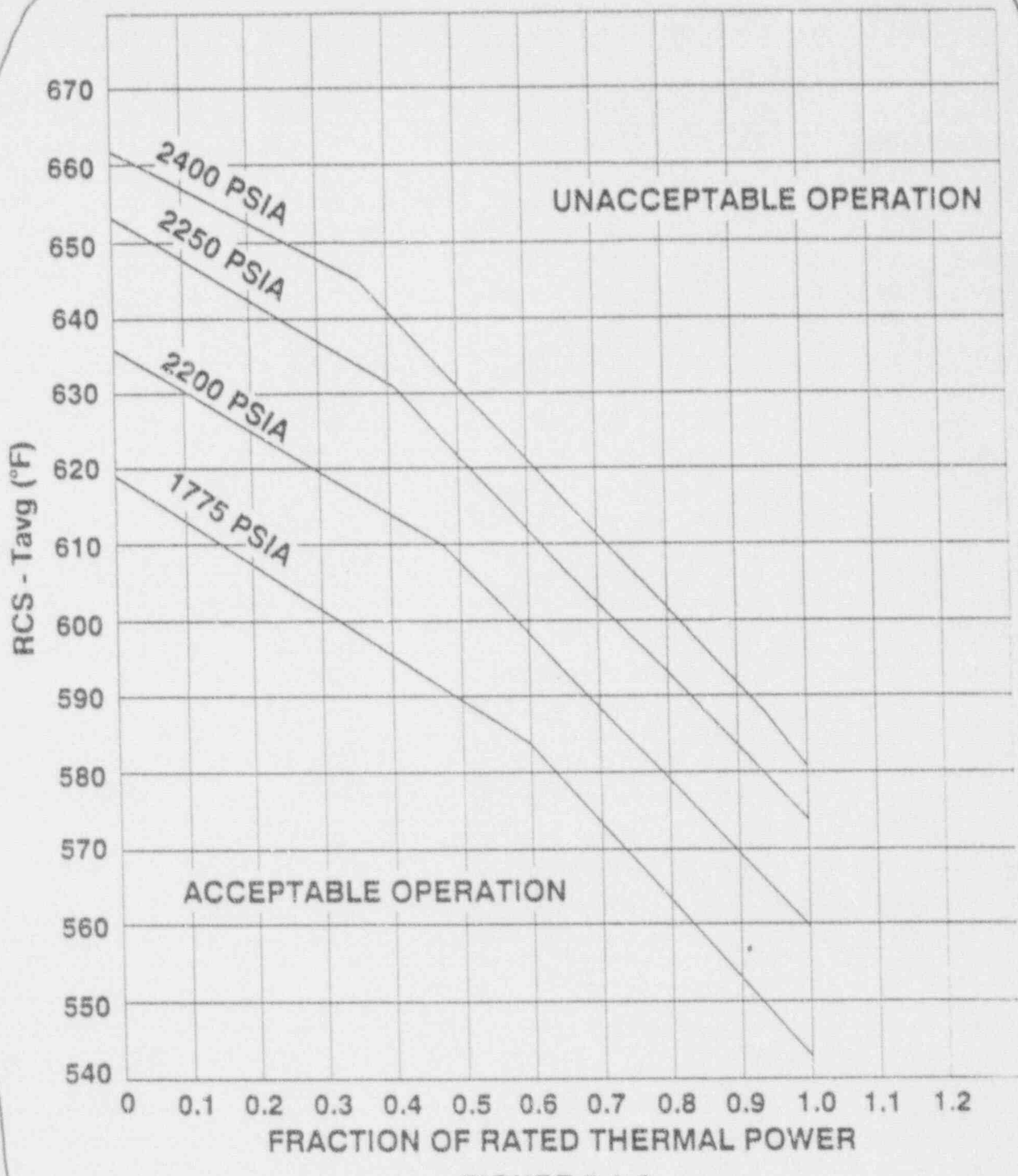

FIGURE 2.1-3

REACTOR CORE SAFETY LIMIT
TWO LOOPS IN OPERATION (NO ISOLATED LOOP)

DELETE

(Proposed Wording)

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER High Setpoint - ≤ 109% of RATED THERMAL POWER	Low Setpoint - ≤ 27.3% of RATED THERMAL POWER High Setpoint - ≤ 111.3% 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% 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% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 31.1% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.4 x 10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	≥ 1945 psig	≥ 1934 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2394 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93.9% of instrument span
12. Loss of Flow	≥ 90% of design flow* per loop	≥ ~~88.9%~~ 89.0% of design flow* per loop

*Design flow is ~~88,500~~ 87,200 gpm per loop.

(Proposed Wording)

LIMITING SAFETY SYSTEM SETTINGS

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 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. 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~~, Figure 2.1-2, and Figure 2.1-3~~. 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 2.2-1.

DELETE

(Proposed Wording)

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)

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1. (2)

ACTION:

With any of the above parameters exceeding its limit, restore the 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.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be indicating 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 by measurement at least once per 18 months.

---

(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

(proposed wording)

DPR-66

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	3 Loops In Operation	LIMITS: 2 Loops In Operation & Loop Stop Valves Open	LIMITS: 2 Loops in Operation & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	≤ ~~581°F~~ 580.7°F	≤ 570°F	≤ 570°F
Pressurizer Pressure	≥ 2220 psia* (1)	≥ 2220 psia*	≥ 2220 psia*
Reactor Coolant System Total Flow Rate	≥ ~~265,500 gpm~~ 261,600 gpm	≥ 189,000 gpm	≥ 187,800 gpm

DELETE

(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.

(Proposed Wording)

ATTACHMENT A-2

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Change No. 74

Revise the Technical Specification as follows:

Remove Pages	Insert Pages
2-2	2-2
2-4	2-4
3/4 2-11	3/4 2-11
3/4 2-12	3/4 2-12

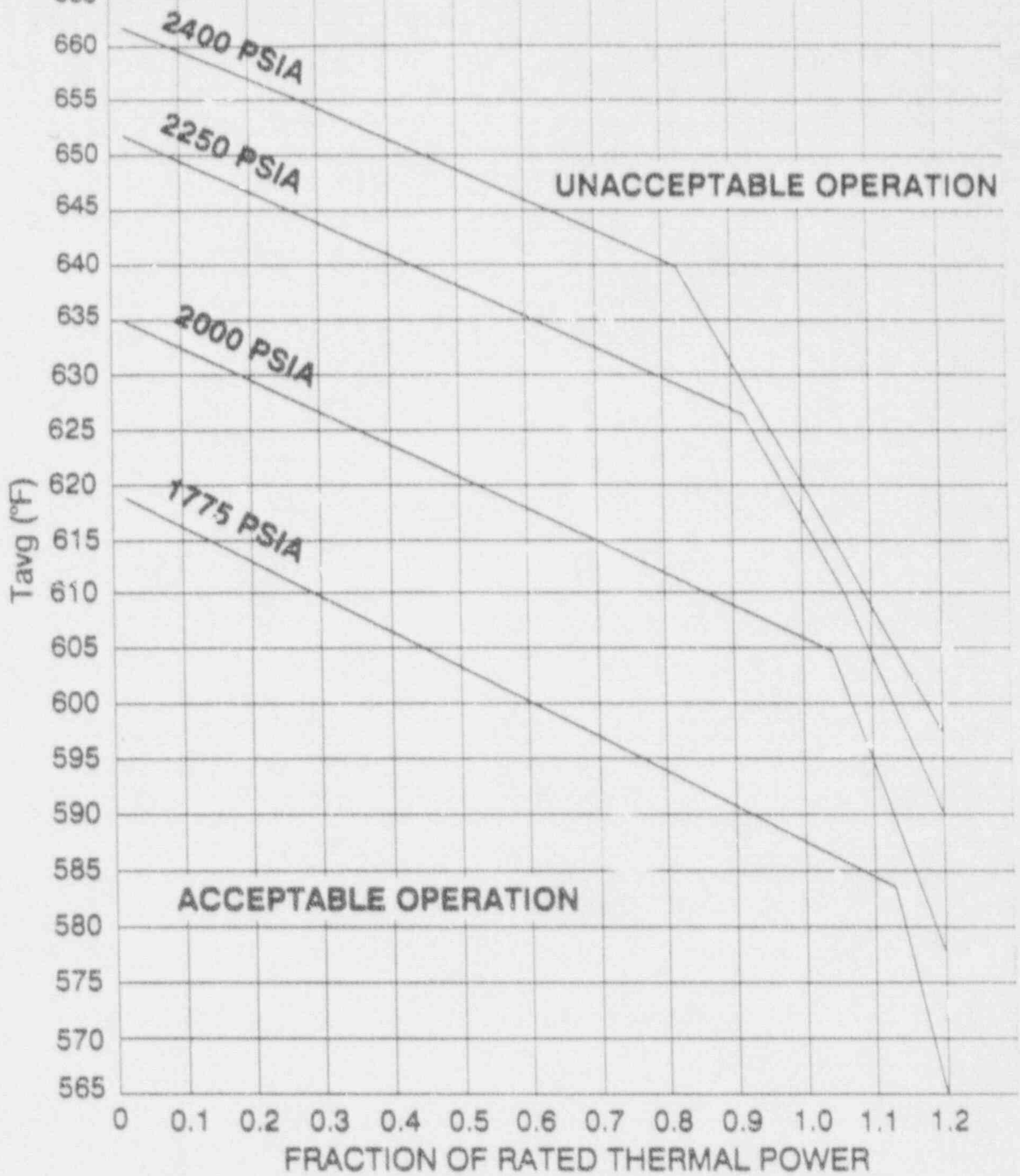

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT
THREE LOOPS IN OPERATION

REPLACE WITH INSERT "B"

(Proposed Wording)

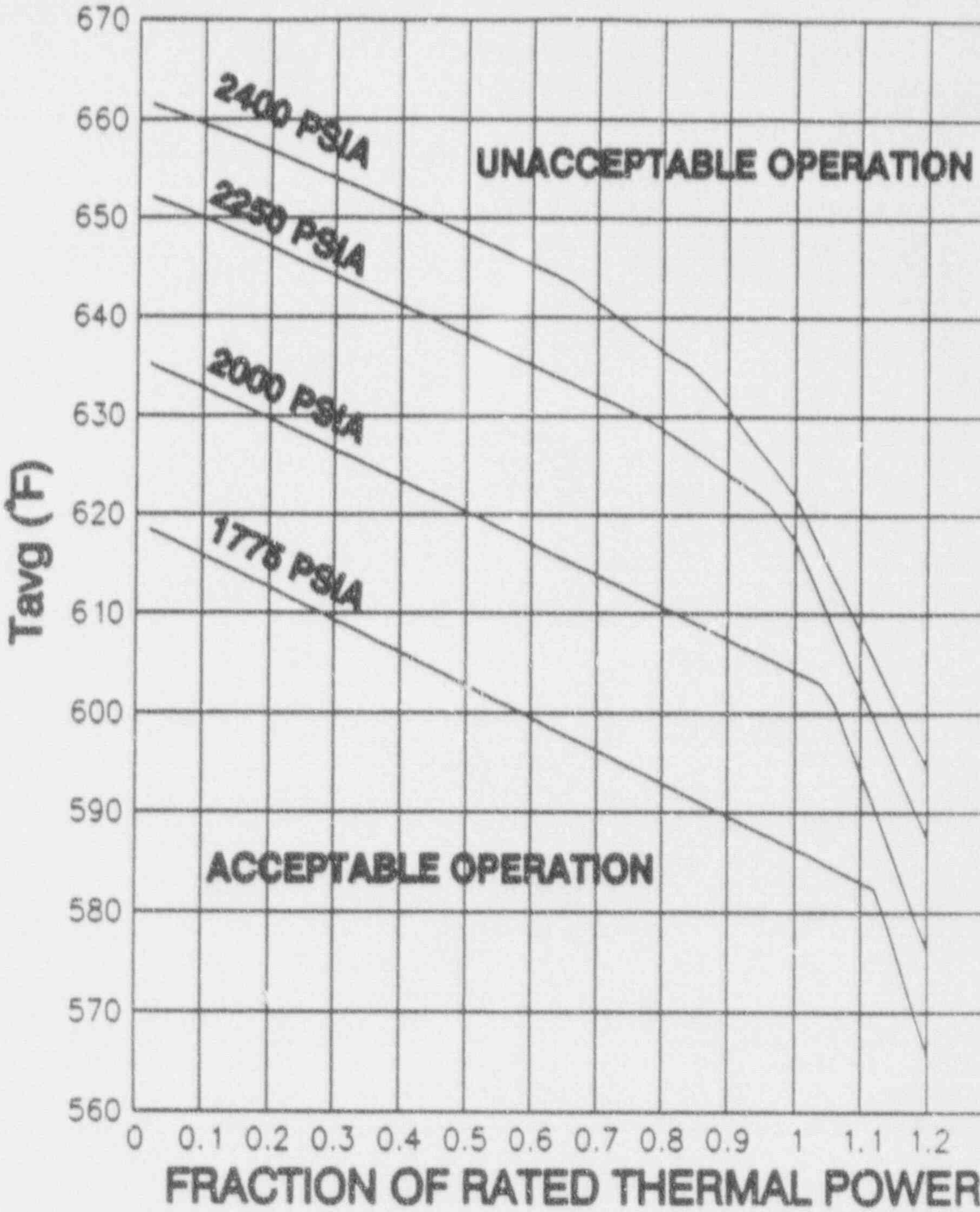

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT
THREE LOOP OPERATION

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	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% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤27.1% 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% 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% of RTP* with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP*	≤ 30.9% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4×10^5 cps
7. Overtemperature ΔT	7.0	5.10	See Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.9	1.71	1.49	See Note 3	See Note 4
9. Pressurizer Pressure-Low	3.1	0.71	1.67	≥ 1945 psig***	≥ 1935 psig***
10. Pressurizer Pressure-High	6.2	4.96	0.67	≤ 2375 psig	≤ 2383 psig
11. Pressurizer Water Level-High	8.0	2.18	1.67	≤ 92% of instrument span	≤ 93.8% of instrument span
12. Loss of Flow	2.5	1.39	0.60	≥ 90% of loop design flow**	≥ ~~88.8%~~ 88.9% of loop design flow**

* = RATED THERMAL POWER
** Loop design flow = ~~88,500~~ 87,200 gpm
*** Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to those values.

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)

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1~~*~~ (2)

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5 percent of RATED THERMAL POWER within the next 4 hours.

DELETE

SURVEILLANCE REQUIREMENTS

4.2.5.~~1.1~~ Each of the parameters of Table 3.2-1 shall be verified to be indicating within their limits at least once per 12 hours.

~~4.2.5.1.2 The provisions of Specification 4.0.3 and 4.0.4 are not applicable for the reactor startups following the initial fueling 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.~~

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

DELETE

---

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

ADD

(2) ~~*~~The provisions of Specification ~~3.0.2~~ 4.0.4 are not applicable ~~for the reactor startup following the initial fueling~~ 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.

(Proposed Wording)

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	3 Loops in Operation
Reactor Coolant System T_{avg}	≤ ~~580.3°F~~ 580.2°F
Pressurizer Pressure	≥ 2220 psia* (11)
Reactor Coolant System Total Flow Rate	≥ ~~270,850 gpm**~~ 261,600 gpm

(11) * 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.

~~** Includes a 2.0% flow measurement uncertainty.~~ DELETE

(Proposed Wording)

ATTACHMENT B

Beaver Valley Power Station, Unit Nos. 1 and 2
Proposed Technical Specification Change No. 208 and 74
REVISION OF SPECIFICATIONS 2.1.1, 2.2.1, AND 3.2.5

A. DESCRIPTION OF AMENDMENT REQUEST

The proposed change request would reduce the minimum required reactor coolant system (RCS) total flow rate by approximately 1.5 percent for Beaver Valley Power Station (BVPS) Unit No. 1 and Unit No. 2.

For BVPS Unit No. 1, Figure 2.1-1 would be revised to support the 1.5 percent reduction in required RCS total flow. Table 2.2-1 would be revised by changing the value for design flow per loop from 88,500 gpm to 87,200 gpm. Also, the allowable value for table item 12 titled "Loss of Flow" would be changed from 88.9 percent to 89.0 percent. Limiting Condition for Operation (LCO) 3.2.5 would be revised by adding a footnote, designated by the number one, pertaining to Table 3.2-1. The footnote would state that the values contained in Table 3.2-1 correspond to analytical limits used in the safety analyses. The Applicability for LCO 3.2.5 would also be revised by adding a footnote designated by the number two. This footnote would state that the provisions of Specification 4.0.4 are not applicable for RCS total flow rate. Table 3.2-1 would be revised by changing the value of RCS Tavg from 581°F to 580.7°F. The value for RCS total flow rate would also be changed from a 265,500 gpm to 261,600 gpm. The existing footnote on Table 3.2-1 would be designated by the number one instead of a single asterisk. The references to two loop operation would also be removed from Specification 2.2-1 and Table 3.2-1. Figures 2.1-2 and 2.1-3, which also pertain to two loop operation, would be deleted. The Bases section for Specification 2.2-1 and the Figure Index would be revised to reflect the deletion of Figures 2.1-2 and 2.1-3.

For BVPS Unit No. 2, Figure 2.1-1 would be revised to support the 1.5 percent reduction in required RCS total flow. Table 2.2-1 would be revised by changing the value for design flow per loop from 88,500 gpm to 87,200 gpm. Also, the allowable value for table item 12 titled "Loss of Flow" would be changed from 88.8 percent to 88.9 percent. LCO 3.2.5 would be revised by adding a footnote, designated by the number one, pertaining to Table 3.2-1. The footnote would state that the values contained in Table 3.2-1 correspond to analytical limits used in the safety analyses. The existing Surveillance Requirement (SR) 4.2.5.1.1 would be designated by SR 4.2.5.1. The existing SR 4.2.5.1.2 would be deleted. The current footnote designated by a single asterisk would be designated by the number one. Also, this footnote would be modified by changing the reference of Specification 3.0.2 to Specification 4.0.4 and by deleting the words "for the reactor startup following the initial fueling." Table 3.2-1 would be revised by changing the value of RCS Tavg from 580.3°F to 580.2°F. The value for RCS total flow rate would also be changed from 270,850 gpm to 261,600 gpm. The existing footnote, designated by a single asterisk, would be designated by the number one. The footnote, pertaining to flow measurement uncertainty, would be deleted.

B. BACKGROUND

Technical Specification 3.2.5 requires that the RCS flow be maintained within a limit greater than or equal to the thermal design flow (TDF) assumed in the current safety analyses. The current safety analyses for both Units assumes a total TDF of greater than or equal to 265,500 gpm. The value specified in Unit 2's Table 3.2-1 for RCS total flow rate includes 2.0 percent flow measurement uncertainty. With the addition of the 2.0 percent uncertainty factor, the value specified in Table 3.2-1 is 270,850 gpm (i.e., 265,500 gpm plus 2.0 percent approximately equals 270,850 gpm). The value specified in Unit 1's Table 3.2-1 for RCS total flow rate does not include any uncertainty factor. Therefore, the value specified in Unit 1's table is 265,500 gpm.

The limits specified in Table 3.2-1 on RCS flow, coolant temperature, and pressurizer pressure ensures that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed in the safety analyses.

The Loss of Flow reactor trip setpoint, as specified in Table 2.2-1, ensures that protection is provided against violating the DNBR limit due to a low flow condition in the RCS loop(s). Figure 2.1-1 provides a loci of points of thermal power, RCS pressure, and average temperature for which the minimum DNBR is no less than the safety analyses DNBR limit, or average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The predictions of future cycle steam generator tube plugging are such that the RCS total flow rate may not continue to meet the current technical specification requirement. The reduction in RCS total flow rate is due to the increase in loop resistance associated with increasing the number of steam generator tube plugs. Therefore, safety analyses and evaluations have been performed which supports an approximate 1.5 percent reduction in the minimum RCS total flow rate limit. The proposed technical specification changes implement a reduced minimum total flow rate requirement which is intended to bound future measured flow values with predicted levels of steam generator tube plugging.

C. JUSTIFICATION

The proposed revisions to the technical specifications will reduce the required RCS total flow rate by approximately 1.5 percent. This reduction in flow is necessary due to the predictions of future cycle steam generator tube plugging. Without the reduction in the minimum required measured RCS flow rate, the likelihood is increased for not satisfying the technical specification requirements. Therefore, safety analyses and evaluations have been performed which support a 1.5 percent

reduction in the minimum required RCS total flow rate. Similar change requests to reduce the RCS total flow have been reviewed and approved by the staff for North Anna Unit No. 1, Vogtle Unit Nos. 1 and 2, Wolf Creek and Farley Unit No. 1.

The proposed changes to Figure 2.1-1 will be more restrictive by reducing the acceptable operation range for a given Tavg and fraction of rated thermal power. This reduction in the acceptable operation range is necessary in order to ensure analysis limits are met with a reduced TDF.

The deletion of Figures 2.1-2 and 2.1-3 (for Unit No. 1 only) and their references in the Figure Index, Specifications 2.1.1 and associated Bases is administrative in nature. These two figures are no longer required. Plant operation with less than three RCS loops is not permitted by our current license. Also, the current safety analyses do not take into account two loop operation. Therefore, technical specification Figures 2.1-2 and 2.1-3 can be deleted since they are no longer applicable for Unit No. 1 operation.

The revision to Table 2.2-1, pertaining to RCS design flow rate, reflects a 1.5% reduction in design loop flow rate. This change is consistent with the purpose of this change request to reduce TDF by 1.5%. The proposed increase in allowable value, for the loss of flow trip contained on Table 2.2-1, is more conservative. This change is necessary in order to ensure analysis limits continue to be met for a loss of RCS flow event with a reduced TDF.

The proposed addition of footnote number one to LCO 3.2.5 will ensure that the parameters stated in Table 3.2-1 are recognized to be analysis limits and not indicated values. The proposed addition of footnote number two will allow the plant to enter operational Mode 1 should the surveillances for RCS total flow rate be beyond the allowable surveillance intervals. Surveillance Requirement 4.2.5.2 requires that the RCS flow rate be determined by measurement. This measurement is performed by utilizing a precision heat balance at a reactor power of at least 90%. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% reactor power to obtain the required RCS flow accuracies. Once the total RCS flow rate has been determined by a heat balance, the control room RCS flow indicators, which have a scale of 0 to 100%, are then correlated to an actual flow value. The control room RCS flow indicators are used to meet the twelve hour surveillance requirement as specified by SR 4.2.5.1. Without an exclusion to Specification 4.0.4, the plant would not be able to enter Mode 1 if SR 4.2.5.2 was beyond the allowable surveillance interval. Therefore, the plant would not be able to commence power operation without a technical specification change to LCO 3.2.5. BVPS Unit No. 2 currently has similar wording contained in SR 4.2.5.1.2 pertaining to reactor startups following the initial fueling. Also, the current version of NUREG 1431 titled

"Standard Technical Specifications for Westinghouse Plants" contains provisions which allow entry into Mode 1 without the RCS total flow rate surveillance being performed. SR 4.2.5.1.2 (on Unit 2 only) can be deleted since the wording pertains only to the initial fuel loading. The proposed footnote number two will incorporate the exclusion to Specification 4.0.4 contained in SR 4.2.5.1.2.

The deletion of limits contained on Table 3.2-1 pertaining to two loop operation (for Unit No. 1 only) is proposed for the same reasons and described for the deletion of Figures 2.1-2 and 2.1-3. The proposed reductio' of the value specified for Tavg, contained in Table 3.2-1, is more conservative. This reduction in the allowable Tavg is necessary to ensure that the plant is operated within the bounds assumed in the accident analyses with a reduced TDF. The reduction of the value for RCS total flow rate, contained on Table 3.2-1, reflects a 1.5% reduction in TDF. This change is consistent with the purpose of the change request. The deletion of the footnote (for Unit No. 2 only), pertaining to flow measurement uncertainty, will make all the parameters specified on Table 3.2-1 analyses values. Presently, the parameters specified for Tavg and Pressurizer Pressure reflect analytical limits used in the safety analyses, while the value stated for RCS total flow rate reflects an indicated value. Currently, RCS flow uncertainty is administratively added to the Unit 1's technical specification value of 265,500 gpm to account for instrument inaccuracies. By deleting the uncertainty value in the Unit No. 2 technical specifications, both Units will have consistent technical specification values for RCS total flows. The addition of flow measurement uncertainty for BVPS Unit No. 2 will be handled administratively in the same manner as currently used for BVPS Unit No. 1.

The designation of the existing footnotes on Table 3.2-1 with numbers instead of symbols is administrative in nature and does not change the intent or application of the footnotes.

D. SAFETY ANALYSIS

Duquesne Light Company contracted Westinghouse Electric Corporation to perform analyses for both BVPS Unit Nos. 1 and 2 to support operation with a TDF of 87,200 gpm per loop (261,600 gpm total). A summary of the analyses and evaluations for BVPS Unit NO. 1 is contained in Attachment 1. A summary of the analyses and evaluation for BVPS Unit No. 2 is contained in Attachment 2. This technical specification change only addresses lowering the thermal design flow. However, some of the analyses and evaluations described in Attachments 1 and 2 were undertaken to support both lowering the thermal design flow and increasing the steam generator tube plugging limit. All of the analyses and evaluations needed to support the lowering of thermal design flow are complete. However, since certain analyses considered both issues, the words pertaining to increased tube plugging limits appear in conjunction with reducing thermal design flow.

The results of the analyses presented in Attachments 1 and 2 show that all of the acceptance criteria previously established in the UFSAR continue to be met for each reanalyzed event. A review of the accident analyses presented in UFSAR Chapter 14 or 15, for Unit No. 1 and 2, respectively, has demonstrated that a reduction in TDF for Unit No. 1 and No. 2 to 261,600 gpm is accommodated by current analysis margins or by the assessment of a penalty against available retained DNBR margin for all accidents. The current Engineered Safety Features and Reactor Protection System setpoints set forth in each Units technical specifications have been demonstrated to provide adequate plant protection at the reduced flow rate condition or revised to ensure adequate plant protection. The core thermal limits have been revised to account for the reduction in TDF. Only the exit boiling portions of the core limits were revised since the current DNB limits, based on the W-3 R-grid DNB correlation, are more limiting than DNB limits based on the WRB-1 correlation and mini Reduced Thermal Design Procedure (the current design basis). The current OT▲T and OP▲T analysis setpoint equations were confirmed to provide protection for the revised core limits.

The addition of exclusion to Specification 4.0.4 for LCO 3.2.5 will not adversely impact the safety of the plant. RCS flow indication will continue to be available to plant operators. The RCS low flow trip will continue to provide core protection should RCS flow drop below 90% of design flow. The exclusion to Specification 4.0.4 will allow RCS flow to be verified at the plant condition which provides the highest degree of measurement accuracy. If RCS flow would be measured below the allowable value, then the LCO action statement will continue to be followed. The deletion of the flow measurement uncertainty from the Unit No. 2 RCS total flow value will not change the requirements for the minimum allowable RCS flow. Administrative controls will ensure that the flow measurement uncertainty factor is added to ensure that actual RCS total flow rate is above the value assumed in the safety analyses and as specified in the proposed wording for Table 3.2-1.

Therefore, the proposed changes are considered safe based on a review of plant specific analyses and the evaluations performed to ensure that the reduction in TDF will not adversely impact the adequacy of the auxiliary systems and components. The reduction in TDF does not affect any of the mechanisms postulated in the UFSAR to cause non-LOCA and SGTR design basis events. The LOCA and LOCA-related events maintain conformance with analysis acceptance criteria (10 CFR 50.46) regulations. The design requirements on both units continue to be met. The integrity of the RCS pressure boundary is not challenged. The assumptions employed in the calculation of the offsite radiological doses remain valid. Therefore, the consequences of the accidents considered in the Beaver Valley Unit 1 and 2 licensing basis remain unchanged.

E. NO SIGNIFICANT HAZARDS EVALUATION

The no significant hazard considerations involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) 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 evaluation is provided for the no significant hazards consideration standards.

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

   An assessment of the NSSS primary components, including the reactor pressure vessel system, reactor coolant pump, steam generator, pressurizer, Control Rod Drive Mechanisms, and RCS piping, concluded that the integrity of the components will be unaffected by the reduction in thermal design flow. Also, evaluations of the Reactor Coolant System, Chemical and Volume Control System, Residual Heat Removal System, and Safety Injection System concluded that the reduced thermal design flow will not adversely impact the adequacy of the auxiliary systems and components. The reduction in thermal design flow does not affect any of the mechanisms postulated in the UFSAR to cause non-LOCA and SGTR design basis events. Also, the LOCA and LOCA-related events maintain conformance with analysis acceptance criteria (10 CFR 50.46) regulations. Therefore, the probability and consequences of an accident previously analyzed in the UFSAR will not be increased. Since design requirements continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid and the consequences of the accidents considered in the Beaver Valley Unit 1 and 2 licensing basis remain unchanged.

The proposed deletion of the RCS flow uncertainty value does not involve a significant increase in the probability or consequences of an accident previously evaluated. The RCS flow will continue to be monitored once per 12 hours in accordance with Surveillance Requirement 4.2.5.1. The required RCS total measured flow rate will be administratively controlled to ensure that the actual flow rate is above the value assumed in the safety analyses. No new performance requirements are being imposed on the RCS due to the deletion of flow uncertainty value. RCS flow is an assumed initial condition in the safety analyses and does not act as an initiator for any transient. The accident analyses are not affected by this proposed deletion and therefore no additional fuel failures or mass releases will result.

Therefore, the proposed changes do 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 reduced thermal design flow and the deletion of the flow measurement uncertainty value does not change the plant configuration in a way which introduces a new potential hazard to the plant. Since design requirements continue to be met and the integrity of the reactor coolant system pressure boundary is not challenged, no new failure mode has been created. Therefore, an accident which is different than any already evaluated in the UFSAR will not be created as a result of this change.

Therefore, the proposed changes do 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 with respect to primary pressure boundary is provided, in part, by the safety factors included in the appropriate design codes. Since the components remain in compliance with the codes and standards in effect when Beaver Valley Unit 1 and 2 were originally licensed and the safety analyses acceptance criteria continue to be met, the margin of safety is not reduced by the reduction in thermal design flow.

The proposed deletion of the RCS flow uncertainty does not involve a significant reduction in the margin of safety. The current flow uncertainty value was derived from a plant specific evaluation which includes a review of calibration procedures and in-plant equipment. The flow uncertainty

value will be administratively added to the proposed technical specification value. This will ensure that the actual RCS flow is at least equal to the flow assumed in the accident analyses.

Therefore, the proposed changes do not involve a significant reduction in the 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 satisfies the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. UFSAR CHANGES

Attachment D provides changes to the UFSAR to accommodate the proposed revision to the RCS thermal design flow. The UFSAR changes are provided for information only and will be incorporated following approval of the proposed technical specification changes.

ATTACHMENT 1

Summary of Analyses and Evaluations
Which Support a Reduced Minimum
RCS Total Flow Rate For
Beaver Valley Power Station Unit No. 1

EVALUATION

1.0 Non-LOCA Evaluation

The current non-LOCA safety analysis licensing basis for Beaver Valley Unit 1 assumes a total RCS Thermal Design Flow (TDF) of 265,500 gpm (88,500 gpm per loop) and includes an evaluation supporting a maximum plugging level of 20% per steam generator at this TDF rate (total and per loop).

The non-LOCA evaluation considers a reduction in the TDF to 261,600 gpm (87,200 gpm per loop) and continues to support up to 20% steam generator tube plugging (SGTP).

All non-LOCA transients were examined to determine the effect of the reduced TDF. The non-LOCA accident analysis can be affected in the following ways by a reduction in TDF.

- Reduction in core thermal limits and calculated DNBR
- Change in plant normal operation conditions
- Reduce margin to non-DNB Acceptance Criteria

For evaluation purposes, the non-LOCA transient analyses have been reviewed on the basis of both DNB and non-DNB acceptance criteria.

1.1 DNB Considerations

The affect of the TDF reduction on the following events has been evaluated to assure that the DNB design basis continues to be met:

- Feedwater System Malfunctions Causing an Increase in Feedwater Flow (UFSAR 14.1.9)
- Excessive Increase in Secondary Steam Flow (UFSAR 14.1.10)
- Loss of External Electrical Load/Turbine Trip (UFSAR 14.1.7)
- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (UFSAR 14.1.2)
- Start Up of an Inactive Reactor Coolant Loop (UFSAR 14.1.6)
- Spurious Operation of the Safety Injection System at Power (UFSAR 14.1.16)
- Accidental Depressurization of the Reactor Coolant System (UFSAR 14.1.15)
- Accidental Depressurization of the Main Steam System (UFSAR 14.1.13), Major Secondary Side Pipe Rupture (UFSAR 14.2.5)
- Partial Loss of Forced Reactor Coolant Flow (UFSAR 14.1.5)
- Complete Loss of Forced Reactor Coolant Flow (UFSAR 14.2.9)

- Reactor Coolant Pump Shaft Seizure (Locked Rotor Rods-in DNB) (UFSAR 14.2.7)

- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition (UFSAR 14.1.1)

A reduction in the thermal design flow has an adverse effect on the core thermal limits (DNB, quality and exit boiling) and consequentially the overtemperature (OT) and overpower (OP) ▲T setpoint equations. The core thermal limits were revised to account for the reduction in TDF. Only the exit boiling portions of the core limits change since the current DNB limits, based on the W-3 R-grid DNB correlation, are more limiting than DNB limits based on the WRB-1 DNB correlation and mini-Reduced Thermal Design Procedure (the current design basis). The current OT▲T and OP▲T setpoint equations were confirmed to provide protection for the revised core limits.

The reduced TDF, steam pressure and steam temperature result in a decrease in the initial mass in the steam generators. The combination of reducing TDF and increasing tube plugging would result in a reduction in the steam generator mass on the order of < 0.5% from the analysis values. Only the loss of heatsink transients (Loss of Non-Emergency AC Power, Loss of Normal Feedwater and Feedwater Line Break) are potentially impacted by this minimal decrease in initial steam generator mass. The remaining non-LOCA transients (including all of those analyzed for DNB considerations) are insensitive to minor changes to steam generator inventory. The "non-DNB" events are discussed below.

Small changes in plant operating conditions such as TDF and SGTP will not significantly affect the transient statepoints used in the DNBR calculations. Hence, the transient conditions used to calculate the minimum DNBRs are still valid for the reduced TDF. A decrease in the RCS flow rate potentially decreases the minimum DNBR calculated during the event. Existing conservatism in the DNB calculations bound the affect on DNB due to the 1.5% flow reduction. For the Rod Cluster Control Assembly Misoperation transient, (UFSAR 14.1.3) generic DNBR margin has been allocated to ensure that the DNB design basis continues to be met with the reduced TDF. For the Reactor Coolant Pump Shaft (RCP) Seizure (Locked Rotor) Rods-in-DNB transient (UFSAR 14.2.7), generic DNBR margin has been allocated to ensure that the limit of 18% rods-in-DNB continue to be met with the reduced TDF.

1.2 Non-DNB Considerations

The effect of the TDF reduction has been evaluated to assure that the design basis continues to be met for the following events which are either not DNB related or for which DNBR is not the only relevant safety criterion:

- Loss of External Electrical Load/Turbine Trip (UFSAR 14.1.7)

- Loss of Offsite Power to the Station Auxiliaries (UFSAR 14.1.11), Loss of Normal Feedwater (UFSAR 14.1.8)
- Feedwater System Pipe Break (UFSAR 14.2.5.2)
- Reactor Coolant Pump Shaft Seizure (Locked Rotor) (UFSAR 14.2.7)
- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (UFSAR 14.1.2)
- Uncontrolled Boron Dilution (UFSAR 14.1.4)
- Rupture of a Control Rod Drive Mechanism Housing Rod Cluster Control Assembly (UFSAR 14.2.6)
- Steamline Break Mass/Energy Release - Inside/Outside Containment

1.2.1 Loss of External Electrical Load/Turbine Trip (UFSAR 14.1.7)

In addition to the DNBR requirement, the UFSAR analysis for this event must demonstrate that the primary and secondary system pressures remain below 110% of the design values. Whether from loss of external load or turbine trip, this transient is characterized by a core power which exceeds the secondary side power extraction. This results in a primary side heat up and RCS pressure and temperature increase. Existing analyses have shown this transient to be insensitive, with respect to the pressure limits, to small changes in RCS flow, steam pressure and steam generator mass. Sufficient margin exists to the acceptance criteria. Therefore, the conclusions of the UFSAR remain valid.

1.2.2 Loss of Offsite Power to the Station Auxiliaries (UFSAR 14.1.11), Loss of Normal Feedwater (UFSAR 14.1.8)

These transients are analyzed to demonstrate that the primary and secondary sides do not overpressurize and that the pressurizer does not overfill. This demonstrates the adequate auxiliary feedwater and steam generator inventory exists to remove decay heat and stored energy. These analyses are not impacted by small changes in nominal plant operating conditions (i.e., steam generator mass, RCS flow, and steam pressure). The slightly reduced mass in the steam generators could adversely impact the results of the transient, however, a sensitivity analysis has shown that sufficient margin exists to the limit to accommodate the penalty incurred due to the reduced mass. Therefore, the conclusions of the UFSAR remain valid.

1.2.3 Feedwater System Pipe Break (UFSAR 14.2.5.2)

The UFSAR analysis demonstrates that adequate auxiliary feedwater exists to remove core decay heat and stored energy following a

reactor trip from full power and that the core remains in a coolable geometry and covered with water. For ease of interpreting the transient, Westinghouse has adopted the restrictive criterion that no bulk boiling occurs in the primary coolant system following a Feedwater Pipe Break prior to the time that the heat removal capacity of the steam generators, being fed auxiliary feedwater, exceeds NSSS heat generation. This is determined by verifying that the RCS coolant remains subcooled. The analysis is not impacted by small changes in nominal plant operating conditions (i.e., steam generator mass, RCS flow, and steam pressure). The slightly reduced mass in the steam generators could adversely impact the results of the transient, however, a sensitivity analysis has shown that sufficient margin exists to the acceptance criteria to accommodate the penalty incurred due to the reduced mass. Therefore, the conclusions of the UFSAR remain valid.

1.2.4 Reactor Coolant Pump Shaft Seizure (Locked Rotor) (UFSAR 14.2.7)

This event is analyzed under full power conditions assuming the instantaneous seizure of one RCP rotor. This results in a rapid RCS flow reduction which may lead to DNB. The reactor is tripped promptly on a low flow signal. The analysis demonstrates that the maximum reactor coolant system pressure is less than 110% of design pressure, the maximum fuel clad temperature is less than 2700°F and the amount of zirconium-water reaction is small. In addition a calculation is made to predict the number of rods-in-DNB. The impact on the rods-in-DNB calculation has been discussed in Section 1.1 above. The system transient is not significantly impacted by the small (1.5%) reduction in TDF, steam pressure, or steam generator mass. The licensing basis analysis reports a PCT and Peak Pressure well below the limits of 2700°F and 2750 psia respectively. Therefore, there is sufficient margin to accommodate the small changes that may result from the TDF reduction. Thus, the conclusions of the UFSAR remain valid.

1.2.5 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (UFSAR 14.1.2)

In addition to the DNBR requirement, the UFSAR analysis for this event must demonstrate that the pressurizer does not overfill. The peak pressurizer water volume is expected to increase with the reduction in TDF and increased tube plugging, since the RCS will heatup more than in the current analysis, due to the reduced heat transfer capability. The increased heatup results in a decrease in the coolant density which in turn would increase the pressurizer insurge. Existing analyses have shown the transient to be insensitive, with respect to the pressurizer volume limits, to small changes in RCS flow, steam pressure and steam generator mass. Sufficient margin exists to the acceptance criteria. Therefore, the conclusions of UFSAR remain valid.

1.2.6 Uncontrolled Boron Dilution (UFSAR 14.1.4)

This event was reanalyzed to demonstrate that sufficient shutdown margin exists, such that, should a dilution event occur, there is sufficient time to allow operator action and termination of the event prior to a complete loss of shutdown margin. The event is analyzed in Modes 1, 2 and 6. The flow reduction does not adversely impact the calculations. Therefore, the conclusions of the UFSAR remain valid.

1.2.7 Rupture of a Control Rod Drive Mechanism Housing Rod Cluster Control Assembly (UFSAR 14.2.6)

In this event, a rapid reactivity insertion and increase in core power leads to high local fuel and clad temperatures and possible fuel and/or clad damage. The Rod Ejection event is analyzed at four conditions: beginning and end of life core physics characteristics (BOL, EOL) at hot zero power and full power (HZP, HFP). The analysis demonstrates that gross fuel damage will not occur, that the core remains in a coolable geometry and that the RCS will remain intact. The Rod Ejection event is characterized by a rapid excursion terminated by Doppler feedback. The reactor trips on High Neutron Flux. A reduction in the RCS flow will result in a reduction in the fuel rod to coolant heat transfer. This may result in an increase in the calculated fuel clad temperatures as well as the stored fuel energy. An existing sensitivity analysis has shown negligible impact on the analysis results (PCT, Fuel Temperatures) to a small change in RCS flow. Therefore, the conclusions of the UFSAR remain valid.

1.2.8 Steamline Break Mass/Energy Release - Inside/Outside Containment

The objective of these analyses is to maximize the release of high energy fluid. The reduction in TDF and increase in SGTP reduce the initial mass in the steam generators resulting in earlier tube uncovery. However, the TDF reduction and increased SGTP also reduces the primary to secondary heat transfer and the reactivity inserted due to the negative moderator temperature coefficient. Also, the reduction in initial secondary temperature and pressure would tend to lessen the mass and energy releases. These offsetting effects would not adversely affect the steamline break mass and energy releases inside or outside containment. Therefore, the steamline break mass and energy release inside and outside containment are considered to remain valid for the reduced TDF and increase SGTP.

1.3 Non-LOCA Results/Conclusions

Operation of Beaver Valley Unit 1 with a reduced thermal design flow of 261,600 gpm (87,200 gpm per loop) and a maximum plugging level of 20% per steam generator is acceptable from the standpoint of the non-LOCA analyses.

2.0 STEAM GENERATOR TUBE RUPTURE (SGTR) EVALUATION

The Steam Generator Tube Rupture analysis in the Beaver Valley Unit 1 UFSAR was performed to evaluate the radiological consequences due to the event. The major factors that affect the radiological doses for a SGTR event are the amount of radioactivity assumed to be available in the reactor coolant, the amount of reactor coolant transferred to the secondary side of the faulted steam generator through the ruptured tube, and the amount of steam released from the ruptured steam generator to the atmosphere.

For the UFSAR analysis, it was assumed that the primary to secondary break flow and the steam release from the faulted steam generator would be terminated within 30 minutes after the accident. The loss of reactor coolant due to the break flow is assumed to result in reactor trip and SI actuation due to low pressurizer pressure. After reactor trip and SI actuation, the break flow rate is assumed to reach equilibrium at the RCS pressure when the incoming SI flow rate equals the outgoing break flow rate. The equilibrium break flow rate is assumed to persist until 30 minutes after the initiation of the accident. The total primary to secondary break flow is then determined for the 30 minute period. The amount of steam released from the faulted steam generator is calculated based on a mass and energy balance for the RCS and the steam generators for the 30 minute period. An evaluation has been completed for a reduced thermal design flow of 261,600 gpm (87,200 gpm/loop) and up to 20% steam generator tube plugging to determine the impact on the UFSAR SGTR analysis.

The conservative fuel failure assumption of 1% defective fuel for the Beaver Valley Unit 1 SGTR analysis will not change due to the reduced TDF. The reduced TDF will change the steam generator operating parameters which will affect the break flow prior to reactor trip and also the steam release from the faulted steam generator. However, the amount of radioactivity released to the atmosphere from for the Beaver Valley Unit 1 SGTR was conservatively calculated independent of the amount of steam released from the faulted steam generator, and thus, the SGTR consequences are primarily dependent upon the primary to secondary break flow.

The Unit 1 SGTR analysis was evaluated for the reduced TDF of 261,600 gpm and a SGTP level of 20%. The results of the evaluation indicate that reduced TDF result in a slight increase in the calculated break flow and consequently in the calculated radiation does for an SGTR. However, due to the conservatism in the calculated results for the SGTR reported in the Beaver Valley Unit 1 UFSAR, the UFSAR results remain bounding. Thus the conclusions presented in the Beaver Valley Unit 1 UFSAR remain valid for a TDF of 261,600 gpm and up to 20% SGTP.

3.0 LOCA

The following UFSAR LOCA related events were evaluated:

- Large Break LOCA (UFSAR Section 14.3.2.2)
- Small Break LOCA (UFSAR Section 14.3.1)
- Blowdown Reactor Vessel Forces (UFSAR Section 14.3.3 & Appendix B)
- Post-LOCA Long-Term Cooling, Subcriticality Evaluation (related to UFSAR Section 14.3.2)
- Reactor Coolant Loop LOCA Forcing Functions
- Hot Leg Switchover to Prevent Potential Boron Precipitation/Long Term SI Verification
- Reactor Coolant Loop Stress Reconciliation

3.1 Large and Small Break LOCA

The Beaver Valley Unit 1 Large Break LOCA (LBLOCA) analysis of record, which is presented in the UFSAR, is a BASH Evaluation Model analysis with a Peak Clad Temperature (PCT) of 1918°F. Additional PCT penalties have been assigned which resulted in a cumulative PCT of 2151°F.

The Beaver Valley Unit 1 Small Break LOCA (SBLOCA) analysis of record, which is presented in the UFSAR, is a NOTRUMP Evaluation Model analysis with a PCT of 1802°F. Additional PCT penalties have been assigned which resulted in a cumulative PCT of 2182°F.

A recent evaluation of the temperature uncertainty associated with the value for design Tavg was performed. This evaluation determined that the value used for RCS Tavg uncertainty should be increased by 0.5°F. Using existing sensitivity studies for the BASH and NOTRUMP Evaluation Models, the 0.5°F increase in RCS Tavg uncertainty results in the following PCT penalties:

SBLOCA: 5°F
LBLOCA: 2°F

There are primarily two aspects of the Emergency Core Cooling System (ECCS) LOCA analyses which should be addressed as a result of the reduction in TDF:

(1) Consideration of the RCS Flow
(2) Consideration of effects of RCS Temperature distribution

Within reasonable limits, such as the reduction from 88,500 gpm/loop to 87,200 gpm/loop being considered, RCS flow (1) has a generally insignificant effect because the break flow dominates the transient almost immediately for both SBLOCA and LBLOCA. Therefore, the majority of the effect is realized through (2) any changes to RCS Tavg that result. LOCA ECCS analyses are performed at 102% Power as directed by 10CFR50 Appendix K. The initial RCS Temperature

distribution assumed by the LOCA analyses is determined using a complex methodology based upon 100% power design RCS conditions. Applying this methodology to the TDF reduction sequence, no change to LOCA ECCS 102% power RCS Tavg is predicted. Therefore, no PCT penalty or benefit is incurred.

The cumulative LBLOCA PCT is revised as follows:

2151°F	Current PCT With Assigned Penalties
+ 2°F	RCS Tavg Uncertainty Penalty
=2153°F	Revised Cumulative PCT

For SBLOCA, the SPIKE PCT Penalty must be re-investigated since the penalty is highly PCT dependent. For the new PCT conditions, the new SPIKE PCT penalty is increased by 10°F. Changes to the PCT total are as follows:

2182°F	Current PCT With Assigned Penalties
+ 5°F	RCS Tavg Uncertainty Penalty
+ 10°F	Increase in SPIKE Penalty
=2197°F	Revised Cumulative PCT

Therefore, conformance with the 10 CFR 50.46 PCT Limit of 2200°F is maintained for both SBLOCA and LBLOCA.

3.2 Blowdown Reactor Vessel and Loop Forces

The Reactor Vessel LOCA forces conclusions are currently presented in WCAP-11556. Blowdown forces are typically limiting immediately after the break and are influenced primarily by design Tcold. Design Tcold decreases slightly for the reduced TDF condition and LOCA forces slightly increase. However, the increase is accommodated within the margin available in the overall structural integrity evaluation. Therefore, the TDF reduction does not change the WCAP-11556 conclusions.

Westinghouse has performed an evaluation on the loop forcing function (LFF) analyses performed in 1980 assuming a reduced TDF. LFF are primarily influenced by RCS temperature, break size and break opening time. The reactor pressure vessel outlet nozzle (RPVON) break is governed by RCS Thot, which increases as a result of the TDF reduction. Thererore, the RPVON LFF remains bounding for the RPVON break. For the remaining 10 break locations, a bounding 0.35% increase in LFF is imposed by the TDF reduction causing a small increase in RCS Tcold. The impact of the small increase in LFF was evaluated and determined to have an insignificant impact on the loop pipe stress and support calculations.

3.3 Post-LOCA Long-Term Cooling, Subcriticality Evaluation

The Westinghouse position for satisfying the requirements of 10 CFR 50.46(b)(5) 'Long Term Cooling' is defined in WCAP-8339, WCAP-8472, and Technical Bulletin NSID-TB-86-08. The Westinghouse commitment is that the reactor will remain shutdown by borated ECCS water alone after a LOCA. Since credit for the control rods is not taken for a LBLOCA, the borated ECCS water provided by the accumulators and the RWST must have a concentration that, when mixed with other sources of borated and non-borated water, will result in the reactor core remaining subcritical assuming all control rods out. The TDF reduction does not alter the conclusion of the evaluation, which is checked by Westinghouse on a cycle by cycle basis at the time of the Reload Safety Evaluation (RSE), most recently the Cycle 9 RSE.

3.4 Hot Leg Switchover to Prevent Potential Boron Precipitation/ Long-Term SI Verification

Hot leg switchover time is dependent upon power level and upon RCS, RWST, and accumulator water volumes and boron concentrations. The TDF reduction has no effect on the listed parameters and, therefore, post-LOCA hot leg switchover time is not affected. Similarly, SI long-term performance is also unaffected.

3.5 Reactor Coolant Loop Stress Reconciliation

The impact of the TDF reduction on reactor coolant loop stresses was evaluated. Lowering RCS flow resulted in a 0.5 degree hot leg increase and cold leg decrease. Based upon the slight increase in hot leg temperature (0.08%) and the reduction in cold leg temperature, it has been determined that the change in system parameters will have an insignificant affect on the design margin for the piping systems and supports.

4.0 LOCA MASS AND ENERGY CALCULATIONS

LOCA mass and energy calculations were evaluated for the impact of reducing core flow. The controlling input to the calculations, in this case, is RCS mass average temperature. Programmed Tavg remains the same even though the temperature difference goes up. Since the cold leg is larger than the hot leg, the coolant mass average temperature goes down. Since the temperature change is in the conservative direction, the current UFSAR values remain bounding.

5.0 NSSS PRIMARY COMPONENTS

5.1 Reactor Pressure Vessel System

The reactor pressure vessel system consists of the reactor vessel, the reactor upper and lower internals assemblies and the reactor core. Since these components are interdependent from a thermal-hydraulic and structural viewpoint, they were evaluated as a system. The reactor pressure vessel system is sensitive to variations in the reactor coolant system flowrate. Therefore, the reactor pressure vessel system was evaluated with respect to the reduction in the thermal design flow.

New flows and pressure drops were calculated for the various flow paths within the reactor pressure vessel system. The results showed that the changes in pressure drops associated with the new operating conditions are evenly distributed throughout the reactor internals, and that the total pressure drop across the internals would decrease an insignificant amount. Since the internals flow and pressure drop changes are not changed significantly by the new operating conditions, detailed calculations of the effect on core bypass flow, hydraulic lift forces, flow induced vibration and Rod Control Cluster Assembly (RCCA) rod drop times were not necessary. Additionally, the temperature rise across the reactor vessel is bounded by the original structural analyses of the Beaver Valley Unit 1 internals.

The evaluation of the reactor pressure vessel system demonstrated that there would be no adverse impact on the performance of the system by the proposed reduction in thermal design flow.

5.2 Control Rod Drive Mechanism (CRDM) and Capped Latch Housing (CLH)

A review of the design values shows that the changes which would affect the CRDM and CLH are very small. The small temperature change would have a negligible effect on the analysis of the pressure boundary components, and there is no change in pressure. Therefore, it is concluded that compliance with the design criteria is not affected.

5.3 Reactor Coolant Pump and RCP Motor

The current design transients remain bounding, therefore only the effects of the changes to the design values were evaluated. A review of the design values shows that the changes which would affect the RCP are very small. The reactor coolant temperature change is small, and there is no change in pressure. Compliance with the design criteria is not affected.

The RCP motor evaluation shows that operation with the revised loads, caused by the revised design values, will not exceed NEMA temperature rise limits. Also, the rotor winding temperature rises, during worst case starting scenarios with the revised loads, have been evaluated. The calculated rotor winding temperature rises (based on a conservative all heat stored analysis) exceed the design allowances

for bars and for rings. The consequence of exceeding the design allowances for rotor winding temperature rise is an accelerated rate of mechanical aging (fatigue) which could result in a rotor winding failure before the 40 year design life of the motor has been achieved. It should be noted that failure in this case means a crack developing in the resistance ring which would, if not corrected, eventually cause a failure of the motor to start. There is no impact on the safety-related function of the motor (i.e., coastdown).

5.4 Pressurizer

The proposed change in the thermal design flow affects the temperatures to which the pressurizer is exposed. The evaluation concluded that the pressurizer components continue to meet the ASME Code, Section III stress analysis and fatigue analysis requirements.

5.5 Reactor Coolant Loop Piping and Primary Equipment Supports

The design values, thermal design transients, and LOCA loop forces are parameters that have a potential impact on the qualification of the reactor coolant loop piping and primary equipment supports. The change in these input parameters for the thermal design flow reduction for Beaver Valley Unit 1 is negligible as far as the loop structural analysis is concerned. The reduced thermal design flow is not expected to have an adverse impact on the design basis evaluation of the loop piping, the primary equipment supports, and the primary equipment nozzles.

6.0 STEAM GENERATOR

6.1 Thermal-Hydraulic Evaluation

The results of a thermal/hydraulic evaluation concluded that operation with the thermal design flow reduction was acceptable with the current hardware. Previous analyses were based on a power level of 887 MWt per steam generator and a steam pressure of 760 psia. These principal parameters, that is the power level and the secondary side steam pressure, are unchanged from previous analyses performed. Thus, the acceptability of the thermal/hydraulic operating characteristics continues to be applicable for the reduced thermal design flow conditions.

6.2 U-Bend Vibration

The primary parameters affecting U-bend vibration are the power level and the steam pressure. Earlier analyses for U-bend stability ratio were performed at the design values which were considered in the current analysis. Therefore, the fatigue usages are not affected. No remedial action is needed to prevent U-bend fatigue.

6.3 Structural Analysis

Previous structural analyses were based on a steam pressure of 790 psia. For the present study, the steam pressure was reduced to 760 psia. The structural analyses focused on the effects of reduced steam pressure resulting in an increased primary to secondary side pressure differential. The results indicated that the stresses are not significantly increased. The stress predictions are conservative due to the conservatism in the assumed pressure differential. Fatigue analyses performed show that acceptable fatigue usage factors can be demonstrated for the conditions encompassing the reduced thermal design flow.

7.0 AUXILIARY EQUIPMENT

7.1 Auxiliary Heat Exchanger/Tanks

The regenerative heat exchanger, residual heat exchanger, seal water heat exchanger, excess letdown heat exchanger, and letdown heat exchanger were evaluated for the reduced thermal design flow. In addition to the auxiliary heat exchangers, the only tanks that have transients identified are the boron injection tank (BIT) and the safety injection accumulators. As a result of the BIT boron concentration reduction program at Beaver Valley Unit 1, the original design transients of the BIT are no longer applicable. Therefore, the BIT is not impacted by the reduction in thermal design flow. Also, since the safety injection accumulator vessels do not have significant design transients requiring a fatigue analysis, they also are not impacted by the reduction in thermal design flow.

A review of the original design and qualification requirements for the Beaver Valley Unit 1 heat exchangers showed that the rerating parameters for the regenerative heat exchangers, the letdown heat exchangers, excess letdown heat exchangers, and residual heat exchangers are bounded by the original design parameters. The seal water heat exchangers were not required to be qualified for pressure or temperature transients. The transients were not included in the design, as they were not expected to have an effect on these components. Therefore, the equipment is designed for only maximum steady state pressures and temperatures, and the reduced thermal design flow will not impose any new limitations on the seal water heat exchangers.

7.2 Auxiliary Valves

The original design and qualification requirements of the auxiliary valves at Beaver Valley Unit 1 were evaluated, and it was concluded that the rerating parameters are bounded by the original design parameters.

7.3 Auxiliary Pumps

The charging/safety injection pumps, residual heat removal pumps, low pressure safety injection pumps, boron injection recirculation pumps, and boric acid transfer pumps were evaluated for the reduced thermal design flow. The specifications require the pumps to be qualified for pressure and temperature transients, or, if the equipment was not expected to be significantly affected by the transients, it was designed for maximum steady state pressures and temperatures only. The evaluation concluded that the design qualification for the charging/safety injection pumps, residual heat removal pumps, low pressure safety injection pumps, boron injection recirculation pumps, and boric acid transfer pumps remains bounding for the conditions of reduced thermal design flow.

8.0 FLUID SYSTEMS

The Reactor Coolant System, Chemical and Volume Control System, Residual Heat Removal System, and Safety Injection System were evaluated to determine if any parameters which changed as a result of the reduction in thermal design flow would affect the design adequacy of those systems. The result of the evaluation showed that the systems reviewed are adequate and acceptable for 100% power operation with the reduced thermal design flowrate of 87,200 gpm per loop.

ATTACHMENT 2

Summary of Analyses and Evaluations

Which Support a Reduced Minimum

RCS Total Flow Rate

For Beaver Valley Power Station Unit No. 2

EVALUATION

1.0 Non-LOCA Evaluation

The current non-LOCA safety analysis licensing basis for Beaver Valley Unit 2 assumes a total RCS thermal design flow (TDF) of 265,500 gpm (88,500 gpm per loop) and includes an evaluation supporting a maximum plugging level of 20% per steam generator at this TDF rate (total and per loop).

The non-LOCA evaluation considers a reduction in the Thermal Design Flow (TDF) to 261,600 gpm (87,200 gpm per loop) and continues to support up to 20% steam generator tube plugging.

All non-LOCA transients were examined to determine the effect of the reduced TDF. The non-LOCA accident analysis can be affected in the following ways by a reduction in TDF:

- Reduction in core thermal limits and calculated DNBR
- Change in plant normal operation conditions
- Impact on non-DNB Acceptance Criteria

For evaluation purposes, the non-LOCA transient analyses have been reviewed on the basis of both DNB and non-DNB acceptance criteria.

1.1 DNB Considerations

The affect of the TDF reduction on the following events has been evaluated to assure that the DNB design basis continues to be met:

- Feedwater System Malfunctions Causing an Increase in Feedwater Flow (UFSAR 15.1.2)

- Excessive Increase in Secondary Steam Flow (UFSAR 15.1.3)

- Loss of External Electrical Load (UFSAR 15.2.2), Turbine Trip (UFSAR 15.2.3)

- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (UFSAR 15.4.2)

- Start Up of an Inactive Reactor Coolant Loop (UFSAR 15.4.4)

- Inadvertent Operation of Emergency Core Cooling System During Power Operation (UFSAR 15.5.1)

- Inadvertent Opening of a Pressurizer Relief Valve (UFSAR 15.6.1)

- Inadvertent Opening of a Steam Generator Relief or Safety Valve Causing a Depressurization of the Main Steam System (UFSAR 15.1.4), Spectrum of Steam System Piping Failures Inside and Outside Containment (UFSAR 15.1.5)

- Partial Loss of Forced Reactor Coolant Flow (UFSAR 15.3.1)

- Complete Loss of Forced Reactor Coolant Flow (UFSAR 15.3.2)

- Reactor Coolant Pump Shaft Seizure (Locked Rotor Rod-in DNB) (UFSAR 15.3.3)

- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Start-up Condition (UFSAR 15.4.1)

A reduction in the thermal design flow has an adverse effect on the core thermal limits (DNB, quality and exit boiling) and consequentially the overtemperature and overpower ΔT analysis setpoint equations. The core thermal limits were revised to account for the reduction in TDF. Only the exit boiling portions of the core limits change since the current DNB limits, based on the W-3 R-grid DNB correlation, are more limiting than DNB limits based on the WRB-1 DNB correlation and mini-RTDP (the current design basis). The current OTΔT and OPΔT analysis setpoint equations were confirmed to provide protection for the revised core limits.

Small changes in plant operating conditions will not affect the transient statepoints used in the DNBR calculations, therefore, the transients conditions used to calculate the minimum DNBRs are still valid for the reduced TDF. A decrease in the RCS flow rate potentially decreases the minimum DNBR calculated during the event. Existing conservatism in the DNB calculations bound the effect on DNB due to the 1.5% flow reduction. For the Rod Cluster Control Assembly Misoperation transient (UFSAR 15.4.3), generic DNBR margin has been allocated to ensure that the DNB design basis continues to be met with the reduced TDF. For the Reactor Coolant Pump Shaft Seizure (Locked Rotor) transient (UFSAR 15.3.3), generic DNBR margin has been allocated to ensure that the limit of 18% rods-in-DNB continues to be met with the reduced TDF.

The reduced TDF, steam pressure and steam temperature result in a decrease in the initial mass in the steam generators. The combination of reducing TDF and increasing tube plugging would result in a reduction in the steam generator mass on the order of < 0.5% from the analysis values. Only the loss of heatsink transients (Loss of Non-Emergency AC Power, Loss of Normal Feedwater and Feedwater Line Break) are potentially impacted by this minimal decrease in initial steam generator mass. The remaining non-LOCA transients (including all of those analyzed for DNB considerations) are insensitive to minor changes to steam generator inventory.

1.2 Non-DNB Considerations

The effect of the TDF reduction has been evaluated to assure that the design basis continues to be met for the following events which are either not DNB related or for which DNBR is not the only relevant safety criterion:

- Loss of External Electrical Load (UFSAR 15.2.2), Turbine Trip (UFSAR 15.2.3)
- Loss of Non-Emergency AC Power to the Station Auxiliaries (Loss of Offsite Power) (UFSAR 15.2.6), Loss of Normal Feedwater (UFSAR 15.2.7)
- Feedwater System Pipe Break (UFSAR 15.2.8)
- Reactor Coolant Pump Shaft Seizure (Locked Rotor) (UFSAR 15.3.3)
- Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (UFSAR 15.4.2)
- Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (UFSAR 15.4.6)
- Spectrum of Rod Cluster Control Assembly Ejection Accidents (UFSAR 15.4.8)

1.2.1 Loss of External Electrical Load (UFSAR 15.2.2), Turbine Trip (UFSAR 15.2.3)

In addition to the DNBR requirement, the UFSAR analysis for this event must demonstrate that the primary and secondary system pressures remain below 110% of the design values. Whether from loss of external load or turbine trip, this transient is characterized by an increase in core power which exceeds the secondary side power extraction. This results in a primary side heat up and RCS pressure increase. Existing analysis has shown this transient to be insensitive with respect to the pressure limits, to a small change in RCS flow. Sufficient margin exists to the acceptance criteria. Therefore, the conclusions of the UFSAR remain valid.

1.2.2 Loss of Non-Emergency AC Power to the Station Auxiliaries (Loss of Offsite Power) (UFSAR 15.2.6), Loss of Normal Feedwater (UFSAR 15.2.7)

These transients are analyzed to demonstrate that the primary and secondary sides do not overpressurize and that the pressurizer does not overfill. This demonstrates that adequate auxiliary feedwater and steam generator inventory exists to remove decay heat and stored energy. These analyses are not impacted by small changes in nominal plant operating conditions. The reduced mass in the steam generators could adversely impact the results of the transient, however, a sensitivity analysis has shown that sufficient margin exists to the limit to accommodate the penalty incurred due to the reduced mass. Therefore, the conclusions of the UFSAR remain valid.

1.2.3 Feedwater System Pipe Break (UFSAR 15.2.8)

The UFSAR analysis demonstrates that adequate auxiliary feedwater exists to remove core decay heat and stored energy following a reactor trip from full power and that the core remains in a coolable geometry and covered with water. For ease of interpreting the transient, Westinghouse has adopted the restrictive criterion that no bulk boiling occurs in the primary coolant system following a Feedwater Pipe Break prior to the time that the heat removal capacity of the steam generators, being fed auxiliary feedwater, exceeds NSSS heat generation. This is determined by verifying that the RCS coolant remains subcooled. The analysis is not impacted by small changes in nominal plant operating conditions. The reduced mass in the steam generators could adversely impact the results of the transient, however, a sensitivity analysis has shown that sufficient margin exists to the limit to accommodate the penalty incurred due to the reduced mass. Therefore, the conclusions of the UFSAR remain valid.

1.2.4 Reactor Coolant Pump Shaft Seizure (Locked Rotor) (UFSAR 15.3.3)

This event is analyzed under full power conditions assuming the instantaneous seizure of one RCP rotor. This results in a rapid RCS flow reduction which may lead to DNB. The reactor is tripped promptly on a low flow signal. The analysis demonstrates that the maximum reactor coolant system pressure is less than 110% of design pressure, the maximum fuel clad temperature is less than 2700°F and the amount of zirconium-water reaction is small. In addition a calculation is made to predict the number of rods-in-DNB. The impact on the rods-in-DNB calculation has been discussed in Section 1.1 above. The system transient is not significantly impacted by the small (1.5%) reduction in TDF. The licensing basis analysis reports a PCT and peak pressure well below the limits of 2700°F and 2750 psia. Therefore, there is sufficient margin to accommodate the small changes that may result from the TDF reduction. Thus, the conclusions of the UFSAR remain valid.

1.2.5 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (UFSAR 15.4.2)

In addition to the DNBR requirement, the UFSAR analysis for this event must demonstrate that the pressurizer does not overfill. The peak pressurizer water volume is expected to increase with the reduction in TDF and increased tube plugging, since the RCS will heatup more than in the current analysis, due to the reduced heat transfer capability. The increased heatup results in a decrease in the coolant density which in turn would increase the pressurizer insurge. However, this effect is small. The UFSAR analysis shows that sufficient margin exists to accommodate the small changes that result from the TDF reduction. Therefore, the conclusions of UFSAR remain valid.

1.2.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (UFSAR 15.4.6)

This analysis demonstrates that sufficient shutdown margin exists, such that, should a dilution event occur, there is sufficient time to allow operator action and termination of the event prior to a complete loss of shutdown margin. The event is analyzed in Modes 1, 2 and 3. The flow reduction does not adversely impact the calculations. Therefore, the conclusions of the UFSAR remain valid.

1.2.7 Spectrum of Rod Cluster Control Assembly Ejection Accidents (UFSAR 15.4.8)

In this event a rapid reactivity insertion and increase in core power leads to high local fuel and clad temperatures and possible fuel and/or clad damage. The Rod Ejection event is analyzed at four conditions: beginning and end of life core physics characteristics (BOL, EOL) at hot zero power and full power (HZP, HFP). The analysis demonstrates that gross fuel damage will not occur, that the core remains in a coolable geometry and that the RCS will remain intact. The Rod Ejection event is characterized by a rapid excursion terminated by Doppler feedback. The reactor trips on High Neutron Flux. A reduction in the RCS flow will result in a reduction in the fuel rod to coolant heat transfer. This may result in an increase in the calculated fuel clad temperatures as well as the stored fuel energy. A sensitivity analysis has shown negligible impact on the analysis results (PCT, fuel temperatures) to a small change in RCS flow. Therefore, the conclusions of the UFSAR remain valid.

1.2.8 Steamline Break Mass/Energy Release - Inside/Outside Containment

The objective of these analyses is to maximize the release of high energy fluid. The reduction in TDF and increase in SGTP reduce the initial mass in the steam generators resulting in earlier tube uncovery. However, the TDF reduction and increased SGTP also reduces the primary to secondary heat transfer and the reactivity inserted due to the negative moderator temperature coefficient. Also, the reduction in initial secondary temperature and pressure would tend to lessen the mass and energy releases. These offsetting effects would not adversely affect the steamline break mass and energy releases inside or outside containment. Therefore, the steamline break mass and energy release inside and outside containment are considered to remain valid for the reduced TDF and increase SGTP.

1.3 Non-LOCA Results/Conclusions

Operation of Beaver Valley Unit 2 with a reduced thermal design flow of 261,600 gpm (87,200 gpm per loop) and a maximum plugging level of 20% per steam generator is acceptable from the standpoint of the non-LOCA analyses.

2.0 STEAM GENERATOR TUBE RUPTURE (SGTR) EVALUATION

For the Steam Generator Tube Rupture (SGTR) event, the Beaver Valley Unit 2 UFSAR SGTR analysis was performed using the LOFTRAN computer code. The primary to secondary break flow was assumed to be terminated at 30 minutes after the initiation of the SGTR event. The major factors that affect the radiological doses of the SGTR event are the amount of fuel failure, the amount of primary coolant transferred to the secondary side of the faulted steam generator through the faulted steam generator tube, and the steam released from the faulted steam generator to the atmosphere. An evaluation has been completed for the reduction in thermal design flow to 261,600 gpm together with up to 20% steam generator tube plugging to determine the impact on the UFSAR SGTR analysis.

Since the conservative technical specification coolant activities assumed for the Beaver Valley Unit 2 SGTR analysis will not change due to the reduced TDF, the major factors which impact the offsite radiation doses calculated for the UFSAR SGTR analysis are the primary to secondary break flow and the steam released from the faulted steam generator to the atmosphere. The parameters which are affected by the reduced TDF include the following: RCS flow, steam temperature and the initial mass and volume in the faulted steam generator.

Taken alone, a reduction in RCS flow would be expected to result in an earlier trip and earlier SI actuation. If the reduced flow would result in an earlier reactor trip and SI actuation, the break flow would be expected to increase. However, for the flow reduction being considered, the impact on reactor trip time is expected to be insignificant.

The reduction in flow, steam temperature and steam generator mass are competing effects on the heat transfer capability of the steam generator. Reduced heat transfer would tend to decrease the amount of steam released through the ruptured steam generators safety valve. Since the changes in these parameters are all minor, and calculated to maintain the same heat transfer capability (i.e., the nominal power level is unchanged), it is expected that the impact on the steam releases would be insignificant. There is sufficient margin between the analysis values of steam release and break flow and those reported in the UFSAR to conclude that the results of a 30 minute LOFTRAN analysis incorporating the reduced flow, and resulting parameter changes, would remain bounded by the UFSAR.

The overtemperature ΔT setpoints have been confirmed to protect the core limits with the revised flow. No changes to the setpoint will be made. Therefore, it is concluded that the reactor would trip on the low pressurizer pressure signal, as in the UFSAR analysis.

The reduction in thermal design flow from 88,500 to 87,200 gpm/loop and the associated changes in operating parameters will not result in a more limiting SGTR event than that presented in the UFSAR. Operation of Beaver Valley Unit 2 with the reduced flow and up to 20% steam generator tube plugging is acceptable with respect to the 30 minute LOFTRAN SGTR analysis.

3.0 LOCA

The following UFSAR LOCA related events were evaluated:

- Large Break LOCA (UFSAR Section 15.6.5)
- Small Break LOCA (UFSAR Section 15.6.5)
- Blowdown Reactor Vessel and Loop Forces (UFSAR Section 3.9N)
- Post-LOCA Long-Term Cooling, Subcriticality Evaluation (related to UFSAR Section 15.6.5)
- Hot Leg Switchover to Prevent Potential Boron Precipitation/Long Term SI Verification (UFSAR 6.3.2.5/Table 6.3-7)

3.1 Large and Small Break LOCA

The BVPS-2 LBLOCA analysis of record, which is presented in the UFSAR, is a BART Evaluation Model analysis with a PCT of 2120°F. Including Peak Clad Temperature (PCT) penalties which have been assigned; the most recent cumulative PCT is 2191°F.

The BVPS-2 SBLOCA analysis of record, which is presented in the UFSAR, is a NOTRUMP Evaluation Model analysis with a PCT of 1399°F. Including PCT penalties which have been assigned; the most recent cumulative PCT is 2119°F.

There are two main facets to the TDF reduction for ECCS LOCA analyses:

(1) Consideration of the RCS Flow
(2) Consideration of effects of RCS Temperature distribution

Within reasonable limits, such as the reduction from 88,500 gpm/loop to 87,200 gpm/loop being considered, RCS flow is a generally insignificant effect because the break flow dominates the transient almost immediately for both SBLOCA and LBLOCA. Therefore, the majority of the effect is realized through any changes to RCS Tavg that result. LOCA ECCS analysis are performed at 102% power as directed by 10 CFR 50 Appendix K. The initial RCS temperature distribution assumed by the LOCA analyses is determined using a complex methodology based upon 100% power design RCS conditions. Applying this methodology to the TDF reduction sequence, a small LOCA ECCS initial RCS Tavg reduction is predicted.

The available data for BART EM analyses indicates that an increase in RCS Tavg is limiting. Therefore, the TDF reduction results in an unquantified benefit for BVPS-2 LBLOCA analysis.

The available sensitivity analysis data for NOTRUMP EM analyses indicates that the limiting direction for RCS Tavg cannot be established without plant specific analysis. Sensitivities have been observed in either direction. The magnitudes of the available sensitivities indicate that a 1°F PCT penalty could be incurred for TDF reduction and has been conservatively assessed to the cumulative PCT summary.

During the evaluation process, anomalies were discovered in the interaction between the RCS temperature distribution methodology and the actual analysis inputs for both SBLOCA & LBLOCA. Action was taken to investigate and evaluate the anomalies. A portion of the apparent discrepancy is attributed to slight changes in the LOCA inputs that result from miscellaneous evolutionary changes to the plant characteristics such as the steam generator fouling factor, which was recently recalculated. Other differences are attributed to deviations that occurred in the LOCA analyses themselves such as an extraction error resulting in incorrect LBLOCA input. The evaluations for LBLOCA & SBLOCA follow.

> The LBLOCA analysis RCS Tavg inputs are somewhat greater than the value that is current using the ECCS methodology to either current TDF or reduced TDF cases, and thus the analysis remains bounding, though the benefit is unquantified.
>
> The SBLOCA analysis RCS Tavg inputs are somewhat greater than the value that is current using the ECCS methodology to either current TDF or reduced TDF cases, and thus a PCT penalty of 20°F is assigned.

Because a SBLOCA PCT penalty has been assessed, the 'Spike Burst & Blockage' PCT penalty that was transmitted recently to DLCo in the Cycle 4 RSE had to be re-evaluated. As discussed in the RSE, this item is associated with NRC Interim Report Issue 91-005 (ET-NRC-91-3647). The evaluation is repeated because the evaluation technique is highly PCT dependent, which generally reflects the exponential nature of the Zirc-Water reaction model employed in the NOTRUMP EM. For the revised cumulative PCT condition, the penalty for this item increases by 36°F.

The cumulative SBLOCA PCT is as follows:

2119°F	Current PCT With Assigned Penalties
+ 1°F	TDF Reduction
+ 20°F	Analysis RCS Tavg
+ 36°F	Spike Burst & Blockage Feedback (incremental, up to 1976°F Baseline)
=2176°F	Revised Cumulative PCT

The cumulative LBLOCA PCT is as follows:

2191°F	Current PCT With Assigned Penalties

Therefore, conformance with 10 CFR 50.46 PCT limit of 2200°F is maintained for both SBLOCA and LBLOCA.

3.2 Blowdown Reactor Vessel and Loop Forces

The Reactor Vessel LOCA forces conclusions are currently presented in WCAP-11523. Blowdown forces are typically limiting immediately after the break, and are influenced primarily by design Tcold. Design Tcold decreases slightly for the reduced TDF condition and LOCA forces slightly increase. However, the increase is accommodated within the margin available in the overall structural integrity evaluation. Therefore, the TDF reduction does not change the WCAP-11523 conclusions.

For the TDF reduction, these Blowdown Loop Forcing Functions would increase by 0.4%. The loop functions together with the remaining aspects of the TDF reduction program (RCS initial conditions, thermal design transients) have a potential impact on the qualification of the RCS loop piping and primary equipment supports. The change in all these input parameters is negligible as far as the loop structural analysis is concerned and will have negligible impact on the design basis evaluation of the loop piping, the primary equipment supports and the primary equipment nozzles.

3.3 Post-LOCA Long-Term Cooling, Subcriticality Evaluation

The Westinghouse position for satisfying the requirements of 10 CFR 50.46(b)(5) 'Long Term Cooling' is defined in WCAP-8339, WCAP-8472, and Technical Bulletin NSID-TB-86-08. The Westinghouse commitment is that the reactor will remain shutdown by borated ECCS water alone after a LOCA. Since credit for the control rods is not taken for a LBLOCA, the borated ECCS water provided by the accumulators and the RWST must have a concentration that, when mixed with other sources of borated and non-borated water, will result in the reactor core remaining subcritical assuming all control rods out. The TDF reduction does not alter the conclusion of the evaluation, which is checked by Westinghouse on a cycle by cycle basis at the time of the RSE, most recently the Cycle 4 RSE.

3.4 Hot Leg Switchover to Prevent Potential Boron Precipitation/Long Term SI Verification

Post-LOCA hot leg recirculation time is determined for inclusion in emergency procedures to ensure no boron precipitation in the reactor vessel following boiling in the core. This recirculation time is dependent upon power level, and the RCS, RWST, and accumulator water

volumes and boron concentrations. The TDF reduction has no effect on the post-LOCA hot leg switchover time. The long-term SI verification for Unit 2 is documented in Westinghouse letter to Duquesne Light Company titled "BVPS Unit 2 Recirc Spray Mod Safety Evaluation." Since the hot leg switchover time is unaffected, and SI performance is also unaffected by the TDF Reduction, the conclusions stated in the above mentioned letter are unaffected.

4.0 LOCA MASS AND ENERGY RELEASE CALCULATIONS

The current design basis LOCA mass and energy release calculations for Beaver Valley Unit 2 were reviewed for adequacy considering the following effects:

1) 11% Steam Generator Tube Plugging without any changes in thermal design flow rate.

2) 20% Steam Generator Tube Plugging with a total design flow reduction from 88,500 gpm to 87,200 gpm per loop.

3) Asymmetric loop flow under the following guidelines:

   The flow in any RCS loop shall be greater than 82,840 gpm (5% below the new TDF of 87,200 gpm). The combined flow from the two lowest flow loops shall be greater than 170,040 gpm, and the combined flow from all three loops shall be greater than 261,600 gpm.

The effects of an 11% to 20% steam generator tube plugging with and without changes in thermal design flow on postulated mass and energy releases was considered. The short-term release effects were addressed by performing a comparison of the total mass and energy release potential of the design condition versus the postulated changes. The governing design condition considered was a hot leg double-ended rupture. The long-term release effects were addressed by comparing changes in heat transfer rates of the steam generator secondary side to the primary side for the design condition versus the postulated changes. It was concluded that the existing analysis enveloped the proposed conditions for both short-term and long-term energy releases. Therefore, the design basis mass and energy release rates as stated in the current UFSAR remain bounding.

5.0 NSSS PRIMARY COMPONENTS

5.1 Reactor Pressure Vessel System

The reactor pressure vessel system consists of the reactor vessel, the reactor upper and lower internals assemblies and the reactor core. Since these components are interdependent from a thermal-hydraulic and structural viewpoint, they are evaluated as a system. The reactor pressure vessel system is sensitive to variations in the reactor coolant system flowrate. Therefore, the reactor pressure vessel system was evaluated with respect to the reduction in the thermal design flow.

New flows and pressure drops were calculated for the various flow paths within the reactor pressure vessel system. The results showed that the changes in pressure drops associated with the new operating conditions are evenly distributed throughout the reactor internals, and that the total pressure drop across the internals would decrease an insignificant amount. Since the internals flow and pressure drop changes are not changed significantly by the new operating conditions, detailed calculations of the effect on core bypass flow, hydraulic lift forces, flow induced vibration and Rod Control Cluster Assembly (RCCA) rod drop times were not necessary.

The second result of the thermal design flow reduction is an increase in the temperature rise across the reactor vessel (i.e., hot leg temperature - cold leg temperature). For the approximate 1.5% thermal design flow reduction, included in the revised Beaver Valley Unit 2 operating conditions, the delta-T increases by 1 degree F. Temperature variations of this magnitude are bounded by the original structural analyses of the Beaver Valley Unit 2 internals.

The evaluation of the reactor pressure vessel system demonstrated that there would be no adverse impact on the performance of the system by the proposed reduction in thermal design flow.

5.2 Control Rod Drive Mechanism and Capped Latch Housing

A review of the design values shows that the changes which would affect the CRDM and CLH are very small. The small temperature change would have a negligible effect on the analysis of the pressure boundary components, and there is no change in pressure. Therefore, it is concluded that compliance with the design criteria is not affected.

5.3 Reactor Coolant Pump and RCP Motor

The current design transients remain bounding, therefore only the effects of the changes to the design values were evaluated. A review of the design values shows that the changes which would affect the RCP are very small. The reactor coolant temperature change is small, and there is no change in pressure. Compliance with the design criteria is not affected.

The RCP motor evaluation shows that operation with the revised loads, caused by the revised design values, will not exceed NEMA temperature rise limits. Also, the rotor winding temperature rises, during worst case starting scenarios with the revised loads, are less than the design allowances and are, therefore, acceptable.

5.4 Pressurizer

The proposed change in the thermal design flow affects the temperatures to which the pressurizer is exposed. The evaluation concluded that the pressurizer components continue to meet the ASME Code, Section III stress analysis and fatigue analysis requirements.

5.5 Reactor Coolant Loop Piping and Primary Equipment Supports

The design values, thermal design transients, and LOCA loop forces are parameters that have a potential impact on the qualification of the reactor coolant loop piping and primary equipment supports. The change in these input parameters for the thermal design flow reduction for Beaver Valley Unit 2 is negligible as far as the loop structural analysis is concerned. The reduced thermal design flow is not expected to have an adverse impact on the design basis evaluation of the loop piping, the primary equipment supports, and the primary equipment nozzles.

6.0 STEAM GENERATOR

6.1 Thermal-Hydraulic Evaluation

The results of a thermal/hydraulic evaluation concluded that operation with the thermal design flow reduction was acceptable with the current hardware. Previous analyses were based on a power level of 887 MWt per steam generator and a steam pressure of 760 psia. These principal parameters, that is the power level and the secondary side steam pressure, are unchanged from previous analyses performed. Thus, the acceptability of the thermal/hydraulic operating characteristics continues to be applicable for the reduced thermal design flow conditions.

6.2 U-Bend Vibration

The primary parameters affecting U-bend vibration are the power level and the steam pressure. Earlier analyses for U-bend stability ratio were performed at the design values which were considered in the current analysis. Therefore, the fatigue usages are not affected. No remedial action is needed to prevent U-bend fatigue.

6.3 Structural Analysis

Previous structural analyses were based on a steam pressure of 790 psia. For the present study, the steam pressure was reduced to 760 psia. The structural analyses focused on the effects of reduced steam pressure resulting in an increased primary to secondary side pressure differential. The results indicated that the stresses are not significantly increased. The stress predictions are conservative due to the conservatism in the assumed pressure differential. Fatigue analyses performed show that acceptable fatigue usage factors can be demonstrated for the conditions encompassing the reduced thermal design flow.

7.0 AUXILIARY EQUIPMENT

7.1 Auxiliary Heat Exchanger/Tanks

The regenerative heat exchanger, residual heat exchanger, seal water heat exchanger, excess letdown heat exchanger, and letdown heat exchanger were evaluated for the reduced thermal design flow. In addition to the auxiliary heat exchangers, the only tank that has a transient identified is the safety injection accumulators. Since the safety injection accumulator vessels do not have significant design transients, they are not impacted by the reduced thermal design flow.

A review of the original design and qualification requirements for the Beaver Valley Unit 2 heat exchangers showed that the rerating parameters for the regenerative heat exchangers, the letdown heat exchangers, excess letdown heat exchangers, and residual heat exchangers are bounded by the original design parameters. The seal water heat exchangers were not required to be qualified for pressure or temperature transients. The transients were not included in the design, as they were not expected to have an effect on these components. Therefore, the equipment is designed for only maximum steady state pressures and temperatures, and the reduced thermal design flow will not impose any new limitations on the seal water heat exchangers.

7.2 Auxiliary Valves

The original design and qualification requirements of the auxiliary valves at Beaver Valley Unit 2 were evaluated, and it was concluded that the rerating parameters are bounded by the original design parameters.

7.3 Auxiliary Pumps

The charging/safety injection pumps, residual heat removal pumps, low pressure safety injection pumps, boron injection recirculation pumps, and boric acid transfer pumps were evaluated for the reduced thermal design flow. The specifications require the pumps to be qualified for pressure and temperature transients, or, if the equipment was not expected to be significantly affected by the transients, it was designed for maximum steady state pressures and temperatures only. The evaluation concluded that the design qualification for the charging/safety injection pumps, residual heat removal pumps, low pressure safety injection pumps, boron injection recirculation pumps, and boric acid transfer pumps remains bounding for the conditions of reducing thermal design flow.

8.0 FLUID SYSTEMS

The Reactor Coolant System, Chemical and Volume Control System, Residual Heat Removal System, and Safety Injection System were evaluated to determine if any parameters which changed as a result of the reduction in thermal design flow would affect the design adequacy of those systems. The result of the evaluation showed that the systems reviewed are adequate and acceptable for 100% power operation with the reduced thermal design flowrate of 87,200 gpm per loop.

ATTACHMENT C-1

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Change No. 208

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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 in Figure 2.1-1 for 3 loop operation.

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.

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.

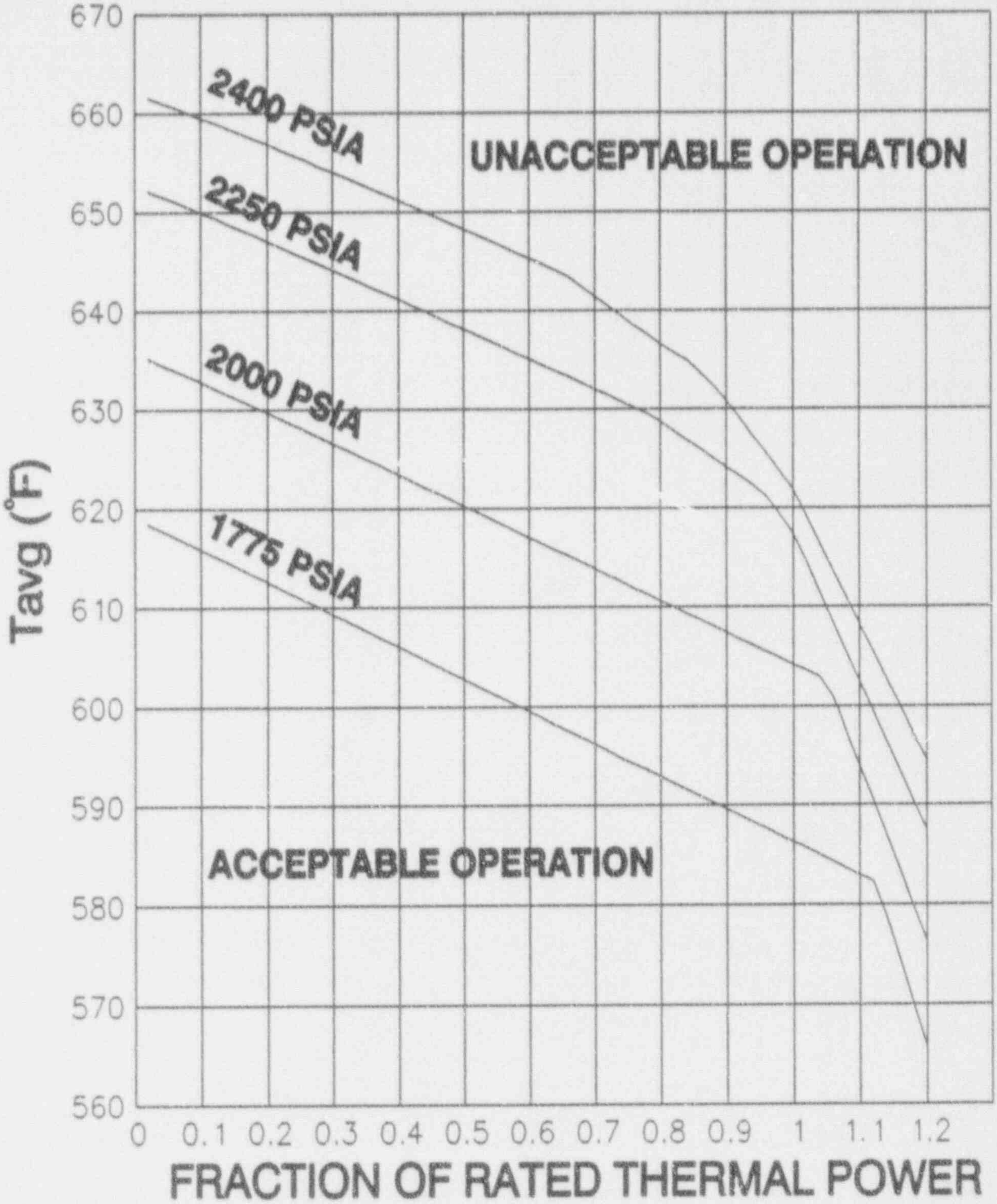

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT
THREE LOOP OPERATION

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - ≤ 25% of RATED THERMAL POWER High Setpoint - ≤ 109% of RATED THERMAL POWER	Low Setpoint - ≤ 27.3% of RATED THERMAL POWER High Setpoint - ≤ 111.3% 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% 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% of RATED THERMAL POWER with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	≤ 25% of RATED THERMAL POWER	≤ 31.1% of RATED THERMAL POWER
6. Source Range, Neutron Flux	≤ 10^5 counts per second	≤ 1.4 x 10^5 counts per second
7. Overtemperature ΔT	See Note 1	See Note 3
8. Overpower ΔT	See Note 2	See Note 4
9. Pressurizer Pressure--Low	≥ 1945 psig	≥ 1934 psig
10. Pressurizer Pressure--High	≤ 2385 psig	≤ 2394 psig
11. Pressurizer Water Level--High	≤ 92% of instrument span	≤ 93.9% of instrument span
12. Loss of Flow	≥ 90% of design flow* per loop	≥ 89.0% of design flow* per loop

*Design flow is 87,200 gpm per loop.

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 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. 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. 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 2.2-1.

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)]:

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1[(2)].

ACTION:

With any of the above parameters exceeding its limit, restore the 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.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be indicating 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 by measurement at least once per 18 months.

---

(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.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	3 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 580.7°F
Pressurizer Pressure	$\geq$ 2220 psia(1)
Reactor Coolant System Total Flow Rate	$\geq$ 261,600 gpm

---

(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.

ATTACHMENT C-2

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Change No. 74

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2-2
2-4
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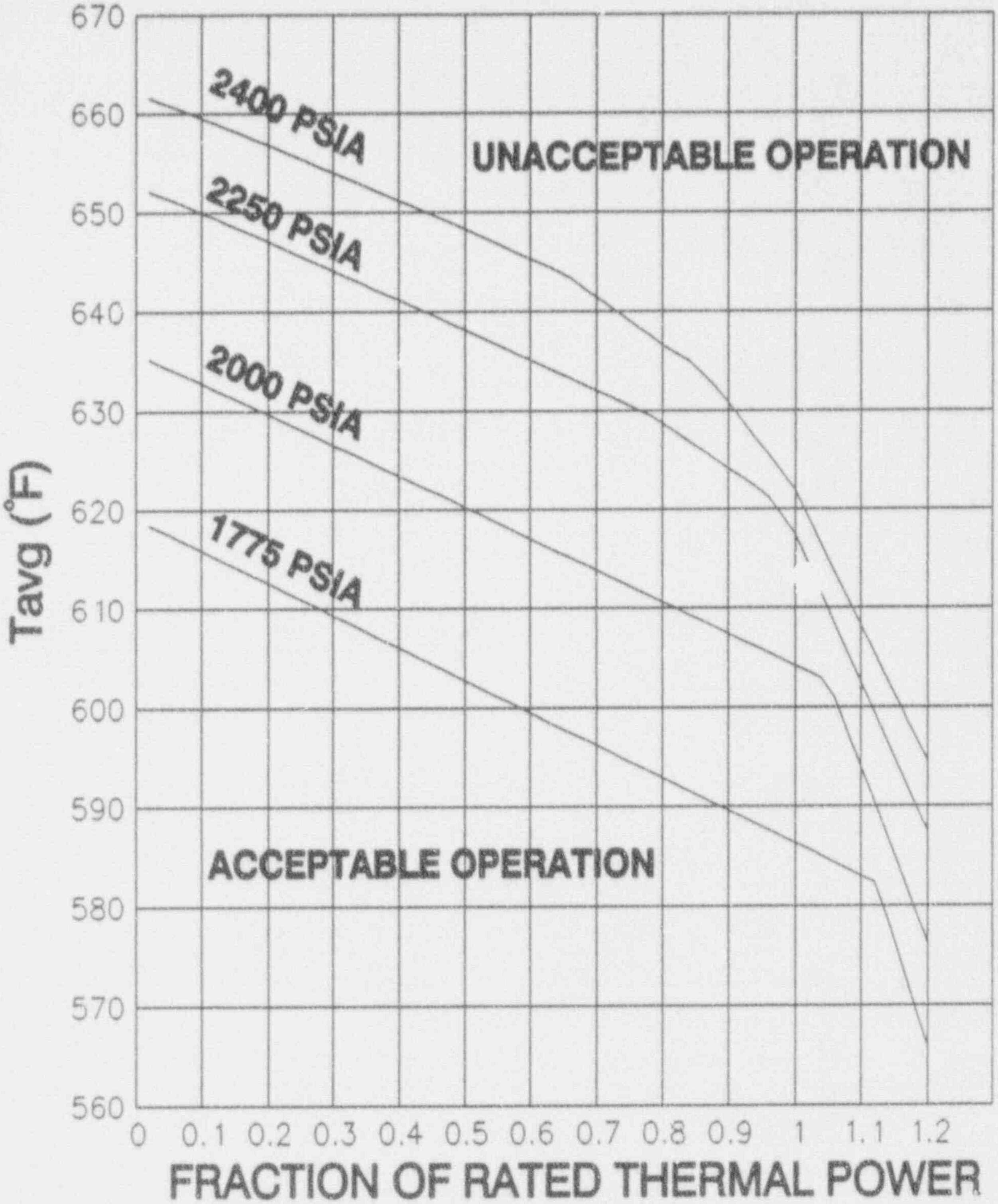

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT
THREE LOOP OPERATION

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	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% of RTP*
b. Low Setpoint	8.3	4.56	0	≤ 25% RTP*	≤ 27.1% of RTP*
3. Power Range, Neutron Flux High Positive Rate	1.6	0.50	0	≤ 5% of RTP* with a time constnat ≥ 2 seconds	≤ 6.3% 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 constnat ≥ 2 seconds	≤ 6.3% of RTP* with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% RTP*	≤ 30.9% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7. Overtemperature ΔT	7.0	5.10	See Note 5	See Note 1	See Note 2
8. Overpower ΔT	4.9	1.71	1.49	See Note 3	See Note 4
9. Pressurizer Pressure-Low	3.1	0.71	1.67	≥ 1945 psig***	≥ 1935 psig***
10. Pressurizer Pressure-High	6.2	4.96	0.67	≤ 2375 psig	≤ 2383 psig
11. Pressurizer Water Level-High	8.0	2.18	1.67	≤ 92% of instrument span	≤ 93.8% of instrument span
12. Loss of Flow	2.5	1.39	0.60	≥ 90% of loop design flow**	≥ 88.9% of loop design flow**

* = RATED THERMAL POWER
** Loop design flow = 87,200 gpm
*** Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are 2 seconds for lead and 1 second for lag. Channel calibration shall ensure that these time constants are adjusted to those values

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]:

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1[2].

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its 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 parameters of Table 3.2-1 shall be verified to be indicating 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 by measurement at least once per 18 months.

---

(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.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	3 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 580.2°F
Pressurizer Pressure	$\geq$ 2220 psia(1)
Reactor Coolant System Total Flow Rate	$\geq$ 261,600 gpm

(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.

ATTACHMENT D-1

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Change No. 208

Applicable UFSAR Changes

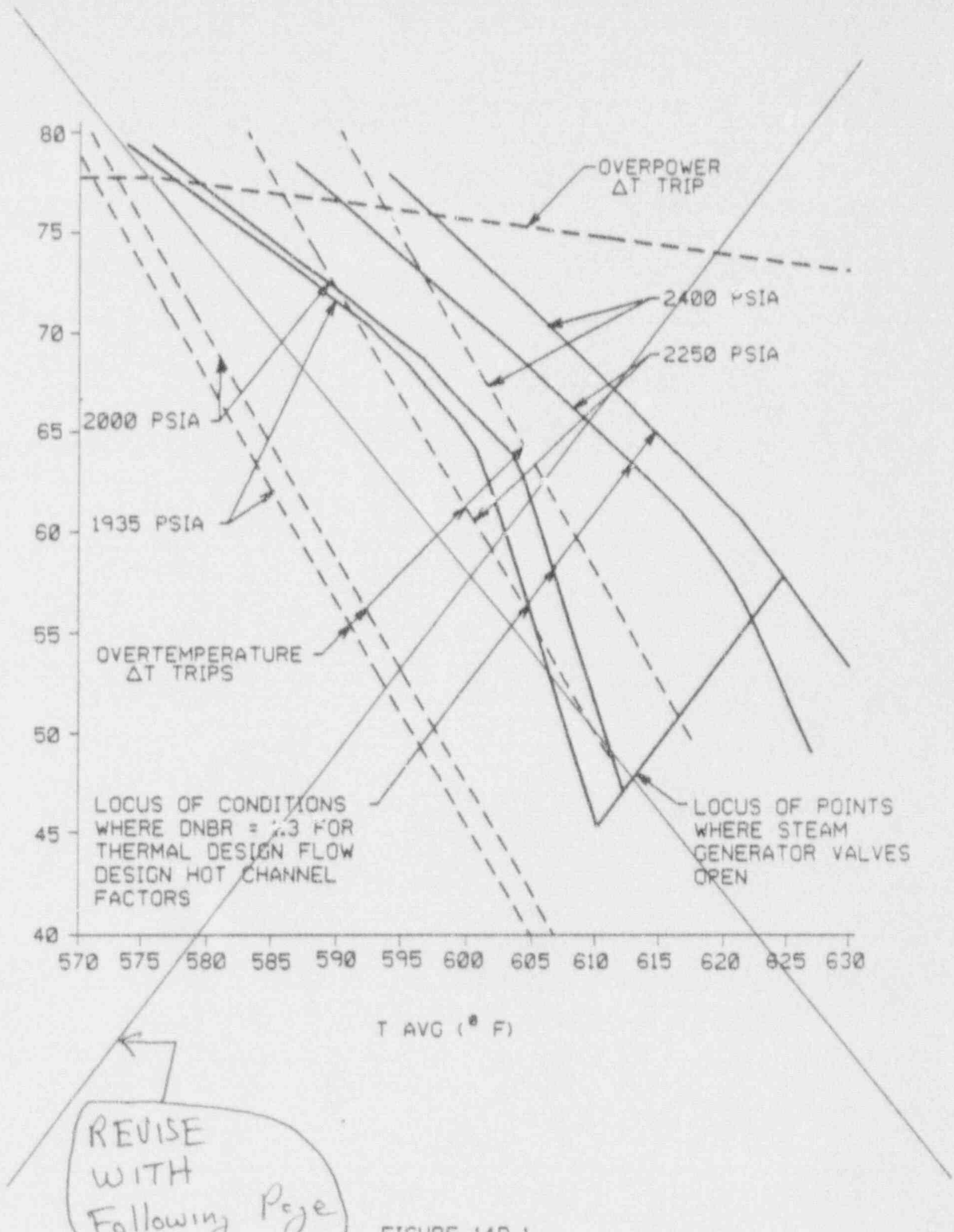

FIGURE 14D-1
ILLUSTRATION OF OVERTEMPERATURE AND OVERPOWER ΔT PROTECTION
BEAVER VALLEY POWER STATION-UNIT 1
UPDATED FINAL SAFETY ANALYSIS REPORT

(Proposed)

ADD

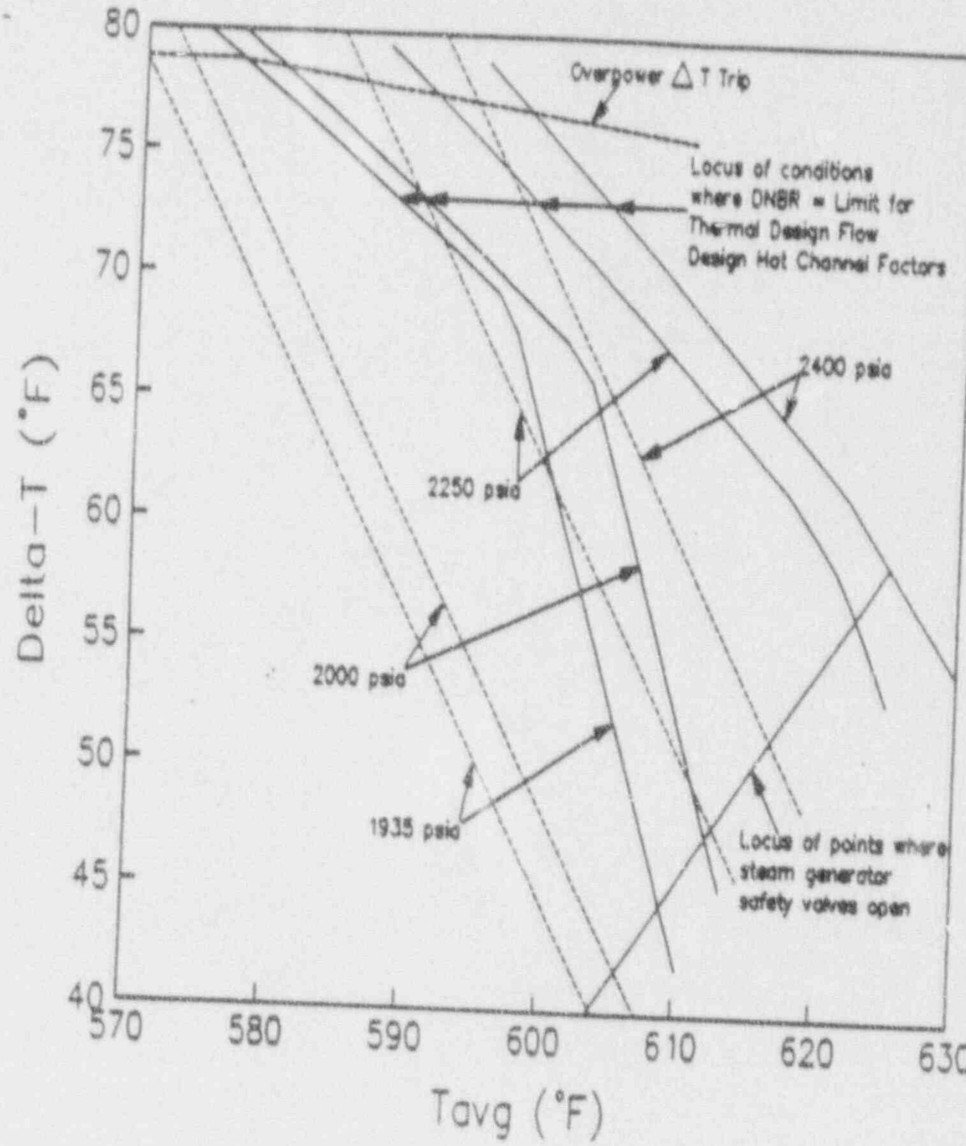

FIGURE 14D-1
ILLUSTRATION OF OVERTEMPERATURE AND OVERPOWER ΔT PROTECTION

BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

(Proposed)

ATTACHMENT D-2

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Change No. 74

Applicable UFSAR Changes

TABLE 5.1-1

REACTOR COOLANT SYSTEM
DESIGN AND OPERATING PARAMETERS

Characteristics	Parameters
Plant design life (years)	40
Nominal operating pressure (psig)	2,235
Total system volume including pressurizer and surge line (ft^3)	9,370
System liquid volume, including pressurizer water at maximum guaranteed power (ft^3)	8,853
Pressurizer spray rate, maximum (gpm)	600
Pressurizer heater capacity (kW)	1,400
Pressurizer relief tank volume (ft^3)	1,300

System Thermal and Hydraulic Data

	3 Pumps Running	One Loop Isolated
NSSS power (MWt)	2,660	1,729
Reactor power (MWt)	2,652	1,724
Thermal design flows (gpm)		
Active loop	87,200 ~~88,500~~	93,900
Idle loop	--	--
Reactor	261,600 ~~265,500~~	187,800
Total reactor flow (lb/hr x 10^6)	99.4 ~~100.8~~	72.0
Temperatures (°F)		
Reactor vessel outlet	610.4 ~~606.0~~	593.8
Reactor vessel inlet	542.0 ~~544.0~~	535.9
Steam generator outlet	~~543.8~~ 541.8	535.7
Steam generator steam	~~516.8~~ 512.1	514.1
Feedwater	437.5	391.0
Steam pressure (psia)	760 ~~790~~	736
Total steam flow (lb/hr x 10^6)	11.6	7.1
Best estimate flows (gpm)		
Active loop	96,800	101,500
Reactor	290,400	203,000
Mechanical design flows (gpm)		
Active loop	100,600	68,300
Reactor	301,800	211,000

(Proposed)

TABLE 15.0-2

BASES FOR VALUES OF PERTINENT PLANT PARAMETERS
UTILIZED IN ACCIDENT ANALYSES

Plant Parameter	N Loop Operation	N-1 Loop Operation
Thermal output of nuclear steam supply system (MWt)	2,660	1,729
Reactor core thermal power output (MWt)	2,652	1,724
Core inlet temperature (°F)	542.0 ~~542.5~~	534.4
Reactor coolant average temperature (°F)	576.2	566.0
Reactor coolant system pressure (psia)	2,250	2,250
Reactor coolant flow per loop (gpm)	87,200 ~~88,500~~	93,900 (active loops) 0 (inactive loop)
Total reactor coolant flow (10^6 lb/hr)	99.4 ~~100.8~~	72.05
Total steam flow from NSSS (10^6 lb/hr)	11.60 ~~11.61~~	7[illegible]0
Steam pressure at steam generator outlet (psia)	760 ~~790~~	772
Maximum steam moisture content (percent)	0.25	0.25
Feedwater temperature at steam generator inlet (°F)	437.5	391
Average core heat flux (1,000 BTU/hr-/ft²)	181	118

(Proposed)

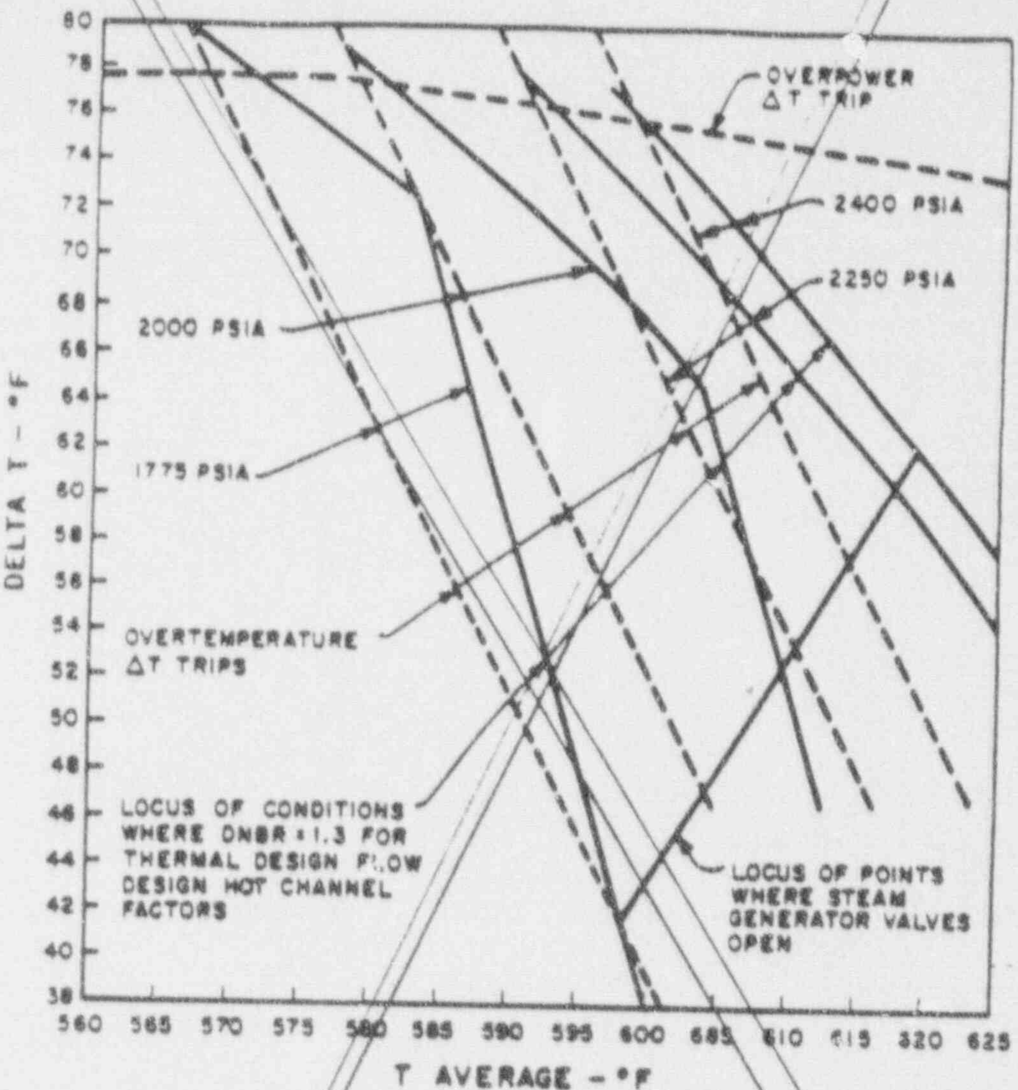

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FIGURE 15.0-1
ILLUSTRATION OF OVERPOWER AND OVERTEMPERATURE ΔT PROTECTION (N LOOP OPERATION)
BEAVER VALLEY POWER STATION- UNIT 2
FINAL SAFETY ANALYSIS REPORT

(Proposed)

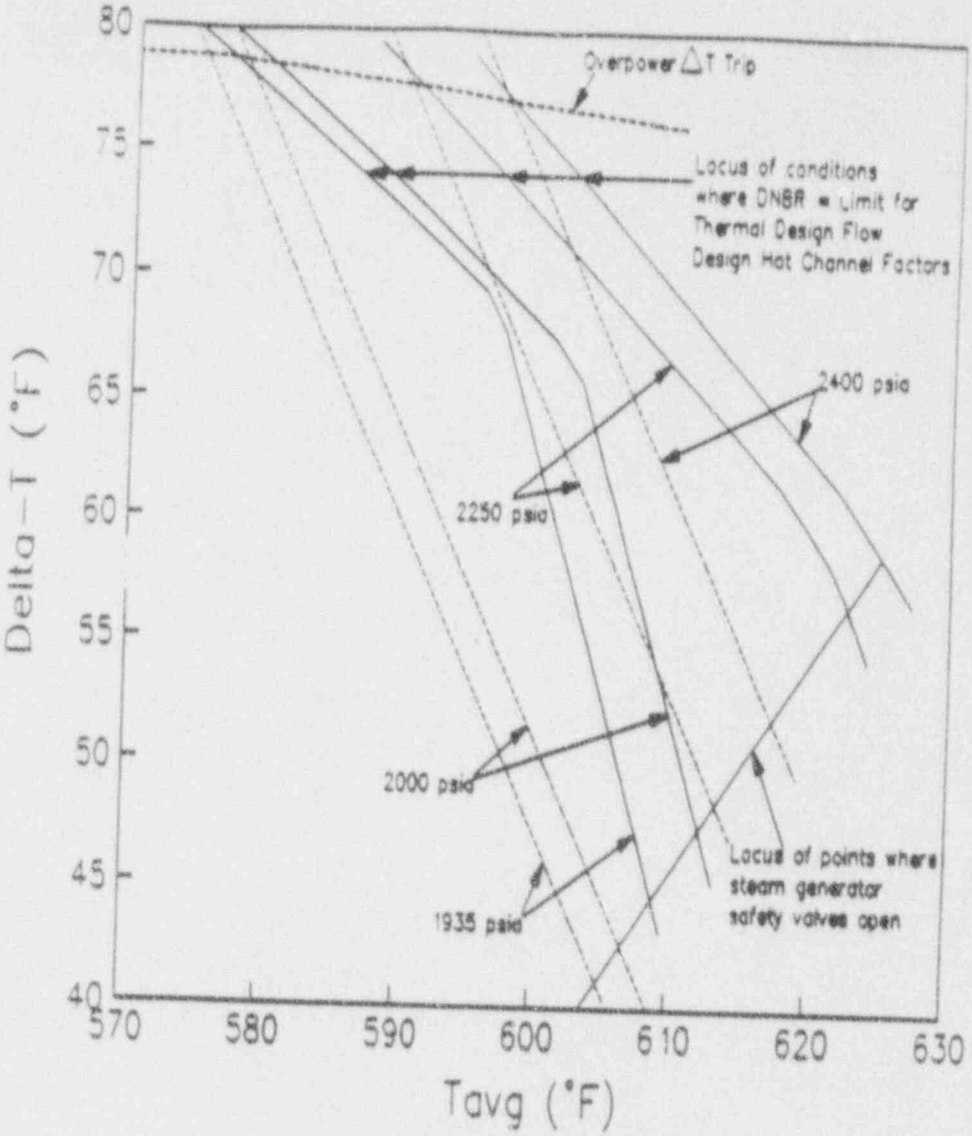

FIGURE 15.0-1
ILLUSTRATION OF OVERTEMPERATURE
AND OVERPOWER ΔT
PROTECTION (N LOOP OPERATION)

BEAVER VALLEY POWER STATION - UNIT 2
FINAL SAFETY ANALYSIS REPORT

(Proposed)