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January 17, 2001
CAW 01-01

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Caldon ER-157P, "Engineering Report – 157P: Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ or LEFM CheckPlus™ System", Rev. 2 enclosure – FENOC Letter, L-01-006, "License Amendment Request Nos. 289 and 161"

Gentlemen:

This application for withholding is submitted by Caldon, Inc. ("Caldon") pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Caldon and customarily held in confidence.

The proprietary information for which withholding is being requested is identified in the subject submittal. In conformance with 10 CFR Section 2.790, Affidavit CAW-01-01 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information, which is proprietary to Caldon, be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW-01-01 and should be addressed to the undersigned.

Very truly yours,

Calvin R Hastings

Calvin R. Hastings
President and CEO

Enclosures

January 17, 2001
CAW-01-01

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Calvin R. Hastings, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Caldon, Inc. ("Caldon") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Calvin R. Hastings

Calvin R. Hastings,
President and CEO
Caldon, Inc.

Sworn to and subscribed before me

this 17th day of

January, 2001

Notarial Seal
Joann B. Thomas, Notary Public
Pittsburgh, Allegheny County
My Commission Expires July 28, 2003

1. I am the President and CEO of Caldon, Inc. and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of Caldon.

2. I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Caldon application for withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by Caldon in designating information as a trade secret, privileged or as confidential commercial or financial information.

4. Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Caldon.

   (ii) The information is of a type customarily held in confidence by Caldon and not customarily disclosed to the public. Caldon has a rational basis for determining the types of information customarily held in confidence by it and, in that connection utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Caldon policy and provides the rational basis required.

   Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Caldon's competitors without license from Caldon constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, and assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Caldon, its customer or suppliers.

(e) It reveals aspects of past, present or future Caldon or customer funded development plans and programs of potential customer value to Caldon.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Caldon system, which include the following:

(a) The use of such information by Caldon gives Caldon a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Caldon competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Caldon ability to sell products or services involving the use of the information.

(c) Use by our competitor would put Caldon at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Caldon of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Caldon in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Caldon capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence, and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in the enclosure (Caldon ER-157P) to FENOC Beaver Valley LLC Letter L-01-006 from Lew W. Myers to the NRC Document Control Desk, "License Amendment Request Nos. 289 and 161". This information is submitted for use by the NRC Staff and is expected to be applicable in other license submittals for justification of the use of Ultrasonic Flow Measurement Instrumentation to increase reactor plants' thermal power.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Caldon because it would enhance the ability of competitors to provide similar flow and temperature measurement systems and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Caldon effort and the expenditure of a considerable sum of money.

In order for competitors of Caldon to duplicate this information, similar products would have to be developed, similar technical programs would have to be performed, and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.

ATTACHMENT A-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 289

The following is a list of the affected pages:

Affected Pages:

Operating License Page 3

Technical Specification Pages
XVII
1-1
3/4 7-1
3/4 7-2
B 3/4 7-1
B 3/4 7-1a
B 3/4 7-1b
B 3/4 7-1c
B 3/4 7-1d
6-18

(3) FENOC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) FENOC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) FENOC, pursuant to the Act and 10 CFR Parts 30, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

FENOC is authorized to operate the facility at a steady state reactor core power level of ~~2652~~ 2689 megawatts thermal.

(2) Technical Specifications

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

(3) Auxiliary River Water System

(Deleted by Amendment No. 8)

(Proposed Wording)

(Proposed Wording)

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~2652~~ 2689 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related safety function(s).

(Proposed Wording)

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

MAIN STEAM SAFETY VALVES (MSSVs)

LIMITING CONDITION FOR OPERATION

Five

3.7.1.1 ~~The~~ MSSVs per steam generator shall be OPERABLE ~~as specified in Table 3.7-1 and Table 3.7-2~~.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

\- - - - - - - - - - - - - GENERAL NOTE - - - - - - - - - - - - - - - -

Separate ACTION entry is allowed for each MSSV.

\- -

INSERT 1

~~a.~~ ~~With one or more required MSSVs inoperable, within 4 hours reduce power to less than or equal to the applicable percent RATED THERMAL POWER listed in Table 3.7-1; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 6 hours.~~

c ~~b.~~ With one or more steam generators with four or more ~~less than two~~ MSSVs inoperable ~~OPERABLE~~ within 6 hours be in HOT STANDBY and in HOT SHUTDOWN within the next 6 hours.

d ~~c.~~ The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Verify ~~(1)~~ (2) each required MSSV lift setpoint per Table 3.7-2 in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1 percent.

(2) ~~(1)~~ Required to be performed only in MODES 1 and 2.

(1) Required to be performed only in MODE 1

ATTACHMENT A-1

Unit 1 Inserts

Unit 1 INSERT 1

a. With one or more steam generators with one MSSV inoperable and the Moderator Temperature Coefficient (MTC) zero or negative at all power levels, within 4 hours reduce THERMAL POWER to less than or equal to 61% RTP; otherwise, be in HOT STANDBY within the next 6 hours, and in HOT SHUTDOWN within the next 6 hours.

b. With one or more steam generators with two or more MSSVs inoperable, or with one or more steam generators with one MSSV inoperable and the MTC positive at any power level, within 4 hours reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs, and reduce the Power Range Neutron Flux-High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs within the next 32 hours(1); otherwise, be in HOT STANDBY within the next 6 hours, and in HOT SHUTDOWN within the next 6 hours.

TABLE 3.7-1

Maximum Allowable

OPERABLE Main Steam Safety Valves versus
~~Applicable~~ Power ~~in Percent of RATED THERMAL POWER (RTP)~~

~~MINIMUM~~ NUMBER OF MSSVs PER STEAM GENERATOR ~~REQUIRED~~ OPERABLE	MAXIMUM ALLOWABLE ~~APPLICABLE~~ POWER (% RTP)
~~5~~	~~≤ 100~~
4	≤ ~~57~~ 56
3	≤ ~~39~~ 40
2	≤ ~~22~~ 24

(Proposed Wording)

No Change
Included for Information

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER	LIFT SETTING*** (+1% -3%)	ORIFICE DIAMETER
a.	SV-MS101A, B & C	1075 psig	4.250 in.
b.	SV-MS102A, B & C	1085 psig	4.515 in.
c.	SV-MS103A, B & C	1095 psig	4.515 in.
d.	SV-MS104A, B & C	1110 psig	4.515 in.
e.	SV-MS105A, B & C	1125 psig	4.515 in.

*** The Lift Setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 MAIN STEAM SAFETY VALVES (MSSVs)

BACKGROUND

The primary purpose of the main steam safety valves (MSSVs) is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the UFSAR, Section 10.3.1. The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 12.8 x 10^6 lbs/hr which is ~~110~~ 108 percent of the total secondary steam flow of ~~11.7~~ 11.8 x 10^6 lbs/hr at 100% RATED THERMAL POWER. The MSSV design includes staggered setpoints, according to Table 3.7-2 in the accompanying limiting condition for operation (LCO), so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip. INSERT 2

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from the ASME Code, Section III and its purpose is to limit the secondary system pressure to less than or equal to 110 percent of design pressure ~~when passing 100 percent of design steam flow. This design basis is sufficient to cope with~~ for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in UFSAR, Section 14.1. Of these, the full power turbine trip without steam dump is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The Safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. INSERT 3 ~~If a minimum reactivity feedback is assumed, the reactor is~~

Unit 1 INSERT 2

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the ASME B&PV Code) for the worst-case loss-of-heat-sink event. Based on this requirement, a conservative criterion was applied that the valves should be sized to relieve 100 percent of the maximum calculated steam flow at an accumulation pressure (5 percent) not exceeding 110 percent of the design pressure.

Unit 1 INSERT 3

One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer relief valves and spray. This analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The UFSAR, Section 14.1 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The UFSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron

Flux-High reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. If the Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions, unless it is demonstrated by analysis that a specified reactor power reduction alone is sufficient to prevent overpressurization of the steam system.

MAIN STEAM SAFETY VALVES (MSSVs) (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

~~tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110 percent of the design value. The MSSVs also open to limit the secondary steam pressure.~~

~~If maximum reactivity feedback is assumed, the reactor is tripped on overtemperature ΔT. The departure from nucleate boiling ratio increases throughout the transient, and never drops below its initial value. Pressurizer relief valves and MSSVs are activated and prevent overpressurization in the primary and secondary systems.~~ ¶ The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

LCO

The accident analysis requires that five ~~four~~ MSSVs and per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at ~~102~~ 100.6 percent RATED THERMAL POWER (RTP). ~~An MSSV will be considered inoperable if it fails to open on demand.~~ The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with the ASME Code, Section III, ~~even though this is not a requirement of~~ the DBA analysis. ~~This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7-1 in the accompanying LCO and associated ACTION.~~

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

~~The lift settings, according to Table 3.7-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure, as identified by a note.~~

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or main steam system integrity.

6

BASES

MAIN STEAM SAFETY VALVES (MSSVs) (Continued)

APPLICABILITY

In ~~MODE 1 above 22% RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7-1 in the accompanying LCO. In MODE 1 below 22% RTP and in~~ MODES 1, 2, and 3 ~~only two~~ five MSSVs per steam generator are required to be OPERABLE, to prevent Main Steam System overpressurization.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS are modified by a General Note indicating that separate condition entry is allowed for each MSSV.

~~a.~~ With one or more MSSVs inoperable, action must be taken ~~reduce power~~ so that the available MSSV relieving capacity meets the ASME Code, Section III requirements ~~for the applicable THERMAL POWER.~~

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

INSERT 4

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined ~~is limited~~ by the governing heat transfer relationship from the equation ~~in the relationship~~ $q = m\Delta h$, where q is the heat input from the primary side, m is the steam flow rate and Δh is the heat of vaporization at the steam relief pressure (assuming no subcooled feedwater). For each steam generator, at a specified pressure, the maximum allowable ~~fractional~~ power level ~~(FPL)~~ is determined as follows:

$$\text{Maximum Allowable Power Level} \le \cancel{FPL =}\ 100/Q \ \frac{(w_s\, h_{fg}\, N)}{K} \ \cancel{- 9}$$

(Proposed Wording)

Unit 1 INSERT 4

a. In the case of only a single inoperable MSSV on one or more steam generators, if the Moderator Temperature Coefficient is not positive, a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. Furthermore, for this case there is sufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, ACTION a. requires an appropriate reduction in reactor power within 4 hours.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as discussed below, with an appropriate allowance for calorimetric power uncertainty.

MAIN STEAM SAFETY VALVES (MSSVs) (Continued)

ACTIONS (Continued)

where:

~~FPL = Fraction of RATED THERMAL POWER equivalent to the safety analysis limit minus 9 percent (to account for typical instrument and channel uncertainties). The uncertainty ensures the maximum plant operating power level will then be lower than the safety analysis limit by an appropriate operating margin.~~

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt

K = Conversion factor, 947.82 $\frac{\text{(Btu/sec)}}{\text{Mwt}}$

w_s = Minimum total steam flow rate capability of the ~~operable~~ MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec. For example, if the maximum number of inoperable MSSVs on any one steam generator is one, then w_s should be a summation of the capacity of the ~~operable~~ MSSVs at the highest ~~operable~~ MSSV operating pressure, excluding the highest capacity MSSV. If the maximum number of inoperable MSSVs per steam generator is three then w_s should be a summation of the capacity of the ~~operable~~ MSSVs at the highest ~~operable~~ MSSV operating pressure, excluding the three highest capacity MSSVs.

OPERABLE

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm

N = Number of loops in plant

INSERT 5

(Proposed Wording)

Unit 1 INSERT 5

b. In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Furthermore, for a single inoperable MSSV on one or more steam generators if the Moderator Temperature Coefficient is positive the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient. The 4 hour completion time to reduce reactor power is consistent with ACTION a. An additional 32 hours is allowed to reduce the Power Range Neutron Flux-High reactor trip setpoints. The total completion time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time to perform the power reduction, operating experience to reset all channels of a protection function, and on the low probability of occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation discussed above, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

ACTION b. is modified by a note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1.1, "Reactor Trip System Instrumentation," provide sufficient protection.

The allowed completion times are reasonable based on operating experience to accomplish the ACTIONS in an orderly manner without challenging unit systems.

BASES

MAIN STEAM SAFETY VALVES (MSSVs) (Continued)

ACTIONS (Continued)

ACTIONS cannot be completed

c ~~b.~~ If the ~~MSSVs cannot be restored to OPERABLE status~~ within the associated completion time~~, or if~~ one or more steam generators have ~~less than two~~ MSSVs ~~OPERABLE~~, four or more inoperable the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

d ~~c.~~ An exception to Specification 3.0.4 is provided since the above ACTION statements require a shutdown if they are not met within a specified period of time.

SURVEILLANCE REQUIREMENTS (SR)

SR 4.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI, requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987. According to ANSI/ASME OM-1-1987, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

The ANSI/ASME Standard requires that all valves be tested every 5 years. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7-2 allows a +1 percent -3 percent setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1 percent during the Surveillance to allow for drift.

INSERT 6

(Proposed Wording)

ATTACHMENT A-1

Unit 1 INSERT 6

The lift settings according to Table 3.7-2 correspond to ambient conditions of the valve at nominal operating temperature and pressure, as identified by a note.

6.9.5 CORE OPERATING LIMITS REPORT (COLR)

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

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

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

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

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

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

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

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

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

INSERT 7

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

(Proposed Wording)

ATTACHMENT A-1

Unit 1 INSERT 7

As described in reference documents listed above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.6% of rated thermal power may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

Caldon, Inc. Engineering Report-80P, "Improving Thermal Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997.

Caldon, Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ or LEFM CheckPlus™ System" Revision 2, December 2000.

ATTACHMENT A-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 161

The following is a list of the affected pages:

Affected Pages:

Operating License Page 3a

Technical Specification Pages

1-1
3/4-31
3/4-32
3/4-32a
3/4-32b
3/4-32c
3/4-32d
3/4 7-1
3/4 7-2
B 3/4 4-7
B 3/4 7-1
B 3/4 7-1a
B 3/4 7-1b
B 3/4 7-1c
B 3/4 7-1d
6-19

transactions shall have no effect on the license for the BVPS Unit 2 facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the term or conditions of any lease agreements executed as part of these transactions; (ii) the BVPS Operating Agreement, (iii) the existing property insurance coverage for BVPS Unit 2, and (iv) any action by a lessor or others that may have adverse effect on the safe operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

a steady state

2689 FENOC is authorized to operate the facility at a steady state reactor core power level~~s not in excess~~ of ~~2652~~ 2689 megawatts thermal. ~~(100 percent power) in accordance with the conditions specified herein.~~

(2) Technical Specifications

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

(proposed wording)

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~2652~~ MWt.

2689

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principal specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related safety function(s).

REPORTABLE EVENT

1.7 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

(Proposed wording)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER ~~15~~ EFPY:	1/4T, 140°F
	3/4T, 128°F

14

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO ~~15~~ EFPY.

14

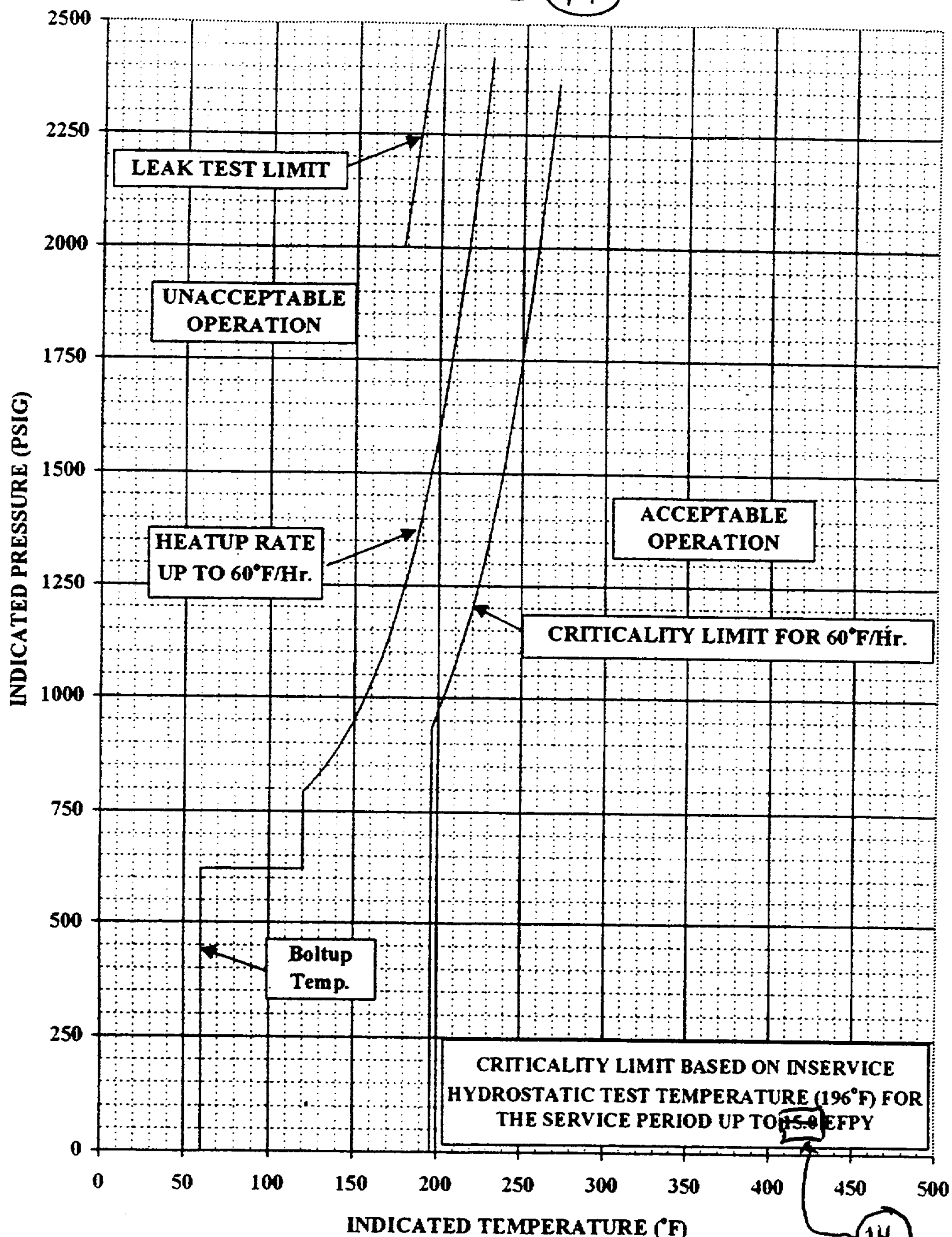

FIGURE 3.4-2
Beaver Valley Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First ~~15~~ EFPY

1

(proposed wording)

MATERIAL PROPERTY BASIS
CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}: 60°F
RT_{NDT} AFTER ~~15~~ EFPY: 1/4T, 140°F
14
3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 0°F/HR FOR THE SERVICE PERIOD UP TO ~~15~~ EFPY.
14

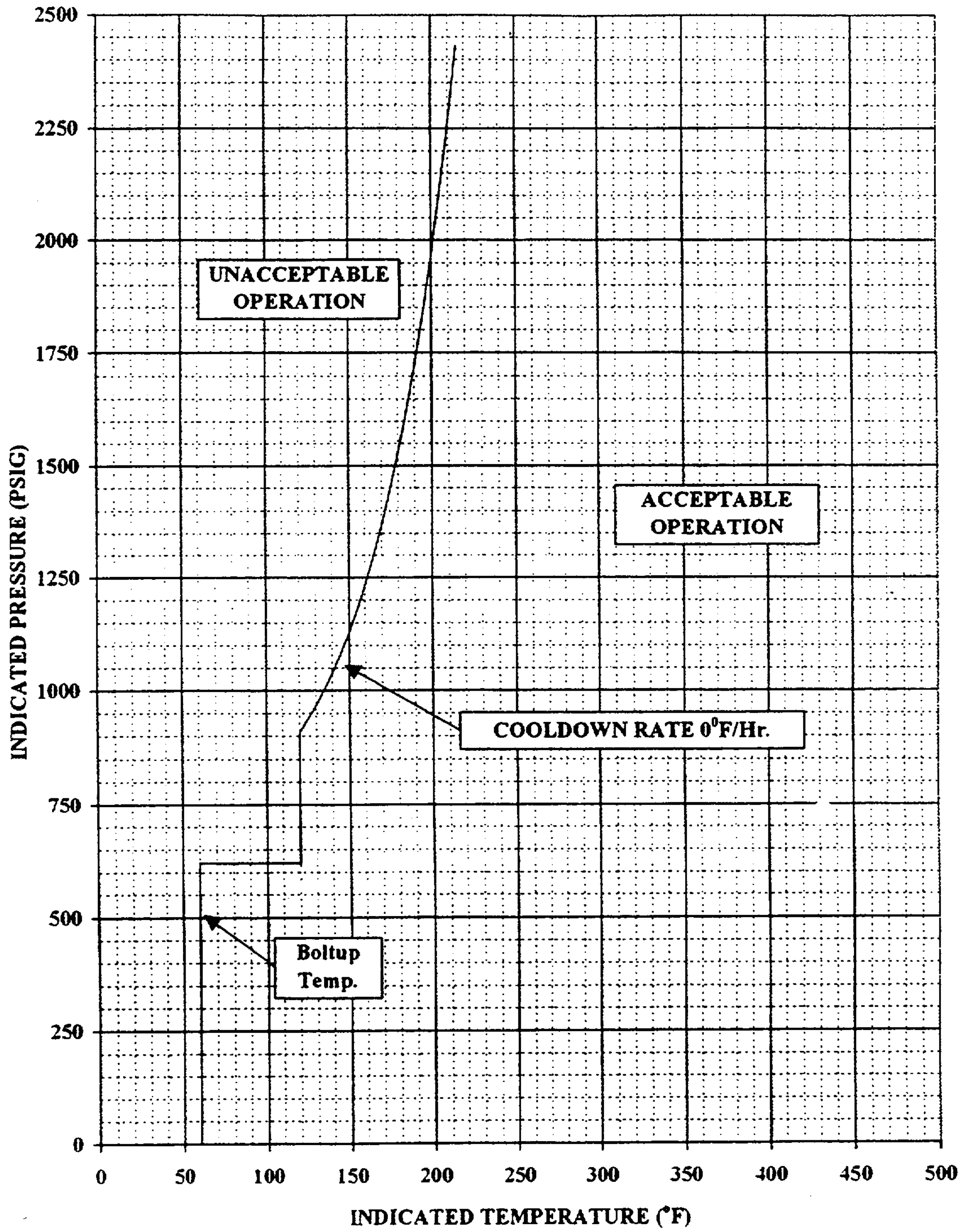

FIGURE 3.4-3 (Sheet 1 of 5)
Beaver Valley Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First ~~15~~ EFPY

(proposed wording)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1

INITIAL RT_{NDT}: 60°F

RT_{NDT} AFTER ~~15~~ EFPY: 1/4T, 140°F

14

3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 20°F/HR FOR THE SERVICE PERIOD UP TO ~~15~~ EFPY.

14

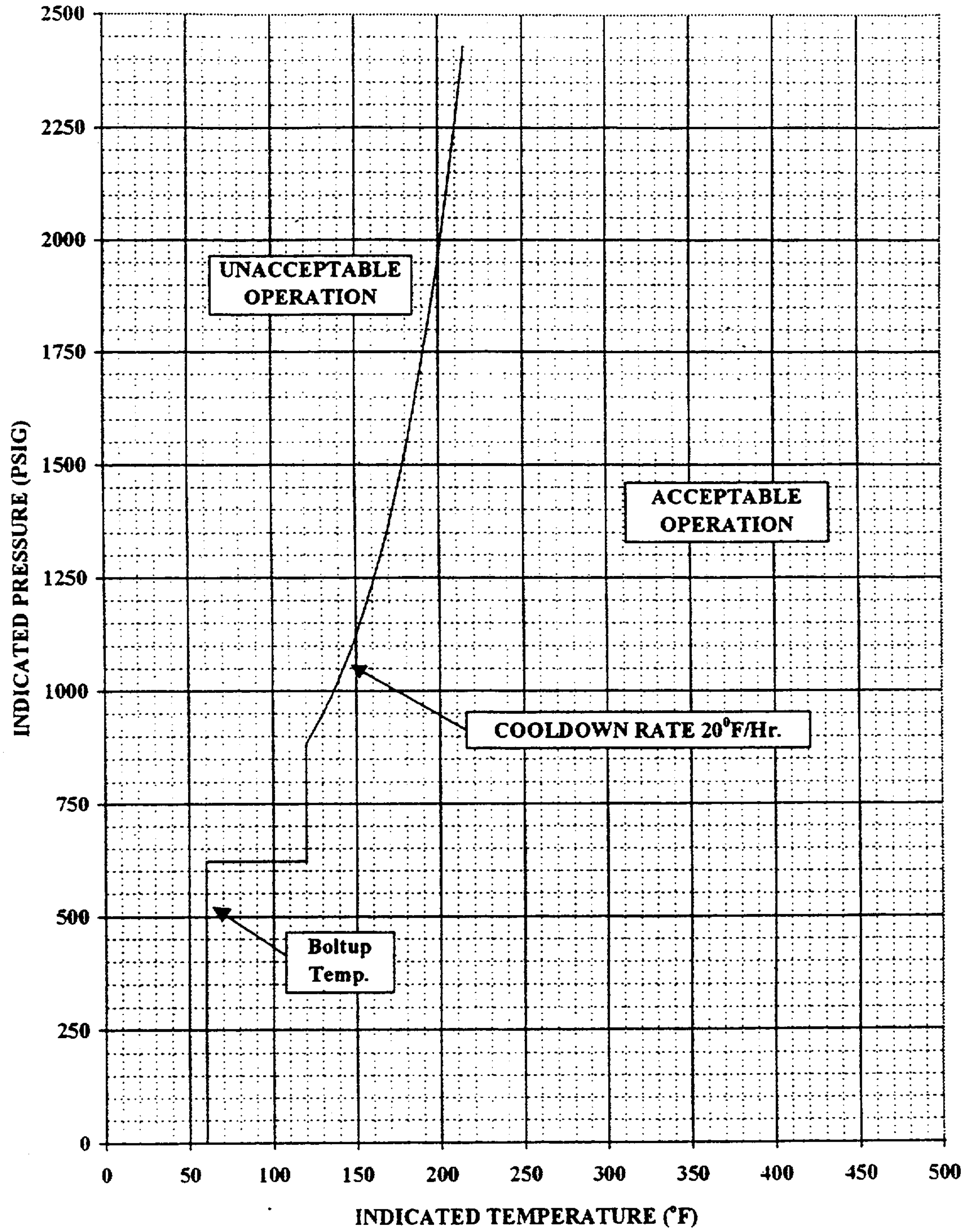

FIGURE 3.4-3 (Sheet 2 of 5)
Beaver Valley Unit 2 Reactor Coolant System Cooldown
Limitations Applicable for the First ~~15~~ EFPY

(proposed wording)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1

INITIAL RT_{NDT}: 60°F

RT_{NDT} AFTER ~~15~~ EFPY: 1/4T, 140°F

14

3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 40°F/HR FOR THE SERVICE PERIOD UP TO ~~15~~ EFPY.

14

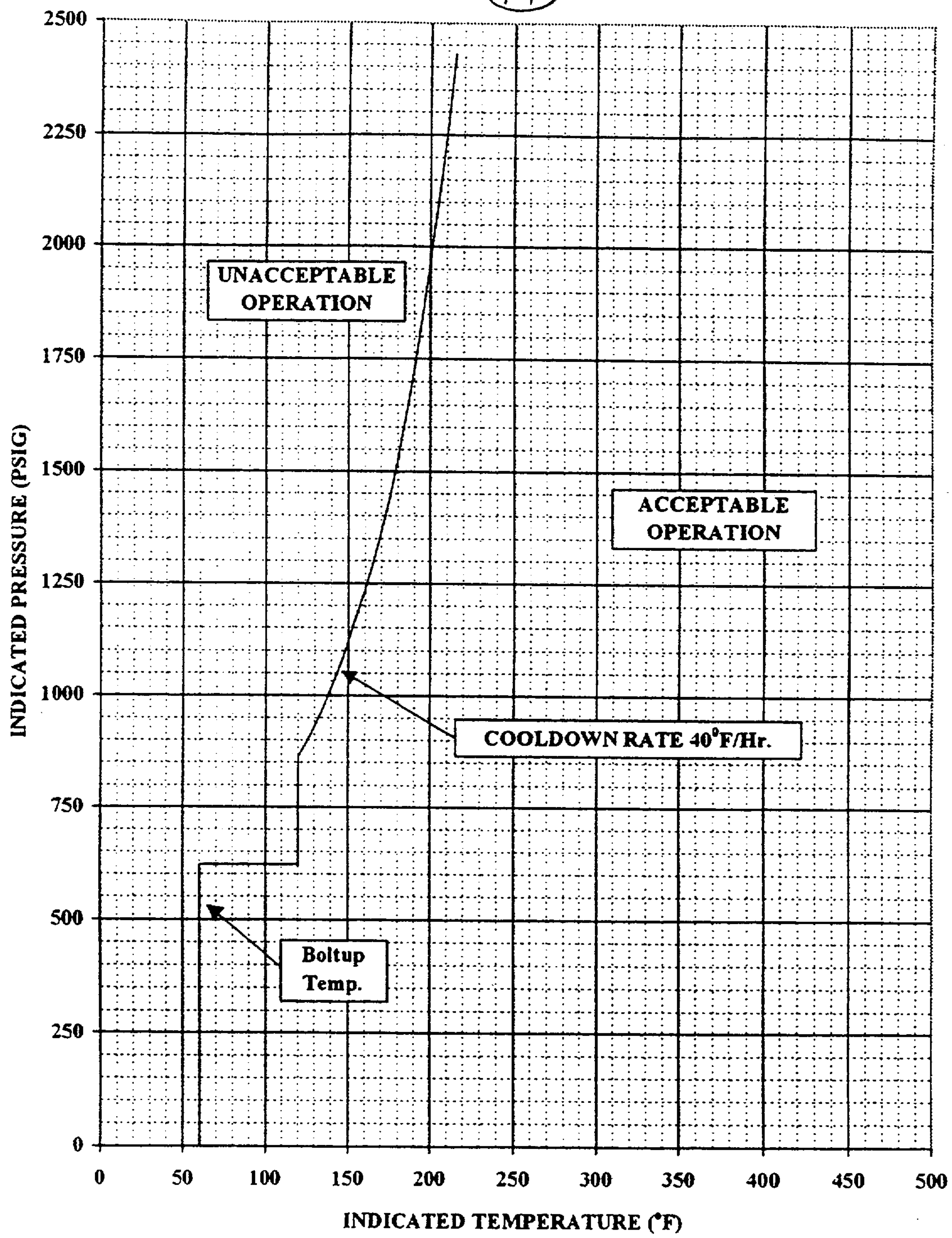

FIGURE 3.4-3 (Sheet 3 of 5)
Beaver Valley Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First ~~15~~ EFPY

(proposed wording)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1

INITIAL RT_{NDT}: 60°F

14 RT_{NDT} AFTER ~~15~~ EFPY: 1/4T, 140°F

3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO ~~15~~ EFPY.

14

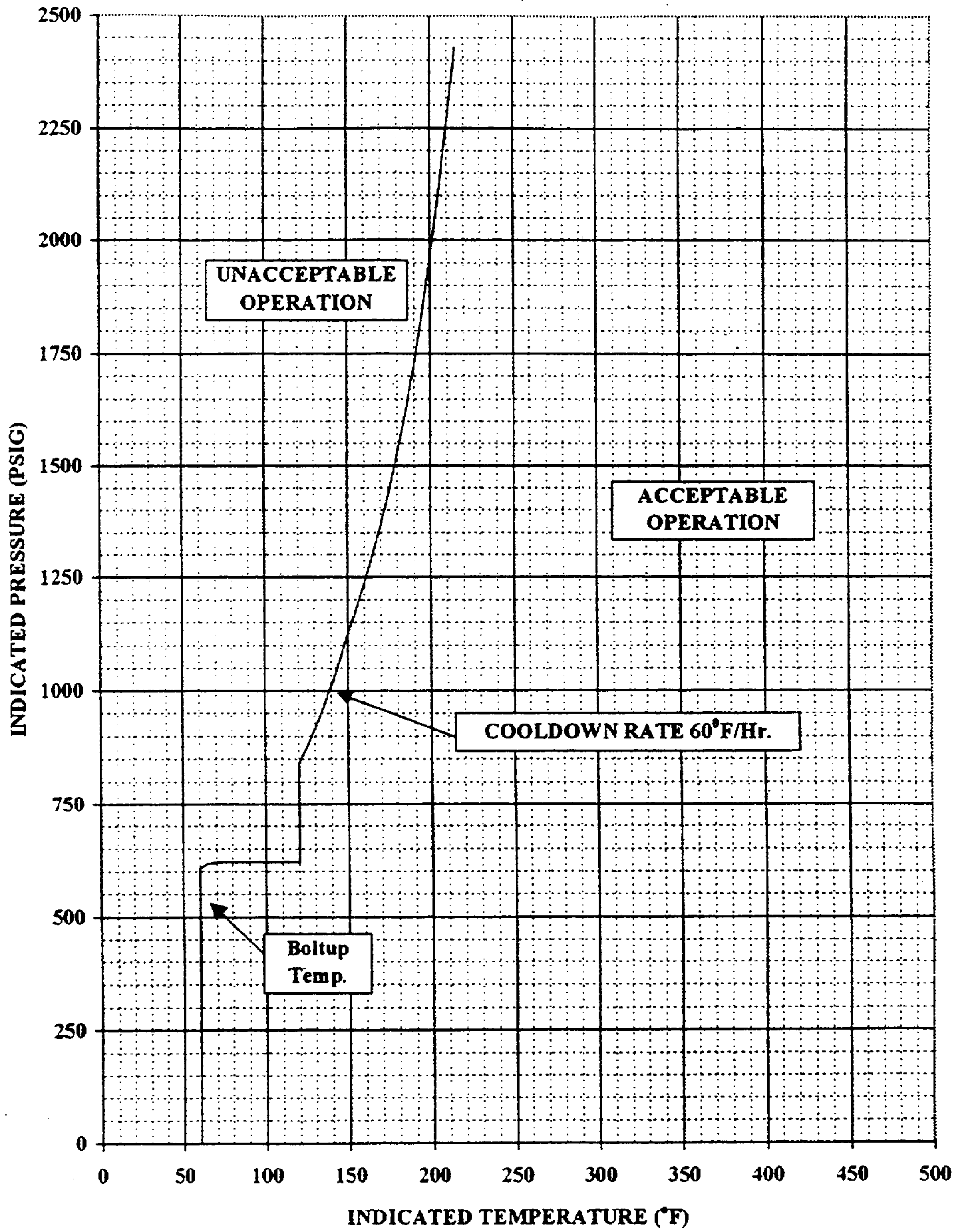

FIGURE 3.4-3 (Sheet 4 of 5)
Beaver Valley Unit 2 Reactor Coolant System Cooldown
Limitations Applicable for the First ~~15~~ EFPY

(proposed wording)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1

INITIAL RT_{NDT}: 60°F

RT_{NDT} AFTER ~~15~~ 14 EFPY: 1/4T, 140°F

3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO ~~15~~ 14 EFPY.

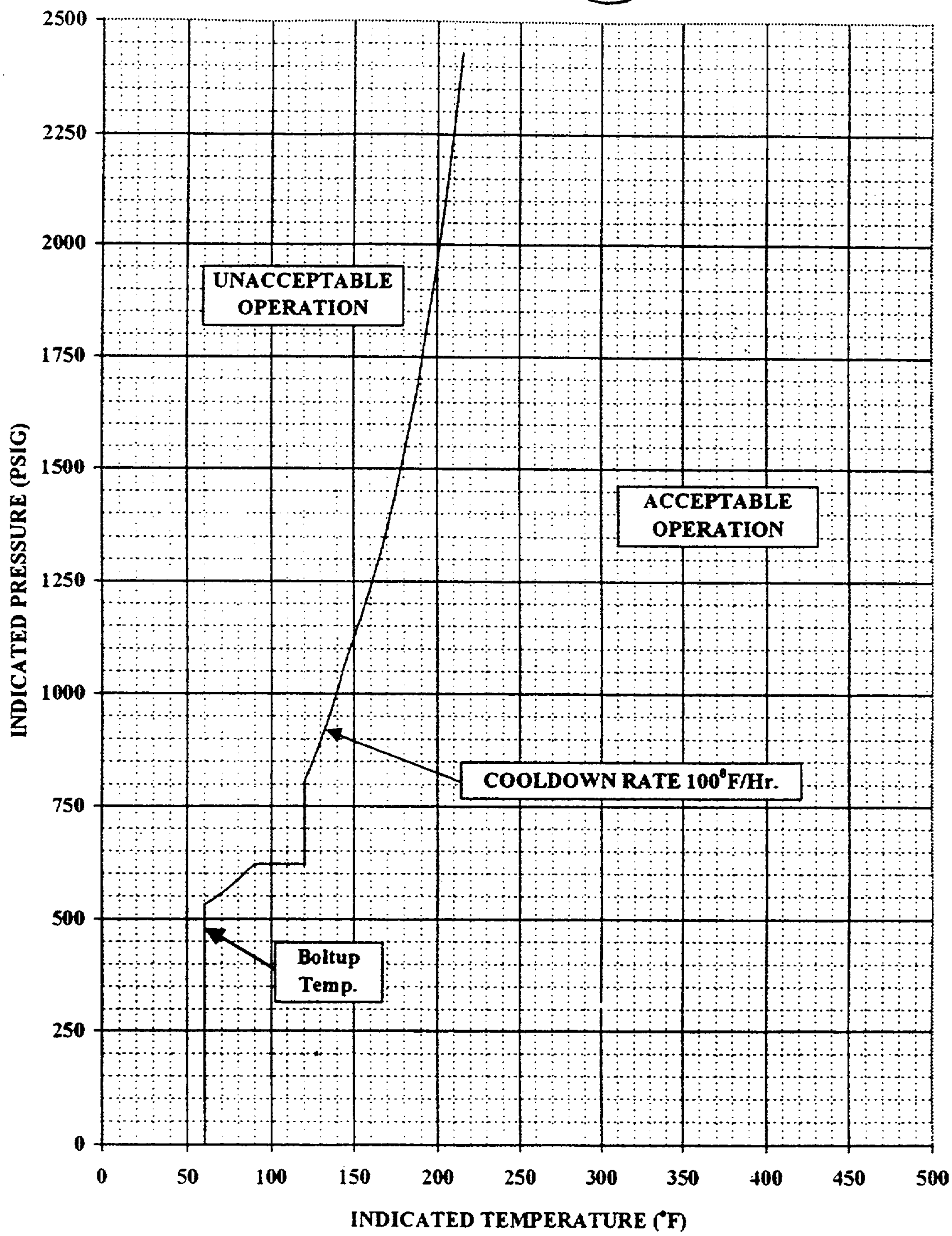

FIGURE 3.4-3 (Sheet 5 of 5)
Beaver Valley Unit 2 Reactor Coolant System Cooldown
Limitations Applicable for the First ~~15~~ 14 EFPY

(NEXT PAGE IS 3/4 4-34)

b

3/4.7.1 TURBINE CYCLE

MAIN STEAM SAFETY VALVES (MSSVs)

LIMITING CONDITION FOR OPERATION

3.7.1.1 ~~The MSSVs~~ Five MSSVs per steam generator shall be OPERABLE ~~as specified in Table 3.7-1 and Table 3.7-2.~~

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

- - - - - - - - - - - - - - GENERAL NOTE - - - - - - - - - - - - - - - -

Separate ACTION entry is allowed for each MSSV.

- -

b. ~~a.~~ INSERT 8 ~~With one or more required MSSVs inoperable, within 4 hours reduce power to less than or equal to the applicable percent RATED THERMAL POWER listed in Table 3.7-1; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 6 hours.~~

c. ~~b.~~ With one or more steam generators with ~~less than two~~ four or more MSSVs ~~OPERABLE~~ inoperable, within 6 hours be in HOT STANDBY and in HOT SHUTDOWN within the next 6 hours.

d. ~~c.~~ The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Verify[1][2] each required MSSV lift setpoint per Table 3.7-2 in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1 percent.

---

(1) Required to be performed only in MODE 1.

(2) ~~(1)~~ Required to be performed only in MODES 1 and 2.

(Proposed Wording)

ATTACHMENT A-2

Unit 2 Inserts

Unit 2 INSERT 8

a. With one or more steam generators with one MSSV inoperable and the Moderator Temperature Coefficient (MTC) zero or negative at all power levels, within 4 hours reduce THERMAL POWER to less than or equal to 63% RTP; otherwise, be in HOT STANDBY within the next 6 hours, and in HOT SHUTDOWN within the next 6 hours.

b. With one or more steam generators with two or more MSSVs inoperable, or with one or more steam generators with one MSSV inoperable and the MTC positive at any power level, within 4 hours reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs, and reduce the Power Range Neutron Flux-High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs within the next 32 hours(1); otherwise, be in HOT STANDBY within the next 6 hours, and in HOT SHUTDOWN within the next 6 hours.

TABLE 3.7-1

Maximum Allowable

OPERABLE Main Steam Safety Valves versus
~~Applicable~~ Power ~~in Percent of RATED THERMAL POWER (RTP)~~

~~MINIMUM~~ NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR ~~REQUIRED~~ OPERABLE	MAXIMUM ALLOWABLE ~~APPLICABLE~~ POWER (% RTP)
~~5~~	~~≤ 100~~
4	≤ 58
3	≤ 41
2	≤ ~~24~~ 25

(Proposed Wording)

No change
Included for Information.

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER	LIFT SETTING* (+1% -3%)	ORIFICE DIAMETER
a.	2MSS-SV101A, B & C	1075 psig	4.515 in.
b.	2MSS-SV102A, B & C	1085 psig	4.515 in.
c.	2MSS-SV103A, B & C	1095 psig	4.515 in.
d.	2MSS-SV104A, B & C	1110 psig	4.515 in.
e.	2MSS-SV105A, B & C	1125 psig	4.515 in.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves, Figures 3.4-3 (Sheets 1 through 5), are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 15 EFPY and revised to 14 EFPY for the uprated condition.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E >1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content and nickel content of the material in question, can be predicted using WCAP-15139 and Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3 (Sheets 1 through 5), include predicted adjustments for this shift in RT_{NDT}.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility temperature). The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material properties and estimating the radiation-induced ΔRT_{NDT}. RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (T_{NDT}) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Thus, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Regulatory Guide 1.99 Revision 2 curves which show the effect of fluence and copper content on upper shelf energy (USE) for reactor vessel steels are shown in Figure B 3/4 4-1.

(proposed wording)

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 MAIN STEAM SAFETY VALVES (MSSVs)

BACKGROUND

The primary purpose of the main steam safety valves (MSSVs) is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the UFSAR, Section 10.3.2. The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition and Winter 1972 Addenda. The total relieving capacity for all valves on all of the steam lines is 12.7 x 10^6 lbs/hr which is ~~110~~ 108 percent of the total secondary steam flow of ~~11.6~~ 11.8 x 10^6 lbs/hr at 100% RATED THERMAL POWER. The MSSV design includes staggered setpoints, according to Table 3.7-2 in the accompanying limiting condition for operation (LCO), so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip. INSERT 9

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from the ASME Code, Section III and its purpose is to limit the secondary system pressure to less than or equal to 110 percent of design pressure ~~when passing 100 percent of design steam flow. This design basis is sufficient to cope with~~ for any anticipated operational occurrence (AOO) or accident ~~considered~~ in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in UFSAR, Section 15.2. Of these, the full power turbine trip without steam dump is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. INSERT 10 ~~If a minimum reactivity feedback is assumed, the reactor is~~

(Proposed Wording)

Unit 2 INSERT 9

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the ASME B&PV Code) for the worst-case loss-of-heat-sink event. Based on this requirement, a conservative criterion was applied that the valves should be sized to relieve 100 percent of the maximum calculated steam flow at an accumulation pressure (3 percent) not exceeding 110 percent of the design pressure.

Unit 2 INSERT 10

One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer relief valves and spray. This analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The UFSAR, Section 15.1 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The UFSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization may be determined by system transient analyses or conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron

Flux-High reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. If the Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions, unless it is demonstrated by analysis that a specified reactor power reduction alone is sufficient to prevent overpressurization of the steam system.

MAIN STEAM SAFETY VALVES (MSSVs) (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

~~tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110 percent of the design value. The MSSVs also open to limit the secondary steam pressure.~~

~~If maximum reactivity feedback is assumed, the reactor is tripped on overtemperature ΔT. The departure from nucleate boiling ratio increases throughout the transient, and never drops below its initial value. Pressurizer relief valves and MSSVs are activated and prevent overpressurization in the primary and secondary systems.~~ The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

LCO

The accident analysis requires ~~four~~ that five MSSVs per steam generator be OPERABLE to provide overpressure protection for design basis transients occurring at ~~102~~ 100.6 percent RATED THERMAL POWER (RTP). ~~An MSSV will be considered inoperable if it fails to open on demand.~~ The LCO requires that five MSSVs per steam generator be OPERABLE in compliance with the ASME Code, Section III, ~~even though this is not a requirement of~~ and the DBA analysis. ~~This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7-1 in the accompanying LCO and associated ACTION.~~

The OPERABILITY of the MSSVs is defined as the ability to open upon demand within the setpoint tolerances, to relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

~~The lift settings, according to Table 3.7-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure, as identified by a note.~~

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB, or Main Steam System Integrity.

MAIN STEAM SAFETY VALVES (MSSVs) (Continued)

APPLICABILITY

~~In MODE 1 above 24% RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7-1 in the accompanying LCO.~~ In ~~MODE 1 below 24% RTP and in~~ MODES 1, 2, and 3 ~~only~~ five ~~two~~ MSSVs per steam generator are required to be OPERABLE to prevent main steam overpressurization.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS are modified by a General Note indicating that separate condition entry is allowed for each MSSV.

~~a.~~ With one or more MSSVs inoperable, action must be taken ~~reduce power~~ so that the available MSSV relieving capacity meets the ASME Code, Section III requirements~~, for the applicable THERMAL POWER.~~

Operation with less than all five MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is ~~proportionally~~ limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

INSERT 11

The maximum THERMAL POWER ~~is limited~~ corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined by the governing heat transfer relationship from the equation ~~in the relationship~~ q = mΔh, where q is the heat input from the primary side, m is the steam flow rate and Δh is the heat of vaporization at the steam relief pressure (assuming no subcooled feedwater). For each steam generator, at a specified pressure, the ~~fractional~~ maximum allowable power level ~~(FPL)~~ is determined as follows:

$$\text{Maximum Allowable Power Level} \le \cancel{FPL =}\ 100/Q\ \frac{(w_s\ h_{fg}\ N)}{K}\ \cancel{\gamma}\ \gamma$$

(Proposed Wording)

Unit 2 INSERT 11

a. In the case of only a single inoperable MSSV on one or more steam generators, if the Moderator Temperature Coefficient is not positive, a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. Furthermore, for this case there is sufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, ACTION a. requires an appropriate reduction in reactor power within 4 hours.

   The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as discussed below, with an appropriate allowance for calorimetric power uncertainty.

BASES

MAIN STEAM SAFETY VALVES (MSSVs) (Continued)

ACTIONS (Continued)

where:

~~FPL = Fraction of RATED THERMAL POWER equivalent to the safety analysis limit minus 9 percent (to account for typical instrument and channel uncertainties). The uncertainty ensures the maximum plant operating power level will then be lower than the safety analysis limit by an appropriate operating margin.~~

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat), Mwt

K = Conversion factor, 947.82 $\frac{(\text{Btu/sec})}{\text{Mwt}}$

w_s = Minimum total steam flow rate capability of the ~~operable~~ MSSVs on any one steam generator at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, in lb/sec. For example, if the maximum number of inoperable MSSVs on any one steam generator is one, then w_s should be a summation of the capacity of the ~~operable~~ MSSVs at the highest ~~operable~~ MSSV operating pressure, excluding the highest capacity MSSV. If the maximum number of inoperable MSSVs per steam generator is three, then w_s should be a summation of the capacity of the ~~operable~~ MSSVs at the highest ~~operable~~ MSSV operating pressure, excluding the three highest capacity MSSVs.

OPERABLE

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm

N = Number of loops in plant

INSERT 12

(Proposed Wording)

Unit 2 INSERT 12

b. In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Furthermore, for a single inoperable MSSV on one or more steam generators if the Moderator Temperature Coefficient is positive the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient. The 4 hour completion time to reduce reactor power is consistent with ACTION a. An additional 32 hours is allowed to reduce the Power Range Neutron Flux-High reactor trip setpoints. The total completion time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time to perform the power reduction, operating experience to reset all channels of a protection function, and on the low probability of occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation discussed above, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

ACTION b. is modified by a note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1.1, "Reactor Trip System Instrumentation," provide sufficient protection.

The allowed completion times are reasonable based on operating experience to accomplish the ACTIONS in an orderly manner without challenging unit systems.

MAIN STEAM SAFETY VALVES (MSSVs) (Continued)

6

ACTIONS (Continued)

ACTIONS are not completed

c ~~b.~~ If the ~~MSSVs cannot be restored to OPERABLE status~~ within the associated completion time, or if one or more steam generators have ~~less than two~~ MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed completion times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

four or more inoperable

d ~~c.~~ An exception to Specification 3.0.4 is provided since the above ACTION statements require a shutdown if they are not met within a specified period of time.

SURVEILLANCE REQUIREMENTS (SR)

SR 4.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI, requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987. According to ANSI/ASME OM-1-1987, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting); and

d. Compliance with owner's seat tightness criteria.

The ANSI/ASME Standard requires that all valves be tested every 5 years. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7-2 allows a +1 percent -3 percent setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1 percent during the Surveillance to allow for drift.

Insert 13

(Proposed Wording)

Unit 2 INSERT 13

The lift settings according to Table 3.7-2 correspond to ambient conditions of the valve at nominal operating temperature and pressure, as identified by a note.

REPORTING REQUIREMENTS (Continued)

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

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

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

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

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

INSERT 14

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

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring

(proposed wording)

ATTACHMENT A-2

Unit 2 INSERT 14

As described in reference documents listed above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.6% of rated thermal power may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

Caldon, Inc. Engineering Report-80P, "Improving Thermal Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997.

Caldon, Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ or LEFM CheckPlus™ System" Revision 2, December 2000.

ATTACHMENT B

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 289 and 161

A. DESCRIPTION OF THE AMENDMENT REQUEST

The Beaver Valley Power Station (BVPS) units are presently licensed for a core rated thermal power (RTP) of 2652 MWt. The proposed license amendment would increase the RTP by 1.4% to 2689 MWt for each BVPS unit. FirstEnergy Nuclear Operating Company (FENOC) has evaluated the impact of a 1.4% uprating to 2689 MWt for applicable systems, structures, components, and safety analyses and determined that such a power uprate is acceptable for BVPS Units 1 and 2.

Markups of the current Technical Specification pages reflecting the proposed changes for each unit are provided in Attachments A-1 and A-2. In summary, the proposed license amendment revises the BVPS Units 1 and 2 Operating Licenses (OL), Technical Specifications (TS), and associated bases to permit increasing the rated core thermal power level by approximately 1.4% to 2689 MWt. Specifically, the following changes are proposed:

Change No.	Change Description
1	The Operating License for Beaver Valley Unit 1 (DPR-66) Section 2.C.(1) identifies the maximum core thermal power level for which FENOC is authorized to operate Beaver Valley Unit 1 as "... at a steady state reactor core power level of 2652 megawatts thermal". It is being proposed that the steady state core power level be changed to 2689 megawatts thermal.
2	The Operating License for Beaver Valley Unit 2 (NPF-73), Section 2.C.(1) identifies the maximum core thermal power level for which FENOC is authorized to operate Beaver Valley Unit 2 as "... not in excess of 2652 megawatts thermal (100 percent power) ...". It is proposed that the wording of this section be revised to be identical with that used in the Unit 1 operating license for the uprated power level. The proposed wording is as follows: "FENOC is authorized to operate the facility at a steady state reactor core power level of 2689 megawatts thermal."

Change No.	Change Description
3	The definition of RATED THERMAL POWER (RTP) in the Unit 1 and Unit 2 Technical Specifications is changed to read: "RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2,689 MWt."
4	BVPS Unit 2 Technical Specification 3/4.4.9, "Pressure/Temperature Limits" contain heatup/cooldown curves, i.e., Figures 3.4-2 and 3.4-3 (sheets 1-5). These curves are being revised from 15 Effective Full Power Years (EFPY) to 14 EFPY. The current curves have been relabeled as applicable to 14 EFPY under the fluence created at uprated conditions. Additionally, the accompanying Bases, on page B 3/4 4-7, is also revised to indicate the 14 EFPY value. There is no corresponding change applicable to BVPS Unit 1.
5	Unit 1 and 2 Section 6.9.5(b), Analytical Methods for Core Operating Limits Report (COLR) is revised by adding the following. As described in reference documents listed above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.6% of rated thermal power may be used when input for reactor thermal power measurement of feedwater mass flow is by the leading edge flow meter (LEFM). Caldon, Inc. Engineering Report-80P, "Improving Thermal Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997. Caldon, Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ or CheckPlus™ System" Revision 2, December 2000.

Change No.	Change Description
6	Technical Specification 3.7.1.1, "Main Steam Safety Valves (MSSVs)", is being revised to be consistent with Technical Specification Traveler Form-235 (TSTF-235) Revision 1 and the Improved Standard Technical Specifications (ISTS). The proposed changes include a rewrite of the Limiting Condition for Operation (LCO) and a change to the title and content of Table 3.7-1 to be consistent with the ISTS, the creation of new Actions to address MSSVs being inoperable and reducing the Power Range Neutron Flux-High reactor trip setpoint to be consistent with TSTF-235, Rev. 1, and changes to the maximum power levels permissible with inoperable MSSVs due the proposed power uprate. The applicable Bases is also changed to be consistent with the revised TS. A clarification is also added to the Bases addressing the determination of the total relieving capacity of the MSSVs.

The applicable Index, TS and Bases will be repaginated as necessary.

B. DESIGN BASES

Power Uprate

The design bases applicable to the power uprate are discussed in detail in Enclosure 1, "Beaver Valley Units 1 and 2, 1.4-Percent Power Uprate Program, FENOC Licensing Submittal, January 2001."

Unit 2 Heatup/Cooldown Curves

Unit 2 Technical Specification 3/4.4.9, "Pressure/Temperature Limits", contains heatup/cooldown curves (Figures 3.4-2 and 3.4-3), that limit the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation. Each heatup/cooldown curve defines an acceptable region for normal operation and is based on the time the core has been critical, based on Effective Full Power Years (EFPY).

The curves are used as operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region. Technical Specification 3/4.4.9 establishes operating Pressure/Temperature (P/T) limits that provide a margin to brittle failure of the reactor vessel. The establishment of P/T limits for specific material fracture toughness requirements of the reactor coolant pressure boundary (RCPB) materials is a requirement of 10 CFR 50, Appendix G, "Fracture Toughness Requirements". Appendix G requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G.

Main Steam Safety Valves

The primary purpose of the Main Steam Safety Valves (MSSVs) is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the RCPB by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink is not available.

Five MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves. The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III. The MSSV design includes staggered setpoints, according to Table 3.7-1 in TS 3.7.1.1, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to insufficient steam pressure to fully open all valves following a turbine reactor trip.

The design bases for the MSSVs is to limit the secondary system pressure to ≤ 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis. The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events. Of these, the full power turbine trip without steam dump is the limiting event.

The total relieving capacity for all valves on Unit 1 steam lines is 12.8×10^6 lbs/hr (when relieving at 1125 psig plus 5% accumulation pressure), which is approximately 110% of the total secondary steam flow of 11.6 x 10^6 lbs/hr at 100% of original RTP. This capacity represents approximately 108% of the uprated secondary steam flow of 11.8 x 10^6 lbs/hr at the uprated RTP of 2689 MWt.

The total relieving capacity for all valves on Unit 2 steam lines is 12.7 x 10^6 lbs/hr (when relieving at 1125 psig plus 3% accumulation pressure), which is approximately 110% of the total secondary steam flow of 11.6 x 10^6 lbs/hr at 100% of original RTP. This capacity represents approximately 108% of the uprated secondary steam flow of 11.8 x 10^6 lbs/hr at the uprated RTP of 2689.

Plant operation with fewer than five Operable MSSVs per steam generator is permissible, but at reduced power levels in accordance with TS Table 3.7-1. The values specified in TS Table 3.7-1 were recalculated, considering the uprating in RTP.

The slightly lower allowable maximum power and Power Range Neutron Flux-High reactor trip setpoint reductions for Unit 1 compared to Unit 2 accounts for the following differences.

While the total specified MSSVs relieving capacity is slightly greater for Unit 1 than Unit 2 (Refer to Enclosure 1, Section 3.7.1), these capacities are specified at the highest safety valve setpoint plus 5% accumulation pressure for Unit 1. For Unit 2 the specified capacities are at the highest safety valve setpoint plus 3% accumulation pressure.

The lowest lift setting MSSV at Unit 1 has a smaller orifice size and reduced rated relieving capacity at its full accumulation pressure compared to the remaining four MSSVs on each steamline. At Unit 2, all MSSVs have the same orifice size and same relief capacity when relieving at the full accumulation pressure of the highest MSSV. This lower capacity MSSV is conservatively always considered one of the remaining operable valves when calculating the required Power Range Neutron Flux-High reactor trip setpoint reduction for number of inoperable MSSVs and therefore slightly reduces the required setpoints on Unit 1 compared to Unit 2.

C. JUSTIFICATION

This power uprating is based on a redistribution of analytical margin originally required by Emergency Core Cooling System (ECCS) evaluation models performed in accordance with the requirements set forth in 10 CFR 50, Appendix K "ECCS Evaluation Models". Appendix K mandated consideration of an assumed reactor operating power level of 102% of the licensed power level for ECCS evaluation models of light water power reactors. The additional 2% was allocated specifically to account for thermal power measurement uncertainties. The Nuclear Regulatory Commission (NRC) approved a change to the requirements of 10 CFR 50, Appendix K, (65 FR 34913, June 1, 2000). This change provide licensees with the option of maintaining the current 2% power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin. For the latter case, the proposed alternative reduced margin must be demonstrated to account for calculated plant-specific instrument uncertainties when measuring power level.

The allowance for power measurement uncertainties can be reduced, based on using improved instrumentation, such as the Caldon Leading Edge Flow Meter (LEFM✓™ in Unit 1 and LEFM CheckPlus™ in Unit 2). Caldon LEFM spool pieces have been installed in both BVPS units, and the power calorimetric uncertainty has been determined for each. The installation and post-modification testing of the LEFM system, including all related plant process computer system changes, will be completed prior to increasing power over 2652 MWt for that unit.

Using the Caldon LEFM equipment, core thermal power level may be determined with a demonstrated calorimetric power measurement uncertainty of less than ±0.6% RTP. FENOC therefore proposes to reduce the licensed power uncertainty required by 10 CFR 50, Appendix K to this value, for an increase of up to 1.4% in the licensed RTP for both units.

This power uprating is being made in concert with related License Amendment Requests incorporating the "Revised Thermal Design Procedure" (RTDP) methodology described in WCAP-15264 (Unit 1) and WCAP-15265 (Unit 2). These reports were submitted as part of License Amendment Requests (LAR) 286 (Unit 1) and 158 (Unit 2), on December 27, 2000. Specifically, these reports describe the current Westinghouse methodology for determining the uncertainties in calorimetric thermal power measurements and reactor coolant system flow measurements. Additionally, these reports calculate the total calorimetric measurement error, both with and without the LEFM, for the two Beaver Valley units. This methodology complies with the recommendations of ANSI/ISA-67.04, and R.G. 1.105, Rev. 2. Certain reactor trip setpoints are revised as a result of adopting the RTDP methodology. The revised setpoints are calculated in Westinghouse WCAP-11419, Revision 2 ("Westinghouse Setpoint Methodology for Protection Systems Beaver Valley Unit 1") and WCAP-11366, Revision 4 ("Westinghouse Setpoint Methodology for Protection Systems Beaver Valley Unit 2"), and are included in the RTDP Amendment request. The reactor trip system and Engineered Safety Features Actuation System (ESFAS) setpoint changes prescribed in the RTDP submittal necessary to support the power uprate will be completed for each unit prior to increasing power above 2652 MWt for that unit.

The uprating is also being made in concert with License Amendment Requests 280 (Unit 1) and 151 (Unit 2), dated May 12, 2000. These amendment requests, which are currently under NRC review, contain revised radiological accident dose evaluations that bound the proposed power uprate.

A requirement will be placed in the Beaver Valley Licensing Requirements Manuals to address LEFM unavailability. The requirement will involve limiting RTP to a maximum steady state power of 2652 MWt (the presently licensed 100% limit) when the LEFM is unavailable.

The proposal to increase the RTP at BVPS to 2689 MWt has been reviewed by FENOC using the methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant". This methodology establishes the general approach and criteria for nuclear power plant uprating projects including the categories that must be addressed. The categories includes items such as Nuclear Steam Supply System (NSSS) performance parameters, design transients, systems, components, accidents and nuclear fuel, as well as interfaces between the NSSS and Balance of Plant (BOP) systems. The results of the review are documented in Enclosure 1.

Section 3.0 of Enclosure 1 provides the design bases and safety analysis applicable to the proposed power uprate. Section 4 contains the results of a review of various programs, as well as information related to previous NRC requests for additional information from the Comanche Peak and Watts Bar uprate reviews. Section 5 contains a review of the environmental impacts associated with the uprate.

In general, the results of the reviews and evaluations performed demonstrate that all acceptance criteria continue to be met following the proposed uprate. In limited cases, the conclusions are based on preliminary assessments that are being confirmed by more detailed calculations or analyses. Commitments made for the completion of these calculations and analyses are identified in Enclosure 1 and are included in Attachment C, "List of Commitments".

The following provides the justification for the proposed changes described in Section A of this License Amendment Request.

For Changes Nos. 1, 2, and 3:

The change to the numerical value of rated thermal power in the operating licenses and Section 1.0 of the TS reflects the limit of core power justified by the 1.4% uprate program. The revised power level is supported by the reviews and evaluations contained in Enclosure 1. In general, the increase in power of approximately 1.4% is based on a plant specific evaluation of reactor power measurement uncertainty using LEFM instrumentation versus the previous mandated 2% uncertainty that was formerly required by 10 CFR 50, Appendix K.

In addition to the revised limit on RTP, a change is being proposed to make the Unit 1 and Unit 2 operating license requirements consistent between the two units regarding how the RTP limit is expressed. Currently, the Unit 1 operating license identifies the maximum power level as a "steady state" level. The Unit 2 license, however, does not incorporate the term "steady state." The proposed change is to modify the Unit 2 license such that it is consistent with Unit 1's license, i.e., to express the maximum power level in terms of a steady state level.

This change achieves greater consistency between the operating licenses for the units, and will not result in any changes to the way the plant is currently operated. It reflects statements regarding maximum power level in the Unit 2

NRC Technical Specifications Safety Evaluation Report (dated October 1985) and the NRC's understanding of maximum power level, as contained in NRC Inspection Manual Procedure 61706, "Core Thermal Power Evaluation", issued July 14, 1986. Specifically, Section 03.02.d contains the following guidance:

> "Core thermal power evaluation is performed on a daily basis for both PWRs and BWRs. The specific requirements can be found in the plant's TS although the plant may follow more stringent guidelines as recommended by the manufacturer. Refer to Inspection Procedure 61705, "Calibration of Nuclear Instrumentation Systems," if calibration is required. In addition, the inspector should check that the average power level over any 8-hour shift did not exceed the "full steady-state licensed power level" (and similarly worded terms). The exact 8-hour periods defined as "shifts" are up to the plant, but should not be varied from day to day (the easiest definition is a normal shift manned by a particular "crew")."

Therefore, based on the above statements, changing the Unit 2 operating license to be consistent with the corresponding statement in the Unit 1 operating license is acceptable.

<u>For Change No. 4:</u>

The Unit 2 reactor coolant system heatup/cooldown curves contained in Figures 3.4-2 and 3.4-3 of TS 3.4.9.1 are revised to reflect a reduction in applicability of the current limits from 15 to 14 EFPY. This is due to the increased neutron fluence associated with the proposed increased power level. Similarly, Bases page B 3/4 4-7 is also revised to reflect the reduced applicability. This change is supported by evaluations provided in Section 3.6.2.2 of Enclosure 1. Unit 2 is currently at approximately 10 EFPY, so this will not impose limitations on operations in the near term.

Amendment 113 for Unit 2 , implemented on September 20, 2000, changed the EFPYs on the heatup/cooldown curves from 10 to 15 EFPY. The amendment also adopted methodology from Revision 2 of Regulatory Guide 1.99 and changed the power operated relief valve setpoints and overpressure protection system enable temperature to be consistent with the change to the EFPY. The proposed change to 14 EFPY for the heatup/cooldown curves is consistent with the changes made by Amendment 113.

Changes to Unit 1's corresponding heatup/cooldown curves are not being proposed at this time. The present curves remain acceptable until 16 EFPY, which is the current limit of these curves. Surveillance Capsule "Y" for Unit 1 was withdrawn during refueling outage 1R13 in the spring of 2000. The capsule test report will be submitted in accordance with the requirements of 10 CFR 50 Appendix H, "Reactor Vessel Materials Surveillance Program Requirements."

<u>For Change No. 5:</u>

Westinghouse has described the methodologies it used in performing design basis accident analyses for BVPS Units 1 and 2 Safety Analysis Reports in a series of topical reports referenced in the Updated Final Safety Analysis Report (UFSAR) and the Technical Specifications. The reports listed in the Technical Specifications describe the analytical methodologies used to determine the core operating limits for the Core Operating Limits Report (COLR). In some of these topical reports, reference is made to use of a 2% uncertainty applied to reactor power. This is consistent with the version of 10 CFR 50, Appendix K that was in effect at the time.

In general, the uprate is accomplished by replacing the prescribed 2% power measurement uncertainty with a plant specific uncertainty value, in effect trading the increased accuracy associated with the LEFM for increased power. Acceptability of the plant to accommodate the increased power level is contained in Enclosure 1. Thus, revision of each of the reports listed in TS Section 6.9.5(b) specifically to accommodate the uprate is considered an administrative burden. To alleviate this administrative burden, FENOC is proposing to modify TS Section 6.9.5(b) to allow the present versions of the reports to apply to the uprated conditions. This proposed modification is conditional on the LEFM being used to measure feedwater mass flow as the input to the reactor thermal power measurement. Consistent with the approach taken by Comanche Peak, a requirement will be placed in the BVPS UFSARs requiring that future, plant-specific revisions of these reports, incorporate consideration of the 1.4% power uprate.

Lastly, the Caldon topical reports ER-80P and ER-157P Rev. 3, will be added to the list of referenced reports in TS Section 6.9.5(b). This is appropriate as these reports provide the basis for acceptability of the LEFM for the power uprate.

For Change No. 6:

Westinghouse Nuclear Safety Advisory Letter, NSAL-94-001, "Operation at Reduced Power Levels with Inoperable MSSVs," dated January 20, 1994. identified a deficiency in the basis for determining the reduced power level as implemented by the Power Range Neutron Flux-High trip setpoints in Table 3.7-1 of BVPS Technical Specification 3/4.7.1.1. This TS addresses continued operation at reduced power levels with inoperable MSSVs. The deficiency was the assumption that the maximum initial power level is a linear function of the available MSSV relief capacity. A loss of load/turbine trip transient initiated during operation at reduced power levels with inoperable MSSVs based on a linear function of the available MSSV relief capacity may result in overpressurization of the main steam system.

During the development of NUREG-1431, "Standard Technical Specifications for Westinghouse Plants", the Action to reduce the Power Range Neutron Flux-High trip setpoints was deleted from NUREG-1431 Technical Specification 3.7.1, "Main Steam Safety Valves". The title of Table 3.7.1-1 was also revised to "Applicable Power," in %RTP, based on the number of operable MSSVs, versus "Maximum Allowable Power Range Neutron Flux-High Setpoint."

Following these modifications to NUREG-1431, Amendments 223 and 99 where implemented for BVPS Units 1 and 2, respectively, on July 28, 1999. These amendments deleted the Action to reduce the Power Range Neutron Flux-High trip setpoints and replaced it with an Action to reduce power. These amendment made the BVPS TS consistent with the NUREG-1431.

Subsequent to the issuance of NSAL-94-001, Westinghouse identified that overpressurization of the main steam system could also occur during operation at reduced power levels with inoperable MSSVs with a positive moderator temperature coefficient (PMTC), or during an uncontrolled Rod Control Cluster Assembly (RCCA) Bank Withdrawal, if the Power Range Neutron Flux-High trip setpoints were not reduced to limit the primary side heat generation. Since this Action was deleted from NUREG-1431, TSTF-235, Rev. 1 "MSSV Changes," was prepared to add the Action back into NUREG-

1431. The change was needed to address potential overpressurization of the main steam system as related to operation with a PMTC or an uncontrolled RCCA Bank Withdrawal. TSTF-235, Rev. 1 was approved by the NRC on January 11, 1999.

During the preparation of the proposed changes necessary to support the BVPS power uprate, the need to add the subject Action back into BVPS Technical Specification 3.7.1.1 was identified. Therefore the addition of this Action is included in the changes being proposed. Changes to the Limiting Condition for Operation (LCO), Actions, Table 3.7-1, and associated Bases of Technical Specification 3.7.1.1 are being made to achieve consistency with NRC approved TSTF-235, Rev. 1.

In addition revised Power Range Neutron Flux-High trip setpoints were recalculated for BVPS Units 1 and 2 to ensure that the maximum power level allowed for operation with inoperable MSSVs would be below the heat removal capability of the operable MSSVs. This is consistent with a recommendation made in NSAL-94-001 and the proposed power uprate.

The Action associated with a PMTC, is being proposed to be added even though BVPS Units 1 and 2 are not currently licensed to operate with a PMTC. The Action associated with a PMTC does not presently apply to BVPS. However approval of the addition of this Action will result in the Action being contained in Technical Specification 3.7.1.1, should BVPS Units 1 and 2 be licensed to operate with a PMTC in the future.

Approval of this proposed change will permit the elimination of administrative controls imposed that revise the Power Range Neutron Flux-High trip setpoints with inoperable MSSVs.

D. SAFETY ANALYSES

Power Uprate

The NSSS performance parameters are the fundamental design parameters used as input in all the NSSS transient and accident analyses. These parameters include the Reactor Coolant System (RCS) and secondary system process conditions (temperatures, pressures, flow) that are used as the basis for the design transient, system, component and accident evaluations. These parameters are established using assumptions that are biased appropriately in order to provide conservative bounding conditions for NSSS analyses. These parameters, as documented in Enclosure 1, were reviewed and evaluated to support the proposed power uprate.

All normal and abnormal plant operating conditions are categorized into four groups according to their anticipated frequency of occurrence and the potential severity of consequences to public health and safety.

1. ANS Condition I - Normal Operation
2. ANS Condition II - Incidents of Moderate Frequency
3. ANS Condition III - Infrequent Faults
4. ANS Condition IV - Limiting Faults

Analyses were performed for a variety of transient and upset events belonging to each of the four conditions to demonstrate that the reactor fuel can be kept safely intact during these events, or if fuel damage occurs, the public health and safety can still be protected.

All Condition I transients were evaluated to confirm that the plant can appropriately respond to these transients without generating a reactor trip or engineered safety feature actuation system (ESFAS) actuation. The analysis methodology for these transients employs a 2% power calorimetric uncertainty to increase the power level to 102%. The improved thermal power measurement accuracy obviates the need for the full 2% power measurement margin assumed in the analysis. The 102% power level bounds the proposed 1.4% uprate conditions. Therefore, the current analyses remain valid and bound the 1.4% uprating conditions.

All analyzed Condition II incidents were reviewed and shown to meet the applicable acceptance criteria at the proposed uprated thermal power rating of 2689 MWt. For each of these transients, minimum Departure from Nucleate Boiling Ratio (DNBR) was greater than the limit, and peak RCS pressure remains below the ASME Code limit of 110% of design pressure. No failed fuel is predicted to result from any Condition II event.

All analyzed Condition III transients and events were reviewed and shown to meet the applicable acceptance criteria. Fewer than 5% of the fuel rods are shown to experience Departure from Nucleate Boiling (DNB) in these transients. Peak RCS pressure remains below 110% of design pressure, and all other core parameters remain within design limits.

All analyzed Condition IV Design Basis Accident (DBA) analyses are performed at an assumed power level of 102% RTP. The additional 2% in power level is a conservatism taken explicitly to account for measurement error in the thermal power calorimetric measurement. Therefore an adequate allowance exists in these analyses to accommodate the proposed power uprate of 1.4% when using the new measurement instrumentation having a demonstrated total uncertainty of less than ±0.6%.

All of the non-LOCA analyses applicable to BVPS Units 1 and 2 were reviewed to determine their continued acceptability for operation considering the 1.4% power uprate conditions. The evaluation of these non-LOCA events was performed concurrently with the recent assessment of the change to the Revised Thermal Design Procedure (RTDP) for both BVPS units. The evaluations and results are presented in the RTDP Analysis Report for Units 1 and 2. All applicable acceptance criteria for each of the analyzed events continue to be met.

The following non-LOCA transients are examples currently documented in the UFSAR having been analyzed using an explicit 2% power measurement uncertainty allowance to increase the assumed initial power level to 102%.

Each of these analyses incorporates other conservatisms taken to account for uncertainties other than power measurement. These are discussed in the BVPS UFSAR accident analysis descriptions. The explicit 2% power uncertainty allowance bounds the 1.4% power uprate since the power uncertainty has been reduced to less than ±0.6%.

- Spurious Operation of the Safety Injection System
- Dropped Rod Cluster Control Assembly (RCCA)
- Loss of Offsite Power
- Loss of Normal Feedwater Flow
- Feedwater System Pipe Break
- Main Steamline Pipe Break Inside Containment, M&E Releases
- Turbine Trip and Loss of External Load
- RCCA Ejection Accident - Full Power Cases

The following non-LOCA transient analyses are presented with the initial power level assumed to be less than full power. This is either because the event is administratively prohibited from occurring at full power conditions, or because it is more limiting to consider the event from less than full power conditions. In either case, an increase in RTP has no effect on the outcome of these analyses.

- Boron dilution accident at Hot Standby
- MSLB at Hot Zero Power (HZP), overcooling event leading to recriticality and return to power
- Inadvertent RCCA withdrawal at shutdown.
- Startup of an Inactive Reactor Coolant Loop
  (The event cannot occur in Mode 1, since reactor operation with an inactive loop is prohibited by Technical Specifications.)

For further details of the safety analysis applicable to the proposed power uprate, see Enclosure 1.

<u>Unit 2 Heatup/Cooldown Curves</u>

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

The neutron embrittlement effect on material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases. The actual shift in the RT_{NDT} of the vessel material is established periodically by removing and evaluating irradiated reactor vessel material specimens, in accordance with ASTM E 185 and Appendix H of 10 CFR 50. The operating P/T limit curves are adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." The consequence of violating Technical Specification 3/4.4.9 limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident.

See Section 3.6.2.2 of Enclosure 1 and the discussion of Change Number 4 in Section C for the justification for the changes to Technical Specification 3/4.4.9.

<u>Main Steam Safety Valves</u>

The MSSV safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip presents no hazard to the integrity of the RCS or the main steam system both prior to and following the proposed power uprate. One turbine trip analysis is performed assuming primary system pressure control via operation of the pressurizer relief valves and spray. This analysis demonstrates that the DNB design basis will continue to be met following the power uprate. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity will continue to be maintained by demonstration that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of design pressure. As discussed in Enclosure 1, the conclusions of these analyses are not changed by the proposed power uprate.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled RCCA bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the

Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO. These conclusions also are not changed by the proposed power uprate provided the Power Range Neutron Flux-High reactor trip setpoint is reduced as described in Change Number 6 in Section C of this License Amendment Request.

E. NO SIGNIFICANT HAZARDS EVALUATION

For Beaver Valley Power Station (BVPS) Unit 1 and 2 the proposed changes consist of the following:

- Section 2.C.(1) of the Operating License (OL) for Beaver Valley Power Station Unit 2 will be revised to be identical with that used in the Unit 1 operating license.
- The definition of RATED THERMAL POWER (RTP) in the Unit 1 and Unit 2 Technical Specifications (TS) will be changed to reflect to the uprated power level.
- Unit 2 TS 3/4.4.9, "Pressure/Temperature Limits" contain heatup/cooldown curves, i.e., Figures 3.4-2 and 3.4-3 (sheets 1-5). These curves are being revised from 15 Effective Full Power Years (EFPY) to 14 EFPY. The applicable Bases pages are also revised to reflect the change in EFPY.
- Unit 1 and 2 Technical Specification Section 6.9.5(b), Analytical Methods for Core Operating Limits Report (COLR), will be revised to state that future revisions of the listed reports will be revised to state that 100.6% of rated thermal power may be used under the appropriate conditions.
- Technical Specification Section 6.9.5(b) is also revised to add references to the following Caldon Reports:

  Caldon, Inc. Engineering Report-80P, "Improving Thermal Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997; and Caldon, Inc. Engineering Report-

157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ or CheckPlus™ System" Revision 2, December 2000.

- Technical Specification 3.7.1.1, "Main Steam Safety Valves (MSSVs)", is being revised to be consistent with Technical Specification Traveler Form-235 (TSTF-235) Revision 1 and the Improved Standard Technical Specifications (ISTS).

  The proposed changes include a rewrite of the Limiting Condition for Operation (LCO) and a change to the title and content of Table 3.7-1 to be consistent with the ISTS, the creation of new Actions to address MSSVs being inoperable and reducing the Power Range Neutron Flux-High reactor trip setpoint to be consistent with TSTF-235, Rev. 1, and changes to the maximum power levels permissible with inoperable MSSVs due the proposed power uprate. The applicable Bases is also changed to be consistent with the revised Technical Specifications. A clarification is also added to the Bases addressing the determination of the total relieving capacity of the MSSVs.

The applicable Index, Technical Specifications and Bases will be augmented and repaginated as necessary to meet format requirements.

The no significant hazards considerations involved with the proposed amendments have been evaluated. The evaluation focused 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 a testing facility involves no significant hazards considerations, 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 proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. Comprehensive analytical efforts performed to support the proposed changes included a review of the Nuclear Steam Supply System (NSSS) systems and components that could be affected by these changes. All systems and components will function as designed and the applicable performance requirements have been evaluated and found to be acceptable.

   The primary loop components (reactor vessel, reactor internals, control rod drive mechanisms (CRDMs), loop piping and supports, reactor coolant pump, steam generator and pressurizer) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

   The Rod Control Cluster Assembly (RCCA) drop time remains within the current limits assumed in the accident analyses. Thus, there is no increase in the consequences of the accidents which credit RCCA drop.

   The Leak-Before-Break analysis conclusions remain valid and the breaks previously exempted from structural considerations remain unchanged.

   All of the NSSS systems will continue to perform their intended design functions during normal and accident conditions. The pressurizer spray flow remains above its design value. Thus, the control system design analyses, which credit the flow, do not require any modification. The auxiliary systems and components continue to comply with applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

   All of the NSSS/Balance of Plant (BOP) interface systems will continue to perform their intended design functions. The steam generator safety valves will provide adequate relief capacity to maintain the steam generators within design limits. The atmospheric dump valves will still relieve at least 10% of the maximum full load steam flow. The steam dump system will still relieve at least 40% of the maximum full load steam flow. The current loss of coolant accident (LOCA) hydraulic forcing functions are still bounding.

Additionally, the reduction in the power measurement uncertainty allows for certain safety analyses to continue to be used, without modification, at the 2705 MWt power level (102% of 2652 MWt). Other safety analyses performed at a nominal power level have been either re-performed or re-evaluated at the 2689 MWt power level and continue to meet their applicable acceptance criteria.

Some existing safety analyses had been previously performed at a power level greater than 2689 MWt, and thus continue to bound the 2689 MWt power level. The effects on accident radiation dose for the power uprate were reanalyzed at 2705 MWt, and therefore are bounding when operating at 2689 MWt using the leading edge flow meter (LEFM) flow instrumentation.

The proposed changes to the Unit 2 reactor coolant system heatup/cooldown curves are being made to impose a conservative projection of the increase in neutron fluence associated with the power uprate. This projection will ensure that the requirements of 10 CFR 50, Appendix G, "Fracture Toughness Requirements", will continue to be met following the power uprate. The proposed changes to the MSSV Technical Specifications will not reduce the valve's capability to provide pressure relief when required. The design basis events that were protected against by the heatup/cooldown curves and the MSSVs have not changed; therefore, the probability of an accident previously evaluated is not increased by these proposed changes. These proposed changes also do not alter any assumptions previously made in the radiological consequence evaluations, nor affect mitigation of the radiological consequences of an accident previously evaluated.

Therefore the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   No new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed changes. All systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design function. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

   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 proposed amendment involve a significant reduction in a margin of safety?

   Operation at the 2689 MWt core power does not involve a reduction in a margin of safety. Extensive analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the Nuclear Regulatory Commission (NRC) or that are in compliance with applicable regulatory review guidance and standards.

   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 evaluated above, it is concluded that the proposed License Amendment Request satisfies the "no significant hazards consideration" standards of 10 CFR 50.92, and accordingly a no significant hazards finding is justified.

G. ENVIRONMENTAL IMPACT CONSIDERATIONS

A review was performed for the proposed power uprate to assess the existing National Pollutant Discharge Elimination System (NPDES) permit and the information contained in the Final Environmental Report (FER). In addition, a review of the Beaver Valley Units 1 and 2 Annual Radioactive Effluent Discharge Reports was conducted to verify that the actual releases from Beaver Valley Power Station (BVPS) are a very small percentage of the allowable limits and the FER estimates.

The BVPS units employ closed-loop natural draft cooling towers to dissipate waste heat to the atmosphere. Make-up for the cooling towers is drawn from the Ohio River. The increase in heat dissipated due to the proposed 1.4% increase in RTP will be approximately 120 million BTU/hr. This increased heat load will result in an increase in the maximum circulating water temperature of approximately 0.5°F. Therefore, the thermal power uprate will have no significant adverse impact on the environment.

The BVPS NPDES permit (No. PA0025615) does not impose any operating limits on cooling tower flow or temperature. Therefore, the power uprating will not result in exceeding any NPDES permit limits.

The FER also assessed other non-radiological impacts of plant operation on the environment and habitat. These assessments, and the assumptions upon which they were based, remain valid and are not impacted as a result of the proposed thermal power uprate.

The proposed changes do not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

H. REFERENCES

Codes & Standards

1. ASME Boiler & Pressure Vessel (B&PV) Code, Section III, "Nuclear Power Plant Components" 1971 Ed; Winter 1972 Addenda
2. ASME Standard NQA-2a-1990; "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications"
3. IEEE Standard 7-4.3.2-1990; "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations"
4. ANSI/ANS 3.5-1985; "Nuclear Power Plant Simulators for Use in Operator Training"
5. ANSI/ANS-67.04-2000; "Setpoints for Nuclear Safety-Related Instrumentation

Caldon Topical and Engineering Reports

6. ER-80P; "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System", Rev. 0, March 1997
7. ER-157P, Rev. 2, "Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System" (Enclosure 2 proprietary; Enclosure 3 non-proprietary ER-157N, Rev. 2)
8. "Responses and Further Clarifications to NRC Questions from September 29, 1998 Meeting [on Topical Report ER-80P] as Applied to Comanche Peak;" December 15, 1998

NRC Generic Industry Guidance

9. 10 CFR 50, Appendix K "ECCS Evaluation Models" as amended (65 FR 34913, June 1, 2000)
10. SECY-2000-057; "Final Rule: Revision of Par 50 Appendix K, 'ECCS Evaluation Models'; 3/3/2000"
11. NUREG-1431, Rev. 01; "Standard Technical Specifications for Westinghouse Plants"
12. NUREG-0800, Rev. 01; "Standard Review Plan for the Review of Safety Analysis Reports For Nuclear Power Plants LWR Edition"

13. R.G. 1.49 "Power Levels Of Nuclear Power Plants"; Revision 1 December 1973
14. R.G. 1.105, Rev. 2; "Instrument Setpoints for Safety-Related Systems"
15. Generic Letter (GL) 95-05; "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking"
16. Information Notice (IN) 86-56; "Reliability of Main Steam Safety Valves"
17. ONR Report; "The Effects of Deregulation of the Electric Power Industry on the Nuclear Plant Offsite Power System: an Evaluation;" June 30, 1999
18. Industry/TSTF Standard Technical Specification Change Traveler, "MSSV Changes", TSTF-235, Rev. 1

<u>Westinghouse Reports</u>

19. WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant"
20. Westinghouse Balance-of-Plant (BOP) Interface Design Criteria Manual;
21. WCAP-15264, Rev. 03 (Unit 1) and WCAP-15265, Rev. 02 (Unit 2) "Revised Thermal Design Procedure - Instrument Uncertainty Methodology for [FENOC] Beaver Valley;" (RTDP)
22. WCAP-11419, Rev. 02 (Unit 1) and WCAP-11366, Rev. 04 (Unit 2); "Westinghouse Setpoint Methodology for Protection Systems [at Beaver Valley Power Station]
23. WCAP-15139, "Beaver Valley Unit 2 Heatup and Cooldown Limit Curves During Normal Operation at 15 EFPY Using Code Case N-626"
24. WCAP-15571; Unit 1 "Surveillance Capsule Y Withdrawal Report;" [draft]
25. WCAP-10858P-A, Rev. 1; "AMSAC Generic Design Package"

ATTACHMENT C

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 289 and 161

LIST OF COMMITMENTS

The following items associated with the uprate will be accomplished as described below:

1. The installation and testing of the new LEFM system in a BVPS unit, including all related plant process computer system changes, will be completed prior to increasing power above 2652 MWt on that unit. Included in this implementation program will be the necessary procedures and documents required for operation, maintenance, testing, and training at the uprated power level with the new LEFM system.

2. FENOC will address the operability requirements for the LEFM system, including the appropriate actions to be taken when the LEFM is unavailable in new requirements to be included within the BVPS Licensing Requirements Manuals (LRM). This will be accomplished prior to raising core power above 2652 MWt. When feedwater flow measurements from the LEFM are unavailable, the originally approved rated thermal power of 2652 shall be used following the next required calorimetric.

3. FENOC will issue changes to the BVPS Unit 1 and Unit 2 UFSARs that will be incorporated with the next regularly scheduled UFSAR update after the amendments are approved and implemented. The changes will stipulate that future revisions of the topical reports listed in TS Section 6.9.5(b) that currently assume 102% of rated power shall reflect 100.6% of rated power only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM).

4. With respect to the 1.4% uprate, the Steam Generator Inspection Program will include consideration of the higher temperatures in crack growth rate analyses. Based on condition monitoring and operational assessments of inspection results, expansion of inspection plans and repairs will be made. Degradation growth rate changes will be incorporated into the operational assessment associated with potential affects of the uprate. FENOC will

confirm that the existing 40% through wall plugging criterion for Steam Generator Tubes will remain adequate for the 1.4% uprate conditions.

5. As part of the implementation program, all required RTS and ESFAS nominal setpoint changes prescribed in the RTDP submittal will be completed for a unit prior to increasing power above 2652 MWt for that unit.

6. The LEFM system software was developed and will be maintained under a Verification and Validation (V&V) program that is compliant with IEEE std. 7-4.3.2-1990 and ASME std. NQA-2a-1990.

7. A review of the training simulator fidelity following the uprating in RTP is being conducted, and revalidation in accordance with ANSI/ANS 3.5-1985 will be completed after benchmarking the plant at the increased power level of 2689 MWt. Included with the design modification package for the uprating will be implementation of all necessary procedures and training documents required to support operation and maintenance at the uprated power level using the LEFM system.

8. An alarm will be provided in the control room to alert operators should the LEFM system require maintenance. The alarm will be installed as a plant modification to be completed prior to the implementation of the power uprate.

9. A grid stability study is being performed at this time, to update the model with system changes that have occurred since 1997. The new study will incorporate the 1.4% power uprate to determine if any stability issues require resolution to support the proposed power uprate. This new study will be completed prior to increasing power above 2652 MWt.

10. Core power will not be increased above 2652 MWt until the NRC has completed its review and approved the revised dose calculations that were performed for the License Amendment Request submitted under LAR 280 for Unit 1, and 151 for Unit 2, forwarded on May 12, 2000 by L-00-008.

11. There are two tubes in Unit 2 and one tube in Unit 1 that will require plugging after an additional cycle of operation, due to fatigue considerations. These tubes will be removed from service no later than the refueling following implementation of the 1.4% uprating.