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L-01-146

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

Subject: Beaver Valley Power Station, Unit No. 2
BV-2 Docket No. 50-412, License No. NPF-73
Response to a Request for Additional Information
In Support of LAR No. 168

This letter provides the FirstEnergy Nuclear Operating Company (FENOC) response to a NRC Request for Additional Information (RAI), dated November 16, 2001, pertaining to FENOC letters L-01-089, dated June 28, 2001, and L-01-112, dated September 13, 2001. FENOC letter L-01-089 submitted License Amendment Requests (LAR) No. 168 that proposed changes to the Beaver Valley Power Station (BVPS), Unit No. 2, to allow operation of the reactor core with a positive moderator temperature coefficient (PMTC) for NRC review and approval. Letter L-01-112 provided the FENOC response to a NRC RAI, dated August 2, 2001, related to the PMTC LAR submittal. The information provided by this letter consists of the following:

- additional information related to the analyses performed including initial conditions,
- elaboration on why the events that were not reanalyzed are unaffected by operations using a PMTC,
- discussion of the affect of using PMTC on other technical specifications,
- further discussion on how BVPS, Unit No. 2, will continue to comply with the Anticipated Transient Without Scram (ATWS) rule, and
- detail on the administrative controls to be put in place in accordance with a new commitment made in support of the LAR.

The FENOC responses to the RAI are provided in Attachment A of this letter.

A001

Rec'd 01/18/02

FENOC requests NRC approval of License Amendment Request No. 168 by February 1, 2002, to support unit startup from the BVPS, Unit No. 2, ninth refueling outage (2R09).

This information does not change the evaluations or conclusions presented in FENOC letter L-01-089. If there are any questions concerning this matter, please contact Mr. Thomas S. Cosgrove, Manager Regulatory Affairs, at 724-682-5203.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 19, 2001.

Sincerely,

Lew W. Myers

Attachment

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

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) POSITIVE MODERATOR TEMPERATURE COEFFICIENT (PMTC) FOR BEAVER VALLEY POWER STATION, UNIT NO. 2 DATED NOVEMBER 16, 2001 (LICENSE AMENDMENT REQUEST [LAR] NO. 168)

During its review of the FirstEnergy Nuclear Operating Company (FENOC) licensing amendment request submitted June 28, 2001, the Nuclear Regulatory Commission (NRC) staff identified several areas that require additional information or clarification in order for [the NRC] to complete [its] review. FENOC responded to an e-mailed request for addition information (RAI) (Agencywide Documents Access and Management System [ADAMS] Accession No. ML012150055) via letter dated September 13, 2001 (ADAMS Accession No. ML012690002). The following is a list of questions identified by the staff after reviewing the responses provided by FENOC.

NRC RAI Question 1

In response to [the August 2, 2001] NRC RAI Question 1 and in information provided by the licensee in the original submittal, the licensee provided the initial conditions for most of the transients. However, the initial conditions for some of the transients are unclear. Please provide, preferably in tabular format, the initial conditions for each of the reanalyzed transients. If the transient was reanalyzed multiple times using different initial conditions each time, provide the values used for each analysis and identify the bounding case. The initial conditions should include the reactor power, moderator temperature coefficient (MTC), and availability of the power operated relief valves at a minimum, with other important values provided as necessary.

FENOC Response

Refer to Table 1. Note that the analyses employ, where applicable, the Revised Thermal Design Procedure (RTDP) (Reference 1). With RTDP, initial condition uncertainties are incorporated into the departure from nucleate boiling ratio (DNBR) design limit value. Thus, analyses that use RTDP (i.e., departure from nucleate boiling [DNB] events) assume an initial power equal to the nominal power of 2697 megawatts thermal (MWt) (2689 MWt plus reactor coolant pump heat). Events that do not use RTDP, include an uncertainty allowance of 0.6% (Reference 2) to cover the power calorimetric uncertainty.

The MTC limit being reviewed is a function of power. The MTC limit between 0% power and 70% power is +2 pcm/°F and decreases linearly from +2 pcm/°F to 0 from 70% power to 100% power. Thus, analyses performed at 100% power using a PMTC of +2 pcm/°F are conservative. Due to the availability of margin and for ease of interpretation, most analyses conservatively assume the PMTC at full power. Where margin was needed, cases were considered both at full power with the full power MTC limit of 0 pcm/°F and at part power with the part power MTC

limit. Analyses were performed in the late 1980s for other Westinghouse designed pressurized water reactors (PWRs) to determine the limiting set of conditions. In all cases, it was shown that the full power/zero MTC combination bounds the part power/PMTC combination – even for 70% power at MTCs as high as +7 pcm/°F. Based on those analyses, using a zero MTC at hot full power (HFP) has become part of the Westinghouse analysis methods. Recently, this question was asked on a Donald C. Cook Nuclear Plant, Unit No. 2, submittal and the effect was again quantified for the Loss of Normal Feedwater and Loss of Non-emergency AC Power events (See Reference 4). For Beaver Valley Power Station (BVPS), Unit No. 2, only the Loss of Flow events required the additional margin and had to be analyzed with the zero MTC at full power. Due to the short time frame of these events (less than 10 seconds), the heatup that occurs for the Loss of Flow events is less than observed for the Loss of Feedwater transients discussed in Reference 4 and therefore are less sensitive to the effects of a PMTC.

NRC RAI Question 2

In response to [the August 2, 2001] NRC RAI Question 1, the licensee provided a series of tables identifying the limits for the accidents analyzed. Some of the limits identified do not reflect the actual safety limits specified in the Beaver Valley Power Station, Unit No. 2 (BVPS-2) technical specifications. Because the staff reviewed BVPS-2 against the Standard Review Plan, the licensee must show the ability to meet the acceptance criteria relating to safety limits for all of the transients using technical specification values and the requested change to a positive moderator temperature coefficient (PMTC). The evaluation of accidents and transients at values not contained in the technical specifications is unacceptable.

FENOC Response

The analyses performed in support of the BVPS, Unit No. 2, RTDP and 1.4% Uprating programs anticipated the potential need for a PMTC and, subsequently, all of the analyses were performed to support a PMTC of +2 pcm/°F. The RTDP and 1.4% Uprating programs have already been approved by the NRC (References 1 and 2). These previously approved analyses are the same analyses that support PMTC.

With respect to limits that appear to be different than those defined in the NUREG-0800, "Standard Review Plan [SRP]," Westinghouse has, in some cases, imposed more restrictive limits than are required per the SRP. For the Main Feedline Rupture event, the SRP requires that the "core remain in place and geometrically intact with no loss of core cooling capability". Westinghouse has imposed a much more restrictive limit of no boiling in the hot legs. As such, the limit given in the summary table in the previous response (Reference 3) is the minimum margin to boiling in the hot leg. Demonstrating that no hot leg boiling occurs conservatively ensures that the core remains covered and geometrically intact. The Condition IV Main Feedline Rupture event must also demonstrate that primary and secondary pressures do not exceed 120% of their respective design pressures. In terms of pressurization, the Condition IV Main Feedline Break event is bounded by the Condition II Loss of Load event, which has been shown to not exceed 110% of the primary and secondary design pressures. This is the same

acceptance criterion that has been used on all previous Main Feedline Break analyses performed for the BVPS units and approved by the NRC.

The summary table in the previous response (Reference 3) gives limits for the Loss of Normal Feedwater and Loss of Non-Emergency Alternating Current (AC) Power events in terms of the maximum pressurizer water volume. These events are both Condition II events and the limits are that the DNB Design Basis is met, the primary and secondary pressures do not exceed 110% of design and that the event does not propagate to a more serious event without other incidents occurring independently. In terms of DNB, the Loss of Normal Feedwater event is bounded by the Loss of Load event and the Loss of Non-Emergency AC Power event is bounded by the Loss of Flow event. In terms of overpressurization, both events are bounded by the Loss of Load event. The most restrictive criterion is that the event does not propagate to a more serious fault. Should the pressurizer fill and water is relieved through the pressurizer safety valves, the potential exists for damage to the safety valves. A damaged safety valve may not reseat which results in an unisolatable breech of the reactor coolant system (RCS) (i.e., a Condition III small break loss-of-cooling accident [LOCA]). As such, Westinghouse has conservatively set the acceptance criterion for these events as the pressurizer shall not become water solid. This is the same acceptance criterion that has been used on all previous Loss of Normal Feedwater and Loss of AC Power analyses performed for the BVPS units and approved by the NRC.

NRC RAI Question 3

In response to [the August 2, 2001] NRC RAI Question 2, the licensee stated that all events not listed on page B-4 of the license amendment requested were unaffected by a PMTC. Please provide a list of those events and the bases to support this statement for each event.

FENOC Response

Refer to Table 1. For those events that do not explicitly model the PMTC, justification is provided in the "Discussion" column of Table 1.

Table 1 – Summary of the Beaver Valley Power Station, Unit No. 2, Non-LOCA Analysis Initial Conditions

Event Name	UFSAR Section	Initial Power (MWt)	Moderator Coefficient	Discussion
Rod Withdrawal from Subcritical	15.4.1	Hot Zero Power (HZP)	+2 pcm/°F	Analysis explicitly assumes the PMTC.
Rod Withdrawal at Power	15.4.2	2697 (100% Rated Thermal Power [RTP]) 1618.2 (60% RTP)	+2 pcm/°F	Analysis explicitly assumes the PMTC. Also, additional cases are analyzed assuming maximum (i.e., end-of-life [EOL]) reactivity feedback including a

Event Name	UFSAR Section	Initial Power (MWt)	Moderator Coefficient	Discussion
		269.7 (10% RTP)		most negative MTC.
Dropped Rod	15.4.3	Not Applicable (N/A)	N/A	This event is evaluated using generic analyses based on the approved methodology discussed in WCAP-11394-P-A. A positive moderator temperature coefficient provides a benefit in the analysis.
Boron Dilution	15.4.6	N/A	N/A	No explicit initial power or MTC assumption is made in this calculation. Boron concentrations and densities are assumed in the analyses, which have an implied MTC.
Loss of Load/Turbine Trip	15.2.3	2697 (DNB Case) 2713.2 (Pressure Case)	+2 pcm/°F	Analysis explicitly assumes the PMTC.
Loss of Normal Feedwater	15.2.7	2713.2	+2 pcm/°F	Analysis explicitly assumes the PMTC.
Loss of AC Power	15.2.6	2713.2	+2 pcm/°F	Analysis explicitly assumes the PMTC.
Feedwater Malfunction	15.1.2	2697 HZP	0.43 Δk/gm/cc	This event results in an RCS cooldown and, thus, the EOL moderator coefficient is conservative.
Excessive Load Increase	15.1.3	2697	N/A	Cases at both beginning-of-life [BOL] and EOL conditions are considered. The transient results in a slight decrease in temperature and the PMTC results in a slight benefit. As such, BOL cases assume an MTC of zero.
RCS Depressurization	15.6.1	2697	+2 pcm/°F	Analysis explicitly assumes the PMTC.
Steamline Break	15.1.5	HZP	See discussion	This event results in an RCS cooldown and, thus, the EOL moderator coefficient is conservative. The reactivity feedback model is verified each cycle. The PMTC would result in less severe analysis results.
Partial Loss of Flow	15.3.1	2697	0 pcm/°F	Analysis at full power with a zero MTC bounds analyses at part power with the PMTC. See discussion in response to RAI Question 1.
Complete Loss of Flow	15.3.2	2697	0 pcm/°F	Analysis at full power with a zero MTC

Event Name	UFSAR Section	Initial Power (MWt)	Moderator Coefficient	Discussion
				bounds analyses at part power with the PMTC. See discussion in response to RAI Question 1.
Locked Rotor	15.3.3	2697 (DNB Case) 2713.2 (Pressure Case)	0 pcm/°F	Analysis at full power with a zero MTC bounds analyses at part power with the PMTC. See discussion in response to RAI Question 1.
Rod Ejection	15.4.8	2705.0 HZP	See discussion	The power level given is 100.6% of the nominal core power (2689 MWt). Also, this analysis models an isothermal temperature coefficient (ITC) which bounds the PMTC. This is based on the approved methodology which is discussed in WCAP-7588, Rev. 1-A.
Feedline Break	15.2.8	2713.2	+2 pcm/°F 0.43 Δk/gm/cc	Analysis explicitly assumes the PMTC. Also, additional cases are analyzed assuming maximum (i.e., EOL) reactivity feedback including a most negative MTC.
Inadvertent ECCS	15.5.1	2713.2	0.43 Δk/gm/cc	This event results in an RCS cooldown and, thus, the EOL moderator coefficient is conservative.

NRC RAI Question 4

In response to [the August 2, 2001] NRC RAI Question 4, the licensee stated that BVPS-2 would have no unfavorable exposure time due to operation with a PMTC. The justification for this statement proposed by the licensee is that the MTC at hot full power (HFP) conditions will be less than -5.5 pcm/°F at all times in core life. The staff does not find the licensee's response and technical bases sufficient to support its position. In its initial RAIs, the staff noted there may be periods of time at the beginning of each cycle where the PMTC can cause BVPS-2 to exceed the ASME Stress Level C Limit of 3200 pounds-per-square inch absolute (psia) in the reactor coolant system (RCS) following an anticipated transient without scram (ATWS). This period of time will most likely occur during the initial startup of the plant following the refueling outage or any shutdown early in the core cycle that permits Xenon to decay. Please provide a quantitative analysis of the percentage of the cycle that BVPS-2 may be susceptible to exceeding the 3200 psia limit. In addition please provide a detailed descriptive summary of the methodology employed by FENOC to answer this question.

FENOC Response

This information is intended to address RAI Questions 4 through 7 related to the subject of Anticipated Transients Without Scram (ATWS). This response is primarily based on information discussed with the NRC via a telephone conference conducted on November 28, 2001, regarding these RAI questions.

Regarding the subject of ATWS, the only existing regulatory requirement applicable to BVPS, Unit No. 2, is that specified in the Final ATWS Rule, 10 CFR 50.62(b), which is specifically applicable to Westinghouse designed PWRs. The requirement of 10 CFR 50.62(b), which is the installation of an ATWS mitigation system, is met for BVPS, Unit No. 2, via the installation of AMSAC (ATWS Mitigation System Actuation Circuitry). The implementation of AMSAC for BVPS, Unit No. 2, was reviewed and approved by the NRC via Reference 5.

Notwithstanding, the analysis basis of the Final ATWS Rule, as documented in SECY-83-293 (Reference 6), are Westinghouse generic ATWS analyses documented in Westinghouse letter NS-TMA-2182 (Reference 7) and performed in response to NUREG-0460, "Anticipated Transient Without Scram for Light Water Reactors" (Reference 8). In these analyses, the assumed reference conditions for the MTC at HFP conditions is -8 pcm/°F, an upper limit HFP MTC condition corresponding to plant operation 95% of the cycle. With this MTC condition, it was adequately demonstrated that the peak RCS pressure following the RCS pressure limiting ATWS events (i.e., Loss of Load ATWS and Loss of Normal Feedwater ATWS) remained below a pressure of 3200 psig, the pressure corresponding to the ASME Service Level C stress limit as prescribed in NUREG-0460. Hence, the Westinghouse generic ATWS analyses documented in Reference 7 satisfactorily demonstrated that on a deterministic analysis basis, the ATWS peak RCS pressure limit of 3200 psig was met for 95% of the cycle. In recent years, the 5% of the cycle that unfavorable reactivity feedback conditions could exist has been termed as unfavorable exposure time (UET). Hence, in assessing plant changes and licensing amendment requests, one of the primary focuses for evaluating compliance with the analysis basis of the Final ATWS Rule is ensuring that UET in the reference case is limited to ≤5% of the cycle. This is most evident in the review and approval of Commonwealth Edison's (currently Exelon Nuclear) Byron and Braidwood LARs for a +7 pcm/°F part-power positive MTC (Reference 9).

For the subject BVPS, Unit No. 2, LAR for a +2 pcm/°F part-power positive MTC, FENOC has committed to limiting the HFP MTC to a maximum value of -5.5 pcm/°F for 100% of the cycle. As documented in Reference 10, a HFP MTC of -5.5 pcm/°F corresponds to conditions in the reference generic Westinghouse ATWS analyses that equate to a peak RCS pressure of 3200 psig. Hence, by conservatively meeting this HFP MTC requirement, the licensee is essentially committing to meeting a 0% UET. It should be noted that the plant configuration and conditions associated with the generic Westinghouse ATWS analysis used to establish the -5.5 pcm/°F HFP MTC value corresponding to 3200 psig are those for a 4-Loop Westinghouse PWR at 3423 MWt Nuclear Steam Supply System (NSSS) power. BVPS, Unit No. 2, is 3-Loop plant configuration with a NSSS power of 2697 MWt. As documented in Reference 7, the 4-Loop, 3423 MWt plant configuration is conservative relative to a 3-Loop plant configuration at 2785 MWt. Hence, the application of the generic 4-Loop ATWS analysis in establishing the -5.5 pcm/°F limit is conservative relative to BVPS, Unit No. 2. As indicated in response to

NRC RAI Question 5 in the initial round of questions (Reference 11), FENOC intends to meet the commitment of designing the core for BVPS, Unit No. 2, to a maximum HFP MTC of -5.5 pcm/°F and will make this limit a reload core design constraint. This reload design constraint will be reflected as a limit in the BVPS, Unit No. 2, Reload Safety Analysis Checklist (RSAC) which is employed as part of the NRC approved Westinghouse Reload Safety Evaluation Methodology (Reference 12). This is what was previously meant with the terminology of "administrative controls." It should be noted the approach of limiting the HFP MTC to a value of –5.5 pcm/°F for ATWS concerns and placing this as a design constraint on the reload core design is consistent with that previously approved by the NRC and currently in use for Millstone Nuclear Power Station, Unit No. 3, operation with a +5 pcm/°F part-power MTC Technical Specification (Reference 13).

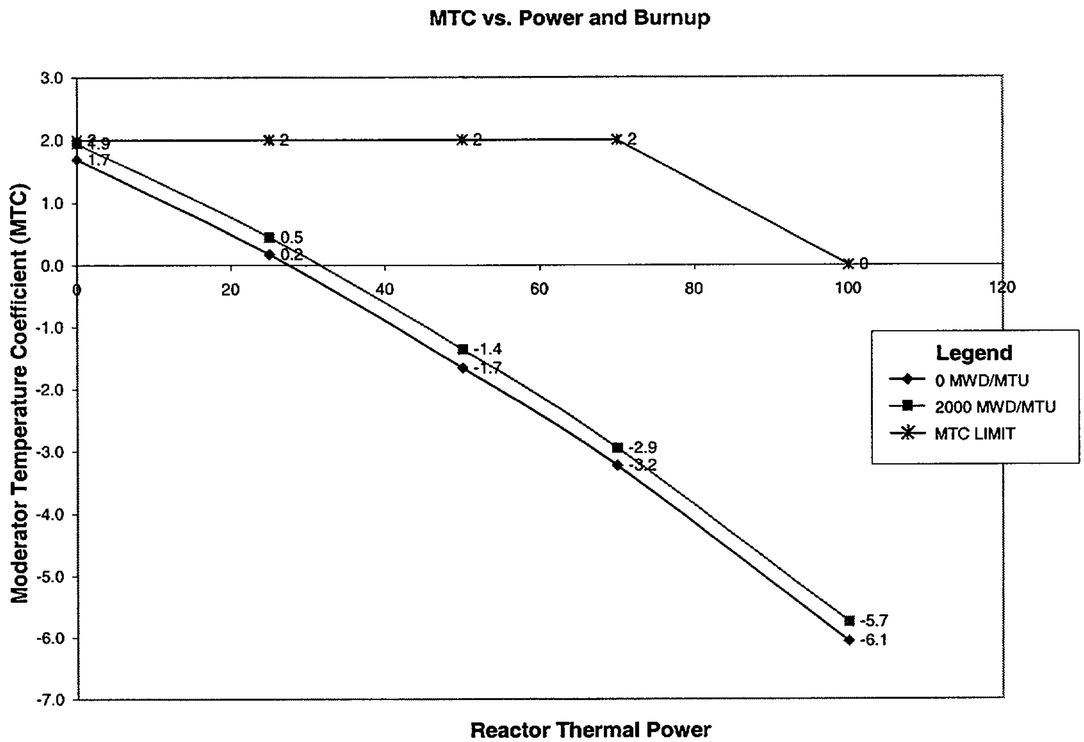
There are several other items that should be considered relative to ATWS:

First, as indicated by the subject LAR, the proposed amendment limits the MTC to +2 pcm/°F at part-power conditions from 0 to 70% power and linearly decreases from +2 pcm/°F at 70% power to 0 pcm/°F at 100% power. The MTC limit at 100% power remains the same as currently licensed for BVPS, Unit No. 2. This part-power positive MTC limit is illustrated in Figure 1 and is shown in comparison to the actual planned reload core MTC at the most limiting times in the cycle (i.e., between 0 megawatt days per metric ton uranium [MWD/MTU] and 2000 MWD/MTU burnup). As can be seen from Figure 1, the actual MTC meets the proposed +2 pcm/°F limit at 0% power conditions and becomes negative at approximately 30% power. At 100% HFP conditions, the MTC is more negative than -5.5 pcm/°F. Further note that the MTC versus power reflected in Figure 1 conservatively includes no Xenon.

The reference ATWS analysis assumed in the determination of the -5.5 pcm/°F as described earlier assumes operation of the pressurizer power-operated relief values (PORVs) consistent with the guidelines of NUREG-0460. As documented in Reference 7, the unavailability of one or more PORVs is adverse for ATWS RCS pressure concerns. However, a review of past operation of BVPS, Unit No. 2, over the past eight years shows that it has been over 5 years since a Unit 2 PORV isolation has been required in the first two months of an operating cycle. Since PORV isolation early in the cycle when MTC is least favorable for ATWS has not occurred in the past five years and, since no change in this operating strategy is anticipated, the impact on ATWS due to the potential for blocking a PORV is not considered significant.

Finally, FENOC operates BVPS, Unit No. 2, at power conditions with rod control in automatic. In practice, reactor startups are performed with the control rods in manual up to 15% power at which time the operators may place the rod control system in operation. The selection of the power level to place the control rods in automatic is based on Senior Reactor Operator preference. However, the control rods are placed in automatic control prior to reaching full power and typically around 50% power. Operation with rods in automatic is a benefit for ATWS and serves to further limit the peak RCS pressure reached during the pressure limiting Loss of Load and Loss of Normal Feedwater ATWS events.

Figure 1



NRC RAI Question 5

When the ATWS rule was written, the worst case condition occurred when the following three initial conditions were evaluated: 1) the most positive MTC, 2) HFP conditions, and 3) beginning of cycle. The staff believes that these conditions may no longer be the most limiting because the use of a power-dependent MTC may result in peak reactor coolant system pressures that exceed the ASME Stress Level C Limit at lower power levels which have a more positive MTC. The licensee has not provided any technical analyses to confirm that their analysis of record is bounding. Please provide the analyses for ATWS events that demonstrate the ability of BVPS-2 to meet the ASME Stress Level C Limit over the spectrum of reactor power levels with the power-dependent MTC proposed in the new technical specifications.

FENOC Response

As agreed to during the November 28, 2001 telecommunication conference, the response discussed for Question Number 4 above addresses this question.

NRC RAI Question 6

In response to [the August 2, 2001] NRC RAI Question 5, the licensee identified "administrative controls" which will be in place to ensure the MTC at HFP conditions will be less than -5.5 pcm/°F. The use of administrative controls to limit or restrict operation of a plant to less than the technical specification values is contrary to the requirements of 10 CFR 50.36 in that the technical specifications must establish the minimum safety limits and initial accident conditions for plant operation. The staff has also developed the following additional questions related to the "administrative controls".

a) Please identify any surveillance requirements or procedures which will be used to monitor this self-imposed limit during power operations and the frequency for their performance. In addition, describe all conditions where BVPS-2 would be permitted to exceed or be exempt from the limit set forth in the commitment. Finally, provide a detailed description identifying actions to be taken if the limit specified in the commitment is violated.

b) In attachment C of the June 28, 2001 submittal, FENOC committed to use of "administrative controls" to ensure the MTC at HFP conditions will be less than or equal to -5.5 pcm/°F at all times during core life prior to first entry into Mode 2 for BVPS-2 Cycle 10 operations. In order for the technical specification modifications to be valid for future core reloads the commitment to use these administrative controls must be made for all future BVPS-2 cycles.

c) FENOC identified the need for "administrative controls" to provide protection for BVPS-2 for an ATWS event. The staff believes the best way to ensure compliance with the FENOC limit (-5.5 pcm/°F) proposed for safety is to include it in the license. Please provide a detailed justification for the use of administrative controls and your interpretation of 10 CFR 50.36 in this area. Provide specific language for a license condition that implements the "administrative controls" requirements.

FENOC Response

As agreed to during the November 28, 2001 telecommunication conference, the response discussed for Question Number 4 above addresses this question.

NRC RAI Question 7

The current technical specifications for BVPS-2 permit continued operation of the unit with one power operated relief valve (PORV) blocked. Operation with one PORV blocked or inoperable and a -5.5 pcm/°F MTC at HFP conditions could result in exceeding the ASME Stress Level C Limit of 3200 psia during ATWS events. This situation is presented in NS-EPR-2833 (LAR No. 168, Reference 3) and demonstrates the susceptibility of Westinghouse plants to an ATWS event when one or more PORVs are blocked while operating at or below the MTC limit

proposed in the licensee's administrative controls. FENOC must either modify the BVPS-2 technical specifications to prevent or limit operation with a blocked or inoperable PORV or adjust their commitment consistent with NS-EPR-2833 to prevent exceeding the ATWS 3200 psia limit with PORVs blocked or inoperable.)

FENOC Response

As agreed to during the November 28, 2001 telecommunication conference, the response discussed for Question Number 4 above addresses this question.

NRC RAI Question 8

In the BVPS-2 submittal, the effects of a PMTC on other technical specifications were not addressed. The boron concentration in the ECCS accumulators and refueling water storage tank (RWST) provides the necessary negative reactivity to ensure adequate shutdown margin. Please provide the technical information necessary to support the view that an increase in the boron concentration in the ECCS accumulators and RWST is not necessary for those periods when the plant would operate at its technical specification MTC limits. If increased boron concentration is needed, the necessary concentration should be submitted and the technical specification values changed. If no increased boron concentration is needed, a detailed technical justification should be provided to prove that current boron concentrations in the RWST and ECCS accumulators provide sufficient negative reactivity to meet the requirements for shutdown margin specified in the technical specifications for BVPS-2.

FENOC Response

During the preparation of the PMTC LAR, FENOC evaluated the potential affect of the proposed change to a PMTC on other technical specifications (TSs) as well as other licensing basis documents. It was determined that the proposed change to a PMTC did not impact other TSs. With respect to the NRC's specific concern regarding boron concentration, there are no boron concentration changes required to implement the proposed PMTC. However, changes to the RWST and accumulator boron concentrations have been proposed to address higher reactivity levels associated with reactor core operation at higher plant capacity factors. Fuel cycle design would be restricted by imposing less than full power operating limitations during future operating cycles if operating margins for these boron concentration requirements are not increased. The RCS boron concentration limit in Mode 6 during refueling is also being revised for consistency with the RWST and accumulator changes. These boron concentration changes were requested through LAR No. 164 for BVPS Unit No. 2, which was submitted as FENOC letter L-01-099 for NRC approval on July 25, 2001. LAR No. 164 is in the final stages of the NRC review and approval process.

References:

1) Letter from L. J. Burkhart (NRC) to L. W. Myers (FENOC), "Beaver Valley Power Station, Unit Nos. 1 and 2 – Issuance of Amendment Re: Implementation of the Revised Thermal Design Procedure, Etc. (TAC Nos. MB0848 and MB0849)," July 20, 2001.

2) Letter from L. J. Burkhart (NRC) to L. W. Myers (FENOC), "Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and BVPS-2) – Issuance of Amendment Re: 1.4-Percent Power Uprate and Revised BVPS-2 Heatup and Cooldown Curves (TAC Nos. MB0996, MB0997 and MB2557)," September 24, 2001.

3) Letter from L. W. Myers (FENOC) to the NRC, "Beaver Valley Power Station Unit No. 2 BV-2 Docket No.50-412, License No. NPF-73 Response to a Request for Additional Information In Support of LAR No. 168," FENOC Letter Number L-01-112, September 13, 2001.

4) Letter from J. F. Stang (NRC) to R. P. Powers (Indiana Michigan Power Company), "Donald C. Cook Nuclear Plant, Unit 2 – Issuance of Amendment (TAC Nos. MA9870 and MA9871)," August 23, 2001.

5) Letter from P. S. Tam (NRC) to J. D. Sieber (Duquesne Light Company), "Beaver Valley Power Station, Units 1 & 2 – Implementation of the ATWS Rule (TAC Nos. 59070 and 62943)," May 31, 1988.

6) SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events", W. J. Dircks, July 19, 1983.

7) NS-TMA-2182, Letter from T. M. Anderson (Westinghouse) to Dr. S. H. Hanauer (NRC) dated December 30, 1979, "ATWS Submittal."

8) NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," December, 1978.

10) Docket Nos. 50-454, 50-455, 50-456, 50-457, Letter from G. F. Dick Jr. (NRC) to D. L. Farrar (Commonwealth Edison Company) dated July 27, 1995, "Issuance of Amendments (TAC NOS. M89092, M89093, M89072 and M89091)."

11) NS-EPR-2833, Letter from E. P. Rahe (Westinghouse) to S. J. Chilk (NRC) dated October 3, 1983, "Rulemaking on Anticipated Transients Without Scram."

12) L-01-112, Letter from L. W. Myers (FENOC) to US-NRC dated September 13, 2001, "Beaver Valley Power Station, Unit No. 2, BV-2 Docket No. 5-412, License No. NPF-73, Response to a Request for Additional Information In Support of LAR No. 168."

13) WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

14) Docket No. 50-423, Letter from R. L. Ferguson (NRC) to Mr. E. J. Mroczka (Northeast Nuclear Energy Company) dated January 20, 1988, "Issuance of Amendment (TAC NOS. 60651, 66023, and 66024)."