FENOC

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December 21, 2000
L-00-141

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

Subject: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
License Amendment Request No. 248

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company requests an amendment to the Beaver Valley Power Station (BVPS) Unit 1 license in the form of a change to the technical specifications and bases. The proposed amendment will revise the Unit 1 Technical Specifications 2.2.1 "Reactor Trip System Instrumentation Setpoints" and 3/4.3.1 "Reactor Trip System Instrumentation" and associated bases. License Amendment Request (LAR) 248 proposes to eliminate the Steam/Feed Flow Mismatch and Low SG Water Level Reactor Trip from the technical specifications.

The Steam/Feed Flow Mismatch and Low SG Water Level Reactor Trip was included in the BVPS Unit 1 design in order to meet the requirements of the Institute of Electric and Electronic Engineers Standard 279, 1971, "Criteria for Protection Systems for Nuclear Power Generating Stations (IEEE-279). Specifically, this reactor trip was necessary to meet the IEEE-279 requirements for potential adverse control and protection system interactions. LAR 248 proposes to take credit for the Steam Generator Level Median Selector Switch (MSS) installed in 1997 to meet the requirements of IEEE-279 for control and protection system interactions. The MSS eliminates the potential for an adverse control and protection system interaction described by IEEE-279 and, therefore, eliminates the design requirement for the Steam/Feed Flow Mismatch and Low SG Level reactor trip. The MSS is a proven design modification for Westinghouse plant feedwater control systems that has been implemented at other Westinghouse plants including BVPS Unit 2.

The proposed technical specification changes for BVPS Unit No. 1 are presented in Attachment A-1. The safety analysis (including the no significant hazards evaluation) is presented in Attachment B.

This change is requested to be approved prior to the start of the next Unit 1 refueling outage (1R14).

A001

This change has been reviewed by the Beaver Valley review committees. The change was determined to be safe and does not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety analysis. An implementation period of up to the first entry into Mode 2 following the 1R14 refueling outage is requested following the effective date of this amendment.

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

Sincerely,

Marc Pearson for
Lew W. Myers

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

Subject: **Beaver Valley Power Station, Unit No. 1**
BV-1 Docket No. 50-334, License No. DPR-66
License Amendment Request No. 248

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

FirstEnergy Nuclear Operating Company

Marc P. Pearson

Marc P. Pearson
Director, Plant Services - FENOC

COMMONWEALTH OF PENNSYLVANIA

COUNTY OF BEAVER

Subscribed and sworn to me, a Notary Public, in and for the County and State above named, this 21st ~~th~~ day of December, 2000.

Tracey A Baczek

My Commission Expires:

Notarial Seal
Tracey A. Baczek, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Aug. 16, 2001
... Association of Notaries

ATTACHMENT A

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Change No. 248

The following is a list of the affected pages:

2-7
B 2-6
B 2-7
3/4 3-3
3/4 3-12
B 3/4 3-1

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level-Low-Low	≥ 12% of narrow range instrument span-each steam generator	≥ 10.7% of narrow range instrument span-each steam generator
14. Deleted ~~Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level~~	~~≤ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level~~ ~~≥ 25% of narrow range instrument span-each steam generator~~	~~≤ 43.4% of full steam flow at RATED THERMAL POWER coincident with steam generator water level~~ ~~≥ 23.1% of narrow range instrument span-each steam generator~~
15. Undervoltage-Reactor Coolant Pumps	≥ 2750 volts-each bus	≥ 2687 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 57.5 Hz - each bus	≥ 57.4 Hz - each bus
17. Turbine Trip		
A. Auto stop oil pressure	45 psig	± 5 psig
B. Turbine Stop Valve	≥ 1% open	≥ 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable
20. Reactor Trip System Interlocks		
A. Intermediate Range Neutron Flux, P-6	≥ 1×10^{-10} Amps	≥ 6×10^{-11} Amps

(Proposed Wording)

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	Functional Unit	Channel Check	Channel Calibration	Channel Functional Test	Modes in Which Surveillance Required
12.	Loss of Flow - Single Loop	S	R	Q	1
13.	Loss of Flow - Two Loops	S	R	Q	1
14.	Steam/Generator Water Level-Low-Low	S	R	Q	1, 2
Deleted 15.	~~Steam Feedwater Flow Mismatch and Low Steam Generator Water Level~~	~~S~~	~~R~~	~~Q~~	~~1, 2~~
16.	Undervoltage-Reactor Coolant Pumps	N.A.	R	M	1
17.	Underfrequency-Reactor Coolant Pumps	N.A.	R	M	1
18.	Turbine Trip				
	a. Auto Stop Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
	b. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19.	Safety Injection Input from ESF	N.A.	N.A.	R	1, 2
20.	Reactor Coolant Pump Breaker Position Trip	N.A	N.A.	R	N.A
21.	Reactor Trip Breaker	N.A.	N.A.	M(5, 11) and S/U(1)	1, 2, 3(14), 4(14), 5(14)

(Proposed Wording)

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

Loss of Flow

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

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

Steam Generator Water level

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

~~Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level~~

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

(Proposed Wording)

~~reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by $\geq$ 1.55 x 10^6 lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below 25 percent, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.~~

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

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

Turbine Trip

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

(proposed Wording)

BASES

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

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

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

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

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

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

(proposed Wording)

ATTACHMENT B

Beaver Valley Power Station, Unit No. 1
Proposed License Amendment Request No. 248
STEAM/FEEDWATER FLOW MISMATCH DELETION

A. DESCRIPTION OF AMENDMENT REQUEST

The proposed amendment would delete the Steam/Feedwater Flow Mismatch and Low Steam Generator (SG) Water Level reactor trip function from the Unit 1 Beaver Valley Power Station (BVPS) technical specifications (TS). Reference to this trip function would be deleted from the following TS: Table 2.2-1 Item 14, Table 3.3-1 Item 15, and Table 4.3-1 Item 15. The word "Deleted" would be inserted into the tables as necessary to maintain the current numbering sequence. The TS bases discussion of this trip function on Pages B 2-6, B 2-7, and B 3/4 3-1 would be deleted as well. This change is similar to the change previously approved by the NRC for BVPS Unit 2 in TS Amendment No. 27 issued on 2/20/90 (TAC No. 74573).

B. DESIGN BASES

Each SG has three independent water level instrument channels which provide input to the Reactor Protection System (RPS) for a reactor trip on two out of three low-low levels as described in the Updated Final Safety Analysis Report (UFSAR) Section 7.2.1.1.5. The feedwater control system, which controls the feedwater regulating valves and hence feedwater flow into the SG also uses one of the three SG water level channels for input. Therefore, common instrument channels (separated electrically by isolation devices) are used for both reactor protection and feedwater control. This design assures that the plant will be controlled by the same measurements with which it is protected. This design precludes any sensor deviation between control and protection functions thereby reducing the likelihood of spurious reactor trips and channel calibration and maintenance tasks. However, this design introduced the potential for adverse control and protection system interaction.

In order to meet the required industry standard regarding the prevention of adverse control and protection system interaction, this design incorporated the Steam/Feedwater Flow Mismatch and Low SG Water Level reactor trip. An adverse control and protection system interaction could result from certain failures that could negate a particular water level channel and simultaneously cause a

control system action that would require a subsequent protective action in order to prevent exceeding design safety limits. For example, a postulated failure (high) of one SG water level channel would send a one out of three high level signal to the RPS and, if being used for feedwater control, the failed channel would also send a high level signal to feedwater control circuits causing the feedwater regulating valve to shut. This would cause the SG water level to decrease. In such a scenario, the Institute of Electric and Electronic Engineers Standard 279, 1971, "Criteria for Protection Systems for Nuclear Power Generating Stations" (IEEE-279), imposes the requirement for degradation by a second random failure. The underlying logic is that the initial protection system failure is considered the initiating event for the transient. Therefore, the initial failed channel does not constitute the "single failure" imposed on the protection system by IEEE-279. As such, an additional protection failure must be postulated to occur, and the protection system must continue to be capable of initiating the appropriate protective action. The limiting single failure in this instance would be a failure (fail as-is) of one of the remaining two SG level channels. This leaves only one operating channel which is insufficient to satisfy the 2/3 logic needed for a SG Water Level Low-Low reactor trip. However, with the diverse Steam/Feedwater Flow Mismatch and Low SG Water Level trip function included in the design, the necessary protective action (i.e., reactor trip) needed to meet safety limits would still function. The Steam/Feedwater Flow Mismatch and Low SG Water Level trip occurs upon a one of two channel steam/feedwater mismatch coupled with a one of two channel low SG water level. The SG water level channels that input to this trip do not include the channel used for control. Therefore, the reduction in water level in the affected SG sensed by the remaining operable level channel, combined with the feedflow/steam flow mismatch signal, would result in the required reactor trip. The original BVPS design described above has been modified by the installation of new equipment designed to prevent adverse control and protection system interaction from taking place.

In 1997, a median signal selector (MSS) was installed in the feedwater level control system for each Unit 1 SG. The MSS addresses the control/protection system interaction previously discussed and eliminates the need for the Steam/Feedwater Flow Mismatch and Low SG Water Level reactor trip. The purpose of the MSS is to prevent a failed instrument channel from causing a disturbance in the controlled system that could initiate a plant transient. Formerly, the feedwater control system received SG water level input from a single channel;

failure of that channel could cause the adverse control system action of concern. With the MSS, all three SG level measurement channels are input to the control system and compared by the MSS. The MSS selects the median signal for use by the control system. By rejecting the high and low signals, the MSS prevents the control system from acting on any single failed protection system instrument channel. Since no adverse control system action may now result from a single, failed protection instrument channel, a second random protection system failure (as would otherwise be required by IEEE-279) need not be considered. Signals resulting from a single failed high or low SG level channel will be rejected for control purposes and, therefore, will not affect the system. The MSS eliminates the control and protection system interaction mechanism.

In addition, the BVPS Unit 1 instrumentation taps for the SG water level and main steam flow instrument channels are independent of each other. An independent tap design for this instrumentation addresses the concern of meeting the IEEE-279 standard for adverse protection and control interaction with an MSS when a SG instrument tap is postulated to fail. This concern was identified in Westinghouse Nuclear Safety Advisory Letter (NSAL) 96-004, "Control and Protection Systems Interactions." In the NSAL, Westinghouse identified a potential control and protection system interaction scenario, applicable to plants with a common instrument tap for SG water level and main steam flow instrumentation, for which an MSS alone would not be capable of meeting the requirements of IEEE-279. In this case, the Steam/Feedwater Flow Mismatch and Low SG Water Level reactor trip or other plant modifications may be necessary to meet the requirements of IEEE-279. However, BVPS Unit 1 has independent instrument taps for the SG water level and main steam flow instrument channels and, therefore, the concern identified in NSAL 96-004 is not applicable to BVPS Unit 1.

Thus, the MSS can be substituted for the reactor trip on Steam/Feedwater Flow Mismatch and Low SG Level to meet the control/protection system interaction requirements of IEEE-279. Westinghouse Electric Corporation WCAP-11484, "Feedwater Control System Median Signal Selector," July 1987 confirms the acceptability of the MSS for this purpose.

C. JUSTIFICATION

The MSS currently installed in BVPS Unit 2 provides similar protection from adverse control and protection system interaction as proposed in this amendment for BVPS Unit 1 and has demonstrated reliable operation since 1987. Similar in design to the Unit 2 MSS, the Unit 1 MSS is also not part of the protection system and is designed to reduce the frequency of system failures through utilization of highly reliable components and a minimum of additional equipment. It should be noted that failure of the MSS does not directly compromise the ability of the protection system to perform its safety related functions (i.e., failure of the MSS will not disable any protection channel). Furthermore, the design provides the capability for complete on-line MSS testing to provide unambiguous determination of system failures. The performance of periodic tests verify the functionality and calibration of each MSS on each SG.

BVPS Unit 2 has utilized the MSS in its SG water level instrumentation since 1987. The MSS installed in Unit 1 is equivalent to the Unit 2 MSS in terms of quality, testability and use of highly reliable components. The combination of demonstrated performance, low likelihood of failure, and the ability to detect failures through continued periodic testing provide the necessary degree of confidence relative to MSS operational readiness and reliability. In addition to eliminating control/protection system interaction concerns, the reliability of the feedwater control system is enhanced with the use of the MSS since it eliminates potential plant transients (SG level excursions) that could result from the failure of a single SG level instrument channel.

Removal of the Steam/Feedwater Flow Mismatch and Low SG Water Level trip from the BVPS design provides additional benefits. The most significant contribution would be the reduced challenges to the overall plant safety systems by eliminating potential spurious and inadvertent reactor trips that could result from this trip function. The plant has already experienced unnecessary inadvertent reactor trips during calibration/surveillance activities on the Steam/Feedwater Flow Mismatch and Low SG Water Level trip function. Some of the trips could have been avoided if the Steam/Feedwater Flow Mismatch and Low SG Water Level trip had been removed. Greater operational flexibility due to additional SG level margin (the low level trip setpoint is higher than the low-low level setpoint) will reduce the potential for SG level related trips. As such, removal of this trip

function will actually enhance safe operation of the plant by reducing the potential for challenges to safety systems and unnecessary plant transients. Elimination of the Steam/Feedwater Flow Mismatch and Low SG Water Level trip function also reduces the work level and system complexity of the plant. Removal of this trip function will reduce the required surveillance/maintenance work needed for the RPS and will reduce the active systems that have the potential to cause unnecessary plant transients and require additional training and operating precautions.

Elimination of the Steam/Feedwater Flow Mismatch and Low SG Water Level reactor trip from the Unit 1 BVPS TS and plant design is acceptable since no credit is taken for the Steam/Feedwater Flow Mismatch and Low SG Water Level reactor trip in any accident analyses described in the UFSAR. The MSS provides sufficient substitute protection against adverse control/protection system interaction, which is the basis for including the Steam/Feedwater Flow Mismatch and Low SG Water Level reactor trip in the original plant design. Elimination of this unnecessary reactor trip will enhance plant safety by 1) reducing the potential for unnecessary and unplanned plant transients, 2) reducing plant surveillance/calibration activity in protection and control systems, and 3) provide human factor benefits derived from reducing plant complexity.

D. SAFETY ANALYSIS

The TS Bases for the Steam/Feedwater Flow Mismatch and Low SG Water Level trip states that this trip is not used in the transient and accident analyses but is included in the TS to ensure the functional capability of the specified trip settings and enhance the reliability of the RPS. In addition, this trip provides a function that is redundant to the SG Water Level Low-Low trip.

The Loss of Normal Feedwater analysis in UFSAR Section 14.1.8 and Major Rupture of Main Feedwater Pipe Analysis in Section 14.2.5.2 both list the Steam/Feedwater Flow Mismatch and Low SG Water Level trip as protection for these events. However, UFSAR Section 7.2.1.1.5 states that no credit is taken in the accident analyses for this trip. As shown in the sequence of events Table 14.1-2 for the Loss of Normal Feedwater and Table 14.2-2 for the Major Rupture of Main Feedwater, both with and without offsite power, the SG Water Level Low-Low trip is credited for reactor trip. No credit was taken for the Steam/Feedwater Flow Mismatch and Low SG Water Level trip in these accident analyses. No

other accident analyses take credit for a reactor trip on low SG water level for protection.

The initial BVPS design utilized the Steam/Feedwater Flow Mismatch and Low SG Water Level trip to address adverse control and protection system interaction in the case where the initiating transient of an event was postulated to be a failure of one SG water level channel with a single failure being the second SG water level channel, rendering the 2/3 low-low water level trip inoperable. However, the current BVPS Unit 1 design, as described in the last paragraph of UFSAR Section 7.2.2.3.5, utilizes the MSS to satisfy the control and protection system interaction requirements of IEEE-279.

Thus, as no credit is taken for the Steam/Feedwater Flow Mismatch and Low SG Water Level trip in the accident analyses described in the UFSAR and the MSS provides an adequate substitute to address adverse control/protection system interaction, there would be no change nor any adverse effect to any safety analyses addressed in the UFSAR upon removal of the Steam/Feedwater Flow Mismatch and Low SG Water Level trip function from the BVPS Unit 1 TS and design. Therefore, the proposed change does not adversely affect the safe operation of the plant.

E. NO SIGNIFICANT HAZARDS EVALUATION

The proposed license amendment would eliminate the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level reactor trip from the Unit 1 Beaver Valley Power Station technical specifications. The Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level reactor trip is not credited in any accident analyses described in the Updated Final Safety Analyses Report. The trip was designed to provide protection from adverse control and protection system interaction in accordance with the requirements of the Institute of Electric and Electronics Engineers Standard 279. The elimination of this reactor trip is acceptable due to the Median Selector Switch installed in the Unit 1 feedwater control system. The operation of the Median Selector Switch prevents a failed instrument channel from causing a disturbance in the feedwater control system that could initiate a plant transient. As such, the Median Selector Switch prevents adverse control and protection system interaction such that it replaces the need for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

reactor trip and satisfies the requirements of the Institute of Electric and Electronic Engineers Standard 279. The proposed change is similar to the change previously approved by the NRC for the Beaver Valley Power Station Unit 2.

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

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

The following evaluation is provided for the no significant hazards consideration standards.

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

   The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The initiating conditions and assumptions for accidents described in the Updated Final Safety Analyses Report remain as previously analyzed. The proposed change does not introduce a new accident initiator nor does it introduce changes to any existing accident initiators or scenarios described in the Updated Final Safety Analyses Report. The Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level reactor trip is not credited for accident mitigation in any accident analyses described in the Updated Final Safety Analyses Report. The Steam/Feedwater Flow Mismatch and

Low Steam Generator Water Level trip was designed to meet the control and protection systems interaction criteria of the Institute of Electric and Electronic Engineers Standard 279. The Median Selector Switch prevents adverse control and protection system interaction such that it replaces the need for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level reactor trip and satisfies the Institute of Electric and Electronic Engineers Standard 279 requirements. As such, the affected control and protection systems will continue to perform their required functions without adverse interaction and the capability to shut down the reactor when required on Low-Low Steam Generator water level to mitigate an accident previously evaluated is unaffected.

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

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. The substitution of the Median Selector Switch for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip will not introduce any new failure modes to the required protection functions. The Median Selector Switch only interacts with the feedwater control system and the Steam Generator Water Level Low-Low protection function is not affected by this change. Isolation devices in the Median Selector Switch circuitry ensure that the Steam Generator Water Level Low-Low protection function is not affected. The Median Selector Switch is designed to reduce the frequency of system failures through utilization of highly reliable components in a design that relies on a minimum of additional equipment. Components utilized in the Median Selector Switch are of a quality consistent with low failure rates and minimum maintenance requirements, and conform to protection system requirements. Furthermore, the design provides the capability for complete unit testing that provides unambiguous determination of credible system failures. It is through these features that the overall design of the Median Selector Switch minimizes the occurrence of undetected failures that may exist between test intervals. Additionally, the reliability of the Median Selector Switch has been shown by Unit 2 operating experience to be acceptable.

3. Does the change involve a significant reduction in a margin of safety?

The margin of safety depends on the maintenance of specific operating parameters and systems within design requirements and safety analysis assumptions.

The proposed amendment does not involve revisions to any safety limits or safety system setting that would adversely impact plant safety. The proposed amendment does not alter the functional capabilities assumed in a safety analysis for any system, structure, or component important to the mitigation and control of design bases accident conditions within the facility. Nor does this amendment revise any parameters or operating restrictions that are assumptions of a design basis accident. In addition, the proposed amendment does not affect the ability of safety systems to ensure that the facility can be placed and maintained in a shutdown condition for extended periods of time.

The ability of the Steam Generator Water Level Low-Low reactor trip function credited in the safety analysis to protect against a sudden loss of heat sink event is not affected by the proposed change. Since the Steam Generator Low-Low Level trip provides complete protection for all accident transients that result in low steam generator level, eliminating the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip will not change any safety analysis conclusion for any analyzed accident described in the Updated Final Safety Analyses Report.

The Median Selector Switch prevents adverse control and protection system interaction such that it replaces the need for the Steam/Feedwater Flow Mismatch and low Steam Generator Water Level reactor trip and satisfies the Institute of Electric and Electronic Engineers Standard 279 requirements. The proposed change will enhance safe operation since the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level trip function removal decreases the challenges to the plant safety systems, decreases the plant surveillance/maintenance activity, and reduces the plant complexity; all resulting in a reduction in the potential for unnecessary plant transients.

The technical specifications continue to assure the applicable operating parameters and systems are maintained within the design requirements and

safety analysis assumptions. Therefore, the elimination of this trip function will not result in a significant reduction in the margin of safety as defined in the Updated Final Safety Analyses Report or technical specifications.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the considerations expressed above, it is concluded that the activities associated with this license amendment request satisfy the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. ENVIRONMENTAL CONSIDERATION

The proposed amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. It has been determined that the proposed amendment involves no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this proposed amendment.