FENOC
FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077-0004

L. William Pearce
Vice President

724-682-5234
Fax: 724-643-8069

February 22, 2005
L-05-022

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

Subject: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
License Amendment Request Nos. 314 (Unit 1) and 187 (Unit 2)

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) hereby requests an amendment to the license for Beaver Valley Power Station (BVPS) Unit No. 1 and Unit No. 2 in the form of changes to the Technical Specifications. The proposed amendments will revise the Radiation and Accident Monitoring Technical Specification requirements applicable to post accident monitoring instrumentation. The proposed changes are based on WCAP-15981-NP (Non-Proprietary), Rev. 0, "Post Accident Monitoring Instrumentation Re-definition for Westinghouse NSSS Plants." WCAP-15981 provides the technical justification for identifying the appropriate post accident monitoring instrumentation to be included in the Technical Specifications for Westinghouse NSSS plants. BVPS is the Westinghouse Owners Group lead plant for this program. WCAP-15981-NP was submitted to the NRC by letter WOG-04-474, dated September 17, 2004.

During the development of the license amendment requests, the important operator actions required to prevent core damage for BVPS Units 1 and 2 were derived from the PRAs for each unit. In the BVPS Units 1 and 2 PRAs, the instrumentation is modeled as part of the human reliability analysis, which models the important operator actions to prevent core damage. These PRAs were updated in 2003, and provide an accurate representation of the design and operation of BVPS Units 1 and 2. The update included internal initiating events, as well as seismic and fire initiating events.

The important operator actions for BVPS Units 1 and 2 were identified using the same risk importance measures and criteria as that used in the generic assessment contained in WCAP-15981. The important operator actions for BVPS Units 1 and 2 were found to be consistent with those identified in the generic assessment contained in WCAP-15981. Therefore, the conclusions in WCAP-15981 are applicable to BVPS Units 1 and 2.

The enclosed License Amendment Request (LAR) contains seven attachments. The proposed Technical Specification changes are provided in Attachments A-1 and A-2 for Unit Nos. 1 and 2, respectively. The proposed changes to the Technical Specification Bases are provided for information in Attachments B-1 and B-2 for Unit Nos. 1 and 2, respectively. In order to facilitate the review of the Technical Specification changes, clean typed copies of the affected Technical Specification pages are provided for information in Attachments C-1 and C-2 for Unit Nos. 1 and 2, respectively. Attachment D provides a summary (Table 1) of the BVPS Unit 1 and 2 Regulatory Guide 1.97 instrumentation including the classification, disposition, and references that identify the source documents (i.e., submittals and Safety Evaluation Reports) regarding the BVPS Regulatory Guide 1.97 instrumentation.

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 and no significant hazard evaluation.

FENOC requests approval of the proposed amendment by February 28, 2006. Once approved, the amendment shall be implemented within 90 days.

No new commitments are contained in this submittal. If there are any questions concerning this matter, please contact Mr. Henry L. Hegrat, Supervisor, Licensing at 330-315-6944.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 22, 2005.

Sincerely,

L.W. Pearce

L. William Pearce

Enclosure:

FENOC Evaluation of the Proposed Change(s)

Enclosure: Beaver Valley Power Station, Unit Nos. 1 and 2,
License Amendment Request Nos. 314 and 187

Attachments:

A-1. Proposed Unit 1 Technical Specification Changes

A-2. Proposed Unit 2 Technical Specification Changes

B-1. Proposed Unit 1 Technical Specification Bases Changes

B-2. Proposed Unit 2 Technical Specification Bases Changes

C-1. Unit 1 Draft Typed Technical Specification Pages

C-2. Unit 2 Draft Typed Technical Specification Pages

D. Table 1, BVPS Units 1 and 2 Regulatory Guide 1.97, Instrumentation Summary Disposition

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Senior Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

ENCLOSURE

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 314 and 187

FENOC Evaluation of the Proposed Changes

Subject: Application for Amendment to the Radiation and Accident Monitoring Technical Specifications - Implementation of WCAP-15981, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants"

Table of Contents

Attachments

* Provided for information only

1.0 DESCRIPTION

This is a request to amend Operating Licenses DPR-66 (Beaver Valley Power Station Unit 1) and NPF-73 (Beaver Valley Power Station Unit 2).

The proposed changes will revise the requirements of the Unit 1 and Unit 2 Technical Specifications (TS) 3/4.3.3.8, "Accident Monitoring Instrumentation." The proposed change revises the Regulatory Guide 1.97 (Reference 1) instrumentation contained in these TS to be consistent with the technical basis for accident monitoring instrumentation identified in WCAP-15981, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," (Reference 2). This change includes evaluating the current Regulatory Guide 1.97 classification of the affected instrumentation with respect to its function as a post accident monitoring instrument based on WCAP-15981. The results of the WCAP-15981 evaluations performed in this LAR are for the sole purpose of determining the most appropriate BVPS instrumentation to be included in Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation." The current BVPS response to Regulatory Guide 1.97 (including the instrument type and category classifications) documented in the references on Table 1 (Attachment D) will not be changed as a result of this LAR. Therefore, there are no changes to the BVPS response to Regulatory Guide 1.97 or the plant design associated with this LAR.

The proposed change will also make the content of the current TS 3/4.3.3.8, "Accident Monitoring Instrumentation" more consistent with the requirements of the corresponding Improved Standard Technical Specifications (ISTS) (Reference 3) 3.3.3, "Post Accident Monitoring Instrumentation".

Table 1 in Attachment D of this LAR contains a summary of the evaluations performed and disposition of the BVPS Unit 1 and Unit 2 Regulatory Guide 1.97 instrumentation with respect to inclusion in Post Accident Monitoring Technical Specification 3/4.3.3.8.

Additionally, the proposed changes include the relocation of the alarm function requirements for the Unit 1 and Unit 2 containment area radiation monitors and the Unit 2 main steam discharge process radiation monitors contained in Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation," to the Licensing Requirements Manual (LRM) and Offsite Dose Calculation Manual (ODCM), respectively. The applicable indication function requirements for the Unit 1 and Unit 2 containment area radiation monitors will be moved to Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation." These radiation monitors are addressed in this License Amendment Request (LAR), since the only reason for retaining these monitors in the Technical Specifications is their Regulatory Guide 1.97 classification and potential use as a post accident monitoring instrument. The classification of the radiation monitors was re-evaluated in this LAR, along with the other post accident monitoring instruments in Technical Specification 3/4.3.3.8 consistent with WCAP-15981.

Additional changes to the Unit 1 and Unit 2 Technical Specifications are proposed in this LAR to provide consistency with the ISTS. Format and editorial changes necessary to support the technical changes described above are also included in this LAR. To meet format requirements the Index, Technical Specifications, and Bases pages will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

2.0 PROPOSED CHANGES

The proposed Technical Specification changes, which are submitted for NRC review and approval, are provided in Attachments A-1 and A-2 for Units 1 and 2, respectively. The changes to the Technical Specification Bases are provided in Attachments B-1 and B-2 for Units 1 and 2, respectively. The Technical Specification Bases changes do not require NRC approval. The Beaver Valley Power Station (BVPS) Technical Specification Bases Control Program (TS 6.18) controls the review, approval and implementation of Technical Specification Bases changes. The Technical Specification Bases changes are provided for information only. In addition, draft clean TS typed pages are provided in Attachments C-1 and C-2 for Units 1 and 2, respectively. These draft typed TS pages are provided for information only to assist in understanding the marked-up Technical Specifications provided in Attachments A-1 and A-2.

The proposed changes to the Technical Specifications and Technical Specification Bases have been prepared electronically. Deletions are shown with a strike-through and insertions are shown by providing a separate text insertion. This presentation allows the reviewer to readily identify the information that has been deleted and added.

To meet format requirements, the Index and Technical Specifications pages will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

Included in this LAR is the relocation of certain Technical Specification requirements to one of two manuals, the LRM or ODCM. The LRM is incorporated by reference in the BVPS Unit 1 and 2 Updated Final Safety Analysis Report (UFSAR). Changes to the ODCM are controlled by the 10 CFR 50.59 in accordance with NRC Safety Evaluation Report for BVPS Amendment Nos. 188 (Unit 1) and 70 (Unit 2) dated June 12, 1995. Therefore, all changes to the relocated requirements will be controlled by the 10 CFR 50.59 process. In addition, the control of changes to the ODCM is also specified in the Technical Specifications (Section 6.14). The NRC has approved both of these manuals for the relocation of Technical Specification requirements in previous BVPS LARs. All requirements proposed for relocation outside of the Technical Specifications in this LAR have been evaluated against the Technical Specification criteria of 10 CFR 50.36(c)(2)(ii) and the specific criteria for post accident monitoring (PAM) instrumentation identified in WCAP-15981. All requirements being proposed for relocation will be relocated outside of the Technical Specifications without making technical changes to the requirements. Only format and editorial changes will be made to the relocated requirements that are necessary to fit the format and presentation of similar requirements in the LRM or ODCM as applicable. Future changes to these requirements, once they are relocated, will be made under the 10 CFR 50.59 process. All relocated requirements will be contained in the LRM or ODCM as identified in the following discussions.

The following provides a description of the proposed changes. Unless otherwise stated, each change described below affects the Technical Specifications of both Unit 1 and Unit 2.

Change No. 1

This change consists of two components. Each component of the change addresses a different function performed by the containment area radiation monitors. One component of this change relocates the Technical Specification requirements in 3/4.3.3.1, "Radiation Monitoring Instrumentation," that are required to preserve the alarm function of the monitors to the LRM. The other component of the change moves the applicable indication requirements of the monitors to Technical

Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," to retain the PAM function of the radiation monitors. Only the PAM function has been determined to be required in the Technical Specifications.

Specifically for the relocation portion of this change, the proposed change will relocate the LCO, Applicability, Action Statements, Action (#35) and Surveillance Requirement (4.3.3.1) associated with the containment area radiation monitors contained in Tables 3.3-6 and 4.3-3 (Instrument 1. b. ii., Unit 1) and Tables 3.3-6 and 4.3-3 (Instrument 1. b., Unit 2) from Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation," to the Licensing Requirements Manual (LRM). The relocated requirements will retain the current requirements that address the alarm function (i.e., setpoint) of the containment area radiation monitors. The alarm function of the monitors is not required to support the PAM (indication only) function of the monitors.

For the indication requirements that are being moved into the PAM Technical Specification, the proposed change will also move the Minimum Channels Operable requirement of 2 channels (and revise it to Required Channels), the Applicable Modes of 1, 2, and 3, and the Channel Check (including revising the Channel Check surveillance frequency from 12 hours to monthly) and the Channel Calibration Surveillance Requirements of SR 4.3.3.1, associated with the containment area radiation monitor indication function contained in Tables 3.3-6 and 4.3-3 (Instrument 1. b. ii., Unit 1) and Tables 3.3-6 and 4.3-3 (Instrument 1. b., Unit 2) from Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation," to Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," Table 3.3-11 and Surveillance Requirement 4.3.3.8. In addition, Action #35 of Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation," is replaced by the applicable Actions of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" for the monitoring Function being moved to Technical Specification 3/4.3.3.8. The movement of these radiation monitoring requirements to Technical Specification 3/4.3.3.8 includes the relocation of the containment area radiation monitor identification numbers (specified in Tables 3.3-6 and 4.3-3) to the Bases of Technical Specification 3/4.3.3.8. The retention of these requirements in Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," will maintain the post accident monitoring indication function of the containment area radiation monitors in the Technical Specifications consistent with the BVPS plant-specific application of the generic methodology for developing a technical basis for accident monitoring instrumentation identified in WCAP-15981.

Change No. 2

This change only affects Unit 2. The proposed change will relocate the Unit 2 LCO, Applicability, Action Statements, Action (#35) and Surveillance Requirement (4.3.3.1) associated with the main steam discharge radiation monitor contained in Tables 3.3-6 and 4.3-3 (Instrument 2. c. iii., Unit 2) in Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation," to the ODCM. These Unit 2 radiation monitors are not required to support any PAM function identified in the BVPS plant-specific application of the generic methodology in WCAP-15981. As these Unit 2 radiation monitors are effluent type monitors, they are being relocated to the ODCM consistent with the location of the requirements for other relocated effluent radiation monitors, including the corresponding Unit 1 main steam effluent monitors. In order to continue to support the operability of these monitors (outside of the Technical Specifications) the current Technical Specification requirements for the Unit 2 main steam discharge radiation monitors will be relocated from Technical Specification 3/4.3.3.1 to the ODCM.

Change No. 3

The title of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," is revised to "Post Accident Monitoring (PAM) Instrumentation." The LCO is revised from referencing "channels" to referencing "each Function," since the LCO and associated Actions are presented on a Function basis in Table 3.3-11. These proposed changes are consistent with the corresponding LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation LCO," in the ISTS. The applicable Index pages (IV and XVII (Unit 1) and V (Unit 2)) will also be revised to reflect the proposed change to the title of Technical Specification 3/4.3.3.8.

Change No. 4

A footnote is added to the Mode 3 Applicability of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," that states:

"Power Range Neutron Flux PAM Function is not required in Mode 3."

This exception to the applicable Modes of the PAM instrumentation is consistent with the Mode of Applicability required for the Power Range Neutron Flux instrumentation to perform its PAM function as identified by the BVPS plant-specific application of the generic methodology in WCAP-15981.

<u>Change No. 5</u>

A new General Note is added to Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," Actions that states, "Separate Action statement entry is allowed for each Function." The new General Note clarifies that the Actions may be entered on a Function basis and is consistent with the change to the LCO to reference "each Function," instead of "channels".

The proposed change is consistent with the corresponding Note to the Actions in Technical Specification 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," in the ISTS.

<u>Change No. 6</u>

The following changes are made to the Technical Specification 3/4.3.3.8 "Accident Monitoring Instrumentation," Actions:

For the instruments relocated to the LRM from Technical Specification 3/4.3.3.8, the current Technical Specification Actions a. and b., are also relocated to the LRM.

For the instruments retained in Technical Specifications 3/4.3.3.8, Actions a. and b., of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," are replaced by the following Actions:

"a.1 One or more PAM Functions with one required channel inoperable, restore the required channel to OPERABLE status within 30 days, or

a.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 14 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channel of the Function to OPERABLE status, or a.3 Be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b.1 One or more PAM Functions with two or more required channels inoperable, restore the required channels to OPERABLE status within 7 days, or

b.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 7 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status, or

b.3 Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours."

Action a. of Technical Specification 3/4.3.3.8 is revised from referencing "With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-11," to reference "One or more PAM Functions with one required channel inoperable." The restoration Completion Time of Action a. (new Action a.1) is revised from 7 days to 30 days. The reference to restoring the "inoperable channel" is revised to restoring the "required channel." A new Action is added to Action a (a.2). The new Action is applicable if the affected channel is not restored to operable status within the allowed 30 days. The new Action requires that a report be submitted to the NRC within the following 14 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channel to OPERABLE status. The shutdown Action (new Action a.3) is revised to require that the unit must be in at least HOT STANDBY in 6 hours, and in HOT SHUTDOWN within the following 6 hours if one of the other provisions of the Action are not met.

Action b. of Technical Specification 3/4.3.3.8 is revised from referencing "With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11," to reference "One or more PAM Functions with two or more required channels inoperable." The restoration Completion Time of Action b (new Action b.1) is revised from 48 hours to 7 days. The reference to restoring the "inoperable channel(s)" is revised to restore "required channels." If the required channels are not restored to operable status within the allowed 7 days, a new requirement is added to Action b. (new Action b.2) to submit a report to the NRC within the following 7 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status. The shutdown Action (new Action b.3) is revised to require that the unit must be in at least HOT STANDBY in 6 hours, and in HOT SHUTDOWN within the following 6 hours.

Technical Specification 3/4.3.3.8 Action c. (Unit 2 only) is applicable to the Reactor Coolant System Subcooling Margin Monitor instrumentation and is relocated to the LRM with the Reactor Coolant System Subcooling Margin Monitor instrumentation requirements in Table 3.3-11.

The proposed changes to Technical Specification 3/4.3.3.8 Actions a. and b. are consistent with the Actions of Technical Specification 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," in the ISTS (and the associated reporting requirements in Section 5.6 of the ISTS) except for the addition of the shutdown action to Action a. (new Action a.3) and the new action in Action b (i.e., Action b.2) that allows for a report to be submitted to the NRC and the reliance on an alternate method of accomplishing the PAM Function for continued operation with two or more required channels inoperable in one or more PAM Functions.

<u>Change No. 7</u>

The following changes are made to the Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," Surveillances:

A General Note is added to the Surveillance Requirements of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," as stated below.

"Surveillance Requirement 4.3.3.8.1 applies to each PAM Function in Table 3.3-11. Surveillance Requirement 4.3.3.8.2 applies to each PAM Function in Table 3.3-11 except for the Penetration Flow Path Containment Isolation Valve Position Function. Surveillance Requirement 4.3.3.8.3 applies only to the Penetration Flow Path Containment Isolation Valve Position Function in Table 3.3-11."

This change identifies the SRs that are required to be performed for each PAM instrument Function and supports the deletion of Technical Specification 3/4.3.3.8

Table 4.3-7 for the PAM instrument Functions that are retained in the Technical Specifications. Current Technical Specification Table 4.3-7 serves to identify the surveillances required to be performed for each PAM instrument function and is no longer needed with the General Note being added to the surveillances.

Surveillance Requirement 4.3.3.8 of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," is also separated into three Surveillance Requirements and revised as stated below.

"4.3.3.8.1 Perform a CHANNEL CHECK at least once every 31 days.

4.3.3.8.2 Perform a CHANNEL CALIBRATION at least once every 18 months.**

4.3.3.8.3 Perform a CHANNEL FUNCTIONAL TEST at least once every 18 months."

A ** footnote is added to the Surveillance for the Channel Calibration that states "Neutron detectors are excluded from the Channel Calibration." This change clarifies that the neutron detectors are excluded from the Channel Calibration and is consistent with Footnote 6 in Table 4.3-1, "Reactor Trip System Instrumentation Surveillance Requirements," in the current BVPS Unit 1 and 2 Technical Specifications.

<u>Change No. 8</u>

The following changes are made to Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," Table 3.3-11 requirements:

The title of Table 3.3-11 of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," is revised to "Post Accident Monitoring Instrumentation" consistent with the ISTS.

The heading of "Function" is added to Table 3.3-11 (Unit 1) above the list of PAM instruments. The heading above the list of PAM instruments in Table 3.3-11 (Unit 2) is revised from "Instrument" to "Function."

The "Minimum Channels" column in Table 3.3-11 is deleted, and the "Total No. of Channels" column is revised to "Required Channels," for the PAM instrument Functions that are retained in the Technical Specifications. These changes clarify the number of PAM instrument channels required for each PAM Function to satisfy the LCO.

The number of "Required Channels" specified in Table 3.3-11 for the Pressurizer Water Level Function is revised from 3 to 2. Only two channels are necessary to maintain the redundancy required by the PAM Technical Specification.

The "Required Channels" for the Auxiliary Feedwater Flow Rate PAM Function for Unit 1 is clarified to reflect that there are 3 required channels (1 per steam generator).

The "Required Channels" for the Auxiliary Feedwater Flow Rate PAM Function for Unit 2 is revised to reflect that there are 2 required channels per steam generator, and separated into individual PAM Functions for each steam generator. Providing a separate Function for each steam generator is consistent with the Unit 2 design that includes redundant monitoring channels for the Auxiliary Feedwater Flow Rate PAM Function for each steam generator.

The Action Column is deleted from Table 3.3-11 (Unit 2 only). The relocation of the Unit 2 Reactor Coolant System Subcooling Margin Monitor instrumentation and associated Action c. to the LRM leaves only Actions a. and b. which apply to all PAM Functions. Therefore, the identification of each Action for each function on Unit 2 Table 3.3-11 is no longer required.

The "Required Channels" requirement for the Core Exit Thermocouple PAM Function is revised from 4 channels /core quadrant to simply 2 channels. This change is justified in WCAP-15981. This change is based on information in WCAP-14696-A (Reference 4), which was approved by the NRC in their Safety Evaluation dated September 2, 1999, and described further in the discussion of this change in Section 4.0.

A footnote is added to the Core Exit Thermocouple PAM Function "Required Channels" that states "A channel consists of two core exit thermocouples." This change clarifies the channel requirement for the Core Exit Thermocouples. This footnote is consistent with the footnote contained in Technical Specification 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," in the ISTS.

Two footnotes are added to the Penetration Flow Path Containment Isolation Valve Position "Required Channels" that state:

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

These footnotes provide necessary clarifications to the Containment Isolation Valve Position requirement and are consistent with the footnotes contained in Technical Specification 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," in the ISTS.

Change No. 9

The following Changes are made to Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," surveillance requirement Table 4.3-7:

Table 4.3-7 has been deleted for the PAM Functions that will be retained in Technical Specification 3/4.3.3.8, Table 3.3-11. The addition of the General Note to the PAM surveillances requires that these Surveillances be performed on the appropriate PAM Function and obviates the need for listing the surveillances in Table 4.3-7.

The elimination of Table 4.3-7 is consistent with presentation of surveillance requirements in Technical Specification 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," in the ISTS. Unit 1 Index page XVII will also be revised to reflect the deletion of Table 4.3-7.

The "S/U" frequency for the Channel Check on the Auxiliary Feedwater Flow Rate PAM Function is revised to an "M" frequency. Additionally, the footnote in Table 4.3-7 that specifies the Auxiliary Feedwater Flow Rate PAM Function Channel Check is to be performed in conjunction with Surveillance Requirement 4.7.1.2.7 following an extended plant outage is deleted. These changes result in a Channel Check being performed on the Auxiliary Feedwater Flow Rate PAM Function consistent with the Frequency of all other PAM functions.

The N/A frequency for the Channel Check on the Containment Wide-Range Pressure PAM Function is deleted, and a Channel Check will be required to be performed on the Containment Wide-Range Pressure PAM Function consistent with the Channel Check Frequency of the other PAM functions.

Change No. 10

The BVPS Regulatory Guide 1.97 and Technical Specification 3/4.3.3.8 PAM Functions have been evaluated consistent with the methodology in WCAP-15981. This change includes the evaluations performed to assess the BVPS PAM instrumentation and describes the following changes:

The PAM Functions and "Required Channels" that are added to Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," Table 3.3-11 are:

Function	Required Channels
Power Range Neutron Flux	2
High Head Safety Injection Flow	1
SG Pressure	
a) SG "A"	2
b) SG "B"	2
c) SG "C"	2
Refueling Water Storage Tank Level (Unit 2 only, Wide Range)	2
RCS Pressure (Wide Range)	2
SG Water Level (Wide Range)	3 (1 per SG)
Penetration Flow Path Containment Isolation Valve Position	2 per penetration flow path [(a)] [(b)]

These Unit 1 and Unit 2 PAM Functions satisfy the criteria of 10 CFR 50.36 (c)(2)(ii) and the BVPS plant-specific application of the generic methodology for developing a technical basis for accident monitoring instrumentation identified in WCAP-15981.

The following Unit 1 accident monitoring instrumentation contained in Table 3.3-11 and all of the associated Technical Specification requirements from Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," are relocated to the Unit 1 LRM:

3. Reactor Coolant System Subcooling Margin Monitor

5. PORV Limit Switch Position Indicator

6. PORV Block Valve Limit Switch Position Indicator

7. Safety Valve Acoustical Detector Position Indicator

9. Containment Sump Wide Range Water Level

12. Reactor Vessel Level Indicating System

The following Unit 2 accident monitoring instrumentation contained in Table 3.3-11 and all of the associated Technical Specification requirements from Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," are relocated to the Unit 2 LRM:

3. Reactor Coolant System Subcooling Margin Monitor

4. PORV Limit Switch Position Indicator

5. PORV Block Valve Limit Switch Position Indicator

6. Safety Valve Position Indicator

8. Containment Sump Wide Range Water Level

10. Reactor Vessel Level Indication System

<u>**Change No. 11**</u>

This change addresses the administrative changes made to the Technical Specifications in this LAR. In addition to the technical changes described above, various editorial and format changes are made to the Technical Specifications. These editorial and format changes are made to accommodate the technical changes described above or to conform more closely to the format and presentation of the corresponding ISTS requirements. The editorial and format changes include such things as re-numbering, repagination, changes to index pages, and the insertion of text such as "deleted" or the phrase "this Action not used" to replace relocated requirements.

3.0 BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during an accident. This information provides the necessary support for the control room operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions during Design Basis Accidents (DBAs).

The PAM instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and assess the unit status and behavior following an accident.

The availability of the PAM instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments were originally identified by addressing the recommendations of Regulatory Guide 1.97 as required by Supplement 1 to NUREG-0737 (Reference 5). The current BVPS-specific Regulatory Guide 1.97 PAM instrumentation was identified to the NRC and approved by the NRC in the documents referenced in the Notes associated with Table 1 in Attachment D of this LAR for Unit 1 and Unit 2.

Regulatory Guide 1.97 Type A variables provide the primary information required for the control room operator to take specific manual actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions as assumed in the DBA analyses.

In addition to Type A variables, Regulatory Guide 1.97 identified Category 1 variables as significant to safety. Regulatory Guide 1.97 Category 1 variables were provided to determine whether other systems important to safety are performing their intended functions.

Typically, Regulatory Guide 1.97 Type A variables are also Category 1 variables. However, not all Category 1 variables are also classified as Type A.

Technical Specification 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," in the ISTS specifies (in a reviewers note) that a plant should include all of their Regulatory Guide 1.97 Type A and all of their Regulatory Guide 1.97 Category 1 instrumentation in the PAM Specification. The list of generic PAM Functions identified in Technical Specification 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," in the ISTS was developed in the late 1980's based on design basis accident requirements and generic insights from Probabilistic Risk Assessments (PRA) available at that time.

The PAM instrumentation was included in the Technical Specifications to ensure that the instrumentation required by the operators to respond to an accident and bring the plant to a safe stable state is operable if required during an accident. The inclusion of PAM instrumentation functions in the ISTS was determined based on the Technical Specification Criteria in 10 CFR 50.36 (c)(2)(ii).

The four Criteria for determining Technical Specification content were codified in the Federal Regulations by an amendment to 10 CFR 50.36 on July 19, 1995 (60 CFR 36953). The 10 CFR 50.36 (c)(2)(ii) Criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The fourth Criterion reflects the insights obtained from PRA studies. As discussed below, the PAM instrumentation contained in Technical Specification 3.3.3, "Post

Accident Monitoring (PAM) Instrumentation," in the ISTS is based primarily on the first three Criteria of 10 CFR 50.36(c)(2)(ii). Insights from PRA studies were not widely known or available when Technical Specification 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," was issued in Revision 0 of NUREG-1431 in September 1992.

The original basis for determining the instrumentation to be included in Technical Specification 3.3.3, "PAM Instrumentation," contained in the ISTS is defined in WCAP-11618 (Reference 6). WCAP-11618 was submitted to the NRC in November 1987, and identified certain PAM Instrumentation that satisfied 10 CFR 50.36(c)(2)(ii) Criterion 3. The justification discussed in WCAP-11618 for satisfying Criterion 3 for the PAM Tech Spec is as follows:

"Specific Accident Monitoring Instrumentation provides the operator with the information needed to perform the required manual actions to bring the plant to a stable condition following an accident. This instrumentation is part of the primary success path and which functions or actuates to mitigate a DBA or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Specific Accident Monitoring Instrumentation satisfies criterion 3."

Therefore, WCAP-11618 limited the content of the proposed ISTS PAM Technical Specification to Regulatory Guide 1.97 Type A instruments. Non Type A Category 1 instrumentation was not identified as satisfying any of the criteria for inclusion in the Technical Specifications.

The NRC letter to the Owners Groups (Reference 7), which documented the review of WCAP-11618, stated that PAM Instrumentation satisfies the definition of Type A variables in Regulatory Guide 1.97, and meets Criterion 3. The NRC justification for retaining Type A variables states: "Type A variables provide primary information (i.e., information that is essential for the direct accomplishment of the specified manual actions (including long-term recovery actions) for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs or transients)." It also discusses that since only Type A variables meet Criterion 3, the Standard Technical Specifications should contain a narrative statement that indicates that individual plant Tech Specs should contain a list of PAM Instrumentation that includes Type A variables.

However, regarding the non-Type A Category 1 variables, the NRC stated in Reference 7, that: "the staff is unable to confirm the Owners Groups' conclusion that Category 1 Post-Accident Monitoring Instrumentation is not of prime importance in limiting risk [Criterion 4]. Recent PRAs have shown the risk significance of operator recovery actions which would require a knowledge of Category 1 variables. Furthermore, recent severe accident studies have shown significant potential for risk reduction from accident management. The Owners Groups' should develop further risk-based justification in support of relocating any or all Category 1 variables from the Standard Technical Specifications." The Owners Groups' participating in the development of the ISTS choose not to evaluate the inclusion of Regulatory Guide 1.97 Non Type A, Category 1 instrumentation in the PAM Technical Specification at that time. Therefore, the ISTS PAM Technical Specification was issued with the requirement that all plant-specific Regulatory Guide 1.97 Type A, and all plant-specific Regulatory Guide 1.97 Category 1 instrumentation be included in the PAM Technical Specification.

WCAP-15981 was developed to specifically address the NRC request to further evaluate the inclusion of Regulatory Guide 1.97 Category 1 variables in the PAM Technical Specification. In addition, WCAP-15981 provides a generic methodology for developing a technical basis for relocating certain Post Accident Monitoring instruments from the Technical Specifications. The conclusions contained in WCAP-15981 are based on generic risk insights (i.e., evaluations against 10 CFR 50.36 (c)(2)(ii) Criterion 4) and a re-evaluation of the overall basis for Accident Monitoring instrumentation with respect to the first three Criteria of 10 CFR 50.36 (c)(2)(ii). WCAP-15981 also includes the consideration of the reliance on the instrumentation not specifically evaluated when the list of PAM instrumentation was originally developed in the ISTS. These additional considerations include instrumentation required to mitigate the consequences of beyond design basis accidents, such as those that are important for Severe Accident Management (e.g., SAMG), and offsite emergency radiological protection actions (e.g., Emergency Action Level declarations and Offsite Dose Calculations). A plant-specific assessment, using the methodology contained in WCAP-15981 is also required to ensure that the generic conclusions are applicable to each plant implementing the methodology in the WCAP. This LAR contains the results of the plant-specific assessment that confirmed that the WCAP-15981 generic conclusions are applicable to BVPS Units 1 and 2.

Regulatory Guide 1.97 provides guidance on the classification of plant parameters (that are indicated by instrumentation) according to their importance. There are different classes of variables identified in Regulatory Guide 1.97 according to the type of information that is provided by that variable. Type A variables provide primary information needed to permit control room operating personnel to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis events. Regulatory Guide 1.97 does not include a listing of Type A variables, since they are to be determined on a plant-specific basis. However, Regulatory Guide 1.97 includes a list of Category 1 variables. The instrumentation associated with those Category 1 variables is identified in Table 2 below:

Table 2: Regulatory Guide 1.97 Category 1 Variables	
Power Range Neutron Flux	Hydrogen Monitors
Source Range Neutron Flux	Pressurizer Level
Reactor Coolant System Hot Leg Temperature	Steam Generator Water Level (Wide Range)
Reactor Coolant System Cold Leg Temperature	Condensate Storage Tank Level
Reactor Coolant System Pressure (Wide Range)	Core Exit Temperature – Quadrant [1]
Reactor Vessel Water Level	Core Exit Temperature – Quadrant [2]
Containment Sump Water Level (Wide Range)	Core Exit Temperature – Quadrant [3]
Containment Pressure (Wide Range)	Core Exit Temperature – Quadrant [4]
Containment Isolation Valve Position	RCS Radiation (no instrumentation available for direct measurement)
Containment Area Radiation (High Range)	

With the exception of the RCS Radiation indication and Hydrogen Monitors, these instruments are included in the Technical Specification 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," in the ISTS based on the NRC's 1987 conclusion contained in Reference 7 these instruments may be important in limiting risk, based on a limited perspective of available PRA results. The results of the assessment contained in WCAP-15981 show that some of the Category 1 non-Type A variables are not important in limiting risk as indicated from a wide-reaching survey of WOG plant PRA results. Thus, WCAP-15981 recommends that only instrumentation that satisfies criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii) be included in the Technical Specifications.

A similar re-evaluation of the basis and classification of the hydrogen monitor PAM Function was performed in the rulemaking to revise 10 CFR 50.44, where the Commission determined that the hydrogen monitors no longer met the definition of Category 1 in Regulatory Guide 1.97. The Commission concluded that Category 3 as defined in Regulatory Guide 1.97 is an appropriate categorization for the hydrogen monitors because they are only required to diagnose the course of beyond design basis accidents.

The NRC approved the elimination of the Post Accident Sampling System (PASS) on June 14, 2000. The PASS requirements were based on the knowledge of severe accidents shortly after the accident at Three Mile Island Unit 2 in 1979. The justification for eliminating the PASS was based on a better understanding of severe accidents due to significant research and analysis after the requirements for the PASS were developed. The basis for the NRC approval for PASS elimination is contained in WCAP-14986-A (Reference 8). Eliminating the PASS was based on the accident progression as implemented in the Abnormal and Emergency Operating Procedures, Severe Accident Management Guidelines, Core Damage Assessment Guidelines, Emergency Plan, and Emergency Plan Implementing Procedures. The WCAP-15981 methodology utilizes a similar approach to evaluate the required PAM instruments based on these procedures.

The proposed changes in the PAM instrumentation Functions specified in Technical Specification 3/4.3.3.8, Table 3.3-11 have been evaluated and selected in accordance with the screening criteria contained in WCAP-15981. The screening criteria were used to identify the PAM instrumentation important to safety (i.e., monitor plant parameters that are the basis for important operator actions to bring the unit to a safe stable state in the event of an accident). The selected instrument Functions satisfy Criterion 3 and/or 4 of 10 CFR 50.36(c)(2)(ii), and include Regulatory Guide 1.97 monitoring instrumentation for parameters identified as important to safety in accordance with the methodology contained in WCAP-15981.

The PAM instrumentation selected in accordance with the methodology in WCAP-15981 provides the capability to monitor plant parameters necessary for safety significant operator actions so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures (EOPs) (these variables are restricted to preplanned actions for the primary success path of DBAs), e.g., loss of coolant accident (LOCA),
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function,
- Implement procedures or guidance that has been shown to have an important role in preventing core damage or early fission product releases,
- Determine the likelihood of a gross breach of the barriers that prevent radioactivity release,
- Determine if a gross breach of a barrier has occurred, and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

4.0 TECHNICAL ANALYSIS

The following discussions provide an evaluation of each of the proposed changes.

Change No. 1

Relocate the LCO, Applicability, Action Statements, Action (35) and Surveillance Requirement (4.3.3.1) associated with the containment area radiation monitor alarm function contained in Tables 3.3-6 and 4.3-3 (Instrument 1. b. ii., Unit 1) and Tables 3.3-6 and 4.3-3 (Instrument 1. b., Unit 2) from Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation," to the Licensing Requirements Manual (LRM). This change results in the Technical Specification 3/4.3.3.1 requirements that pertain to the containment area radiation monitors being relocated to the LRM.

Although the Technical Specification 3/4.3.3.1 requirements applicable to the containment area radiation monitors are relocated to the LRM to support the alarm Function of the containment area radiation monitors, certain requirements applicable to the PAM Function of these radiation monitors are also retained in Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation."

As Change 1 and Change 2 are similar (involving Technical Specification relocations) the technical analysis for Change 1 is presented below together with the technical analysis for Change 2.

<u>Change No. 2</u>

For Unit 2 only, the LCO, Applicability, Action Statements, Action (35) and Surveillance Requirement (4.3.3.1) associated with the main steam discharge radiation monitor contained in Tables 3.3-6 and 4.3-3 (Instrument 2. c. iii., Unit 2) in Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation," are relocated to the ODCM.

The following discussions provide information regarding the Unit 1 and 2 containment area radiation monitors and the Unit 2 main steam discharge radiation monitor proposed to be relocated to the LRM and ODCM respectively. In addition, the changes involved in the retention of the applicable requirements for the containment area radiation monitor indication Function in Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" is discussed.

The 10 CFR 50.36 (c)(2)(ii) Criteria used to evaluate Technical Specification requirements for retention are as follows:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Criterion 1 applies to instrumentation used to detect RCS leakage and is satisfied by the instrumentation included in the RCS Leakage Detection Instrumentation Specification. Criterion 2 applies to a process variable, design feature, or operating restriction that must be maintained within limits by a Technical Specification requirement to preserve an initial condition assumed in a design basis accident or transient analysis. The purpose of the PAM instrumentation is to function in a post accident environment to provide indications necessary for the operators to take manual actions to mitigate the consequences of an accident, or indications that have been determined to be risk significant. Therefore, only Criteria 3 and 4 of 10 CFR 50.36 (c)(2)(ii) are applicable when evaluating instruments for retention in the PAM Technical Specification.

The corresponding ISTS 3.3.3, "Post Accident Monitoring (PAM) Instrumentation" Bases discusses that all Regulatory Guide 1.97 Type A and all Regulatory Guide 1.97 Category 1 instrumentation must be included in the PAM Technical Specification. Therefore, in addition to evaluating each instrument against the 10 CFR 50.36 (c)(2)(ii) Criteria 3 and 4 above, each instrument is evaluated based on the methodology in WCAP-15981 to determine the applicable Regulatory Guide 1.97 Type and Category. Each PAM instrument proposed for relocation from the Technical Specifications has been determined not to fulfill the function of Regulatory Guide 1.97 Type A or Category 1 instrument. The evaluations performed in accordance with WCAP-15981 also support the conclusions that the affected instrumentation does not satisfy 10 CFR 50.36 (c)(2)(ii) Criterion 3 (i.e., it was not determined to be a Type A instrument) or Criterion 4 (i.e., it was not determined to be significant to risk).

The containment area radiation monitors provide continuous surveillance of radiation levels in containment, where personnel may be present and where significant radiation levels may occur. The alarms associated with the containment area radiation monitors provide warning of high radiation levels and/or abnormal conditions to operating personnel. Additionally, the containment area radiation monitors provide indications used to assess selected plant parameters following an accident consistent with the recommendations of NUREG-0737, "Clarification of

TMI Action Plan Requirements" October 1980. The evaluation of the containment area radiation monitors with respect to an accident (i.e., the PAM Function) is addressed in Section 4.7 of this LAR.

The Unit 2 main steam discharge radiation monitor analyzes effluent gases from the steam discharge piping. The monitor provides information to trend and control plant effluents to protect the health of plant personnel and to limit plant effluent releases to within the limits of 10 CFR 20. The alarms associated with the effluent monitors also warn plant personnel of abnormal releases. Additionally, the main steam discharge radiation monitors provide indications used to assess selected plant parameters following an accident consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements" October 1980. Although the main steam discharge radiation monitors may provide an indication of steam generator tube leakage, other Technical Specification requirements (i.e., RCS Operational Leakage) provide specific limits and Actions (including a unit shutdown within 4 hours of exceeding a limit) to assure that steam generator tube leakage is monitored and controlled.

The containment area radiation monitors and main steam discharge radiation monitors provide alarms and indications to alert plant personnel of high radiation conditions and to assist in evaluating and trending plant effluents. The TS Actions applicable if these monitors are inoperable require that the channel be restored to Operable status within 72 hours, or a preplanned alternate method of monitoring the parameter be initiated and the channel to be restored to Operable status within 30 days or that an explanation be provided in the next Annual Effluent Release Report why the channel was not restored to Operable status in a timely manner. The TS Actions do not impact or reference the operability of other systems or require a unit shutdown. Additionally, the alarm function of the Unit 1 and Unit 2 containment area radiation monitors and all functions of the Unit 2 main steam discharge radiation monitors proposed for relocation do not:

- Provide an automatic initiation function assumed in the safety analysis for any design basis accident described in Unit 1 UFSAR Chapter 14 or Unit 2 UFSAR Chapter 15.

- Provide indication or alarm functions relied on by operators to take manual actions that are assumed in the safety analyses for any design basis accident described in Unit 1 UFSAR Chapter 14 or Unit 2 UFSAR Chapter 15.

- Provide the primary indication that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary, or

- Monitor variables which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The following discussion evaluates the Unit 2 Main Steam Discharge Radiation indication with respect to its function as a PAM instrument. The Unit 2 Main Steam Discharge Radiation indication may be used for the diagnosis of a steam generator tube rupture accident, which prompts an operator action for which no automatic actuation is provided. However, with the low fuel rod leakage history of the current operating plants, secondary side radiation is not a reliable indicator of a steam generator tube rupture accident. The history of diagnosis and response to a steam generator tube rupture accident has typically been based on increased RCS inventory losses (e.g., decreasing pressurizer level and pressure) and increasing water level in the affected steam generator. This indication provides the most reliable diagnosis of a steam generator tube rupture accident to prompt the appropriate operator actions. As such, the Unit 2 Main Steam Discharge Radiation indication is not the primary indication relied on to diagnose or mitigate a steam generator tube rupture accident. The BVPS specific steam generator tube rupture accident analysis assumptions are discussed in more detail in Section 4.1, "Design Basis Accidents" under Change 10.

Therefore, based on the discussions above, the Unit 2 Main Steam Discharge Radiation indication does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in the PAM Technical Specification.

The Main Steam Discharge Radiation indication is currently classified as a Type A, Category 1 variable for Unit 2 based on its potential use in diagnosing a steam generator tube rupture. Based on the methodology in WCAP-15981, the Unit 2 Main Steam Discharge Radiation indication is determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type C and Category 3 variable (i.e., the same classification currently assigned to the corresponding Unit 1 radiation monitor). Consistent with the guidance of Regulatory Guide 1.97 the Main Steam Discharge Radiation indication should be Type C, since it provides information to indicate the breach of the steam generator tube fission product barrier and Category 3 because it provides backup diagnostic indications for a steam generator tube rupture accident.

Based on the above discussions, the alarm function of the Unit 1 and Unit 2 containment area radiation monitors and all functions of the Unit 2 main steam discharge radiation monitors do not satisfy any of the 10 CFR 50.36(c)(2)(ii) criteria for retention in the Technical Specifications. Therefore, the proposed change to relocate the Technical Specification requirements for these radiation monitors is acceptable.

The requirements are being relocated to the LRM or ODCM, respectively. Therefore, changes to the relocated material will be controlled in the same manner as changes to the UFSAR, i.e., in accordance with 10CFR 50.59. As such, the relocation of Technical Specification requirements to the LRM and ODCM is acceptable, as changes to these documents will be adequately controlled by 10 CFR 50.59. The provisions of 10 CFR 50.59 establish adequate controls over requirements removed from the TS, and assure future changes to these requirements will continue to be consistent with safe plant operation.

The Unit 1 and 2 containment area radiation monitor (alarm function) and Unit 2 main steam discharge radiation monitors and all associated Technical Specification 3/4.3.3 requirements (i.e., LCO, Actions, and Surveillances) will be relocated together to form a complete radiation monitor requirement in the LRM and ODCM respectively. The containment area radiation monitors will be relocated to the LRM and the Unit 2 main steam discharge radiation monitors will be relocated to the ODCM. The main steam discharge radiation monitors are being relocated to the ODCM instead of the LRM to be consistent with the location selected for effluent radiation monitors previously removed from the TS in accordance with Generic Letter 89-01. In addition, the relocation of the Unit 2 main steam discharge radiation monitor requirements to the ODCM is consistent with the current location of the corresponding Unit 1 radiation monitor requirements.

The evaluation of the containment area radiation monitors with respect to a Post Accident Monitoring indication function is addressed in Section 4.7 of this LAR. The evaluation results in the retention of certain Technical Specification requirements for the PAM indication Function of the containment area radiation monitors.

The containment area radiation monitor requirements retained in proposed Technical Specification 3/4.3.3.8, "Post Accident Monitoring Instrumentation," include the requirement for two operable channels which is the same as the minimum channels operable requirement in Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation." However, the Mode of applicability for these radiation monitors is revised from Modes 1, 2, 3, and 4 in Technical Specification 3/4.3.3.1, to Modes 1, 2, and 3 in Technical Specification 3/4.3.3.8. The Bases for the PAM Function Mode of applicability is related to the Design Basis Accidents associated with the PAM Functions. In Mode 4, the probability and consequences of potential accidents are reduced such that the PAM requirements of Technical Specification 3/4.3.3.8 are no longer necessary to assure the safe unit operation. Therefore, based on the PAM Function requirements, the proposed change to revise the containment area radiation monitors Mode of applicability in the Technical Specifications is acceptable. It should be noted that the Mode 4 applicability for these monitors will continue to be specified in the LRM in support of the alarm function provided by these monitors.

In addition to the revision in the Mode of applicability described above, the monthly Channel Functional Test currently required by Technical Specification 3/4.3.3.1 for the containment area radiation monitors is not part of the surveillance requirements specified for the PAM Functions in Technical Specification 3/4.3.3.8. Channel Functional Tests are intended to confirm the required actuation, alarm, or interlock function of an instrument channel. PAM Functions are retained in the Technical Specifications based on their ability to indicate certain plant parameters important to the safe operation of the plant in a post accident environment. PAM Functions are not retained in the Technical Specifications for the actuation, alarm, or interlock Functions that these instruments may also provide. For the containment area radiation monitors, the Channel Functional Test requirement is being relocated to the LRM in support of the alarm function provided by these monitors (which is being relocated to the LRM). Based on the retention of the containment area radiation monitors in the Technical Specifications for their PAM indication Function and not their alarm function, the monthly Channel Functional Test requirement is no longer necessary to confirm the indication capability associated with these monitors. Therefore, the proposed change is acceptable, since the Channel Check and Channel Calibration surveillance requirements specified for the PAM Functions are sufficient to ensure the continued operability of the containment area radiation monitor indications.

The Channel Check currently required by Technical Specification 3/4.3.3.1 for the containment area radiation monitors is required to be performed once per shift or once per 12 hours. The Channel Check specified for the PAM Functions in Technical Specification 3/4.3.3.8 is required to be performed on a monthly basis. Therefore, the proposed change to move the containment area radiation monitor indication function to the PAM Technical Specification revises the applicable Channel Check surveillance from a 12-hour interval to a monthly interval. The proposed change is acceptable, since based on the operating experience for the other PAM indication Functions, the monthly interval for the Channel Check continues to provide adequate assurance that the required indications are maintained operable. The proposed change results in the ISTS requirements applicable to the PAM instrumentation being applied to the containment area radiation monitors. In addition, the proposed change is acceptable because the 12-hour Channel Checks required by Technical Specification 3/4.3.3.1 for the containment area radiation monitors will continue to be specified in the LRM requirement associated with the relocated containment area radiation monitor alarm function. Therefore, the proposed change provides a more appropriate surveillance interval based on the reason that the instrumentation is retained in the Technical Specifications (i.e., the PAM Function it serves). As the proposed change continues to provide adequate assurance that the required indications are maintained operable, the change does not adversely affect the safe operation of the plant.

Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation," Action #35 applies to less than the minimum channels inoperable (i.e., 1 or 2 channels inoperable) and requires that Action be taken to restore a channel to operable status within 72 hours or initiate an alternate method of monitoring and restore the channel to operable status within 30 days or explain in the next annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

The proposed Action of Technical Specification 3/4.3.3.8, "Post Accident Monitoring Instrumentation," requires that with a single inoperable channel, the channel be restored to operable status within 30 days. Failure to restore to operable status results in a report being made to the NRC within the following 14 days detailing the alternate method being utilized to monitor the affected PAM channel and the plans and schedule for restoring the inoperable channel to operable status. With two or more inoperable channels, the proposed Actions of Technical Specification 3/4.3.3.8 require that a required channel be restored within 7 days. Failure to restore at least one channel to operable status results in a report being made to the NRC within the following 7 days detailing the alternate method being utilized to monitor the affected PAM Function and the plans and schedule for restoring the inoperable Function to operable status.

Although current Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation," Action # 35 requires that action be taken within 72 hours, while the corresponding PAM Action a would allow 30 days, the basis for the 72 hour Action is not the same as the Actions required for inoperable PAM Functions. Technical Specification 3/4.3.3.1, as stated in the Bases, addresses the more routine plant need to continuously monitor radiation levels in various areas of the plant. Maintaining the capability to continuously monitor radiation levels in the plant is important for personnel safety and diagnosing equipment and system problems. The 72-hour Action provides assurance that this routine capability is maintained. Additionally, the more routine nature of these Technical Specification monitoring requirements is confirmed in that the monitoring Function could be lost for up to one year, before the Actions require a report be sent to the NRC. Further, the reporting of the inoperable Function only has to be included in the routine Annual Radioactive Effluent Release Report. However, the Technical Specification requirements for routine monitoring functions do not satisfy the criteria for retention in the Technical Specifications. Action #35 is being relocated to the LRM with the other requirements for the containment area radiation monitors. The LRM requirements will continue to provide adequate assurance that the radiation levels around the plant continue to be routinely monitored.

The proposed change, which would allow up to 30 days for one inoperable channel and 7 days for two or more inoperable channels (instead of the 72 hours allowed by Technical Specification 3/4.3.3.1) to take initial action is acceptable because the basis for retaining the affected monitors in the Technical Specifications is revised to be consistent with the PAM Function of the monitors. Requirements for PAM instruments are retained in the Technical Specifications in order to assure the required indication function is available in the unlikely event of a design basis accident. Due to the low probability of a design basis accident, the Action completion times allowed in the PAM Technical Specification are longer than those allowed in Technical Specification 3/4.3.3.1 for routine radiation monitoring functions. In addition, the proposed change is acceptable because the extended Action times allowed by the PAM Technical Specification require more restrictive NRC reporting requirements. The proposed change includes the application of all of the PAM Technical Specification Actions to the containment area radiation monitors. Therefore, in addition to the extended Action times, the PAM Actions also require that a special report be submitted to the NRC within 14 days for one inoperable channel and within 7 days for two or more inoperable channels not restored to operable status. The current Technical Specifications would allow up to a year to submit a similar report (i.e., documenting inoperable channels) as part of the Annual Radioactive Effluent Release Report. The reporting requirement must detail the alternate method(s) being used and the plan and schedule for restoring the inoperable PAM instrumentation to operable status. Thus, the proposed change includes more prompt reporting requirements than those specified in the current Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation," Action #35. Therefore, the proposed change includes more stringent reporting requirements to inform the NRC of the status of any inoperable PAM functions that are not restored to operable status within the required Action time.

In addition to the changes described above, Technical Specification 3/4.3.3.1 Tables 3.3-6 and 4.3-3 specify the equipment identification numbers for the containment area radiation monitors. The requirements retained in the PAM Technical Specification for these radiation monitors do not specify equipment identification numbers. This changes the Technical Specification requirements by moving the equipment identification numbers for these instruments to the Bases of the PAM Technical Specification.

The removal of the equipment identification numbers from the Technical Specifications is acceptable because this type of information is not necessary to be included in the Technical Specifications to provide adequate protection of public health and safety. The PAM Technical Specification retains the requirement for the instruments to be operable in the specified Modes to assure that they can perform their required safety function. In addition, this change is acceptable because the relocated information will be retained within the Technical Specification Bases and changes to the Bases are controlled by the Bases Control Program specified in the Administrative Controls Section of the Technical Specifications. This program requires the evaluation of Bases changes to ensure that the Bases are properly controlled and that prior NRC review and approval is requested when required.

Change No. 3

Revise the title of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," to "Post Accident Monitoring (PAM) Instrumentation." The LCO is revised from referencing "channels" to referencing "each Function," since Table 3.3-11 and the associated Actions are presented on a Function basis.

The revision to the applicable Index page is required in order to accurately reflect the proposed title change to Technical Specification 3/4.3.3.8. The revision to the title of Technical Specification 3/4.3.3.8 and associated index pages are editorial changes made to be consistent with the title of the corresponding ISTS Technical Specification and to better describe the actual functional requirements of the specification (i.e., "post" accident monitoring). The change from referencing "channels" to referencing "Functions" is also consistent with the presentation contained in the corresponding ISTS, and supports the changes to the Actions and Surveillance Requirements that are written on a Function basis.

The proposed changes are acceptable because they are administrative in nature and serve to better integrate the LCO, Action, and specific instrumentation requirements consistent with the terminology used in the corresponding ISTS requirements. In addition, the proposed changes only involve the format and presentation of the Technical Specification requirements and do not result in any technical changes.

Change No. 4

The addition of a footnote to the Applicability of the Power Range Neutron Flux PAM Function that states the function is not required in Mode 3.

The PAM Technical Specification is applicable in Modes 1, 2, and 3. The BVPS Unit 1 and 2 PRA shows that power range neutron flux is a key indication for accident management operator actions to initiate a manual reactor trip to bring the reactor to a subcritical condition. This is consistent with the keff of $\geq$ 0.99 specified for Mode 2 and for power operation in Mode 1. Subsequent operator actions (in Mode 3 after a reactor trip) to assure that the reactor remains in a subcritical state, where the power range neutron flux monitor may be inoperable, such as during RCS depressurization, were not determined to be important for long term core cooling. Therefore, for the required post accident monitoring indication function (i.e., confirming reactor trip from Modes 1 and 2) the Power Range Neutron Flux monitor is only required to be operable in Modes 1 and 2. The proposed change addresses this Mode of Applicability by the addition of a Note that excludes the Mode 3 applicability requirement for the Power Range Neutron Flux monitors. In addition, the proposed change makes the PAM Technical Specification requirements for the Power Range Neutron Flux monitors more consistent with the corresponding Mode requirements for this instrumentation in the Reactor Trip System Technical Specification.

The proposed change is part of the larger more restrictive change that adds the power range neutron flux indication to the Unit 1 and Unit 2 PAM Instrumentation Technical Specifications. The addition of the power range neutron flux indication to the PAM Technical Specification is consistent with the results of evaluations performed in accordance with WCAP-15981 and is discussed in Section 4.7 of this LAR. The proposed change (which limits the applicability of the power range neutron flux indication to Modes 1 and 2) is acceptable because it is necessary to properly define the applicable Modes in which the power range neutron flux indication is required to perform its PAM function and to exclude Mode 3 where the power range instrumentation may not be operable and is not required for post accident monitoring.

<u>Change No. 5</u>

A new General Note that states, "Separate Action statement entry is allowed for each Function." is added to the Actions of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation."

The addition of the General Note clarifies that the Actions may be entered on a Function basis and is consistent with the change to reference "each Function," in the LCO. The change is consistent with the presentation of the corresponding Note contained in the ISTS, and supports the changes to the LCO and Surveillance Requirements that are written on a Function basis.

The addition of the General Note is acceptable because the existing Technical Specification Actions do not preclude applying the Actions to multiple Functions specified on PAM Technical Specification Table 3.3-11. The current Actions can be applied separately for each inoperable PAM Function listed on Table 3.3-11. The Actions provide appropriate compensatory measures for each inoperable PAM Function that may be applied individually to each Function. Compliance with the Actions for each inoperable Function provides assurance that the plant continues to be operated in a safe manner or requires the plant to be placed in a Mode where the PAM function is no longer required to be operable. Therefore, the addition of the Note is not a technical change and is considered a clarification that improves consistency with the corresponding ISTS requirements. As such, the proposed change is considered administrative in nature and does not adversely affect the safe operation of the plant.

Change No. 6

Revisions to Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" Actions.

Current Technical Specification 3/4.3.3.8 Action a. is revised from being based on "less than the Total Number of Channels," to be based on a "Function with one required channel inoperable". The Action is revised to refer to the "required" channel instead of an "inoperable" channel. The proposed changes are consistent with the standard presentation of instrumentation Action requirements throughout the ISTS. As such, the change makes the Technical Specification Actions more consistent with the corresponding ISTS Actions, and supports the changes to the LCO and Surveillance Requirements that are written on a Function basis with the number of "Required Channels" identified for each function.

The proposed changes are acceptable because the Actions continue to provide the appropriate remedial measures to assure that the redundancy of indication is maintained (i.e., the affected indication is restored or the use of alternate indication(s) is implemented) or the plant is placed in a Mode where the PAM Functions are no longer required to be operable.

The restoration Completion Time of Action a. is revised from 7 days to 30 days (in proposed Action a.1). This is a less restrictive change. The Completion Time is consistent with the ISTS and is acceptable in cases where there are more than one channel because it considers that there is a remaining channel available to provide the monitoring Function, and the fact that the Function does not provide any automatic protective action. For the Functions where only one channel is provided, the Completion Time is acceptable because it considers that other control room indications will be available to accomplish the Function. In any case, the Completion Time extension is justified based on the low likelihood of an event occurring within the allowed Completion Time that would require the affected PAM Function to be operable. If the channel can not be restored to Operable status within 30 days, proposed Action a.2 requires that an alternate method of monitoring the Function be identified and a report submitted to the NRC in 14 days detailing the plans and schedule for using the alternate indication method. This change is acceptable, as it takes into account that there are alternate Functions available to monitor the inoperable channel prior to the loss of functional capability, and the likelihood that this information would be necessary during the time allowed by the Action. Section 4.8 of this LAR discusses the alternate Functions available to monitor inoperable PAM Functions.

In addition, the Action is retained to bring the unit to a Mode where the requirements of the LCO do not apply if the other Actions are not or can not be met. Proposed shutdown Action a.3 is not consistent with the corresponding ISTS Actions for a single inoperable channel but is retained in lieu of the ISTS Actions that would default to Specification 3.0.3. The retention of this Action is acceptable because it provides clear guidance as to the applicable course of action if the other requirements of Action a.1 and a.2 discussed above are not met.

Current Technical Specification 3/4.3.3.8 Action b. is revised from being based on less than the "Minimum Channels Operable," to be based on a "Function with two or more required channels inoperable" in proposed Action b.1. The change results in a presentation that is more consistent with the corresponding ISTS presentation of these requirements, and supports the changes to the LCO and Surveillance Requirements that are written on a Function basis with the number of "Required Channels" identified for each Function. The Action is also revised to refer to restoring "required" channels instead of restoring "inoperable" channels. The proposed changes are consistent with the ISTS presentation of instrumentation Action requirements throughout the Technical Specifications. The proposed Action b.1 is slightly different than the corresponding ISTS PAM Action in that it addresses two or more required channels inoperable" instead of simply "two required channels inoperable." This difference from the ISTS is made to accommodate the BVPS PAM Functions that have three required channels. The ISTS Actions do not address the design of PAM Functions with three channels, and the potential for three inoperable channels.

The proposed changes are acceptable because the Actions continue to provide the appropriate remedial measures to assure that the PAM Function is maintained (i.e., the affected indication is restored or the use of an alternate indication(s) is implemented) or the plant is placed in a Mode where the PAM Functions are no longer required to be operable.

The completion time of Action b. is revised from 48 hours to 7 days in proposed Action b.1. This is a less restrictive change. The completion time is consistent with the corresponding ISTS completion time and is acceptable because it considers the relatively low likelihood of an event that would require the affected PAM Function, the availability of an alternate means to obtain the information, and the fact that the Function is passive and does not provide any automatic protective actions.

If the affected channel(s) can not be restored to operable status within 7 days, proposed Action b.2 requires an alternate method of monitoring the Function to be identified and a report submitted to the NRC in the following 7 days detailing the plans and schedule for using the alternate indication method. Proposed Action b.2 is not consistent with the corresponding ISTS PAM Actions for all Functions. However, similar to proposed Action a.2, the proposed alternative of Action b.2 is acceptable because it takes into account the passive nature of the PAM Function and that there are alternate Functions available to accomplish the required PAM Function, and the low likelihood that the PAM Function would be required during the time that an alternate method of monitoring the PAM Function is employed. The proposed Completion Time for new Action b.2 is half the time allowed in corresponding Action a.2 and the ISTS Actions for implementing alternate PAM methods and reporting the condition to the NRC. If an acceptable alternate method of monitoring the PAM Function is not available, proposed Action b.3 continues to require that the plant be placed in a Mode where the PAM Function is no longer required to be operable.

The allowance to rely on an alternate PAM Function is also acceptable because in most cases, the alternate Functions identified for a given PAM Function are Regulatory Guide 1.97 qualified or associated with a safety related system with a reliable power supply (e.g., pump or header discharge pressure or motor amperes) and are also displayed in the control room. Additionally, the alternate indications that may not be Regulatory Guide 1.97 qualified (e.g., rod bottom lights, rod position indicators, pump/header pressures, and motor amperage) may also be utilized in the EOPs to confirm protective system actuation. Section 4.8 of this LAR discusses the specific alternate instrumentation available to perform certain PAM Functions.

Proposed Action b.2 also requires that a report be made to the NRC that outlines the alternate method being used to accomplish the PAM function and the duration that the alternate method will be employed, as well as the cause of the inoperability, and the plans and schedule for restoring the affected channel(s) to operable status. Therefore, the use of an alternate method to accomplish a required PAM Function and the duration that alternate method may be employed and the plans and schedule for restoring the affected channel(s) to operable status are subject to NRC review. This reporting requirement provides the NRC with the opportunity to judge the acceptability of the proposed alternate method(s) and, if necessary, question the continued use of the proposed methods.

The allowance for continued unit operation with an alternate PAM indication based on submitting a report to the NRC was previously approved by the NRC in the ISTS PAM Technical Specification for the containment area radiation monitors and the reactor vessel water level indication system. In addition, the current Technical Specification requirements also provide the allowance for continued unit operation with both the containment area radiation monitors inoperable. As such, an established precedence exists for allowing the option to use an alternate indication (if available and acceptable) in lieu of requiring a unit shutdown for the loss of indication instrumentation that provides no protective actuation functions and that only would be required in the unlikely event of a Design Basis Accident. Based on the above discussions, the proposed change continues to assure that the unit is operated in a safe manner (i.e., a forced unit shutdown would not always be required for the loss of instrumentation that only provides an indication function, and for which there are acceptable alternative monitoring methods). The additional safety benefit provided by the option to use an alternate PAM Function in lieu of a unit shutdown is based on the fact that additional risk of a plant shutdown/restart transient is avoided. In addition, the proposed change (via the required NRC report) continues to assure that adequate regulatory control is maintained when alternate PAM methods are employed.

The current Unit 2 Technical Specification 3/4.3.3.8 Action c pertains to the Reactor Coolant System Subcooling Margin Monitor. This Unit 2 Action will be relocated to the LRM along with the PAM requirement for the Reactor Coolant System Subcooling Margin Monitor. The acceptability of relocating this Unit 2 Action is based on the justification for the relocation of the Reactor Coolant System Subcooling Margin Monitor PAM Function from Unit 2 Technical Specification 3/4.3.3.8. The technical basis for relocating the Reactor Coolant System Subcooling Margin Monitor PAM Function from Unit 2 Technical Specification 3/4.3.3.8 is discussed in Section 4.9 of this LAR.

Change No. 7

Revisions to the Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," Surveillance Requirements.

Surveillance Requirement 4.3.3.8 of Unit 1 and Unit 2 Technical Specification 3/4.3.3.8 is revised by the addition of a General Note that specifies which surveillance requirements are applicable to each PAM instrument Function specified in Table 3.3-11. The adoption of this ISTS type of note, establishes the requirement that the specified surveillances must be performed for the associated PAM Functions. The inclusion of this Note serves to conform more closely with the corresponding ISTS PAM requirements and eliminates the need for a separate Technical Specification Table (4.3-7) to specify the required surveillances for each PAM Function. As such, the addition of the General Note revises the format and presentation of the surveillance requirements to be more consistent with the ISTS (i.e., the need for a separate surveillance Table is eliminated). The proposed change is acceptable because replacing the Table with this new General Note is administrative in nature, and does not introduce a technical change to the surveillance requirements.

Current surveillance requirement 4.3.3.8 specifies that both a Channel Check and a Channel Calibration be performed on the required PAM Functions as specified by

Table 4.3-7. The current surveillance is revised by separating the Channel Check and Channel Calibration into individual surveillance requirements (4.3.3.8.1 and 4.3.3.8.2). The proposed change is acceptable, since the two requirements are distinctly different surveillances that are addressed by separate procedural requirements. The proposed change is acceptable because it is administrative in nature and only reformats the current requirements to be more consistent with the corresponding ISTS surveillance requirements. In addition, the proposed change does not introduce a technical change to the current surveillance requirements.

The proposed PAM Technical Specification Channel Check surveillance does not include the provision from the ISTS that only requires the Channel Check to be performed on normally energized channels. This ISTS provision is only needed if the proposed Technical Specification includes PAM instrumentation that is normally de-energized. The performance of a Channel Check on de-energized instrumentation would not provide a very useful indication of channel operability. As all the BVPS specific instrumentation proposed for inclusion in the PAM Technical Specification is normally energized, the ISTS provision for performing the surveillance only on normally energized channels is unnecessary.

As part of the larger and more restrictive change that adds the Penetration Flow Path Containment Isolation Valve Position Indication to the PAM Functions required to be operable by Technical Specification 3/4.3.3.8, a new surveillance is added to the PAM Technical Specification. Proposed surveillance 4.3.3.8.3 specifies that a Channel Functional Test be performed every 18 months. The General Note to the surveillance requirements (added by this LAR) describes that this surveillance is only applicable to the Penetration Flow Path Containment Isolation Valve Position PAM Function. The proposed new surveillance is necessary because the existing PAM Channel Calibration surveillance (which pertains to the adjustment of sensors etc. in each channel) is not the most appropriate surveillance to be performed on the Penetration Flow Path Containment Isolation Valve Position channels. The Channel Calibration surveillance is intended for channels with a sensing device, associated setpoint, and electronic signal processing. As such, the Channel Calibration surveillance is not directly applicable to a valve position indication circuit comprised, in most cases, of simply a limit switch and indicating light. A Channel Functional Test is a more appropriate verification of a valve position indication circuit. The Channel Functional Test is adequate to verify the operation of the required valve position indication circuits and assure that these required PAM indications are maintained operable and available when needed. The proposed change is acceptable because it specifies a more appropriate surveillance to ensure the continued operability of the required containment isolation valve position indication.

As part of the larger and more restrictive change that adds the Power Range Neutron Flux indication to the PAM Functions required operable by Technical Specification 3/4.3.3.8, a ** footnote is added to the Channel Calibration surveillance requirement that excludes the neutron detectors from the Channel Calibration. The proposed change adds the standard exclusion for neutron detectors that is applied throughout the Technical Specifications when a Channel Calibration is applied to channels containing neutron detectors. The addition of such notes to the Channel Calibration surveillance requirements is acceptable as the neutron detectors are not adjustable as are other sensors associated with more typical instrumentation channels (e.g., channels containing pressure sensors). The adjustment of the channel sensor is required by the definition of the Channel Calibration surveillance. Therefore, the exclusion of neutron detectors is a necessary clarification in order to comply with the requirement to perform a Channel Calibration as defined in the Technical Specifications. The proposed change is consistent with the corresponding ISTS surveillance requirements.

Change No. 8

Table 3.3-11 of Specification 3/4.3.3.8 lists the applicable PAM Functions, the number of instrument channels required, and in the case of Unit 2 the applicable Actions. The following changes are made to the Accident Monitoring Instrumentation requirements on Table 3.3-11 of Specification 3/4.3.3.8.

Changes are made to the title of the Technical Specification and to the Table 3.3-11 headings. The changes are made to conform to the corresponding ISTS table headings and Technical Specification title. The changes revise the title of the Technical Specification to introduce the term "Post" to the existing Accident Monitoring title and revise the heading for the list of instrumentation on Table 3.3-11 to Function, rather than instrument (Unit 2) or no heading (Unit 1). The proposed changes are acceptable because they do not affect the technical requirements of the Technical Specification and are made to conform more closely to the corresponding ISTS terminology.

The Table 3.3-11 heading for "Total No. of Channels" is revised to "Required Channels" and the "Minimum Channel Operable" heading is deleted. The proposed change in Table headings is consistent with the corresponding ISTS PAM Table headings. The intent of the specified channels in Table 3.3-11 is to ensure that redundant channels be maintained operable where possible (not all Functions are designed with redundant channels) and provide appropriate Actions when that redundancy is not maintained. The proposed change is acceptable because it continues to require that redundant channels be maintained operable (where plant design permits) and specifies the appropriate Actions when the required redundancy is not maintained. Although, the presentation of the Table (and associated Actions) are revised to conform more closely to the corresponding ISTS requirements (i.e., based on the required number of channels) the revised Table accomplishes the same function with the single "Required Channels" heading as the current Table 3.3-11 accomplishes with the multiple Table headings of Total and Minimum channels. As such, the proposed change is administrative in nature and does not introduce a technical change to the current requirements.

The Total Number of channels for the Pressurizer Water Level PAM Function on Table 3.3-11 is currently specified as 3. The revised Table 3.3-11 "Required Channels" specified for the Pressurizer Water Level PAM Function is 2. This change reduces the number of Pressurizer Water Level channels required to be operable on Table 3.3-11. The basis of the PAM requirements for multiple channel Functions is that two channels are maintained operable to assure at least one channel remains operable to accomplish the required PAM Function. As such, the proposed change is acceptable because it continues to require that two channels are maintained operable such that in the event of a single failure, one channel will still be operable to provide the required PAM Function. The current requirement for 3 channels is overly conservative for this purpose and is not necessary to assure the availability of the Function in the event of a single failure. Therefore, the proposed change continues to provide adequate assurance that the PAM Function will be available, when required, consistent with similar two channel requirements for other PAM Functions.

The number of channels specified on Table 3.3-11 for the Unit 1 and Unit 2 Auxiliary Feedwater Flow (AFW) Rate indication is revised to more accurately reflect the different unit designs and to support the change to present a single Table heading of "Required Channels".

The Unit 1 control room design only includes a single AFW flow rate indication for each steam generator. Therefore, in order to ensure adequate indication capability is maintained the Unit 1 "Required Channels" are specified as 3 (1 per steam generator). This presentation supports the application of the Technical Specification Actions that provide for one and two inoperable channels to the Unit 1 design of 1 channel per steam generator. The Technical Specification Actions will continue to assure that the redundancy provided by the multiple steam generator AFW flow rate indication channels is maintained, or that appropriate remedial measures are taken.

The Unit 2 control room design includes two channels of AFW flow rate indication per steam generator. Unit 2 not only has the redundancy provided by multiple steam generators (as does Unit 1) but Unit 2 has redundant indication on each steam generator. In order to more accurately show the additional redundancy provided by the Unit 2 design, the Unit 2 AFW Flow Rate Function presentation on Table 3.3-11 is separated into individual Functions for each steam generator consisting of two channels per steam generator. This presentation supports the application of the Technical Specification Actions that provide for one and two inoperable channels and allows these Actions to be applied on a per steam generator basis. The proposed Technical Specification Actions will continue to assure that the redundancy provided by the AFW flow rate indication channels is maintained, or that appropriate remedial measures are taken.

The proposed changes to the method of specifying the required channels of Unit 1 and Unit 2 AFW flow indication on Table 3.3-11 (described above) is less restrictive than in the current Table 3.3-11, but more accurately reflects the individual Unit designs, and better supports the "Required Channel" presentation of the revised Table and the application of the associated Actions. The proposed changes are acceptable, because they continue to require the redundancy available in each Unit's design to be fully operable and provide appropriate Actions to assure redundancy is maintained or appropriate remedial measures are taken to assure the safe operation of the unit.

The Unit 2 Table 3.3-11 is further revised by the elimination of the Action column from the Table. The Unit 2 Action column provides guidance as to which PAM Action(s) are applicable to each PAM Function. Due to other changes made in this LAR (to the contents of Table 3.3-11 and the applicable PAM Actions) the Unit 2 Action column on Table 3.3-11 is no longer necessary. In the PAM Technical Specification, as revised by this LAR, all Actions are applicable to all PAM Functions listed on the Table. Therefore, the elimination of the guidance provided by the Action column in the Unit 2 Table is administrative in nature and supports other changes in this LAR which change the Table content and simplify the application of the associated Actions. The proposed change is acceptable as it simplifies the presentation of Table 3.3-11 and does not (by itself) introduce a technical change to the Technical Specification.

The number of required channels specified on Table 3.3-11 for the Core Exit Thermocouples (CETs) is revised from 4 per core quadrant to simply 2 channels.

The risk importance of the core exit thermocouples is derived from the operator actions to respond to inadequate core cooling conditions in the PRA and from the Emergency Plan notifications of plant conditions that may influence offsite emergency radiological protective actions. Inadequate core cooling for BVPS Units 1 and 2 is identified when the core exit thermocouples indicate greater than 1200°F. The core heatup assessment in WCAP-14696-A (pages 5-1 through 5-7) shows that there is a radial temperature gradient in the core during core heatup due to inadequate core cooling. For the purpose of the timely diagnosis of an inadequate core cooling condition, the "central" core exit thermocouple locations provide the most timely indication. The "central" core exit thermocouple locations are defined as all core locations excluding core exit thermocouples in the outermost three rows of fuel assemblies on each side of the core.

The proposed change includes an addition to the PAM Technical Specification Bases that describes the acceptable core locations for the two required CET channels consistent with WCAP-14696-A. The inclusion of this CET location requirement in the Bases is acceptable because the Bases Control Program in the Administrative Control Section of the Technical Specifications provides assurance that changes to the Bases will be evaluated and prior NRC review and approval requested when necessary.

This above information was not available when the ISTS PAM Technical Specification was issued with NUREG-1431, Revision 0, in 1992. The required number of CETs channels is therefore revised based on the information contained in WCAP-14696-A, which was approved by the NRC in September 1999.

A footnote is added to Table 3.3-11 that is applicable to the "Required Channels" of the CET Function listed on the Table. The new footnote defines a channel as consisting of two CETs and is consistent with the corresponding ISTS definition of a CET PAM channel. The proposed change results in an administrative change that clarifies the channel requirement for CETs consistent with the ISTS. The proposed change is acceptable because it conforms to the ISTS definition of a CET channel and provides additional assurance that the required temperature indication provided by the CETs will be available when required to assess the adequacy of core cooling. As such, the proposed requirement for operable CETs does not adversely affect the safe operation of the plant.

As part of the more restrictive change that adds the Penetration Flow Path Containment Isolation Valve (CIV) Position Indication to the PAM Functions listed on Table 3.3-11, two footnotes are added to Table 3.3-11 to provide the necessary clarifications to the requirement of two CIV positions per penetration flow path. The additional Notes clarify that the CIV Position Indication is not required for penetrations that are isolated and that only one indication channel is required for penetrations with only one CIV position indication in the control room. The addition of these Notes is consistent with the corresponding ISTS requirements for the CIV Position Indication PAM Function.

The Note regarding isolated penetrations is acceptable because the primary purpose of the indication function is to confirm containment isolation. Therefore, if a penetration is already isolated, the CIV Position Indication PAM Function is no longer required to confirm a successful actuation of the affected isolation valve. The other Note is also acceptable because it recognizes that certain penetrations may only be designed with a single CIV indication in the control room. Penetrations that utilize check valves or closed systems inside containment or a single isolation valve outside of containment do not have two CIV position indicators in the control room. Therefore, this Note is necessary for the Technical Specification requirements to account for the design of those penetrations that only include a single CIV position indication.

Change No. 9

Table 4.3-7 contains the current surveillance requirements associated with each PAM Function required operable in Specification 3/4.3.3.8. The following changes are made to the surveillance requirement Table 4.3-7 of PAM Specification 3/4.3.3.8.

Table 4.3-7 currently exempts the Containment Wide Range Pressure PAM Function from a monthly Channel Check with the annotation of N/A. The CTS exception for the performance of a Channel Check is due to the wide indication range of the instruments (0-200 psia for Unit 1 and 0-180 psia for Unit 2) and the expected normal operating indication being at the very low end of the indicating range. With normal operating pressure in containment, accurate pressure readings from the wide range instruments would be difficult. Therefore, the CTS did not require the performance of Channel Checks for this instrumentation. However, the CTS requirements result in no formal (Technical Specification) operability verifications of the wide range instrumentation between the required 18 month Channel Calibrations.

The exemption provided by the current Technical Specification is deleted consistent with the ISTS. The proposed change is more restrictive in nature as it makes the performance of a Channel Check applicable to the required containment pressure indicators on a monthly basis consistent with the other PAM Functions. Although the containment pressure indicated on the wide range instrumentation may not be read with a high degree of accuracy, the performance of a Channel Check of the indicators may still detect a potential instrument failure between 18 month Channel Calibrations. As discussed in the ISTS Bases for a Channel Check, the Channel Check ensures a gross instrumentation failure has not occurred between Channel Calibrations. Since the Channel Check is a qualitative assessment (as defined in the Technical Specifications) and is not intended to be a precision calibration, it can be performed on instruments when the measured parameter normally indicates in the extreme low end of the indicating range (i.e., with less than optimum accuracy). For example, a containment wide range pressure indicator should not be indicating 100 psia when atmospheric pressure is known to exist in containment and the other containment pressure indicators (both wide and narrow) indicate at or near atmospheric pressure. In this case, the performance of the Channel Check surveillance functions, as intended, to detect a gross failure of a pressure indicator. As such, the proposed change is acceptable because it provides an additional safety benefit by requiring formal operability verifications of the containment wide range pressure indicators that could detect a gross instrument failure between Channel Calibrations consistent with the purpose of a Channel Check.

The Channel Check requirement on current Technical Specification Table 4.3-7 for the AFW Flow Rate PAM Function specifies that the surveillance is only required during startup (S/U). The Channel Check requirement is further modified by an asterisk footnote which specifies that the Channel Check need only be performed in conjunction with Surveillance Requirement 4.7.1.2.7 following an extended outage. Surveillance 4.7.1.2.7 is contained in AFW Technical Specification 3/4.7.1.2. The surveillance requires that AFW flow to the steam generators be verified after an extended shutdown (i.e., shutdown in Modes 5 or 6 for greater than 30 days). The CTS requirement is intended to specify the performance of a Channel Check only when the AFW system is in service (i.e., in conjunction with Surveillance Requirement 4.7.1.2.7 following an extended outage). The CTS requirement could result in a Channel Check only being performed once every 18 months after the normal extended outage due to refueling.

The Channel Check surveillance requirement for the AFW Flow Rate PAM Function is revised to be required every month without any footnote exception consistent with all the other PAM Functions in Technical Specification 3/4.3.3.8. The proposed change is more restrictive in nature as it requires more frequent Channel Checks to be performed on the AFW Flow Rate indicators even when the AFW system may not be in operation. Although the AFW system may not normally be supplying water to the steam generators, the performance of a Channel Check of the indicators may still detect a potential instrument failure between the 18-month Channel Calibrations. As discussed in the ISTS Bases for a Channel Check, the Channel Check ensures that a gross instrumentation failure has not occurred between Channel Calibrations. As the Channel Check is a qualitative assessment (as defined in the Technical Specifications) and is not intended to be a precision calibration, it can be performed on instruments when the measured parameter indicates zero. For example, an AFW flow rate indicator should not be indicating 100 gpm, when the AFW system is not in service. In this case, the performance of the Channel Check surveillance functions, as intended, to detect a gross failure of a flow indicator. The associated AFW system does not have to be in operation supplying water to the steam generators to make this operability determination. As such, the proposed change is acceptable, because it provides an additional safety benefit by requiring formal operability verifications of the AFW Flow Rate indicators (even when the AFW system may not be in service) that could detect a gross instrument failure between Channel Calibrations consistent with the purpose of a Channel Check.

Due to the changes made to the surveillance requirements of Technical Specification 3/4.3.3.8 (i.e., the addition of a General Note to define the applicability of each surveillance) and the changes to Table 4.3-7 discussed above, Table 4.3-7 is no longer necessary to specify the required surveillances for each PAM Function, and (along with its reference in the Unit 1 Table index page) is deleted. As the applicability of the surveillance requirements to the PAM functions are identified in the General Note that modifies the surveillances, listing each surveillance and PAM Function in Table 4.3-7 is unnecessary and inconsistent with the presentation of the corresponding surveillance requirements in the ISTS. As such, the proposed change to delete Table 4.3-7 is acceptable because it is administrative in nature, conforms more closely to the corresponding ISTS requirements, and does not introduce a technical change to the current surveillance requirements.

Change No. 10

The BVPS Regulatory Guide 1.97 and Technical Specification 3/4.3.3.8 PAM Functions have been evaluated consistent with the guidance provided by WCAP-15981. The plant-specific implementation of WCAP-15981 requires a review of the; 1) Design Basis Accidents, 2) Emergency Operating Procedures, 3) Probabilistic Risk Assessment, 4) Severe Accident Management Guidelines, and 5) Emergency Plan. These reviews and a summary are described in Sections 4.1 through 4.6 below. The following Sections of the LAR describe the evaluations performed to assess the BVPS specific PAM instrumentation and discuss the following:

- **The PAM instrumentation selected for inclusion in Table 3.3-11 of revised Technical Specification 3/4.3.3.8 is discussed in Section 4.7,**

- **The acceptable alternate indications instrumentation for certain PAM Functions selected for inclusion in Technical Specification 3/4.3.3.8 is discussed in Section 4.8,**

- **The PAM instrumentation selected for relocation from current Technical Specification 3/4.3.3.8 Table 3.3-11 is discussed in Section 4.9,**

- **The BVPS Regulatory Guide 1.97 Type A PAM instrumentation that is not currently included in the Technical Specifications is discussed in Section 4.10, and**

- **The BVPS Regulatory Guide 1.97 Category 1 PAM instrumentation that is not currently included in the Technical Specifications is discussed in Section 4.11.**

The following Subsections (4.1 through 4.11) address Change # 10.

4.1 Design Basis Accidents

The instrumentation that provides the primary information that is essential for the direct accomplishment of the specified manual actions (including long-term recovery actions) for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs or transients was identified from the design basis accident analyses documented in BVPS Unit 1 UFSAR Chapter 14 and BVPS Unit 2 UFSAR Chapter 15.

The manual actions specifically credited in the design basis analyses for which no automatic control is provided were found to be identical for BVPS Unit 1 and Unit 2, and are described in detail in the following paragraphs.

Loss of Coolant Accidents – In the event of a LOCA, the Design Basis Accident analyses do not require that the control room staff take any actions for the transfer to the Emergency Core Cooling System (ECCS) recirculation mode of operation. Specific RWST level transmitters are provided to automatically actuate the ECCS switchover from the injection mode to recirculation mode following an accident. All of the required changes in the system configuration are accomplished automatically. For Unit 1, in the recirculation mode, a minimum of one of the two low head SI pumps provides flow to the cold legs; for Unit 2, in the recirculation mode, a minimum of one of the two containment spray recirculation pumps, that also serve as low head SI pumps (during recirculation mode) provides flow to the cold legs.

At a specified time after the initiation of the accident, a switchover to hot leg recirculation is required to limit the potential for boron precipitation in the reactor vessel. This is performed manually by the control room operators. However, the cue to perform the hot leg recirculation switchover is based on time, as opposed to any plant variables.

Steam Generator Tube Rupture – The steam generator tube rupture accident requires operator actions for which no automatic control is provided to bring the plant to a safe stable state. The operator actions are discussed in the UFSAR. For BVPS Unit 2, the operator actions assumed in the Design Basis Accident analysis are given in Table 15.6-5 of the UFSAR as (these identical actions are described in Section 14.2.4.2.3 of the BVPS Unit 1 UFSAR):

- Identify and isolate ruptured SG – based on SG level (diagnose rupture) and SG pressure (identify stuck open relief valve which affects the recovery strategy),

- Operator action to initiate cooldown using the intact SGs – based on ruptured SG pressure (cooldown target) and RCS pressure and temperature (maintain subcooling),

- Operator action to initiate RCS depressurization using the pressurizer PORVs – based on RCS pressure, RCS temperature, pressurizer level, and RCS subcooling, and

- Operator action to initiate SI termination – based on SI termination criteria of RCS pressure and temperature and pressurizer level.

The SG pressure indication is used for identifying and isolating a ruptured SG to identify if a coincident uncontrolled depressurization has also occurred which affects the recovery strategy.

Main Steam Line Break – The main steam line break accident described in the BVPS UFSARs (Section 14.2.5.1 for Unit 1 and Section 15.1.5.2 for Unit 2) does not credit any operator actions.

Main Feed Line Break - The main feed line break accident is discussed in BVPS UFSAR, Section 14.2.5.2 for Unit 1 and Section 15.2.8 for Unit 2. The DBA analyses credit operator action to isolate the faulted SG (terminate AFW to the faulted SG). This operator action would utilize the SG pressure and AFW flow rate indications.

Other Design Basis Accidents – All of the remaining Design Basis Accident analyses documented in the BVPS UFSARs do not rely on explicit operator actions. However, inherent in all of these remaining design basis accident analyses are two operator actions to establish and maintain long term core cooling: controlling auxiliary feedwater flow to maintain a heat sink and prevent SG overfill, and termination of SI to prevent pressurizer overfill. The control of the auxiliary feedwater flow to prevent SG overfill is based on SG level indication. Termination of safety injection to prevent pressurizer overfill is based on a combination of pressurizer level and RCS subcooling, which is derived from RCS pressure and RCS temperature.

4.2 Emergency Operating Procedures (EOPs)

The BVPS Emergency Operating Procedures (EOPs) provide instructions for the optimal recovery from plant events that result in either a reactor trip or a safety injection signal. While the EOPs are consistent with the plant design and licensing basis, they also provide recovery instructions for events that are outside the plant design and licensing basis. The EOPs also utilize a broad range of plant instrumentation to diagnose plant conditions and monitor the performance of equipment and systems.

The EOPs for BVPS Unit 1 and Unit 2 were reviewed to identify key operator actions that would be implemented during an optimal recovery from Design Basis Accidents. The following key operator actions were identified:

Small Break LOCA – The optimal response to a small loss of coolant accident involves operator actions for RCS cooldown and depressurization by dumping steam from the SG secondary side. These operator actions are not required for the design basis response, but are included in the EOPs to describe a set of actions to bring the plant to cold shutdown conditions. The operator actions include:

- Operator actions to initiate a controlled cooldown of the RCS by dumping steam from the SGs, and
- Operator actions to limit RCS pressure and pressurizer level by reducing or terminating SI flow.

Steam Generator Tube Rupture – The optimal response to a steam generator tube rupture is the same as that described for the Design Basis Accident response, including:

- Identify and isolate ruptured SG – based on SG level (diagnose rupture) and SG pressure (identify stuck open relief valve which affects the recovery strategy),
- Operator action to initiate cooldown using the intact SGs – based on ruptured SG pressure (cooldown target) and RCS pressure and temperature (maintain subcooling),
- Operator action to initiate RCS depressurization using the pressurizer PORVs – based on RCS pressure, RCS temperature, pressurizer level, and RCS subcooling, and
- Operator action to initiate SI termination – based on SI termination criteria of RCS pressure and temperature and pressurizer level.

The SG pressure indication is used for identifying and isolating a ruptured SG to identify if a coincident uncontrolled depressurization has also occurred which affects the recovery strategy.

Main Steam Line Break - There are several key operator actions described in the BVPS EOPs, for which no automatic control is provided, to bring the plant to a safe stable state. However, these actions are not credited in the UFSAR analyses to bring the plant to a safe stable state for these accidents. The key operator actions are:

- Operator action to limit the excessive heat loss through the faulted SG by terminating auxiliary feedwater flow to that SG (detected by SG pressure loss), and
- Operator action to limit RCS pressure and pressurizer level by terminating SI flow and to control steam generator level and RCS coolant temperature using the auxiliary feedwater system.

Functional Restoration Procedures - In addition to these key operator actions, there are several key operator diagnoses regarding beyond design basis accident conditions that are based on plant indications. These operator actions include:

- Subcriticality – Neutron Flux indication
- Core Cooling – Core Exit Temperature indication
- Heat Sink – SG Level indication
- Integrity – RCS Pressure indication
- Containment – Containment Pressure indication

4.3 Probabilistic Risk Assessment (PRA)

The important operator actions to prevent core damage for BVPS Units 1 and 2 can be derived from the Probabilistic Risk Assessments (PRAs) for each unit. The BVPS Unit 1 and Unit 2 PRAs are separate because of design differences that impact the risk assessments. For BVPS Units 1 and 2, the PRA includes an internal events PRA as well as a seismic and fire PRA. Transition and shutdown risk assessments for BVPS are not applicable to this application because the PAM Technical Specification is only applicable in Modes 1 through 3.

The PRAs for BVPS Units 1 and 2 were updated in 2003, and provide an accurate representation of the design and operation of BVPS Units 1 and 2. The update included internal initiating events, as well as seismic and fire initiating events.

In the BVPS Unit 1 and 2 PRAs, the instrumentation is modeled as part of the human reliability analysis, which models the important operator actions to prevent core damage. In the human reliability analysis, instrumentation is assumed to be available to "cue" the operator response (i.e., to diagnose a condition requiring an operator response). Instrumentation is also assumed to be available to "control" the operator response (i.e., to indicate the desired end point of the response).

Internal Events PRA

The Level 1 PRA (quantification of core damage frequency, or CDF) provides a detailed investigation of the important equipment and operator actions for the successful mitigation of accidents. There are a number of risk importance measures that can be used to define the components that are important to risk. The Risk Achievement Worth (RAW) was chosen for this evaluation, because it is a measure of the increase in risk if the component is totally unreliable (i.e., failure rate is 1.0). This is appropriate for this evaluation, because the potential availability of instrumentation can be impacted by removing the instrument from the Technical Specifications and the RAW provides a bounding estimate of the impact of instrument unavailability on risk. The important operator actions and related instrumentation for BVPS, as derived from the Level 1 PRA, are shown in the following tables. A RAW value greater than 2.0 is typically used to indicate high risk significance (e.g., Regulatory Guide 1.174).

No quantification of the important equipment or operator actions to prevent large early releases is provided in the Level 1 PRA. However, some insights into important operator actions are provided in the Level 2 PRA discussion as summarized below.

The Level 2 PRA models the consequences of core damage accidents and provides a quantification of the Large Early Release Frequency, or LERF.

However, operator actions to prevent fission product releases are not modeled in the BVPS Level 2 PRA; which is typical of most Level 2 PRA models in use today. Therefore, the importance of operator actions in preventing large early releases cannot be totally determined from the current Level 2 PRA analyses. However, from other PRA studies, it is known that there are three types of operator actions that can impact LERF: 1) operator actions to preserve the remaining fission product barriers after core damage has occurred, in accordance with the plant Severe Accident Management Guidance (SAMG); 2) operator actions taken before core damage, in accordance with the plant EOPs, that do not impact the core damage frequency, but that help to preserve the remaining fission product barriers; and 3) operator actions that are taken to prevent core damage for containment bypass sequences that, in turn, also impact LERF, because a large fraction of containment bypass core damage sequences are LERF sequences.

The Level 2 PRAs for BVPS Units 1 and 2 indicate that LERF is dominated by containment bypass sequences. For Unit 1, 91.8% of the LERF is due to SGTR core damage sequences, and additional 8.0% is due to interfacing system LOCA core damage sequences. Containment isolation failures and early containment failures contribute less than 0.2%. The Unit 1 PRA indicates a RAW value for high pressure melt ejection of 2.71. From this it can be inferred that operator actions to depressurize the RCS carry a high importance; this operator action depends on the RCS pressure indication for diagnosis. For Unit 2, 75.1% of the LERF is due to SGTR core damage sequences, and additional 24.5% is due to interfacing system LOCA core damage sequences. Containment isolation failures and early containment failures contribute less than 0.6%. The Unit 2 PRA indicates a RAW value for high pressure melt ejection of 5.04. From this it can be inferred that operator actions to depressurize the RCS carry a high importance; this operator action depends on the RCS pressure indication for diagnosis.

In addition, it can be inferred that those operator actions that prevent core damage for containment bypass sequences (e.g., steam generator tube rupture and interfacing system LOCAs) would potentially be important from a quantitative LERF importance assessment. The two operator actions that are important for preventing core damage for these sequences (see Table 3

(Unit 1) and Table 4 (Unit 2) below from Level 1 PRA importances) are RWST refill and RCS depressurization, which depend on RWST level and RCS pressure indications, respectively.

Table 3: BVPS Unit 1 PRA Update
TOP 10 OPERATOR ACTIONS RANKED BY RAW IMPORTANCE

Rank	RAW	Operator Action Description	Key Parameter Indication
1	6.75	Operator aligns makeup to the RWST, given a SGTR with secondary leakage.	RWST Level
2	3.99	Operator manually initiates safety injection given failure of SSPS.	High Head SI Flow
3	2.70	Operator trips the RCPs during a loss of all CC.	Not Applicable (see Note 1)
4	2.62	Operators setup portable fans & open doors to cool Emergency Switchgear.	Not Applicable (see Note 1)
5	2.54	Operator identifies ruptured SG and initiates isolation.	RCS Pressure, SG Level, and SG Pressure
6	2.54	Operator cools down and depressurizes the RCS using atmospheric dumps or RHR valve during a SGTR.	RCS Pressure, RCS Temperature, and Pressurizer Level
7	2.53	Operator depressurizes RCS to RHR and LHSI entry conditions using pressurizer PORVs or sprays.	RCS Pressure, Pressurizer Level, and RCS Temperature
8	2.13	Operator initiates Bleed and Feed when AFW fails; DAFW and MFW restoration not tried.	SG Level
9	1.96	Operator depressurizes RCS to LHSI entry conditions using pressurizer PORVs or sprays; Small Break LOCA and failure of HHSI.	RCS Pressure, Pressurizer Level, and RCS Temperature, and Core-Exit Temperature
10	1.66	Operator manually aligns Auxiliary River Water pump when main RW pumps fail.	Not Applicable (see Note 1)

Note 1: These operator actions are based on the failure of a normally operating system. The failure of the system would be indicated in the control room by multiple indications and alarms. As such, no essential "key" parameter indication is identified, since the operator action is based on the status of the entire system.

Table 4: BVPS Unit 2 PRA Update TOP 10 OPERATOR ACTIONS RANKED BY RAW IMPORTANCE			
Rank	**RAW**	**Operator Action Description**	**Key Parameter Indication**
1	16.6	Operator initiates Bleed and Feed when AFW fails, after attempting to realign MFW.	SG Level
2	5.23	Operator manually actuates AFW following a transient.	AFW Flow
3	5.06	Operator realigns Main Feedwater - no SI signal present.	SG Level
4	4.61	Operator aligns makeup to the RWST, given a SGTR with secondary leakage.	RWST Level
5	2.88	Operator manually trips reactor within 1 minute, given automatic trip failed.	Neutron Flux
6	2.03	Operator identifies ruptured SG and initiates isolation.	RCS Pressure, SG Level, and SG Pressure
7	2.03	Operator cools down and depressurizes the RCS using atmospheric dumps or RHR valve during a SGTR.	RCS Pressure, RCS Temperature, and Pressurizer Level
8	2.03	Operator depressurizes RCS to RHS and LHSI entry conditions using pressurizer PORVs or sprays.	RCS Pressure, Pressurizer Level, RCS Temperature, and Core-Exit Temperature
9	1.89	Operator initiates Bleed and Feed when AFW fails, MFW restoration was not attempted.	SG Level
10	1.59	Operator trips the RCPs during a loss of all CC.	Not Applicable (see Note 1)
Note 1: These operator actions are based on the failure of a normally operating system. The failure of the system would be indicated in the control room by multiple indications and alarms. As such, no essential "key" parameter indication is identified, since the operator action is based on the status of the entire system.			

Fire and Seismic Events PRA

The BVPS Units 1 and 2 PRAs include contributions from seismic and fire initiating events from at-power operation. However, risk importance measures (e.g., RAW values) for the operator actions modeled in those risk assessments are not available. For Unit 1, seismic and fire initiating events represent > 68% of the total core damage frequency; for Unit 2 that value is > 40%.

Expressed in terms of contribution to the large early release frequency, > 99% of the total large early release frequency is due to internal initiating events. Thus, LERF contributors from seismic and fire events are not considered further in this assessment.

For BVPS Unit 1, the dominant core damage sequences for fire initiating events are fires in the cable spreading room which lead to a total loss of emergency switchgear ventilation and consequently all emergency a.c. and all emergency d.c. power. For these sequences, the failure of all emergency d.c. power would result in the unavailability of all control room instrumentation that is important for diagnosing and responding to the event. From the perspective of important instrumentation for operator actions, there would be no operator actions based on instrumentation. Therefore, there is no instrumentation importance input from the fire initiating event PRA. For the BVPS Unit 1 seismic risk assessment, the dominant seismic core damage sequences involve either a loss of all river water, loss of all emergency a.c. and d.c. power or a station blackout. In the case of a loss of all emergency d.c. power, there is no instrumentation to cue operator actions and therefore no risk importance for instrumentation. For the case of the loss of all river water or a loss of all emergency a.c. power, the sequence is very similar to a station blackout already considered in the internal events PRA. For this event, operator actions to control steam generator level and initiate cooldown and depressurize the RCS are risk important operator actions. These actions are already designated as risk important operator actions from the internal events PRA.

For BVPS Unit 2, the dominant fire initiated core damage event is a control room fire that results in the loss of all secondary system decay heat removal capability and therefore requires bleed and feed cooling. The operator actions are identical to those already considered for the internal initiating events. That is, the risk important operator action for this event is based on steam generator wide range level to initiate bleed and feed cooling. For BVPS Unit 2, the dominant seismic event is very similar to the station blackout event from the internal events PRA assessment. For this event, operator actions to control steam generator level and initiate cooldown and depressurize the RCS are risk important operator actions. These actions are already designated as risk important operator actions from the internal events PRA.

Therefore the seismic and fire risk assessments do not identify any new insights with respect to instrumentation risk importance.

4.4 Severe Accident Management Guidance (SAMG)

The Severe Accident Management Guidance (SAMG) for BVPS Units 1 and 2 are identical in content, except to account for design differences between the two units. The SAMG was developed from the generic Westinghouse Owners Group SAMG. Therefore, the conclusions in WCAP-15981 can be applied in a straight-forward manner to BVPS Units 1 and 2.

The SAMG makes use of 6 plant parameters for establishing and/or maintaining a safe stable state following a core damage accident. The parameters and instrumentation used to determine the value of each parameter are:

- SG Water Level – SG Level
- RCS Pressure – RCS Pressure
- Core Temperature – Core Exit Thermocouples
- Containment Water Level – Containment Level
- Containment Pressure – Containment Pressure
- Containment Hydrogen – Hydrogen Monitor

From a PRA perspective, operator actions in accordance with the SAMG are only used after core damage has occurred. Therefore, they do not have a risk importance with respect to preventing core damage and are not modeled in the Level 1 PRA. The SAMG actions can have a risk importance with respect to fission product releases. The SAMG actions are not modeled in the Level 2 PRAs for BVPS Units 1 and 2. Thus, the risk importance of these instruments can only be estimated through qualitative evaluations.

An evaluation of the importance of these SAMG instruments for BVPS Units 1 and 2 was performed with the following conclusions:

Steam Generator Water Level: The BVPS Unit 1 and Unit 2 PRAs identify that steam generator tube rupture bypass sequences are the largest contributor to LERF. The accident sequences are binned as LERF contributors, in part, because fission product scrubbing in the SGs is not credited in these sequences, since AFW is terminated to the ruptured SG. This results in a condition where the fission product releases from the reactor coolant system through the ruptured tube may be above the water level in that SG (BVPS procedures ensure AFW is terminated when sufficient level exists to cover the SG tubes). SAMG operator actions to maintain a water cover over the top of the highest tube would provide a water pool for fission product scrubbing, which would significantly reduce the release quantities. Therefore, operator actions to maintain an adequate steam generator level in the ruptured SG in accordance with the SAMG are considered to be risk important for BVPS Units 1 and 2.

RCS Pressure: The BVPS Unit 1 and Unit 2 PRAs identify that early containment failures resulting from high pressure melt ejection have a RAW value greater than 2.0. Therefore operator actions to depressurize the RCS to avoid a high pressure melt ejection in accordance with the SAMG are considered to be risk important for BVPS Units 1 and 2.

Core Temperature: Operator actions to re-establish core cooling are inherent in the SAMG and would be attempted regardless of instrumentation. Therefore, no instrumentation that indicates core temperature in the SAMG is considered to be important to risk for BVPS Units 1 and 2.

Containment Water Level: The purpose of the containment water level instrumentation as utilized in the SAMG, is to assure that adequate water exists in the containment for cooling a core that has melted and relocated to containment via reactor vessel melt-through. These SAMG operator actions are intended to prevent core-concrete interactions, which could lead to long term loss of containment integrity. The BVPS Unit 1 and Unit 2 PRAs identify that these accident sequences represent a small fraction of the accident sequences that result in a containment challenge following core damage if no operator action is taken. Therefore, although operator actions to inject water to the containment are not modeled in the Level 2 PRA, these actions would not change the LERF risk metric and also would have a negligible impact on long term containment integrity. There are no operator actions based on containment water level that have any impact on CDF or LERF. The SAMG action to add water to containment would only be taken after core damage had already occurred. Therefore, it is concluded that this is not a risk significant operator action and the instrumentation utilized to support this action is not risk significant.

Containment Pressure: The purpose of the containment pressure instrumentation as utilized in the SAMG, is to either: a) indicate the containment pressure to allow the operators to take action to vent the containment to prevent a catastrophic containment failure due to overpressurization, or b) determine if a hydrogen burn challenge to the containment integrity exists requiring consideration of hydrogen control strategies. Since long term containment overpressurization and slow burn overpressurization are the dominant late containment failure modes in the BVPS Units 1 and 2 PRAs, containment pressure is important for SAMG mitigation. The PRA analyses show that SAMG initiated operator actions for hydrogen control strategies would only be required for station blackout events after all instrumentation is lost due to battery depletion. It is not important for any accident in which instrumentation is available and therefore only long term overpressurization is of interest. SAMG initiated operator actions for containment venting due to long term overpressurization would only be considered as the containment pressure approaches the lower bound of the failure pressure. Only the containment high range pressure indication is adequate for this consideration and should be included with respect to the SAMG. Therefore, it is concluded that the containment wide range pressure instrumentation is an important SAMG indication.

Containment Hydrogen: The purpose of the containment hydrogen instrumentation as utilized in the SAMG, is to indicate the containment hydrogen concentration to allow the operators to take action to prevent a containment failure due to a hydrogen burn. For this reason, requirements for containment hydrogen monitor operability are retained in the Licensing Requirements Manual, consistent with TSTF-447 and BVPS License Amendment Numbers 259 (Unit 1) and 142 (Unit 2). The potential for early containment failures due to hydrogen burns was analyzed in the Level 2 PRA and found not to be risk significant. The potential for slow burn overpressurization containment failures is a contributor to the probability of late containment failures, but only for station blackout events with no power recovery in the first day. In this case, there would be no power available for instrumentation so the availability of hydrogen monitors is not relevant to this case. Further, in the rulemaking to revise 10 CFR 50.44, the NRC determined that the importance of containment hydrogen monitoring could be downgraded since it was not important to risk. Therefore, it is concluded that operator action for hydrogen control, based on containment hydrogen indication, is not a risk significant.

SAMG Entry: In order to enter the SAMG, a sustained high indication from the core exit thermocouples is required. Therefore, from a SAMG perspective the core exit thermocouple indications are important to risk for BVPS Units 1 and 2.

After the SAMG are entered, the Steam Generator Water Level, RCS Pressure, and Containment Pressure indications are utilized in the SAMG, and are therefore important from a risk perspective.

4.5 Emergency Plan (E-Plan)

As discussed in WCAP-15981, the Emergency Plan (E-Plan) and the Emergency Plan Implementing Procedures may rely on plant instrumentation for three separate activities: 1) Assessment of the appropriate Emergency Action Level (EAL), 2) Core Damage Assessment (CDA), and 3) Offsite Dose Projections using the Environmental Assessment and Dose Projection Guideline. The role of the instrumentation as utilized in each of these activities for BVPS Units 1 and 2 is discussed below.

Emergency Action Levels (EALs)

The EALs provide a means of determining the severity of events and an emergency classification. Upon an emergency declaration communications are initiated with offsite agencies to protect the health and safety of the public in the vicinity of the plant. Based on the potential for fission product releases from the plant, the offsite authorities would invoke various levels of offsite emergency protective actions for members of the general public. The

BVPS Units 1 and 2 Emergency Preparedness Plan uses the EAL assessment methods in NUMARC/NESP-007. As discussed in WCAP-15981, the most severe potential for fission product releases is associated with the General Emergency level, which is classified as a loss of any two barriers and a potential loss of a third barrier. WCAP-15981 provides an assessment of the important instrumentation used in the Emergency Action Levels. This assessment was reviewed and found to be directly applicable to BVPS Units 1 and 2. Based on this assessment, for the purposes of the protection of the health and safety of the offsite general public, the key indicators of the need to implement offsite emergency protective actions at BVPS Units 1 and 2 are high core exit thermocouple indications, high containment radiation levels, failure of complete containment isolation and/or high containment pressures. The other indications are most useful to validate the loss of barriers, but not as primary indications of the potential for, or the loss of the barrier.

Core Damage Assessment (CDA)

The ability to assess the occurrence of and degree of core damage is a NUREG-0737 requirement. The core damage assessment has been implemented within the EALs based on the containment radiation level at BVPS Units 1 and 2.

Environmental Assessment and Dose Projection Guideline

BVPS does not rely solely on plant instrumentation for offsite dose projections. The best available information is used for projecting offsite dose (primarily from effluent radiation monitors). However, if adequate plant instrumentation is not available, default values based on information contained in the UFSAR are used. Therefore, no essential key indication(s) are identified for inclusion in the Technical Specifications to support offsite dose projections.

4.6 Summary of Instrumentation Importance

A composite list of PAM instrumentation relied upon in the design basis accident analysis, the PRA, accident management (EOPs and SAMG), and offsite emergency protective actions has been determined based on the evaluations above. The PAM instrumentation that was determined to be important from the assessments discussed above, is summarized in Table 5 below.

4.7 BVPS Instrumentation Proposed to be Included in PAM Technical Specification 3/4.3.3.8

The BVPS PAM instrumentation required to monitor the parameters that were determined to be important for operator actions to respond to accidents is discussed below. For each indication, the available PAM instrumentation is discussed and the most appropriate PAM instrumentation to provide the required indications is described. An assessment of the applicability of the 10 CFR 50.36 (c)(2)(ii) criteria and the Regulatory Guide 1.97 classification is also provided with regard to the PAM instrumentation selected.

Table 5: Summary of BVPS Important Indications for Accident Management

Indication / Purpose	DBA	EOP	SAMG	PRA	EP
SG level					
• Diagnose SGTR	✔	✔		✔	
• Maintain SG heat sink	✔	✔		✔	✔
• Prevent SG overfill	✔	✔		✔	
• Initiate Bleed and Feed		✔		✔	
• Scrub Fission Products for SGTR			✔		
SG pressure					
• Diagnose secondary side break or stuck open RV	✔	✔		✔	
• Cooldown target for RCS depressurization SGTR	✔	✔		✔	
RCS pressure and temperature					
• Cooldown target for RCS depressurization	✔	✔		✔	
• Maintain subcooling during RCS cooldown and depressurization	✔	✔		✔	
• SI termination	✔	✔		✔	
• High Pressure Melt Ejection prevention			✔	✔	
Pressurizer level					
• SI termination to prevent pressurizer overfill	✔	✔		✔	

Table 5: Summary of BVPS Important Indications for Accident Management					
Indication / Purpose	**DBA**	**EOP**	**SAMG**	**PRA**	**EP**
Core temperature					
• Diagnose inadequate core cooling		✔	✔	✔	✔
Neutron flux					
• Diagnose subcriticality		✔		✔	✔
Containment Pressure					
• Diagnose inadequate containment cooling		✔	✔		✔
Containment Radiation					
• Diagnose core damage					✔
Containment Isolation					
• Diagnose unisolated containment					✔
RWST Level					
• Diagnose RWST refill				✔	
High Head SI Flow					
• Diagnose manual SI				✔	
Auxiliary feedwater flow					
• Diagnose loss of heat sink				✔	

<u>SG Level indication</u>:

SG level indication can be provided by either the narrow range SG level instrumentation or the wide range level instrumentation. Operator actions for the diagnosis of a steam generator tube rupture, the maintenance of an adequate SG level to provide a heat sink, controlling the SG level to prevent SG overfill, and covering the SG tubes to scrub fission products for a steam generator tube rupture are performed utilizing the SG narrow range level indication. However, the wide range SG indication encompasses the narrow range span and can be used in level ranges where the narrow range indication is not available. As such, the preferred level instrumentation for the initiation of bleed and feed cooling operations is the wide range SG level indication. The narrow range SG level indication could also be used to initiate bleed and feed cooling, but would have to be used in conjunction with additional indication(s) (e.g., AFW Flow) in order to accomplish the same task at all SG levels. Therefore, the SG level indication provided by the wide range level instrumentation is the recommended PAM instrument for BVPS Units 1 and 2.

The BVPS Units 1 and 2 UFSARs discuss that controlling SG level for long term core cooling is an operator action for which no automatic control is provided. Therefore, the SG wide range indication satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii). The PRA shows that the initiation of bleed and feed core cooling is a risk significant operator action for both BVPS Unit 1 and 2. Also, the SAMG assessment shows that maintaining the water level over a ruptured SG tube is a risk significant operator action. Therefore, the SG wide range indication also satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The SG Water Level (Wide Range) indication is currently classified as Regulatory Guide 1.97 Type D, Category 1 variable for BVPS Unit 1 and a Type A, Category 1 variable for Unit 2. Based on the methodology in WCAP-15981, the SG Water Level (Wide Range) indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type A and Category 1 variable for both BVPS Unit 1 and Unit 2.

The SG Water Level (Wide Range) indication provides information for operator actions to maintain a heat sink, for which no automatic response is provided. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type A variable for both Units. The SG Water Level (Wide Range) indication provides direct verification of the heat sink safety function. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is also considered a Category 1 variable for both Units.

<u>SG Pressure indication</u>:

SG pressure indication can be provided by the steam line pressure instrumentation. Each steam line contains pressure indication instrumentation between the SG outlet and the Main Steam Isolation Valve. There is no other indication that can be used to indicate SG pressure. Operator actions for the diagnosis of a secondary side break or stuck open relief valve, and the cooldown target for RCS depressurization for a SGTR are performed utilizing the SG pressure indication.

The BVPS Units 1 and 2 UFSARs discuss that responding to a secondary side depressurization event and RCS cooldown and depressurization for a SGTR to terminate the break flow are operator actions for which no automatic control is provided. Therefore, the SG Pressure indication satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii). Also, the PRAs indicate that RCS depressurization to terminate the break flow for a SGTR event is a risk significant operator action for both BVPS Unit 1 and 2. Therefore, the SG Pressure indication also satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The Steam Generator (SG) Pressure indication is currently classified as a Regulatory Guide 1.97 Type A, Category 1 variable for BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the Steam Generator (SG) Pressure indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type A and Category 1 variable for both BVPS Unit 1 and Unit 2.

The Steam Generator (SG) Pressure indication provides information needed for operator actions for SGTR break flow termination for which no automatic response is provided. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type A variable. The Steam Generator (SG) Pressure indication, together with RCS pressure, provides information to verify that break flow through a ruptured SG tube is terminated, thereby satisfying the inventory safety function. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is also considered a Category 1 variable.

<u>RCS Pressure indication</u>:

The RCS pressure instrumentation provides RCS pressure indication that is used for all accident sequences. Operator actions for a cooldown target for RCS depressurization and for maintaining subcooling during the RCS cooldown and depressurization and for SI termination are performed utilzing the RCS pressure indication. Also, operator actions in the EOPs and SAMGs utilize the RCS pressure indication to diagnose the need to depressurize the RCS to minimize the potential containment integrity challenges from a high pressure melt ejection.

The BVPS Units 1 and 2 UFSARs discuss that RCS cooldown and depressurization for a steam generator tube rupture to terminate break flow and for SI termination to prevent pressurizer overfill are operator actions for which no automatic control is provided. Therefore, the RCS Pressure indication satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii). Additionally, the PRA shows that RCS depressurization to terminate the break flow during a steam generator tube rupture event and depressurization of the RCS after core damage to prevent a high pressure melt ejection that could challenge containment integrity are risk significant operator actions for both BVPS Unit 1 and 2. Therefore, the RCS Pressure indication also satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The Reactor Coolant System Pressure (Wide Range) indication is currently classified as a Regulatory Guide 1.97 Type A Category 1 variable for BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the Reactor Coolant System Pressure (Wide Range) indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type A and Category 1 variable for both BVPS Unit 1 and Unit 2.

The Reactor Coolant System Pressure (Wide Range) indication provides information for operator action for SGTR break flow termination for which no automatic response is provided. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type A variable. The Reactor Coolant System Pressure (Wide Range) indication, together with SG pressure, provides information to verify that break flow through a ruptured SG tube is terminated thereby satisfying the inventory safety function. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is also considered a Category 1 variable.

RCS Temperature indication:

The RCS temperature indication can be provided by a number of instruments. The RCS temperature indication provides information on the degree of subcooling of the reactor coolant fluid for RCS cooldown and depressurization and for SI termination. The core exit thermocouples (CETs) provide the best information regarding the RCS temperature, since they are located at the core outlet and would therefore also provide an indication of the hottest RCS fluid temperatures. The RCS hot leg temperature instrumentation could also be used as an indication of the RCS temperature. However, the CETs are used in the determination of RCS subcooling and provide the most timely indication of temperature changes in the core. Therefore, the CETs provide the best indication for the diagnosis of inadequate core cooling, and are the preferred instrument for indicating the RCS temperature for subcooling assessments.

The BVPS Units 1 and 2 UFSARs discuss that RCS cooldown and depressurization during a steam generator tube rupture to terminate the break flow and for SI termination to prevent pressurizer overfill are operator actions for which no automatic control is provided. Therefore, the CET indication satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii). Additionally, the PRA shows that RCS depressurization to terminate the break flow during a SGTR event is a risk significant operator action for both BVPS Unit 1 and 2. Therefore, the CET indication also satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii). See the Core Temperature indication discussion that follows for more information regarding the CETs.

Pressurizer Level indication:

The pressurizer level instrumentation provides pressurizer level indication that is utilized for all accident sequences. The purpose of pressurizer level indication is for the SI termination criteria to prevent pressurizer overfill, and is the only instrument that provides this indication.

The BVPS Units 1 and 2 UFSARs discuss that SI termination in the event of a steamline break to prevent pressurizer overfill is an operator action for which no automatic control is provided. Therefore, pressurizer level indication satisfies Criterion 3 of 50.36 (c)(2)(ii). The PRAs for BVPS Units 1 and 2 also indicate that SI termination in the event of a steam generator tube rupture is required for long term core cooling. Therefore, the pressurizer level indication satisfies Criterion 4 of 50.36 (c)(2)(ii).

The Pressurizer Water Level indication is currently classified as a Regulatory Guide 1.97 Type A Category 1 variable for BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the Pressurizer Water Level indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type A and Category 1 variable for both BVPS Unit 1 and Unit 2.

The Pressurizer Water Level indication provides the primary information needed for control room operating personnel to take specified manual actions to terminate SI. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type A variable. The Pressurizer Water Level indication provides information related to the accomplishment of the RCS inventory safety function for SI termination. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 1 variable.

Core Temperature indication:

The core exit thermocouples (CETs) provide the most direct measurement of the core temperature since they are located in the core outlet region of the reactor. The purpose of the core temperature indication is for the diagnosis of an inadequate core cooling condition.

The BVPS Units 1 and 2 UFSARs discuss that operator action to initiate RCS depressurization for steam generator tube rupture is an operator action for which no automatic control is provided. The RCS subcooling indication is required for the successful completion of this operator action. Therefore, the core exit temperature indication, which is an input to RCS subcooling, satisfies Criterion 3 of 10 CFR 50.36 (c)(2)(ii).

The BVPS SAMG uses the CET temperature indication as a transition from the EOPs to the SAMG. RCS depressurization to prevent a high pressure melt ejection is contained in the SAMG, which is a risk important action in the BVPS Unit 1 and 2 PRA. Additionally, the Emergency Action Levels in the E-Plan for BVPS Units 1 and 2 utilize the CET temperature as an indication for the potential loss of a fission product barrier, which is important in the assessment of the declaration of a General Emergency level and the potential need for offsite radiological protection actions. Therefore, the CET indication satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The Core Exit Temperature indication is currently classified as a Regulatory Guide 1.97 Type C Category 1 variable for BVPS Unit 1 and Type A, Category 1 variable for Unit 2. Based on the methodology in WCAP-15981, the Core Exit Temperature indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type A and Category 1 variable for both BVPS Unit 1 and Unit 2.

The Core Exit Temperature indication provides information for the operators to initiate RCS depressurization. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type A variable. The Core Exit Temperature indication provides direct verification of the core cooling safety function. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 1 variable.

<u>Neutron Flux indication</u>:

The power range neutron flux indication provides the most direct indication of reactor criticality. The intermediate range and source range indications provide a better indication of sustained subcriticality when the RCS is depressurized.

The BVPS Unit 1 and 2 PRAs indicates that the power range neutron flux is a key indication for accident management operator actions to initiate a manual reactor trip to bring the reactor to a subcritical condition. Subsequent operator actions to assure that the reactor remains in subcritical state, such as during RCS depressurization, were not determined to be important for long term core cooling. Therefore the intermediate range and source range indications are not identified as key instruments in this assessment. Additionally, the Emergency Action Levels in the E-Plan for BVPS Units 1 and 2 utilize the power range neutron flux monitor as an indication of a potential loss of a fission product barrier in the assessment of the declaration of a General Emergency level and the potential need for offsite radiological protection actions. Therefore, the power range neutron flux monitor satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The Unit 1 response to RG 1.97 identified source, intermediate, and power range neutron flux variables that utilize the Westinghouse Nuclear Instrumentation System. The Unit 2 response to RG 1.97 identified upper and lower range neutron flux variables that utilize the full range Gamma-Metrics instrumentation. The NRC subsequently approved both unit responses to RG 1.97. However, based on the evaluations performed in accordance with WCAP-15981, the full range Gamma-Metrics instrumentation is not specifically required to fulfill the key indicating function of the neutron flux variable (i.e., to confirm reactor trip and not to provide long term monitoring during accident conditions). Due to the extensive calibration surveillance requirements of the Reactor Trip System Technical Specifications (daily, monthly, and quarterly adjustments to the indicators), in addition to the 18 month Channel Calibration, the Westinghouse power range instrumentation is highly reliable and sufficiently accurate to accomplish the required PAM function. In addition, the Westinghouse power range indications provide the primary indication of reactor power in the control room during Modes 1 and 2, which are the proposed Modes of applicability for this PAM indication. Therefore, in order to make the Unit 1 and 2 PAM Technical Specifications consistent and to use the most accurate and reliable indication to satisfy the required PAM function, the Westinghouse power range instrumentation is included in the proposed PAM Technical Specifications for both units. The status (not included in the Technical Specifications) and RG 1.97 classification of the Unit 2 Gamma-Metrics instrumentation will remain unchanged.

The Power Range Neutron Flux (Unit 1) and Neutron Flux Upper Range (Unit 2) indication is currently classified as a Regulatory Guide 1.97

Type B, Category 1 variable for BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the Power Range Neutron Flux indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 1 variable for both BVPS Unit 1 and Unit 2.

The Power Range Neutron Flux indication provides information to verify automatic actuation of RPS. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The Power Range Neutron Flux indication provides direct information to verify the accomplishment of the subcriticality safety function. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is also considered a Category 1 variable.

Containment Pressure indication:

The containment pressure indication is provided for assessing inadequate containment cooling and for determining the potential challenge to the containment pressure integrity. The wide range containment pressure instrumentation provides an adequate range and sensitivity for this purpose. Other containment pressure instrumentation does not extend sufficiently beyond the design basis pressure, and therefore does not have an adequate range to provide the required indication.

The containment pressure indication is used as an indicator of the potential loss of a fission product boundary in the Emergency Action Levels in the E-Plan for BVPS Units 1 and 2. Containment pressure is a key indicator in the declaration of a General Emergency level and the potential need for offsite radiological protection actions. Therefore, the containment wide range pressure instrumentation satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The Containment Pressure (Wide Range) indication is currently classified as a Regulatory Guide 1.97 Type C, Category 1 variable for BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the Containment

Pressure (Wide Range) indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type C and Category 1 variable for both BVPS Unit 1 and Unit 2.

The Containment Pressure (Wide Range) indication provides information to identify a fission product barrier challenge. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type C variable. The Containment Pressure (Wide Range) indication provides direct verification of containment cooling to maintain the containment fission product barrier safety function. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is also considered a Category 1 variable.

<u>Containment Radiation indication</u>:

The containment high range area radiation monitors provide an indication of a loss of one or more fission product barriers. Other containment radiation instrumentation is available to indicate radiation levels during normal plant operation and to provide an indication of fission product particulates.

The Emergency Action Levels in the E-Plan for BVPS Units 1 and 2 utilize the containment high range radiation monitors as an indication of a loss of one or more fission product barriers in the assessment of the declaration of a General Emergency level and the potential need for offsite radiological protection actions. The BVPS Unit 1 and 2 Core Damage Assessment also uses the containment high range radiation monitors as an input to the determination of core damage. The containment high range radiation monitors provide an adequate range and sensitivity for the determination of core damage. The containment high range radiation monitor satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The Containment Area Radiation (High Range) indication is currently classified as a Regulatory Guide 1.97 Type A, Category 1 variable for BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the Containment Area Radiation (High Range) indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type C and Category 1 variable for both BVPS Unit 1 and Unit 2.

The Containment Area Radiation (High Range) indication provides information to identify a fission product barrier challenge. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type C variable. The Containment Area Radiation (High Range) indication provides direct verification of the core cooling safety function. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is also considered a Category 1 variable.

<u>Containment Isolation indication</u>:

The purpose of the Penetration Flow Path Containment Isolation Valve position indication is to provide a direct indication of a failure to completely isolate containment following the receipt of a containment isolation signal.

Emergency Action Levels in the E-Plan for BVPS Units 1 and 2 utilize the Containment Isolation Valve Position Indication as an indication of a potential loss of one or more fission product barriers in the assessment of the declaration of a General Emergency level and the potential need for offsite radiological protection actions. Therefore, the Containment Isolation Valve Position Indicator satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

Containment Isolation Valve Position indication is currently classified as a Regulatory Guide 1.97 Type B, Category 1 for BVPS Unit 1 and Type C, Category 2 for Unit 2. Based on the methodology in WCAP-15981, the Containment Isolation Valve Position indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 1 variable for both BVPS Unit 1 and Unit 2.

Containment Isolation Valve Position indication provides verification of the automatic actuation of Phase A and Phase B containment isolation. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The Penetration Flow Path Containment Isolation Valve Position indication provides direct verification of containment isolation to maintain the containment fission product barrier safety function. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 1 variable.

RWST Level indication:

The RWST level (Wide Range for Unit 2) instrumentation provides an indication of the need to initiate make-up to the RWST to maintain long term cooling. The RWST level instrumentation provides an adequate range and sensitivity for this purpose.

The BVPS Unit 1 and 2 PRAs indicates that make-up to the RWST to provide long term core cooling for the steam generator tube rupture and interfacing system LOCA (ISLOCA) accidents are risk significant operator actions that are keyed from the RWST level instrumentation. Therefore, the RWST level instrumentation satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The RWST level indication is currently classified as a Regulatory Guide 1.97 Type A, Category 1 variable for BVPS Unit 1 and Type D, Category 2 variable for Unit 2. Based on the methodology in WCAP-15981, the RWST level indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type D and Category 1 variable for both BVPS Unit 1 and Unit 2.

The RWST level indication provides information to indicate the continued operation of safety injection for continued inventory control. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type D variable. The RWST level indication provides information to indicate the need to refill the RWST to continue inventory control for SGTR and ISLOCA events. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 1 variable.

High Head Safety Injection (SI) Flow indication:

The High Head SI Flow instrumentation provides an indication of the total SI flow to allow the diagnosis of the need for operator actions to manually initiate a safety injection signal. The BVPS Unit 1 and 2 PRAs identify that manual SI initiation, as diagnosed from the High Head SI Flow rate instrumentation, is a risk significant operator action. Therefore, the High Head SI Flow instrumentation satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The High Head SI Flow indication (total SI flow) is currently classified as Regulatory Guide 1.97 Type D, Category 2 for BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the High Head SI Flow indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 1 variable for both BVPS Unit 1 and Unit 2.

The High Head SI Flow indication provides information for the verification of automatic actuation of SI. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The High Head SI Flow indication provides direct information to verify the operation of safety injection to maintain the inventory safety function for core cooling. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 1 variable.

Auxiliary Feedwater Flow Indication:

The Auxiliary Feedwater Flow instrumentation provides the most direct indication of auxiliary feedwater flow to allow the diagnosis of the need for operator actions to initiate an alternative source of feedwater in order to assure a long term heat sink via the steam generators. An alternate indication would be the SG level instrumentation.

The BVPS Unit 1 and 2 PRAs identify that operator actions to initiate alternate feedwater sources to maintain a steam generator heat sink are risk significant operator actions. The success of these actions as modeled in the PRAs requires the prompt diagnosis of the loss of auxiliary feedwater capability. The Auxiliary Feedwater Flow Rate instrumentation provides the most direct indication of this condition. The use of SG level instrumentation would delay the operator actions as modeled in the PRAs and is therefore not the preferred indication for this Function. Therefore, the Auxiliary Feedwater Flow Rate instrumentation satisfies Criterion 4 of 10 CFR 50.36 (c)(2)(ii).

The Auxiliary Feedwater Flow Rate indication is currently classified as a Regulatory Guide 1.97 Type D, Category 2 for BVPS Unit 1 and Type A, Category 1 for Unit 2. Based on the methodology in WCAP-15981, the Auxiliary Feedwater Flow Rate indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 1 variable for both BVPS Unit 1 and Unit 2.

The Auxiliary Feedwater Flow Rate indication provides information used for the verification of the automatic actuation of AFW. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The Auxiliary Feedwater Flow Rate indication provides the direct verification of the heat sink safety function. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is also considered a Category 1 variable.

Based on the application of the WCAP-15981 methodology discussed above, the BVPS instrumentation proposed to be included in the BVPS PAM Technical Specification (3/4.3.3.8) is summarized below.

Pressurizer Water Level
Auxiliary Feedwater Flow Rate
Power Range Neutron Flux
High Head Safety Injection (SI) Flow
Steam Generator (SG) Pressure
Refueling Water Storage Tank Level (for Unit 2 only (Wide Range))
Reactor Coolant System Pressure (Wide Range)

SG Water Level (Wide Range)
Containment Area Radiation (High Range)
Containment Pressure (Wide Range)
Core Exit Temperature
Penetration Flow Path Containment Isolation Valve Position

Table 1 in Attachment D of this LAR provides a summary of the BVPS PAM instrumentation proposed to be included in Technical Specification 3/4.3.3.8, as well as the current Regulatory Guide 1.97 classification, and the results of the WCAP-15981 evaluation of the instrumentation.

4.8 Alternate PAM Indications to the Proposed BVPS PAM Technical Specification 3/4.3.3.8 Instrumentation

The following PAM indications that are proposed to be included in Technical Specification 3/4.3.3.8 contain alternate indications which are discussed below. The acceptable alternate indications for the applicable PAM Functions are also identified in the Bases for PAM Technical Specification 3/4.3.3.8.

Power Range Neutron Flux

Two channels of power range neutron flux are specified for PAM purposes to indicate criticality immediately following an accident or the receipt of a reactor trip signal. For the purposes of providing an indication of the failure to achieve subcriticality, which would result in operator actions to manually trip the reactor, the power range neutron flux monitors are the primary control room indication relied on for this information.

If the power range neutron flux indications are not available, an alternate method of verifying a reactor trip is a combination of either the intermediate range or source range neutron flux indications and either the rod bottom lights or rod position indicators. These alternate indications are acceptable because they can provide the necessary information to the operators so that the need for a manual reactor trip can be determined.

High Head SI Flow

There is one Regulatory Guide 1.97 instrument channel of High Head SI flow installed in the automatic SI flow path that provides an indication of total SI flow in the control room. The total flow indication supports the diagnosis of the need for operator actions to manually initiate an SI signal or start the high head SI pumps if automatic actuation fails.

Unit 1 also has high head SI branch line flow indication in the control room. However, only the high head SI total flow instrumentation is classified as Regulatory Guide 1.97 instrumentation. The Unit 1 branch line flow instrumentation (with transmitters located in containment) was never classified as Regulatory Guide 1.97 instrumentation and therefore, was not considered for inclusion in PAM Technical Specification 3/4.3.3.8 or for use as an alternate indication.

An alternate method of verifying SI initiation can be provided by the High Head SI pump amperage indication, the High Head SI header pressure indication, and the SI automatic valve position indication. The combination of these alternate indications are acceptable because they can provide an adequate verification of automatic SI initiation.

SG Level (Wide Range)

There is only one channel of wide range SG level instrumentation installed on each SG for indicating the SG level required to maintain a heat sink and for the diagnosis of a SGTR accident. The indication for the initiation of bleed and feed requires that the SG indicate a very low level.

If a channel of wide range SG level instrumentation is not available, an alternate method of monitoring the SG level is a combination of one channel of SG narrow range instrumentation and Auxiliary Feedwater Flow Rate indication to that SG. These alternate indications are acceptable because they are adequate to infer that an inventory (heatsink) is available in an intact SG.

Auxiliary Feedwater Flow

The Auxiliary Feedwater Flow instrumentation provides the most direct indication of auxiliary feedwater flow to allow the diagnosis of the need for operator actions to manually start the auxiliary feedwater pumps to initiate an alternate source of feedwater. The risk significant action is to provide an alternate SG feed source if no AFW pumps are available. BVPS Unit 1, has only one channel of auxiliary feedwater flow instrumentation per steam generator. BVPS Unit 2 has two channels of auxiliary feedwater flow rate indication per steam generator.

Alternate methods of determining the need for operator action in the event that the auxiliary feedwater flow rate indication is not available can be provided by the AFW pump amperage instrumentation (motor-driven pumps), flow control valve position indication (SG supply), and automatic turbine steam supply valve position indication. These alternate indications are acceptable because they provide adequate indication to confirm AFW system initiation and assess the need to manually start a pump.

Containment Area Radiation (High Range)

Two channels of containment area radiation monitoring provide an indication of a loss of one or more fission product barriers.

An alternate method of monitoring is radiation monitor RM-1RM-201 (Unit 1) and 2RMR-RQ202B (Unit 2), or a portable radiation monitor could be used outside of containment to infer the order of magnitude of the level of radiation inside the containment, and therefore the state of the reactor core. The Core Damage Assessment methodology in WCAP-14696-A shows that the details of the accident sequence can account for differences in containment radiation levels that are an order of magnitude different. Radiation monitor RM-1RM-201 (Unit 1) and 2RMR-RQ202B (Unit 2), or a portable radiation monitor are capable of providing information for an order of magnitude estimate. These alternate indications are consistent with the current Technical Specification allowance for the use of alternate radiation monitoring instrumentation and the current implementing plant procedures.

4.9 Discussion of the Relocation of the BVPS Regulatory Guide 1.97 Instrumentation Contained in Current Technical Specification 3/4.3.3.8.

The basis for relocating the following BVPS Regulatory Guide 1.97 PAM indications contained in current Technical Specification 3/4.3.3.8 to the LRM is discussed below.

The 10 CFR 50.36 (c)(2)(ii) Criteria for the evaluation of Technical Specification requirements follow:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Criterion 1 applies to instrumentation used to detect RCS leakage and is satisfied by the instrumentation included in the RCS Leakage Detection Instrumentation Technical Specification. Criterion 2 applies to a process variable, design feature, or operating restriction that must be maintained within limits by a Technical Specification requirement to preserve an initial condition assumed in a design basis accident. The purpose of PAM instrumentation is to function in a post accident environment to provide indications necessary for operators to take manual actions to mitigate the consequences of an accident. PAM instrumentation may also include indications that have been determined to be risk significant. Therefore, only Criteria 3 and 4 of 10 CFR 50.36 (c)(2)(ii) are applicable when evaluating instruments for retention in the PAM Technical Specification.

The corresponding ISTS 3.3.3, "Post Accident Monitoring (PAM) Instrumentation" Bases explains that all Regulatory Guide 1.97 Type A and all Regulatory Guide 1.97 Category 1 instrumentation must be included in the PAM Technical Specification. Therefore, in addition to evaluating each instrument against the 10 CFR 50.36 (c)(2)(ii) Criteria 3 and 4 discussed above, each instrument is evaluated based on the methodology in WCAP-15981 to determine the applicable Regulatory Guide 1.97 Type and Category. This evaluation is performed for the sole purpose of determining the most appropriate content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation".

Using the methodology of WCAP 15981, it has been determined that each PAM instrument proposed for relocation from the Technical Specifications does not fulfill the function of Regulatory Guide 1.97 for a Type A or Category 1 instrument. The evaluations performed in accordance with WCAP-15981 also support the conclusions that the affected instrumentation does not meet 10 CFR 50.36 (c)(2)(ii) Criterion 3 (i.e., it was not found to be a Type A instrument) or Criterion 4 (i.e., it was not found to be significant to risk).

The Technical Specification instruments that do not satisfy Criterion 3 or 4 of 10 CFR 50.36 (c)(2)(ii) are being relocated to the LRM. Technical Specifications relocated to the LRM are considered to be incorporated by reference in the BVPS Unit 1 and 2 UFSARs (as applicable). Therefore, changes to the relocated material will be controlled in the same manner as changes to the UFSAR, i.e., in accordance with 10 CFR 50.59. As such, relocation of Technical Specification requirements to the LRM is acceptable as changes to these documents will be adequately controlled by 10 CFR 50.59. The provisions of 10 CFR 50.59 establish adequate controls over requirements removed from the TS, and assure future changes to these requirements will continue to be consistent with safe plant operation.

The Unit 1 and 2 PAM Functions selected for relocation to the LRM will be relocated with the associated Technical Specification 3/4.3.3.8 requirements (i.e., LCO, Actions, and Surveillances) together to form a complete requirement in the LRM.

The Unit 1 and Unit 2 Reactor Coolant System Subcooling Margin Monitor contained in Tables 3.3-11 and 4.3-7 (Instrument 3) of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation."

The RCS subcooling indication provides information to the control room operators related to satisfying one of the SI termination criteria following a steam line break accident. The inputs to the RCS subcooling monitor are the core exit thermocouples for RCS temperature and the wide range RCS pressure indication for RCS pressure. Since both of these indications are independently displayed in the control room and are also included in proposed PAM Instrumentation Technical Specification 3/4.3.3.8, the RCS subcooling monitor only provides a verification of these other primary indications. Therefore, the RCS subcooling monitor does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii), and should not be included in PAM Technical Specification 3/4.3.3.8.

The RCS subcooling indication is currently classified as Regulatory Guide 1.97 Type B, Category 2 for BVPS Unit 1 and Type A, Category 2 for Unit 2. Based on the methodology in WCAP-15981, the RCS subcooling indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 3 variable for both BVPS Unit 1 and Unit 2.

The RCS subcooling indication provides information to indicate whether the core cooling safety function is being accomplished. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The RCS subcooling indication is a backup to the core exit thermocouples and RCS pressure. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 3 variable.

The Unit 1 and Unit 2 PORV Limit Switch Position Indicator contained in Tables 3.3-11 and 4.3-7 (Instrument 5 (Unit 1) and 4 (Unit 2)) of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation."

The PORV Limit Switch Position Indicators provide information to the control room operators related to the position of the pressurizer PORVs. It could be used to diagnose a high RCS pressure or a stuck open PORV (LOCA) at lower RCS pressures. The DBA analysis of an inadvertent opening of the PORV does not rely on operator diagnosis and closure of the PORV or block valve; the DBA analysis assumes that automatic safety injection actuation will provide adequate protection. Since the PORV Limit Switch Position indicator does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The PORV Limit Switch Position Indicator is currently classified as Regulatory Guide 1.97 Type D, Category 2 for BVPS Unit 2 and was not classified as a Regulatory Guide 1.97 instrument for BVPS Unit 1. Based on the methodology in WCAP-15981, the PORV Limit Switch Position Indicator was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type D and Category 2 variable for both BVPS Unit 1 and Unit 2.

The PORV Limit Switch Position Indicator provides information on the operating status for a system important to safety, but not identified as risk significant. The results of the WCAP evaluation are consistent with the Unit 2 Regulatory Guide 1.97 classification of this instrumentation (Type D and Category 2). For the purpose of determining the content of the Unit 1 Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is also considered a Type D, Category 2 variable.

The Unit 1 and Unit 2 PORV Block Valve Limit Switch Position Indicator contained in Tables 3.3-11 and 4.3-7 (Instrument 6 (Unit 1) and 5 (Unit 2)) of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation."

The PORV Block Valve Limit Switch Position Indicator provides information to the control room operators on the position of the pressurizer PORV block valves. It could be used to diagnose the availability of the pressurizer PORVs for use in depressurizing the RCS or to indicate the isolation of a stuck open PORV (LOCA) at lower RCS pressures. Since the PORV Block Valve Limit Switch Position Indicator does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The PORV Block Valve Limit Switch Position Indicator was not classified as Regulatory Guide 1.97 instrumentation for BVPS Unit 1 and 2. Based on the methodology in WCAP-15981, the PORV Block Valve Limit Switch Position Indicator was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type D and Category 2 variable for both BVPS Unit 1 and Unit 2.

The PORV Block Valve Limit Switch Position Indicator provides information on the operating status of a system important to safety, but not identified as risk significant. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type D, Category 2 variable for both BVPS Unit 1 and Unit 2.

The Unit 1 and Unit 2 Safety Valve (Acoustical Detector (Unit 1) Position Indicator contained in Tables 3.3-11 and 4.3-7 (Instrument 7 (Unit 1) and 6 (Unit 2)) of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation."

The Safety Valve (Unit 2) and Acoustical Detector (Unit 1) Position Indicator provides information to the control room operators on the position of the pressurizer safety valves. It could be used to diagnose high RCS pressure or a stuck open safety valve (LOCA) at lower RCS pressures. Since the Position Indicator does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The Safety Valve (Unit 2) and Acoustical Detector (Unit 1) Position Indicator is currently classified as Regulatory Guide 1.97 Type D, Category 2 for both BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the Position Indicator was also determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type D and Category 2 variable for both BVPS Unit 1 and Unit 2.

The Safety Valve (Unit 2) and Acoustical Detector (Unit 1) Position Indicator provides information on the operating status of a system important to safety, but not identified as risk significant. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type D, Category 2 variable.

The Unit 1 and Unit 2 Containment Sump Wide Range Water Level indication contained in Tables 3.3-11 and 4.3-7 (Instrument 10 (Unit 1) and 8 (Unit 2)) of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation."

The Containment Sump Wide Range Water Level indication provides information to indicate whether sufficient water is available in the containment sump at the time that the recirculation spray system is automatically started. It also provides an indication of excessive containment water levels that could result in the flooding of key equipment and instrumentation. Part of the Unit 2 recirculation spray system is switched-over to provide long term low head emergency core cooling in the recirculation mode of operation when the RWST level reaches the extreme low level setpoint. All actions associated with recirculation spray system are automatic and therefore no operator action is required in the design basis analyses based on containment sump level. Since the Containment Sump Wide Range Water Level indication does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The Containment Sump Wide Range Water Level indication is currently classified as Regulatory Guide 1.97 Type A, Category 1 for BVPS Unit 1 and Unit 2. Although there are differences in the design for the recirculation spray and Emergency Core Cooling recirculation operation between BVPS Units 1 and 2 and the evaluation contained in WCAP-15981, the generic conclusions contained in WCAP-15981 are applicable to BVPS Units 1 and 2. Based on the methodology in WCAP-15981, the Containment Sump Wide Range Water Level indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 2 variable for both BVPS Unit 1 and Unit 2.

The Containment Sump Wide Range Water Level indication provides information to indicate whether the core cooling safety function can be accomplished when the Emergency Core Cooling System switchover to the recirculation mode of operation occurs. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The Containment Sump Wide Range Water Level indication provides information on the status of safety injection from the RWST. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 2 variable.

The Reactor Vessel Level Indicating (Unit 1) Indication (Unit 2) System contained in Tables 3.3-11 and 4.3-7 (Instrument 12 (Unit 1) and 10 (Unit 2)) of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation."

The Reactor Vessel Level Indicating (Unit 1) Indication (Unit 2) System provides information to the control room operators on the presence of voids in the reactor vessel head. This information is used in the plant EOPs as an indication of inadequate core cooling and as an indication of the potential for a void formation that can impede natural circulation cooling. The potential for void formation, which can be diagnosed using Reactor Vessel Water Level is not modeled in the BVPS PRAs based on the definition of primary and secondary success paths for all possible initiating events. Since Reactor Vessel Water Level is not modeled in the BVPS PRAs, it is not risk significant. Since the instrument functions described above do not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, Reactor Vessel Level Indicating (Unit 1) Indication (Unit 2) System does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The Reactor Vessel Level Indicating (Unit 1) Indication (Unit 2) System is currently classified as Regulatory Guide 1.97 Type B, Category 1 for BVPS Unit 1 and Type B, Category 2 for Unit 2. Based on the methodology in WCAP-15981, the Reactor Vessel Level Indicating (Unit 1) Indication (Unit 2) System was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 3 variable for both BVPS Unit 1 and Unit 2.

The Reactor Vessel Level Indicating (Unit 1) Indication (Unit 2) System provides information to indicate whether the core cooling safety function is being accomplished. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The Reactor Vessel Level Indicating (Unit 1) Indication (Unit 2) System is a backup to the core exit thermocouples for this purpose. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 3 variable. However, since it does not provide an indication for transition from the EOPs to SAMG, it is not considered an alternate indication for the core exit themocouples.

4.10 Discussion of BVPS Regulatory Guide 1.97 Type A Instrumentation Not Contained in The Current Technical Specifications.

The basis for not including the following current BVPS Regulatory Guide 1.97 Type A PAM instruments in the proposed Technical Specification 3/4.3.3.8 is discussed below.

RCS T Hot (Wide Range)

The RCS T Hot Wide Range indication provides information to indicate the temperature of the RCS hot leg fluid. The RCS T Hot Wide Range indication can be used by the plant operators to verify adequate core cooling, RCS subcooling, and in conjunction with the RCS cold leg temperature indication, the effectiveness of RCS heat removal by the secondary system. However, the RCS T Hot Wide Range indication is not the primary indication used by the plant operators for any of those determinations. Since the RCS T Hot Wide Range indication does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The RCS T Hot Wide Range indication is currently classified as Regulatory Guide 1.97 Type A, Category 1 for both BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the RCS T Hot Wide Range indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 3 variable for both BVPS Unit 1 and Unit 2.

The RCS T Hot Wide Range indication provides information to indicate whether the core cooling safety function is being accomplished. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The RCS T Hot Wide Range indication is a backup to the core exit thermocouples for indicating that the core cooling safety function is being accomplished. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 3 variable.

RCS T Cold (Wide Range)

The RCS T Cold Wide Range indication provides information to indicate the temperature of the RCS cold leg fluid. The RCS T Cold Wide Range indication can be used by the plant operators, in conjunction with the RCS hot leg temperature indication, to verify the effectiveness of RCS heat removal by the secondary system. However, the RCS T Cold Wide Range indication is not the primary indication used by the plant operators for that determination. Since the RCS T Cold Wide Range indication does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The RCS T Cold Wide Range indication is currently classified as Regulatory Guide 1.97 Type A, Category 1 for both BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the RCS T Cold Wide Range indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 3 variable for both BVPS Unit 1 and Unit 2.

The RCS T Cold Wide Range indication provides information to indicate whether the core heat removal safety function is being accomplished. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The RCS T Cold Wide Range indication is also used as a diagnostic indication. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 3 variable.

SG Level (Narrow Range)

The SG Level Narrow Range indication provides information to indicate the level in the steam generator (SG) in the range of interest for maintaining an effective heat sink. The SG Level Narrow Range indication is used by the plant operators to control SG level in the required range for effective decay heat removal for design basis accidents. Since the SG Level Narrow Range indication provides an indication for operator actions for which no automatic control is provided and it is important from a risk perspective, it satisfies both Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii). However, the SG Level Wide Range indication can also be used for this purpose. In addition to level control for decay heat removal, the SG Level Wide Range indication is also used to implement operator actions for the loss of SG heat sink, for which the SG Level Narrow Range indication does not have a sufficient range of indication. Since the SG Level Wide Range indication has been determined to also meet both Criterion 3 and 4 of 10 CFR 50.36 (c)(2)(ii) for the indication of a loss of SG heat sink and it can also serve to indicate SG level for effective heat removal, it is the preferred SG level indication to be included in the PAM Technical Specification. Since the SG Level Narrow Range indication does not have the required range of indication to accomplish all the functions of the SG Level Wide Range indication, it is not included in PAM Technical Specification 3/4.3.3.8.

The SG Level Narrow Range indication is currently classified as Regulatory Guide 1.97 Type A, Category 1 indication for both BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the SG Level Narrow Range indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 2 variable for both BVPS Unit 1 and Unit 2.

The SG Level Narrow Range indication provides information to indicate whether the SG heat sink safety function is being accomplished. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The SG Level Narrow Range indication also provides information on the status of SG feedwater delivery. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 2 variable.

Containment Pressure (Narrow Range)

The Containment Pressure Narrow Range indication provides information to indicate the containment pressure in the range of interest following a design basis accident. It is used by the plant operators to verify the operation of the quench spray system and that the containment pressure has been reduced following a design basis accident. It can also be used by the plant operators to verify that the service water system is adequately removing decay heat for a design basis accident and that the EQ envelope for equipment has not been exceeded. Since the Containment Pressure Narrow Range indication does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The Containment Pressure Narrow Range indication is currently classified as Regulatory Guide 1.97 Type A, Category 1 for both BVPS Unit 1 and Unit 2. Based on the methodology in WCAP-15981, the Containment Pressure Narrow Range indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 3 variable for both BVPS Unit 1 and Unit 2.

The Containment Pressure Narrow Range indication provides information to indicate whether the containment cooling safety function is being accomplished. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B. The Containment Pressure Narrow Range indication is also a backup to the containment pressure wide range indication. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 3 variable.

Containment Water Level (Narrow Range, Unit 2)

The Containment Water Level Narrow Range indication provides information to indicate whether sufficient water is available in the containment sump when the recirculation spray system is automatically started. A portion of the recirculation spray system is switched-over to provide long term low head emergency core cooling when the RWST level reaches the extreme low level setpoint. All actions associated with recirculation spray system are automatic, and therefore no operator action is required in the design basis analyses based on the containment sump level. Since the Containment Water Level Narrow Range indication does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The Unit 2 Containment Water Level Narrow Range indication is currently classified as Regulatory Guide 1.97 Type A, Category 1. Although there are differences in the design of the recirculation spray and Emergency Core Cooling System recirculation between BVPS Unit 2 and the evaluation contained in WCAP-15981, the generic conclusions contained in WCAP-15981 are applicable to BVPS Unit 2. Based on the methodology in WCAP-15981, the Containment Water Level Narrow Range indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 2 variable for BVPS Unit 2.

The Containment Water Level Narrow Range indication provides information to indicate whether the core cooling safety function can be accomplished when RWST switchover occurs. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The Containment Water Level Narrow Range indication provides information on the status of safety injection recirculation delivery. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 2 variable.

Primary Plant DWST Level (AFW System Supply)

The Primary Plant Demineralized Water Storage Tank (DWST) Level indication is used for the diagnosis of the need to refill the Primary Plant DWST to provide a long term steam generator heat sink for decay heat removal. The Primary Plant DWST refill is a long-term action that is not credited in the UFSAR analyses. Additionally, the PRA does not identify that the Primary Plant DWST refill is a risk significant operator action. In addition, there are manual means of obtaining information on the Primary Plant DWST level that can be easily implemented given the time available. Therefore, the Primary Plant DWST Level indication does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The Primary Plant DWST Level indication is currently classified as Regulatory Guide 1.97 Type A, Category 1 for BVPS Unit 2 and Type D, Category 1 for Unit 1. Based on the methodology in WCAP-15981, the Primary Plant DWST Level indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 2 variable for both BVPS Unit 1 and Unit 2.

The Primary Plant DWST Level indication provides information to indicate whether a continued SG heat sink can be maintained. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The Primary Plant DWST Level indication also provides information for the long term AFW system operating status. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 2 variable.

A summary of the current BVPS Regulatory Guide 1.97 Type A PAM indications not included in proposed Technical Specification 3/4.3.3.8 and the results of the WCAP-15981 evaluation of these indications is contained in Table 1 in Attachment D of this LAR.

4.11 Discussion of the BVPS Regulatory Guide 1.97 Category 1 Instrumentation Not Contained in the Current Technical Specifications.

The basis for not including the following BVPS Regulatory Guide 1.97 Category 1 PAM indications in proposed Technical Specification 3/4.3.3.8 is discussed below.

Neutron Flux Source Range (Unit 1) and Neutron Flux Lower Range (Unit 2)

After subcriticality is achieved, the source/lower range neutron flux can be used to confirm continued subcriticality by monitoring the startup rate. A positive startup rate indicates that criticality is being approached. The source/lower range neutron flux can be used as a backup to the power range neutron flux during shutdown to determine whether sufficient negative reactivity (e.g., boron, RCS temperature during RCS cooldown) is available for long term subcriticality. However, the long-term confirmation of subcritical operation is not shown to be important to risk in the BVPS Unit 1 or Unit 2 PRAs. Since the Neutron Flux Source Range (Unit 1) and Neutron Flux Lower Range (Unit 2) do not provide an indication for operator actions for which no automatic control is provided and are not important from a risk perspective, they do not satisfy either Criterion 3 or 4 of 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The Neutron Flux Source Range (Unit 1) and Neutron Flux Lower Range (Unit 2) are currently classified as Regulatory Guide 1.97 Type B, Category 1 for both BVPS units. Based on the methodology in WCAP-15981, the Neutron Flux Source Range (Unit 1) and Neutron Flux Lower Range (Unit 2) were determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 3 variable for both BVPS Unit 1 and Unit 2.

The source/lower range neutron flux provides a verification of the automatic actuation of the RTS. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The source/lower range neutron flux also provides diagnostics for maintaining subcriticality during an RCS cooldown and depressurization. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 3 variable.

Neutron Flux Intermediate Range (Unit 1)

At power levels below approximately 5% of full power, the intermediate range neutron flux monitor can be used to confirm an automatic reactor trip by monitoring the startup rate. A positive startup rate indicates that subcriticality has not been achieved. However, the long term confirmation of subcritical operation is not shown to be important to risk in the BVPS Unit 1 or Unit 2 PRAs. The intermediate range neutron flux monitor can be used as a backup to the power range neutron flux monitor at low power levels to indicate incomplete rod insertion. Since the Neutron Flux Intermediate Range monitor does not provide an indication that is used for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The Neutron Flux Intermediate Range monitor is currently classified as Regulatory Guide 1.97 Type B, Category 1 indication for BVPS Unit 1. Unit 2 does not identify a separate intermediate range neutron flux variable in the RG 1.97 instrumentation. Based on the methodology in WCAP-15981, the Neutron Flux Intermediate Range monitor was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type B and Category 3 variable.

The intermediate range neutron flux monitor provides a verification of the automatic actuation of RTS. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type B variable. The intermediate range neutron flux monitor also provides diagnostics for monitoring the continued subcriticality during an RCS cooldown and depressurization. Therefore, for the purpose of determining the content of

Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 3 variable.

Radiation Level in Primary Coolant (Unit 1)

The Radiation Level in the Primary Coolant is provided by the RCS letdown radiation monitor. These monitors are located in the letdown line, which is isolated upon the receipt of a safety injection signal. As discussed in WCAP-14986-A Revision 2, the reactor coolant radiation level is only important for design basis accidents where there is fuel rod cladding damage without coincident core overheating, such as local reactivity events caused by the withdrawal of a single rod control cluster assembly (RCCA). For these events, the reactor is tripped and shutdown by the Reactor Trip System. The letdown radiation monitor indication would be used by the plant operators to decide whether the declaration of an Unusual Event condition was appropriate. However, this determination is not shown to be important to risk in the BVPS Unit 1 or Unit 2 PRAs. Since the letdown radiation monitor indication does not provide an indication for operator actions for which no automatic control is provided and it is not important from a risk perspective, it does not satisfy either Criterion 3 or 4 of 10 CFR 50.36(c)(2)(ii) and should not be included in PAM Technical Specification 3/4.3.3.8.

The letdown radiation monitor indication is currently classified as Regulatory Guide 1.97 Type C, Category 1 for BVPS Unit 1. The corresponding variable in Unit 2 was never classified as either Type A or Category 1. Based on the methodology in WCAP-15981, the letdown radiation monitor indication was determined to fulfill the post accident monitoring function of a Regulatory Guide 1.97 Type C and Category 3 variable (i.e., the same classification as currently assigned the corresponding Unit 2 radiation monitor).

The Radiation Level in the Primary Coolant provides information to indicate the potential for a breach of the fuel cladding fission product barrier. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Type C variable. The Radiation Level in the Primary

Coolant also provides diagnostic indications of core damage not associated with core overheating. Therefore, for the purpose of determining the content of Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation" this variable is considered a Category 3 variable.

Primary Plant DWST Level (AFW System Supply) (Unit 1)

See discussion of Primary Plant DWST Level (AFW System Supply) in Section 4.10 which addresses both Units.

A summary of the current BVPS Regulatory Guide 1.97 Category 1 PAM indications not included in proposed Technical Specification 3/4.3.3.8 and the results of the WCAP-15981 evaluation of these indications is contained in Table 1 in Attachment D of this LAR.

Change No. 11

Administrative Changes

In addition to the technical changes described above, various editorial and format changes are made to the Technical Specifications. These editorial and format changes are made to accommodate the technical changes described and justified above. In addition, format and presentation changes are made to conform more closely with the corresponding requirements in the ISTS.

The editorial changes include such things as repagination, changes to the index pages to show deletions or name changes as well as the insertion of text such as "deleted" or "this Action not used." These changes are necessary due to renamed, renumbered, reformatted, or relocated Technical Specification requirements. In some cases, these changes are necessary to preserve the format and presentation conventions used in the current Technical Specifications when material is removed from the Specifications.

In some cases, the presentation of the technical requirements is revised based on the presentation used in the ISTS. This type of change includes such things as consolidation of the PAM requirements in one table with a single set of surveillance requirements and the addition of notes to explain the application of the surveillances to the list of Functions on the single table. These changes do not affect the technical requirements of the specification and are made solely to conform more closely to the ISTS.

These changes are acceptable because they are administrative in nature and do not introduce additional technical changes (beyond those described and justified above) to the Technical Specifications.

5.0 REGULATORY SAFETY ANALYSIS

This License Amendment Request (LAR) proposes changes that will revise the instrumentation contained in the Unit 1 and Unit 2 Technical Specifications (TS) 3/4.3.3.8, "Accident Monitoring Instrumentation," to be consistent with the technical basis for accident monitoring instrumentation identified in WCAP-15981. This change includes reevaluating the current Regulatory Guide 1.97 classification of the affected instrumentation with respect to its function as a post accident monitoring instrument based on WCAP-15981. The evaluations contained in this LAR support the identification of the most appropriate instrumentation for inclusion in Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation," consistent with the methodology contained in WCAP-15981. The proposed changes do not revise the current BVPS Unit 1 or 2 response to Regulatory Guide 1.97 and do not include any plant design changes.

The changes proposed by this LAR will make the content of the current TS 3/4.3.3.8, "Accident Monitoring Instrumentation," more consistent with the requirements of the corresponding Improved Standard Technical Specifications (ISTS) 3.3.3, "Post Accident Monitoring Instrumentation."

Additionally, the LAR proposes changes that include the relocation of the alarm function requirements for the Unit 1 and Unit 2 containment area radiation monitor and the Unit 2 main steam discharge process radiation monitors contained in Technical Specification 3/4.3.3.1, "Radiation Monitoring Instrumentation," to the Licensing Requirements Manual and Offsite Dose Calculation Manual, respectively. The applicable indication function requirements for the Unit 1 and Unit 2 containment area radiation monitors will be moved to Technical Specification 3/4.3.3.8, "Accident Monitoring Instrumentation." These radiation monitors are addressed in this LAR, since the only reason for retaining these monitors in the Technical Specifications is their Regulatory Guide 1.97

classification. The classification of the radiation monitors was re-evaluated in this LAR, along with the other post accident monitoring instruments in Technical Specification 3/4.3.3.8 consistent with WCAP-15981.

Additional changes to the Unit 1 and Unit 2 Technical Specifications are proposed in this LAR to provide consistency with the ISTS. Format and editorial changes are also included in this LAR that are necessary due to the deletion and addition of requirements and the moving of requirements.

This LAR also contains some additional changes beyond those made to conform to ISTS 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," as follows:

- The required number of core exit temperature channels is revised from 2 per core quadrant (as specified in ISTS 3.3.3) to 2.
- An Action has been added for two or more inoperable PAM Function channels that allows operation to continue, provided that a report is submitted to the Nuclear Regulatory Commission within 7 days. The report must identify the alternate method used to accomplish the affected PAM Function and the plans for restoring the inoperable channels to operable status.
- The surveillance for the Penetration Flow Path Containment Isolation Valve (CIV) Position Indication PAM is revised from a Channel Calibration to a Channel Functional Test.

5.1 <u>No Significant Hazards Consideration</u>

FirstEnergy Nuclear Operating Company (FENOC) has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

The proposed changes do not involve any changes to plant equipment, system design functions or a change in the methods governing normal plant operation. Therefore, the probability of a malfunction of a structure, system or component to perform its design function will not be significantly increased.

The proposed changes relocate specific requirements associated with the alarm function of radiation monitors, and Regulatory Guide 1.97 instrumentation that do not fulfill the functional requirements of Regulatory Guide 1.97 Type A or Category 1 instrumentation, as evaluated in accordance with the methodology of WCAP-15981. The requirements proposed to be relocated from the Technical Specifications have been found not to satisfy any of the Criteria of 10 CFR 50.36 c)(2)(ii). The proposed changes also make the Accident Monitoring Instrumentation Technical Specification more consistent with the corresponding Improved Standard Technical Specifications (ISTS) PAM Technical Specification. These changes include relaxations to the required actions for inoperable instrumentation and relaxations in required instrument surveillance intervals. The proposed changes are not initiating conditions for any accident previously evaluated. In addition, changes that are consistent with the ISTS have been evaluated and found not to adversely affect the safe operation of Westinghouse plants or initiate any accident previously evaluated. Based on the conclusions of the plant-specific evaluation associated with the changes and the evaluation performed in developing the ISTS, the proposed changes do not result in operating conditions that will significantly increase the probability of initiating an analyzed event. The PAM instruments that are assumed to provide the information for manual operator actions for which no automatic control is provided are retained in the Technical Specifications, and the PAM instruments which also serve this function are added to the PAM Technical Specification. As a result, the consequences of any accident previously evaluated are not significantly increased.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite or significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with the safety analysis assumptions and resultant consequences.

It is concluded that these changes do not significantly increase the probability of occurrence of a malfunction of equipment important to safety. Therefore, the proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed changes do not involve any changes to plant equipment, system design functions or the methods governing normal plant operation. No new accident initiators are introduced by these changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed changes do not result in a change in the manner in which the PAM instruments provide plant information. There are no design changes associated with the license amendment. The proposed changes do not change any existing accident scenarios, nor create any new or different accident scenarios.

   The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different operating requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analysis. The proposed changes are consistent with the safety analysis assumptions and current plant operating practice.

The possibility of a new or different malfunction of safety related equipment is not created. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The changes do not involve a significant reduction in the margin of safety. The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not impacted by these changes. The proposed changes do not include any plant design changes. The proposed changes will not revise the indication provided by the affected instruments, and all operator actions based on these indications that are credited in the accident analyses will remain the same. As such, the proposed changes will not result in plant operation in a configuration outside the design basis or assumptions of the design basis accident analyses.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, FENOC concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

In the following paragraphs applicable criteria as they are related to the proposed changes are discussed. A summary of the applicable criteria and assessment of the impact to the BVPS design conformance are provided in the following tables.

General Design Criteria (GDC)		Assessment
13	*Instrumentation and control*	No Impact
19	*Control room*	No Impact
64	*Monitoring radioactivity releases*	No Impact

10 CFR 50		Assessment
.49	*Environmental qualification of electric equipment important to safety for nuclear power plants*	No Impact
.36	*Technical specifications*	No Impact
.67	*Accident source term*	No Impact
.47	*Emergency plans*	No Impact

10 CFR Part 100	Assessment
Design basis accident dose guidelines	No Impact

GDC 13 includes a design requirement that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to ensure adequate safety. There is no impact on the requirement of GDC 13, since the proposed LAR does not include any plant design changes. The proposed LAR does not eliminate or otherwise alter any existing instrumentation.

GDC 19 includes a requirement that a control room be provided from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including loss-of-coolant accidents, and that equipment, including the necessary instrumentation, at appropriate locations outside the control room be provided with a design capability for prompt hot shutdown of the reactor. In addition, GDC 19 requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of the specified limits. There is no impact on the requirements of GDC 19, since the proposed LAR does not include any design changes or plant modifications. In addition, the LAR does not introduce any changes that could adversely affect the potential radiation exposure of control room personnel.

GDC 64 requires that means be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations and postulated accidents. There is no impact on the requirement of GDC 64, since the proposed changes do not modify the design or otherwise alter the current plant instrumentation used for monitoring radioactive releases. No monitoring instrumentation is being eliminated or modified due to the proposed changes.

10 CFR 50.49 specifies design and performance requirements for safety-related instrumentation exposed to adverse environments during accident conditions. The proposed changes do not impact the requirements of 10 CFR 50.49. The proposed changes do not introduce changes that affect the design or performance of the safety-related instrumentation subject to 10 CFR 50.49. There are no plant design changes or modifications associated with the proposed changes.

10 CFR 50.36 contains requirements applicable to the content of a plant's Technical Specifications. The proposed changes utilize the criteria of 10 CFR 50.36 c)(2)(ii) to evaluate the content of the PAM Technical Specification. The proposed change includes the relocation of certain instruments from the Technical Specifications that do not satisfy any of the criteria of 10 CFR 50.36 c)(2)(ii). In addition, the proposed change includes the addition of several instruments to the PAM Technical Specifications that have been determined to meet one or more criteria of 10 CFR 50.36 c)(2)(ii). The net effect of the proposed change is to increase the number of instruments in the PAM Technical Specification. As such, the proposed changes are consistent with the requirements of 10 CFR 50.36.

10 CFR Part 100 contains the design basis accident (DBA) dose guidelines. The proposed changes do not affect the BVPS compliance with the dose guideline values stated in 10 CFR Part 100. The proposed changes do not affect the assumptions of any radiological consequence analysis.

10 CFR 50.67, "accident source term," contains requirements and dose limits associated with the use of an alternate source term for design basis radiological consequence analyses. The proposed changes do not affect the requirements or

dose limits specified in 10 CFR 50.67. In addition, the proposed changes do not affect the assumptions of any radiological consequence analysis.

10 CFR 50.47, "Emergency Plans," contains requirements for Emergency Plans. The methodology in WCAP-15981 utilizes the Emergency Plan to determine the instrumentation that should be included in the Technical Specifications. Therefore, the proposed changes do not result in any changes to the Emergency Plan, and do not result in a decrease in the effectiveness of the Emergency Plan.

In summary, the proposed changes do not involve any design changes to the PAM instrumentation, or changes to the physical arrangement of PAM instrumentation. The proposed changes result in expanding the scope of the existing PAM Technical Specification consistent with the requirements of 10 CFR 50.36. Therefore, the proposed changes provide enhanced assurance that the required PAM Functions remain capable of performing their accident monitoring function. Thus, the proposed changes do not adversely impact the design or performance characteristics of the PAM system, or any other system.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980.

2. WCAP-15981, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," August 2004.

3. NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3, March 2004.

4. WCAP-14696-A, "Westinghouse Owners Group Core Damage Assessment Guidance," Revision 1, November 1999.

5. Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," December 1982.

6. WCAP-11618, "Methodically Engineered, Restructured and Improved, Technical Specifications, MERITS Program – Phase II Task 5 Criteria Application," November 1987.

7. NRC letter from T. E. Murley (NRC) to W. S. Wilgus (B&W Owners Group), "NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications," May 1988.

8. WCAP-14986-A, "Westinghouse Owners Group Post Accident Sampling System Requirements: A Technical Basis," Revision 2, July 2000.

Attachment A-1

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

License Amendment Request No. 314

The following is a list of the affected pages:

IV
XVII
3/4 3-33
3/4 3-34
3/4 3-35
3/4 3-36
3/4 3-50
3/4 3-51
3/4 3-52

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Post

Table Index (cont.)

Post

Reviewers Note: The requirements on this page remain in the Technical Specifications for Radiation Monitors not affected by this License Amendment Request (LAR). The relocation shown below is only applicable for the Containment Area Radiation Monitor Alarm Function being relocated by this LAR.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

Relocated to the Licensing Requirements Manual (LRM) to support the relocation of the Containment Area Radiation Monitor Alarm Function.

Reviewers Note: The markup on this page includes the changes associated with the relocation of the Containment Area Radiation Monitor Alarm Function to the LRM. The markups are not applicable to other Radiation Monitors shown on this page.

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-1 TO CLARIFY FINAL PAGE CONTENT

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

Relocated to the Licensing Requirements Manual (LRM) to support the relocation of the Containment Area Radiation Monitor Alarm Function.

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT(3)	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Deleted					
b. Containment					
i. Purge & Exhaust Isolation (RMVS 104 A & B)	2	(2)	$\leq 1.6 \times 10^3$ cpm	$10 - 10^6$ cpm	22
~~ii.~~ ~~Area (RM-RM-219 A & B)~~	~~2~~	~~1, 2, 3 & 4~~	~~$\leq 1.5 \times 10^4$ R/hr~~	~~$1 - 10^7$ R/hr~~	~~35~~
c. Control Room Isolation (RM-RM-218 A & B)	2	1,2,3,4 and (4)	$\leq$ 0.47 mR/hr	$10^{-2} - 10^3$ mR/hr	41
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection (RM 215B)	1	1,2,3 & 4	N/A	$10 - 10^6$ cpm	20
ii. Particulate Activity RCS Leakage Detection (RM 215A)	1	1,2,3 & 4	N/A	$10 - 10^6$ cpm	20
b. Deleted					

Reviewers Note: The markups on this page include the changes associated with the retention of the Containment Area Radiation Monitor Post Accident Monitoring (PAM) indication function in the PAM Technical Specification (3/4.3.3.8). Other Radiation Monitors on this page are not affected by the marked changes.

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-1 TO CLARIFY FINAL PAGE CONTENT

Reviewers Note: The unmarked requirements applicable to the Containment Area Radiation Monitors on this page are relocated as shown on the previous page.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	~~MINIMUM~~ REQUIRED CHANNELS ~~OPERABLE~~	APPLICABLE MODES	SETPOINT (3)	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Deleted					
b. Containment					
i. Purge & Exhaust Isolation (RMVS 104 A & B)	2	(2)	≤ 1.6 x 10^3 cpm	10 - 10^6 cpm	22
~~ii.~~ Area (RM-RM-219 A & B)	2	1,2,3 & 4	≤ 1.5 x 10^4 R/hr	1 - 10^7 R/hr	35
c. Control Room Isolation (RM-RM-218 A & B)	2	1,2,3,4 and (4)	≤ 0.47 mR/hr	10^{-2} - 10^3 mR/hr	41
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection (RM 215B)	1	1,2,3 & 4	N/A	10 - 10^6 cpm	20
ii. Particulate Activity RCS Leakage Detection (RM 215A)	1	1,2,3 & 4	N/A	10 - 10^6 cpm	20
b. Deleted					

Relocated to PAM Specification Bases

Moved to PAM Technical Specification 3/4.3.3.8

Replaced by PAM Specification Actions a and b in 3/4.3.3.8

Moved to PAM TS 3/4.3.3.8 and title changed to Containment Area Radiation (High Range)

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-1 TO CLARIFY FINAL PAGE CONTENT

Reviewers Note: The markup on this page includes the changes associated with the relocation of the Containment Area Radiation Monitor Alarm Function to the LRM. The markups are not applicable to other Radiation Monitors in this Specification.

Relocated to LRM to support relocation of Containment Area Radiation Monitor Alarm Function.

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

(1) (Not used)

(2) During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

(3) Above background.

(4) During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies.

ACTION STATEMENTS

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - This Action is not used.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 35 - ~~With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:~~

~~a) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and~~

~~b) Return the channel to OPERABLE status within 30 days, or, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.~~

This Action is not used.

ACTION 41 - a) With the number of Unit 1 OPERABLE channels one less than the Minimum Channels OPERABLE requirement:

1. Verify the respective Unit 2 control room radiation monitor train is OPERABLE within 1 hour and at least once per 31 days.

Relocated to LRM to support relocation of Containment Area Radiation Monitor Alarm Function.

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-1 TO CLARIFY FINAL PAGE CONTENT

Reviewers Note: The markups on this page include the changes associated with the retention of the Containment Area Radiation Monitor Post Accident Monitoring (PAM) indication function in the PAM Technical Specification (3/4.3.3.8).

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

(1) (Not used)

(2) During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

(3) Above background.

(4) During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies.

ACTION STATEMENTS

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - This Action is not used.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

This Action is not used.

ACTION 35 - ~~With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:~~

~~a) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and~~

~~b) Return the channel to OPERABLE status within 30 days, or, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.~~

Replaced by PAM Specification 3/4.3.3.8 Actions a and b (PAM Actions shown on next page)

ACTION 41 - a) With the number of Unit 1 OPERABLE channels one less than the Minimum Channels OPERABLE requirement:

1. Verify the respective Unit 2 control room radiation monitor train is OPERABLE within 1 hour and at least once per 31 days.

POST ACCIDENT MONITORING SPECIFICATION 3/4.3.3.8 ACTIONS a AND b REPLACE CONTAINMENT AREA RADIATION MONITORING ACTION # 35 (on previous page)

a.1 One or more PAM Functions with one required channel inoperable, restore the required channel to OPERABLE status within 30 days, or

a.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 14 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channel of the Function to OPERABLE status, or

a.3 Be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b.1 One or more PAM Functions with two or more required channels inoperable, restore the required channel(s) to OPERABLE status within 7 days, or

b.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 7 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status, or

b.3 Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Reviewers Note: The markup on this page includes the changes associated with the relocation of the Containment Area Radiation Monitor Alarm Function to the LRM. The markups are not applicable to other Radiation Monitors in this Specification.

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-1 TO CLARIFY FINAL PAGE CONTENT

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Relocated to the Licensing Requirements Manual (LRM) to support the relocation of the Containment Area Radiation Monitor Alarm Function.

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Deleted				
b. Containment				
i. Purge & Exhaust Isolation (RMVS 104 A & B)	S	R	M	**
~~ii.~~ ~~Area (RM-RM-219 A & B)~~	~~S~~	~~R~~	~~M~~	~~1,2,3,& 4~~
c. Control Room Isolation (RM-RM-218 A & B)	S	R	M###	1,2,3,4, and ##
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection (RM 215B)	S	R#	M	1,2,3 & 4
ii. Particulate Activity RCS Leakage Detection (RM 215A)	S	R#	M	1,2,3 & 4
b. Deleted				

** During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

\# Surveillance interval may be extended to the upcoming refueling outage if the interval between refueling outages is greater than 18 months.

\## During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies.

\### Control Room intake and exhaust isolation dampers are not actuated.

Reviewers Note: The unmarked requirements applicable to the Containment Area Radiation Monitors on this page are relocated as shown on the previous page.

Reviewers Note: The markups on this page include the changes associated with the retention of the Containment Area Radiation Monitor Post Accident Monitoring (PAM) indication function in the PAM Technical Specification (3/4.3.3.8). Other Radiation Monitors on this page are not affected by the marked changes.

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-1 TO CLARIFY FINAL PAGE CONTENT

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Moved to PAM Specification Surveillances 4.3.3.8.1 and 4.3.3.8.2

Moved to PAM Specification 3/4.3.3.8 Applicability

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Deleted				
b. Containment				
i. Purge & Exhaust Isolation (RMVS 104 A & B)	S	R	M	**
ii. Area (RM-RM-219 A & B)	~~S~~ M	R	M	1,2,3, & 4
c. Control Room Isolation (RM-RM-218 A & B)	S	R	M###	1,2,3,4, and ##
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection (RM 215B)	S	R#	M	1,2,3 & 4
ii. Particulate Activity RCS Leakage Detection (RM 215A)	S	R#	M	1,2,3 & 4
b. Deleted				

** During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

Surveillance interval may be extended to the upcoming refueling outage if the interval between refueling outages is greater than 18 months.

During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies.

Control Room intake and exhaust isolation dampers are not actuated.

(PAM)

INSTRUMENTATION

POST

ACCIDENT MONITORING INSTRUMENTATION

Reviewers Note: The changes marked on this page are only applicable to those PAM Functions being retained in Technical Specification 3/4.3.3.8.

LIMITING CONDITION FOR OPERATION

PAM

3.3.3.8 The ~~accident monitoring~~ instrumentation ~~channels shown~~ in Table 3.3-11 shall be OPERABLE.

for each Function

APPLICABILITY: MODES 1, 2 and 3. *

ACTION:

Insert new General Note - INSERT 1

Replace with revised PAM Actions a and b INSERT 2

~~a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours (follow Specification 3.4.11 when determining ACTIONS for Items 5 and 6).~~

~~b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.~~

SURVEILLANCE REQUIREMENTS

~~4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.~~

Replace with revised PAM Surveillances, new Surveillance General Note, and new ** footnote - **INSERT 3**

* Power Range Neutron Flux PAM Function is not required in MODE 3.

PAM SPECIFICATION INSERTS

INSERT 1. **New General Note**

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

Separate ACTION statement entry is allowed for each Function.

\- -

INSERT 2. **Revised PAM Actions a & b**

a.1 One or more PAM Functions with one required channel inoperable, restore the required channel to OPERABLE status within 30 days, or

a.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 14 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channel of the Function to OPERABLE status, or

a.3 Be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b.1 One or more PAM Functions with two or more required channels inoperable, restore the required channel(s) to OPERABLE status within 7 days, or

b.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 7 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status, or

b.3 Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 3. **Revised PAM Surveillances, new Surveillance General Note, and ** footnote**

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

Surveillance Requirement 4.3.3.8.1 applies to each PAM Function in Table 3.3-11. Surveillance Requirement 4.3.3.8.2 applies to each PAM Function in Table 3.3-11 except for the Penetration Flow Path Containment Isolation Valve Position Function. Surveillance Requirement 4.3.3.8.3 applies only to the Penetration Flow Path Containment Isolation Valve Position Function in Table 3.3-11.

\- -

4.3.3.8.1 Perform a CHANNEL CHECK at least once every 31 days.

4.3.3.8.2 Perform a CHANNEL CALIBRATION at least once every 18 months.**

4.3.3.8.3 Perform a CHANNEL FUNCTIONAL TEST at least once every 18 months.

** Neutron detectors are excluded from the Channel Calibration.

Reviewers Note: The changes marked on this page are only applicable to those PAM Functions being relocated to the Licensing Requirements Manual (LRM).

~~LIMITING CONDITION FOR OPERATION~~

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours (follow Specification 3.4.11 when determining ACTIONS for Items 5 and 6).

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

Relocated to the LRM to support the relocation of those PAM Functions identified as relocated to the LRM.

POST

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

Insert new PAM Functions 3-9 - **INSERT 4**

REQUIRED CHANNELS

FUNCTION

FUNCTION	~~TOTAL NO. OF CHANNELS~~	~~MINIMUM CHANNELS OPERABLE~~
1. Pressurizer Water Level	2 ~~3~~	~~2~~
2. Auxiliary Feedwater Flow Rate to Steam Generator (SG)	3 (1 per SG) ~~1 per steam generator~~	~~1 per steam generator~~
~~3. Reactor Coolant System Subcooling Margin Monitor~~	~~1~~	~~1~~
~~4. Deleted~~		
~~5. PORV Limit Switch Position Indicator~~	~~1/valve~~	~~0/valve~~
~~6. PORV Block Valve Limit Switch Position Indicator~~	~~1/valve~~	~~0/valve~~
~~7. Safety Valve Acoustical Detector Position Indicator~~	~~2/valve*~~	~~1/valve~~
~~8. Deleted~~		
~~9. Containment Sump Wide Range Water Level~~	~~2~~	~~1~~
10. Containment Wide-Range Pressure	2	~~0~~
11. ~~In-Core Thermocouples~~ ~~(~~Core-Exit Temperature ~~Thermocouples)~~	$2^{(c)}$ ~~4/core quadrant~~	~~2/core quadrant~~
12 ~~Reactor Vessel Level Indicating System~~	~~1~~	~~1~~

Insert new PAM Function 12 - **INSERT 5**

delete this hyphen

CURRENT PAM FUNCTION #12 RELOCATED TO THE LRM

~~* One Detector Active, Second Detector Passive~~

PAM FUNCTIONS AND * NOTE RELOCATED TO THE LRM

Insert New Table Notes (a), (b), and (c)
INSERT 6

PAM TABLE 3.3-11 INSERTS

INSERT 4. **New PAM Functions 3-9**

	FUNCTION	REQUIRED CHANNELS
3.	Power Range Neutron Flux	2
4.	High Head Safety Injection Flow	1
5.	SG Pressure	
	a) SG "A"	2
	b) SG "B"	2
	c) SG "C"	2
6.	Refueling Water Storage Tank Level	2
7.	Reactor Coolant System Pressure (Wide Range)	2
8.	SG Water Level (Wide Range)	3 (1 per SG)
9.	Containment Area Radiation (High Range)	2

INSERT 5. **New PAM Function 12**

FUNCTION	REQUIRED CHANNELS
12. Penetration Flow Path Containment Isolation Valve Position	2 per penetration flow path [(a)] [(b)]

INSERT 6. **New Table Notes (a), (b), and (c)**

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) A channel consists of two core exit thermocouples.

Reviewers Note: Table 4.3-7 is not used in the revised PAM Specification. The markup of Table 4.3-7 shows changes to retained surveillance requirements and the relocated surveillance requirements.

~~TABLE 4.3-7~~

~~ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS~~

Surveillance 4.3.3.8.1

Surveillance 4.3.3.8.2

		~~CHANNEL CHECK~~	~~CHANNEL CALIBRATION~~
1.	Pressurizer Water Level	M	R
2.	Auxiliary Feedwater Flow Rate	M ~~S/U(1)~~	R
~~3.~~	~~Reactor Coolant System Subcooling Margin~~	~~M~~	~~R~~
~~4.~~	~~Deleted~~		
~~5.~~	~~PORV Limit Switch Position Indicator~~	~~M~~	~~R~~
~~6.~~	~~PORV Block Valve Limit Switch Position Indicator~~	~~M~~	~~R~~
~~7.~~	~~Safety Valve Acoustical Detector Position Indicator~~	~~M~~	~~R~~
~~8.~~	~~Deleted~~		
~~9.~~	~~Deleted~~		
~~10.~~	~~Containment Sump Wide-Range Water Level~~	~~M~~	~~R~~
~~11.~~ 10	Containment Wide-Range Pressure	M ~~N/A~~	R
~~12.~~ 11	~~In-Core Thermocouples (~~Core-~~E~~xit ~~Thermocouples)~~ Temperature	M	R
~~13.~~	~~Reactor Vessel Level Indicating System~~	~~M~~	~~R~~

PAM FUNCTIONS RELOCATED TO THE LRM

delete this hyphen

~~(1) Channel check to be performed in conjunction with Surveillance Requirement 4.7.1.2.7 following an extended plant outage.~~

Attachment A-2

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

License Amendment Request No. 187

The following is a list of the affected pages:

V
3/4 3-39
3/4 3-40
3/4 3-41
3/4 3-42
3/4 3-43
3/4 3-44
3/4 3-57
3/4 3-58
3/4 3-59

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Post

3/4.4 REACTOR COOLANT SYSTEM

Reviewers Note: The requirements on this page remain in the Technical Specifications for Radiation Monitors not affected by this License Amendment Request (LAR). The relocation shown below is only applicable for the Containment Area Radiation Monitor Alarm Function and Main Steam Discharge Effluent Radiation Monitors being relocated by this LAR.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

Relocated to the Licensing Requirements Manual (LRM) and Offsite Dose Calculation Manual (ODCM) to support the relocation of the Containment Area Radiation Monitor Alarm Function and Main Steam Discharge Effluent Radiation Monitors.

Reviewers Note: The markup on this page includes the changes associated with the relocation of the Containment Area Radiation Monitor Alarm Function to the LRM. The markups are not applicable to other Radiation Monitors shown on this page.

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-2 TO CLARIFY FINAL PAGE CONTENT

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

Relocated to the Licensing Requirements Manual (LRM) to support the relocation of the Containment Area Radiation Monitor Alarm Function.

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT(3)	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Deleted					
b. ~~Containment Area (2RMR-RQ206 & 207)~~ Deleted	~~2~~	~~1,2,3 & 4~~	~~$\leq 2.0 \times 10^4$ R/hr~~	~~1 to 10^7 R/hr~~	~~35~~
c. Control Room Area (2RMC-RQ201 & 202)	2	1,2,3,4, and (4)	≤ 0.476 mR/hr	10^{-2} to 10^3 mR/hr	46,47
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity (Xe-133) RCS Leakage Detection (2RMR-RQ303B)	1	1,2,3 & 4	N/A	10^{-6} to 10^{-1} μCi/cc	20
ii. Particulate Activity (I-131) RCS Leakage Detection (2RMR-RQ303A)	1	1,2,3 & 4	N/A	10^{-10} to 10^{-5} μCi/cc	20
b. Deleted					

Reviewers Note: The markups on this page include the changes associated with the retention of the Containment Area Radiation Monitor Post Accident Monitoring (PAM) indication function in the PAM Technical Specification (3/4.3.3.8). Other Radiation Monitors on this page are not affected by the marked changes.

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-2 TO CLARIFY FINAL PAGE CONTENT

Reviewers Note: The unmarked requirements applicable to the Containment Area Radiation Monitors on this page are relocated as shown on the previous page.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	~~MINIMUM~~ REQUIRED CHANNELS ~~OPERABLE~~	APPLICABLE MODES	SETPOINT(3)	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Deleted					
b. Containment Area (2RMR-RQ206 & 207)	2	1,2,3 & 4	≤ 2.0 x 10^4 R/hr	1 to 10^7 R/hr	35
c. Control Room Area (2RMC-RQ201 & 202)	2	1,2,3,4, and (4)	≤ 0.476 mR/hr	10^{-2} to 10^3 mR/hr	46,47
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity (Xe-133) RCS Leakage Detection (2RMR-RQ303B)	1	1,2,3 & 4	N/A	10^{-6} to 10^{-1} µCi/cc	20
ii. Particulate Activity (I-131) RCS Leakage Detection (2RMR-RQ303A)	1	1,2,3 & 4	N/A	10^{-10} to 10^{-5} µCi/cc	20
b. Deleted					

Deleted

Moved to PAM TS 3/4.3.3.8 and title changed to Containment Area Radiation (High Range)

Relocated to PAM Specification Bases

Moved to PAM Technical Specification 3/4.3.3.8

Replaced by PAM Specification Actions a and b in 3/4.3.3.8

Reviewers Note: The markup on this page includes the changes associated with the relocation of the Main Steam Discharge Effluent Radiation Monitors to the ODCM. The markups are not applicable to other Radiation Monitors shown on this page.

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-2 TO CLARIFY FINAL PAGE CONTENT

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

Relocated to the ODCM.

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT(3)	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS (Continued)					
c. Noble Gas and Effluent Monitors					
i. Deleted					
ii. Containment Purge Exhaust (Xe-133) (2HVR-RQ104A & B)	2	(5)	$\leq 1.01 \times 10^{-3}$ µCi/cc	10^{-6} to 10^{-1} µCi/cc	22
~~iii.~~ Main Steam Discharge (Kr-88) (2MSS-RQ101A, ~~B & C)~~	~~1/SG~~	~~1, 2, 3&4~~	~~$\leq 3.9 \times 10^{-2}$ µCi/cc~~	10^{-2} to 10^{3} µCi/cc	35

NPF-73

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-2 TO CLARIFY FINAL PAGE CONTENT

Reviewers Note: The markup on this page includes the changes associated with the relocation of the Containment Area Radiation Monitor Alarm Function and Main Steam Discharge Effluent Radiation Monitors to the LRM and ODCM. The markups are not applicable to other Radiation Monitors in this Specification.

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

(1) Not used.

(2) Not used.

Relocated to LRM and ODCM to support relocation of Containment Area Radiation Monitor Alarm Function and Main Steam Discharge Effluent Radiation Monitors.

(3) Above background.

(4) During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies.

(5) During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

ACTION STATEMENTS

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - This Action is not used.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

This Action is not used.

ACTION 35 - ~~With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:~~

~~1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and~~

~~2) Return the channel to OPERABLE status within 30 days, or, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.~~

Relocated to LRM and ODCM to support relocation of Containment Area Radiation Monitor Alarm Function and Main Steam Discharge Effluent Radiation Monitors.

SEE DRAFT CLEAN TYPED PAGES IN LAR
ATTACHMENT C-2 TO CLARIFY FINAL PAGE CONTENT

Reviewers Note: The markups on this page include the changes associated with the retention of the Containment Area Radiation Monitor Post Accident Monitoring (PAM) indication function in the PAM Technical Specification (3/4.3.3.8).

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

(1) Not used.

(2) Not used.

(3) Above background.

(4) During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies.

(5) During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

ACTION STATEMENTS

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - This Action is not used.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

This Action is not used.

ACTION 35 - ~~With the number of OPERABLE channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:~~

~~1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and~~

~~2) Return the channel to OPERABLE status within 30 days, or, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.~~

Replaced by PAM Specification 3/4.3.3.8 Actions a and b (PAM Actions shown on next page)

POST ACCIDENT MONITORING SPECIFICATION 3/4.3.3.8 ACTIONS a AND b REPLACE CONTAINMENT AREA RADIATION MONITORING ACTION # 35 (on previous page)

a.1 One or more PAM Functions with one required channel inoperable, restore the required channel to OPERABLE status within 30 days, or

a.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 14 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channel of the Function to OPERABLE status, or

a.3 Be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b.1 One or more PAM Functions with two or more required channels inoperable, restore the required channel(s) to OPERABLE status within 7 days, or

b.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 7 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status, or

b.3 Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Reviewers Note: The markup on this page includes the changes associated with the relocation of the Containment Area Radiation Monitor Alarm Function to the LRM. The markups are not applicable to other Radiation Monitors in this Specification.

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-2 TO CLARIFY FINAL PAGE CONTENT

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Relocated to the Licensing Requirements Manual (LRM) to support the relocation of the Containment Area Radiation Monitor Alarm Function.

Deleted

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Deleted				
b. ~~Containment Area (2RMR-RQ206 & 207)~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~
c. Control Room Area (2RMC-RQ201 & 202)	S	R	M	1, 2, 3, 4, and ##
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection (2RMR-RQ303B)	S	R#	M	1, 2, 3 & 4
ii. Particulate Activity RCS Leakage Detection (2RMR-RQ303A)	S	R#	M	1, 2, 3 & 4
b. Deleted				

Surveillance interval may be extended to the upcoming refueling outage if the interval between refueling outages is greater than 18 months.

During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies.

Reviewers Note: The unmarked requirements applicable to the Containment Area Radiation Monitors on this page are relocated as shown on the previous page.

Reviewers Note: The markups on this page include the changes associated with the retention of the Containment Area Radiation Monitor Post Accident Monitoring (PAM) indication function in the PAM Technical Specification (3/4.3.3.8). Other Radiation Monitors on this page are not affected by the marked changes.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-2 TO CLARIFY FINAL PAGE CONTENT

Moved to PAM Specification Surveillances 4.3.3.8.1 and 4.3.3.8.2

Moved to PAM Specification 3/4.3.3.8 Applicability

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Deleted				
b. Containment Area (2RMR-RQ206 & 207)	~~S~~ M	R	M	1, 2, 3, 4
c. Control Room Area (2RMC-RQ201 & 202)	S	R	M	1, 2, 3, 4, and ##
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection (2RMR-RQ303B)	S	R#	M	1, 2, 3 & 4
ii. Particulate Activity RCS Leakage Detection (2RMR-RQ303A)	S	R#	M	1, 2, 3 & 4
b. Deleted				

Surveillance interval may be extended to the upcoming refueling outage if the interval between refueling outages is greater than 18 months.

During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies.

Reviewers Note: The markup on this page includes the changes associated with the relocation of the Main Steam Discharge Effluent Radiation Monitors to the ODCM. The markups are not applicable to other Radiation Monitors in this Specification.

SEE DRAFT CLEAN TYPED PAGES IN LAR ATTACHMENT C-2 TO CLARIFY FINAL PAGE CONTENT

TABLE 4.3-3 (Continued)

Relocated to the ODCM to support the relocation of the Main Steam Discharge Effluent Radiation Monitors.

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
2. PROCESS MONITORS (Continued)				
c. Noble Gas Effluent Monitors				
i. Deleted				
ii. Containment Purge Exhaust (2HVR-RQ104A & B)	S	R	M	###
~~iii.~~ Main Steam Discharge (2MSS-RQ101A, B & C)	S	R	M	1, 2, 3 & 4

During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

INSTRUMENTATION

(PAM)

POST

ACCIDENT MONITORING INSTRUMENTATION

Reviewers Note: The changes marked on this page are only applicable to those PAM Functions being retained in Technical Specification 3/4.3.3.8.

LIMITING CONDITION FOR OPERATION

PAM

3.3.3.8 The ~~accident monitoring~~ instrumentation ~~channels shown~~ in Table 3.3-11 shall be OPERABLE.

for each Function

*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Insert new General Note - **INSERT 1**

Replace with revised PAM Actions a and b **INSERT 2**

~~a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours (follow Specification 3.4.11 when determining ACTIONS for Items 4 and 5).~~

~~b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.~~

~~c. With the number of OPERABLE Reactor Coolant System Subcooling Margin Monitor instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.~~

Relocated to LRM to support relocation of Subcooling Margin Monitor

SURVEILLANCE REQUIREMENTS

~~4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.~~

Replace with revised PAM Surveillances, new Surveillance General Note, and new ** footnote - **INSERT 3**

* Power Range Neutron Flux PAM Function is not required in MODE 3.

PAM SPECIFICATION INSERTS

INSERT 1 - **New General Note**

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

Separate ACTION statement entry is allowed for each Function.

- -

INSERT 2 - **Revised PAM Actions a & b**

a.1 One or more PAM Functions with one required channel inoperable, restore the required channel to OPERABLE status within 30 days, or

a.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 14 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channel of the Function to OPERABLE status, or

a.3 Be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b.1 One or more PAM Functions with two or more required channels inoperable, restore the required channel(s) to OPERABLE status within 7 days, or

b.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 7 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status, or

b.3 Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

INSERT 3- **Revised PAM Surveillances, new Surveillance General Note, and ** footnote**

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

Surveillance Requirement 4.3.3.8.1 applies to each PAM Function in Table 3.3-11. Surveillance Requirement 4.3.3.8.2 applies to each PAM Function in Table 3.3-11 except for the Penetration Flow Path Containment Isolation Valve Position Function. Surveillance Requirement 4.3.3.8.3 applies only to the Penetration Flow Path Containment Isolation Valve Position Function in Table 3.3-11.

- -

4.3.3.8.1 Perform a CHANNEL CHECK at least once every 31 days.

4.3.3.8.2 Perform a CHANNEL CALIBRATION at least once every 18 months.**

4.3.3.8.3 Perform a CHANNEL FUNCTIONAL TEST at least once every 18 months.

** Neutron detectors are excluded from the Channel Calibration.

Reviewers Note: The changes marked on this page are only applicable to those PAM Functions being relocated to the Licensing Requirements Manual (LRM).

~~LIMITING CONDITION FOR OPERATION~~

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours (follow Specification 3.4.11 when determining ACTIONS for Items 4 and 5).

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE Reactor Coolant System Subcooling Margin Monitor instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

Relocated to the LRM to support the relocation of those PAM Functions identified as relocated to the LRM.

POST

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

REQUIRED CHANNELS

Insert new PAM Functions 3 - 9
INSERT 5

FUNCTION

~~INSTRUMENT~~		~~TOTAL NO. OF CHANNELS~~	~~MINIMUM CHANNELS OPERABLE~~	~~ACTION~~
1.	Pressurizer Water Level	2 ~~3~~	~~2~~	~~a, b~~
~~2.~~	~~Auxiliary Feedwater Flow Rate~~	~~2 per steam generator~~	~~1 per steam generator~~	~~a, b~~
~~3.~~	~~Reactor Coolant System Subcooling Margin Monitor~~	~~2~~	~~1~~	~~c~~
~~4.~~	~~PORV Limit Switch Position Indicator~~	~~1/valve~~	~~0/valve~~	~~a, b~~
~~5.~~	~~PORV Block Valve Limit Switch Position Indicator~~	~~1/valve~~	~~0/valve~~	~~a, b~~
~~6.~~	~~Safety Valve Position Indicator~~	~~1/valve~~	~~0/valve~~	~~a, b~~
~~7.~~	~~Deleted~~			
~~8.~~	~~Containment Sump Wide Range Water Level~~	~~2~~	~~1~~	~~a, b~~
10 ~~9.~~	Containment Wide Range Pressure	2	~~1~~	~~a, b~~
~~10.~~	~~Reactor Vessel Level Indication System~~	~~2~~	~~1~~	~~a, b~~
11.	Core Exit Temperature ~~Thermocouples~~	2(c) ~~4/core quadrant~~	~~2/core quadrant~~	~~a, b~~

INSERT 4

Insert new PAM Function 12 - **INSERT 6**

PAM FUNCTIONS RELOCATED TO THE LRM

Insert New Table Notes (a), (b), and (c)
INSERT 7

PAM TABLE 3.3-11 INSERTS

INSERT 4. **AFW Flow Rate Functions 2. a), b), and c)**

FUNCTION	REQUIRED CHANNELS
2. Auxiliary Feedwater Flow Rate to Steam Generator (SG)	
a) SG "A"	2
b) SG "B"	2
c) SG "C"	2

INSERT 5. **New PAM Functions 3-9**

FUNCTION	REQUIRED CHANNELS
3. Power Range Neutron Flux	2
4. High Head Safety Injection Flow	1
5. SG Pressure	
a) SG "A"	2
b) SG "B"	2
c) SG "C"	2
6. Refueling Water Storage Tank Level (Wide Range)	2
7. Reactor Coolant System Pressure (Wide Range)	2
8. SG Water Level (Wide Range)	3 (1 per SG)
9. Containment Area Radiation (High Range)	2

INSERT 6. **New PAM Function 12**

FUNCTION	REQUIRED CHANNELS
12. Penetration Flow Path Containment Isolation Valve Position	2 per penetration flow path [(a) (b)]

INSERT 7. **New Table Notes (a), (b), and (c)**

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) A channel consists of two core exit thermocouples.

Reviewers Note: Table 4.3-7 is not used in the revised PAM Specification. The markup of Table 4.3-7 shows changes to retained surveillance requirements and the relocated surveillance requirements.

Surveillance 4.3.3.8.1

Surveillance 4.3.3.8.2

~~TABLE 4.3-7~~

~~ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS~~

INSTRUMENT	~~CHANNEL CHECK~~	~~CHANNEL CALIBRATION~~
1. Pressurizer Water Level	M	R
2. Auxiliary Feedwater Flow Rate	M ~~S/U*~~	R
3. Reactor Coolant System Subcooling Margin Monitor	M	R
4. PORV Limit Switch Position Indicator	M	R
5. PORV Block Valve Limit Switch Position Indicator	M	R
6. Safety Valve Position Indicator	M	R
7. Deleted		
8. Containment Sump Wide-Range Water Level	M	R
10 ~~9.~~ Containment Wide Range Pressure	M ~~N/A~~	R
~~10. Reactor Vessel Level Indication System~~	~~M~~	~~R~~
11. Core Exit Temperature ~~Thermocouples~~	M	R

PAM FUNCTIONS RELOCATED TO THE LRM

~~* Channel check to be performed in conjunction with Surveillance Requirement 4.7.1.2.7 following an extended plant outage.~~

Attachment B-1

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Bases Changes

License Amendment Request No. 314

The following is a list of the affected pages:

B-I
B 3/4 3-2
B 3/4 3-3
B 3/4 4-11b
B 3/4 4-11d

TECHNICAL SPECIFICATION BASES INDEX

BASES

Post

Provided for Information Only.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

and

The OPERABILITY of the radiation monitoring channels ensures that: 1) the radiation levels are continually measured in the areas served by the individual channels~~;~~ 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded~~; and 3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," October, 1980~~.

A "recently" irradiated fuel assembly is fuel that has occupied part of a critical reactor core within the previous 100 hours.

3/4.3.3.2 (This Specification number is not used.)

3/4.3.3.3 (This Specification number is not used.)

3/4.3.3.4 (This Specification number is not used.)

Provided for Information Only.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

POST

3/4.3.3.7 (This Specification number is not used.)

INSERT 1, REVISED PAM BASES

3/4.3.3.8 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

Provided for Information Only.

3/4.4.11 RELIEF VALVES (Continued)

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 to minimize challenges to the pressurizer safety valves.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for OPPS in MODES 4 (below the enable temperature specified in the PTLR), 5, and 6 with the reactor vessel head in place. LCO 3.4.9.3 addresses the PORV requirements in these MODES.

ACTION

A General Note provides clarification that all pressurizer PORVs and block valves are treated as separate entities, each with separate completion times (i.e., the completion time is on a component basis).

a. With the PORVs inoperable and capable of being manually cycled, either the PORVs must be restored or the flow path isolated within 1 hour. The block valves should be closed but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems ~~related to PORV accident monitoring instruments identified in LCO 3.3.3.8~~, or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. If the position indication is inoperable, then the PORVs are inoperable. For these reasons, the block valve shall be closed but the ACTION requires power be maintained to the valve. Automatic control problems and related instrumentation problems would not render the PORVs inoperable. Accident analyses assume manual operation of

Provided for Information Only.

3/4.4.11 RELIEF VALVES (Continued)

ACTION (Continued)

plant will be in a less limiting ACTION statement with the time clock started at the original declaration of having three PORVs inoperable. If no PORVs are restored within the completion time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.9.3.

d. If one block valve is inoperable and open, then it is necessary to either restore the block valve to OPERABLE status within the completion time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the required action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. If the block valve in inoperable, it is necessary to restore the block valve to OPERABLE status within 1 hour or close it. If block valve limit switch position indicator instrumentation ~~related to accident monitoring instrumentation identified in LCO 3.3.3.8~~ is determined to be inoperable, then the block valve shall be declared inoperable. Closing the block valve precludes the need to place the PORV in manual control since it is isolated from the system. The completion time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a completion time of 72 hours to restore the inoperable open block valve to OPERABLE status. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply in order to avoid continuous operation without a redundant ability to isolate this PORV flow path. If the block valve is restored within the completion time of 72 hours, the power will be restored and the PORV restored to OPERABLE status. With one block valve inoperable and closed, there

ATTACHMENT B-1
INSERT 1

Provided for Information Only.

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the control room operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs) or that the Probabilistic Risk Assessment (PRA) has shown to be significant to the public health and safety.

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess the unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by addressing the recommendations of Regulatory Guide 1.97 (Ref. 1) as required by Supplement 1 to NUREG-0737 (Ref. 2) and by evaluating the instrumentation consistent with the methodology contained in WCAP-15981 (Ref. 3). This methodology considers the use of the accident monitoring instrumentation in the design basis accident analyses, PRA, Emergency Operating Procedures (EOPs), Severe Accident Management Guidance (SAMG) procedures, and Emergency Plan (E-Plan).

The control room monitoring instrumentation Functions required to be OPERABLE by this LCO have been evaluated and selected in accordance with the screening criteria contained in WCAP-15981. The screening criteria were used to identify the PAM instrumentation important to safety (i.e., monitor plant parameters that are the basis for important operator actions to bring the unit to a safe stable state in the event of an accident). The details and results of this evaluation are documented in License Amendment Request 314 and License Amendment number [Later] and the associated NRC Safety Evaluation Report (Ref. 4).

The selected instrument Functions satisfy Criterion 3 and/or 4 of 10 CFR 50.36(c)(2)(ii), and include Regulatory Guide 1.97 monitoring instrumentation for parameters identified as important to safety in accordance with WCAP-15981. The selected PAM instrument Functions are listed in Table 3.3-11 and are discussed in more detail in the LCO section of the Bases.

APPLICABLE
SAFETY
ANALYSES

The PAM specification ensures the operability of instrumentation to monitor plant parameters necessary for safety significant operator actions so that the control room operating staff can:

- Perform the diagnosis specified in the EOPs (these variables are restricted to preplanned actions for the primary success path of DBAs), e.g., loss of coolant accident (LOCA),
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety

Provided for Information Only.

function,

- Implement procedures or guidance that has been shown to have an important role in preventing core damage or early fission product releases,
- Determine the likelihood of a gross breach of the barriers that prevent radioactivity release,
- Determine if a gross breach of a barrier has occurred, and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

The PAM instrumentation selected in accordance with WCAP-15981 is used to monitor plant parameters necessary for safety significant operator actions and satisfies Criterion 3 and/or 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for the control room monitoring instrumentation Functions important to safety (i.e., monitor plant parameters that are the basis for important operator actions to bring the plant to a safe stable state in the event of an accident).

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident. This capability is consistent with the guidance of Reference 3.

LCO 3.3.3.8 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident. Therefore, where plant design and channel availability permit, the two channels required OPERABLE by the LCO should be supplied from different trains of electrical power.

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

For some PAM Functions, Table 3.3-11 specifies one or three required channels. The following are exceptions to the two-channel requirement:

Three channels of steam generator (SG) wide range level instrumentation are required to be OPERABLE. Each SG has one installed wide range channel that assures the ability to monitor SG level during operating conditions when the level may not be in the normal range. In many accident analyses, two SGs are assumed to be available to provide the necessary heat removal capacity. The requirement for three OPERABLE channels of wide range level indication (one per SG) helps to assure adequate wide range SG level indication remains available (assuming one indication channel fails or a SG is faulted) to monitor SG level and support maintaining the necessary heat removal capacity.

Provided for Information Only.

One channel of high head safety injection (HHSI) total flow is required to be OPERABLE. The normal SI injection flow path (automatically initiated on an SI signal) has a single installed Regulatory Guide 1.97 flow instrument that indicates total SI flow in the control room. This indicator is used to confirm automatic SI flow initiation. The single HHSI total flow indication is adequate considering the alternate control room indications available to confirm the operation of the SI system. An alternate method of verifying SI initiation can be provided by the High Head SI pump amperage indication, the High Head SI header pressure indication, and the SI automatic valve position indication.

Three channels of Auxiliary Feedwater (AFW) Flow indication (1 per SG) are required to be OPERABLE. Each SG has a single AFW flow indicator in the control room. AFW flow is used by the operator to verify that the AFW System is delivering the correct flow to each SG. The single AFW flow indicator per SG is acceptable, considering the alternate indications provided by the SG Water Level Wide Range indication or the SG Water Level Narrow Range indication to ensure adequate SG inventory. In addition, alternate methods of determining the need for operator action in the event that the auxiliary feedwater flow rate indication is not available can be provided by the AFW pump amperage instrumentation (motor-driven pumps), flow control valve position indication (SG supply), and automatic turbine steam supply valve position indication.

Another exception to the two channel requirement is Penetration Flow Path Containment Isolation Valve (CIV) Position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV (i.e., associated penetration not isolated and designed with control room indication per the Table 3.3-11 footnotes modifying the required CIV position indication). The active CIVs addressed by this LCO only include valves designed to close on a Phase A or Phase B containment isolation signal. Valves that open on a Phase A or Phase B containment isolation signal are not required to have their position verified to confirm adequate containment isolation. Thus, the requirements of this LCO are sufficient to redundantly verify the isolation status of each isolable penetration (required to be isolated during accident conditions) either via indicated status of the active valve, or the reliability of containment isolation valves without control room indication (i.e. automatic check valves and relief valves that are not dependent on a external power source or closure signal), or prior knowledge of a passive valve, or via closed system boundary status. If a normally active CIV is known to be closed and deactivated or open under administrative controls in accordance with the provisions of the CIV technical specification, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

Table 3.3-11 provides a list of the control room indications identified as important to safety in accordance with the methodology of WCAP-15981.

The following instrument monitoring Functions are required to be OPERABLE by this LCO:

1. <u>Pressurizer Water Level</u>

   Pressurizer Level indication is used for the SI termination criteria to prevent pressurizer overfill. The termination of SI to prevent pressurizer overfill is an operator action assumed in the design basis steamline break analysis for which no automatic actuation is provided. The PRA also indicates that SI termination in the event of a steam generator tube rupture is required for long term core cooling.

Pressurizer Level indication satisfies both Criteria 3 and 4 of 10 CFR 50.36(c)(2)(ii).

2. Auxiliary Feedwater (AFW) Flow Rate

AFW Flow indication is used by the operator to confirm that the AFW System is in operation and delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level. The PRA shows that AFW Flow indication can be important to safety by providing information necessary for operator action to initiate alternate feedwater sources in the event of a failure of the AFW system.

AFW Flow indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Alternate methods of determining the need for operator action in the event that the AFW flow rate indication is not available can be provided by the AFW pump amperage instrumentation (motor-driven pumps), flow control valve position indication (SG supply), and automatic turbine steam supply valve position indication.

3. Power Range Neutron Flux

The Power Range Neutron Flux indication is used to confirm a reactor shutdown following a design basis accident. The PRA shows that operator actions to manually shutdown the reactor in the event of a failure of the automatic actions, as determined from the Power Range Neutron Flux indication, can be important to safety.

The Power Range Neutron Flux indication satisfies criterion 4 of 10 CFR 50.36(c)(2)(ii).

The PAM Applicability is modified by a footnote that provides an exception to the PAM OPERABILITY requirement for Power Range Neutron Flux indication in MODE 3. The basis for the PAM requirement for Power Range Neutron Flux indication is that it is used to confirm an automatic reactor shutdown from power operation. Therefore, the PAM Power Range Neutron Flux indication requirements are only applicable in MODES 1 and 2 when the power range instrumentation functions to provide the necessary PAM indication.

If the power range neutron flux indications are not available, an alternate method of verifying a reactor trip is a combination of either the intermediate range or source range neutron flux indications and either the rod bottom lights or rod position indicators.

4. High Head Safety Injection (SI) Flow

High Head SI Flow indication is used to confirm automatic safety injection initiation following a design basis accident. Therefore, the required flow indicator for this PAM Function is the total flow indicator installed in the automatic High Head SI flow path. The results of the PRA shows that this is a risk significant operator action. Failure to manually initiate SI flow when the automatic initiation fails can lead to a significant

Provided for Information Only.

increase in core damage frequency. The operator action is based on the ECCS flow indication in the control room. The PRA shows that only high head safety injection is important for all accident sequences except the unlikely double-ended guillotine rupture of the largest reactor coolant pipe. Therefore, only the High Head SI Flow indication is required.

High Head SI Flow indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

If the total High Head SI Flow indication is not available, an alternate method of verifying SI initiation can be provided by the High Head SI pump amperage indication, the High Head SI header pressure indication, and the SI automatic valve position indication.

5. Steam Generator (SG) Pressure

SG Pressure indication is provided as a target for RCS depressurization for the steam generator tube rupture accident to terminate the RCS inventory loss. In the event of a steam generator tube rupture accident, the EOPs instruct the operators to depressurize the RCS to a pressure below the secondary side pressure in the ruptured steam generator. RCS depressurization to a pressure less than the steam generator pressure terminates the RCS inventory loss and terminates the steam generator inventory gain, preventing overfill of the steam generator. The termination of the break flow is an operator action assumed in the design basis steam generator tube rupture analysis for which no automatic action is provided. The PRA shows that failure to depressurize the RCS to a pressure less than the secondary side pressure in the ruptured steam generator is a risk significant operator action.

Two channels of pressure indication are required OPERABLE for each steam generator. Due to the redundant indications required OPERABLE, the indications for each steam generator are treated as separate PAM Functions. Therefore, consistent with General Note 2, separate ACTION statement entry is allowed for Functions 5.a, 5.b, and 5.c.

SG Pressure indication satisfies both Criteria 3 and 4 of 10 CFR 50.36(c)(2)(ii).

6. Refueling Water Storage Tank (RWST) Level

RWST Level provides an indication of the water inventory remaining for use by containment spray and safety injection for core cooling and containment cooling. No operator actions in the design basis accident analysis are based on the RWST Level indication. The switchover from the RWST to the containment sump is performed automatically.

The PRA shows that in the event of an accident in which the RCS inventory losses are outside of containment (e.g., steam generator tube rupture and interfacing system LOCA), the remaining RWST level is an important indication for choosing the appropriate operator actions to maintain core cooling in the EOPs. The PRA shows the importance of diagnosing the need for implementing RWST refill to maintain a sufficient inventory for long term core cooling following these events.

RWST Level indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Provided for Information Only.

7. Reactor Coolant System (RCS) Pressure (Wide Range)

RCS Wide Range Pressure indication provides the information necessary for RCS depressurization for the steam generator tube rupture accident to terminate the RCS inventory loss. In the event of a steam generator tube rupture accident, the EOPs instruct the operators to depressurize the RCS to a pressure below the secondary side pressure in the ruptured steam generator. RCS depressurization to a pressure less than the steam generator pressure terminates the RCS inventory loss and terminates the steam generator inventory gain, preventing overfill of the steam generator. The termination of the break flow is an operator action assumed in the design basis steam generator tube rupture analysis for which no automatic action is provided. RCS pressure is also used for operator action to terminate SI in the event of a steamline break to prevent pressurizer overfill for which no automatic actuation is provided. Additionally, the PRA indicates that RCS pressure is a variable important to safety for RCS cooldown and depressurization following a steam generator tube rupture.

RCS Pressure indication satisfies both Criteria 3 and 4 of 10 CFR 50.36(c)(2)(ii).

The LCO requirement for two channels can be met by using any combination of the RCS Pressure Wide Range transmitter(s) or the Reactor Vessel Level Indicating System Pressure transmitter(s).

8. SG Water Level (Wide Range)

SG Water Level (Wide Range) indication is provided to monitor operation of decay heat removal via the SGs. SG Water Level (Wide Range) indication is used to:

- identify the faulted SG following a steam generator tube rupture,
- verify that the intact SGs are an adequate heat sink for the reactor,
- determine the nature of the accident in progress (e.g., verify a steam generator tube rupture),
- verify unit conditions for the termination of SI during secondary side HELBs outside containment, and
- verify SG tubes are covered before terminating AFW to the faulted SG to assure iodine scrubbing and design basis iodine partitioning in the event of a steam generator tube rupture.

Controlling SG level to maintain a heat sink and the diagnosis of a steam generator tube rupture based on SG level are operator actions assumed in the design basis accident analysis for which no automatic actuation is provided. In addition, the PRA shows that SG Wide Range Level indication can be important to safety by providing information for the initiation of operator actions to establish bleed and feed for a loss of heat sink event.

SG Water Level (Wide Range) indication satisfies both Criteria 3 and 4 of

10 CFR 50.36(c)(2)(ii).

If a channel of wide range SG level instrumentation is not available, an alternate method of monitoring the SG level is a combination of one channel of SG narrow range instrumentation and Auxiliary Feedwater Flow Rate indication to that SG.

9. Containment Area Radiation (High Range)

Containment Area Radiation High Range provides an indication of a loss of one or more fission product barriers. The Emergency Action Levels in the E-Plan utilizes the Containment Area Radiation High Range monitor as an indication of the potential loss of one or more fission product barriers in the assessment of the declaration of a General Emergency level and the potential need for offsite radiological protection actions. The post accident Core Damage Assessment also uses the Containment Area Radiation High Range monitor as an input to the determination of core damage. The required high range monitors are designated RM-1RM-219 A & B.

The Containment Area Radiation High Range indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Radiation monitor RM-1RM-201 or a portable radiation monitor (with appropriate multiplier if necessary) can be used as an alternate method of indication for Containment Area Radiation High Range.

10. Containment Pressure (Wide Range)

Containment Pressure (Wide Range) indication is provided for assessing containment cooling and containment integrity. No operator actions in the design basis accident analysis are based on the Containment Pressure indication. Containment Pressure is an indicator of the potential loss of a fission product boundary in the Emergency Action Levels in the E-Plan. Containment Pressure is a key indicator in the declaration of a General Emergency level and the potential need for offsite radiological protection actions. Containment pressure may also be used in post accident conditions to determine when to vent the containment to prevent overpressurization.

Containment Pressure (Wide Range) indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

11. Core Exit Temperature

Core Exit Temperature indication is provided for verification and long term surveillance of core cooling. The Core Exit Temperature indication provides information for the operators to initiate RCS depressurization following a steam generator tube rupture. The PRA shows that Core Exit Thermocouple indication is important to safety by providing information necessary to maintain subcooling for RCS cooldown and depressurization following steam generator tube rupture and other small LOCA events. It is also used as an indication for the transfer from the EOPs to the Severe Accident Management Guidance, where a greater focus is maintained on preserving the

Provided for Information Only.

remaining fission product barriers.

Table 3.3-11 requires two OPERABLE channels of Core Exit Temperature. Footnote (c) to Table 3.3-11 requires a Core Exit Temperature channel to consist of two core exit thermocouples. Two sets of two thermocouples ensure that a single failure will not affect the ability to determine whether an inadequate core cooling condition exists.

Two OPERABLE channels of Core Exit Temperature from any core location except the three outermost rows of fuel assemblies on each side of the core are required to provide the most timely indication of the coolant temperature rise across the core exit. The three outermost rows of fuel assemblies are identified by counting straight in towards the center of the core from the outermost row consisting of three assemblies on each side of the core. The acceptable central core exit thermocouples can also be identified as being within the core area consisting of up to four fuel assemblies from the center fuel assembly (not counting the center assembly).

Severe accident analyses documented in WCAP-14696-A (Ref. 5) demonstrate that the coolant temperature increase at a central core location (i.e., not in the three outermost rows of fuel assemblies) provides the most rapid indication of inadequate core cooling. Therefore, in order to get the most rapid indication of coolant temperature rise in the core, the two thermocouples in each channel used to meet the LCO requirement must not be located in the three outermost rows of fuel assemblies.

Core Exit Temperature indication satisfies both Criteria 3 and 4 of 10 CFR 50.36(c)(2)(ii).

12. <u>Penetration Flow Path Containment Isolation Valve (CIV) Position</u>

Penetration Flow Path CIV Position indication is provided for verification of Containment Phase A and Phase B isolation. The E-Plan identifies that an elevated emergency action level should be declared following an accident in the event of a failure of automatic containment isolation.

This requirement only applies to containment isolation valves which receive a Phase A and Phase B containment isolation closure signal. This requirement is not applicable to valves that open on receipt of a Containment Phase A or B signal. When used to verify Phase A and Phase B isolation, the important information is the isolation status of the containment penetrations. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two active valves that have control room position indication. For containment penetrations with only one active CIV having control room indication, footnote (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve with control room indication and the reliability of containment isolation valves without control room indication (i.e. automatic check valves and relief valves that are not dependent on a external power source or closure signal), or prior knowledge of a passive valve, or via closed system boundary status. If a normally active CIV is known to be closed and deactivated or open under administrative controls in accordance with the provisions of the CIV technical specification, position indication is

Provided for Information Only.

not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE. Footnote (a) to the Required Channels states that the Function is not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. Each penetration is treated separately and each penetration flow path is considered a separate function. Therefore, separate Action Statement entry is allowed for each inoperable penetration flow path.

Penetration Flow Path CIV Position indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3 except for the Power Range Neutron Flux indication, which is only required to be OPERABLE in MODES 1 and 2. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

The Applicability is modified by a footnote affecting the Power Range Neutron Flux indication requirements. The footnote provides an exception to the PAM OPERABILITY requirement for Power Range Neutron Flux indication in MODE 3. The basis for the PAM requirement for Power Range Neutron Flux indication is that it is used to confirm an automatic reactor shutdown from power operation. Therefore, the PAM Power Range Neutron Flux indication requirements are only applicable in MODES 1 and 2 when the power range instrumentation functions to provide the necessary indication.

ACTIONS

A General Note has been added in the ACTIONS to clarify the application of completion times. The ACTIONS of this Specification may be entered independently for each Function listed on Table 3.3-11. The completion time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Action was entered for that Function.

ACTIONS a.1, a.2, and a.3

ACTIONS a.1, a.2, and a.3 apply when one or more Functions have one required channel that is inoperable.

ACTION a.1 requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day completion time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other control room indications available to accomplish the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments for a PAM function), and the low probability of an event requiring PAM instrumentation during this interval.

If the inoperable channel is anticipated to remain inoperable for an extended time, it is expected that action will be initiated as soon as possible to confirm the availability, functionality, and

Provided for Information Only.

procedure impact of any applicable pre-planned alternate instrumentation required to meet ACTION a.2. This will assure the capability of performing the PAM Function is maintained and that preparations are initiated as soon as possible to meet the requirements of Action a.2.

In accordance with ACTION a.2, continuous operation with one required channel inoperable in a Function is acceptable beyond the initial 30 days provided that acceptable alternate instrumentation is available to monitor the Function(s) with an inoperable channel. If the inoperable channel is not restored to OPERABLE status in 30 days, action must be initiated immediately to establish the alternate method of monitoring the affected Function(s). This includes verification that the alternate indication is functionally capable of performing the required PAM Function and the revision of any applicable procedures necessary to implement the alternate method. In addition, ACTION a.2 requires that a report be submitted to the NRC within 14 days after ACTION a.2 becomes applicable. The report discusses the alternate method of monitoring available, the cause of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability, and given the likelihood of unit conditions that would require information provided by this instrumentation.

ACTION a.3 is applicable if the inoperable channel is not restored to OPERABLE status in 30 days per ACTION a.1 and an alternate method of monitoring the Function(s) is not established and a report is not submitted to the NRC within the following 14 days in accordance with ACTION a.2. In this case, the unit must be placed in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN in the following 6 hours. The allowed Completion Times to place the plant in a condition where the PAM instrumentation is no longer required OPERABLE 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.

ACTIONS b.1, b.2, and b.3

ACTIONS b.1, b.2, and b.3 apply when one or more Functions have two or more inoperable required channels (i.e., two or more channels inoperable in the same Function). When action is completed such that the affected Function(s) only have one inoperable channel, ACTION a is applicable for the remaining inoperable channel not Action b.

ACTION b.1 requires restoring the inoperable channels in the Function(s) to OPERABLE status within 7 days. The completion time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation.

In accordance with ACTION b.2, continuous operation with two or more required channels inoperable in a Function is acceptable beyond the initial 7 days provided that acceptable alternate instrumentation is available to monitor the Function(s) with the inoperable channels. If the required channel(s) (i.e. number of channels necessary to allow exit from action b) is not restored to OPERABLE status in 7 days, action must be initiated immediately to establish the alternate method of monitoring the affected Function(s). This includes verification that the alternate indication is functionally capable of performing the required PAM Function and the revision of any applicable procedures necessary to implement the alternate method. In addition, ACTION b.2 requires that a report must be submitted to the NRC within 7 days after ACTION a.2 becomes applicable. This report discusses the alternate method of monitoring available, the cause of the inoperability, and identifies proposed restorative actions. This action

Provided for Information Only.

is appropriate in lieu of a shutdown requirement since alternative actions are identified that provide a similar functional capability, and given the likelihood of unit conditions that would require information provided by this instrumentation.

ACTION b.3 is applicable if two or more channels in one Function remain inoperable at the end of the 7 days allowed by ACTION b.1 and an alternate method of monitoring the Function(s) is not established or a report is not submitted to the NRC within the following 7 days in accordance with ACTION b.2. In this case, the unit must be placed in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN in the following 6 hours. The allowed Completion Times to place the plant in a condition where the PAM instrumentation is no longer required OPERABLE 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.

SURVEILLANCE
REQUIREMENTS

A General Note has been added to the SRs to clarify the following:

- Surveillance Requirement 4.3.3.8.1 applies to each PAM instrument Function in Table 3.3-11,
- Surveillance Requirement 4.3.3.8.2 applies to each PAM instrument Function in Table 3.3-11 except for the Penetration Flow Path Containment Isolation Valve Position instrument Function, and
- Surveillance Requirement 4.3.3.8.3 applies only to the Penetration Flow Path Containment Isolation Valve Position instrument Function in Table 3.3-11.

SR 4.3.3.8.1

Performance of the CHANNEL CHECK at least once every 31 days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

In addition, it is not necessary to place a system or component in service that is not normally in service (e.g., initiate AFW flow to the SGs) in order to perform the required CHANNEL CHECK. In cases where the required instrumentation may be energized but only a single channel is available (e.g., HHSI Flow) or where there may be no flow (e.g., AFW Flow), the CHANNEL CHECK may be accomplished by comparing the indicated value to the known plant condition (e.g., zero flow). In the case of CIVs, the CHANNEL CHECK may be accomplished by comparing the indicated valve position to the known or expected valve position based on current plant conditions.

Provided for Information Only.

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

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels associated with the LCO required channel indications that are utilized during normal plant operation.

SR 4.3.3.8.2

A CHANNEL CALIBRATION is performed at least once every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter with the necessary range and accuracy.

The Surveillance Requirements are modified by a General Note that states the CHANNEL CALIBRATION surveillance is not applicable to the Containment Isolation Valve Position Function. The Containment Isolation Valve Position Function is verified OPERABLE by a CHANNEL FUNCTIONAL TEST rather than a CHANNEL CALIBRATION.

This SR is also modified by a footnote that excludes neutron detectors from the CHANNEL CALIBRATION. The calibration method for neutron detectors is described in the Bases for 3 / 4.3.1 Reactor Trip System.

Whenever a core exit temperature thermocouple is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors is accomplished by an in-place cross calibration that compares the other sensing elements with the recently installed sensing element. The Frequency of 18-months is based on operating experience and consistency with the typical industry refueling cycle.

SR 4.3.3.8.3

A CHANNEL FUNCTIONAL TEST is only required for the Penetration Flow Path Containment Isolation Valve Position Function. This test is required to be performed at least once every 18 months, or approximately at every refueling. A CHANNEL FUNCTIONAL TEST verifies OPERABILITY of the containment isolation valve position indication instrumentation.

A general Note modifies the surveillance requirements and specifies that the CHANNEL FUNCTIONAL TEST surveillance is only applicable to the Penetration Flow Path Containment Isolation Valve Position Function. Due to the relatively simple instrument circuits involved and the lack of a conventional process sensor to adjust, the CHANNEL FUNCTIONAL TEST, rather than the CHANNEL CALIBRATION, provides the more appropriate OPERABILITY verification of these channels.

The Frequency of 18-months is consistent with the typical industry refueling cycle.

Provided for Information Only.

REFERENCES

1. Regulatory Guide 1.97, Rev. 2, December 1980.

2. NUREG-0737, Supplement 1, "TMI Action Items."

3. WCAP-15981, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants."

4. License Amendment number [Later] and the associated NRC Safety Evaluation Report dated [Later].

5. WCAP-14696-A, Revision 1, "Westinghouse Owners Group Core Damage Assessment Guidance."

Attachment B-2

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Bases Changes

License Amendment Request No. 187

The following is a list of the affected pages:

B-I
B 3/4 3-10
B 3/4 3-11
B 3/4 4-16b
B 3/4 4-16d

Provided for Information Only.

TECHNICAL SPECIFICATION BASES INDEX

BASES

Provided for Information Only.

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

CHANNEL CALIBRATION

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

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

SLAVE RELAY TESTING

The slave relay test frequency can be extended to once per 12 months provided a satisfactory contact loading analysis has been completed and a satisfactory slave relay service life has been established for the slave relay being tested. The frequency of 12 months is justified in WCAP-15887-NP, Revision 2, dated December 2002.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: 1) the radiation levels are continually measured in the areas served by the individual channels; and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded~~; and 3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," October, 1980~~.

A "recently" irradiated fuel assembly is fuel that has occupied part of a critical reactor core within the previous 100 hours.

3/4.3.3.2 (This Specification number is not used.)

BASES

3/4.3.3.3 (This Specification number is not used.)

3/4.3.3.4 (This Specification number is not used.)

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 (This Specification number is not used).

3/4.3.3.7 (This Specification number is not used).

INSERT 1, REVISED PAM BASES

3/4.3.3.8 ACCIDENT MONITORING INSTRUMENTATION

~~The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."~~

3/4.4.11 REACTOR COOLANT SYSTEM RELIEF VALVES (Continued)

APPLICABILITY (Continued)

PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 to minimize challenges to the pressurizer safety valves.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for OPPS in MODES 4 (below the enable temperature specified in the PTLR), 5, and 6 with the reactor vessel head in place. LCO 3.4.9.3 addresses the PORV requirements in these MODES.

ACTION

A General Note provides clarification that all pressurizer PORVs and block valves are treated as separate entities, each with separate completion times (i.e., the completion time is on a component basis).

a. With the PORVs inoperable and capable of being manually cycled, either the PORVs must be restored or the flow path isolated within 1 hour. The block valves should be closed but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. Although a PORV may be designated inoperable, the associated vent path may be manually opened and closed, and the PORV therefore, able to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems ~~related to PORV accident monitoring instruments identified in LCO 3.3.3.8~~, or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. If the position indication is inoperable, then the PORVs are inoperable. For these reasons, the block valve shall be closed but the ACTION requires power be maintained to the valve. Automatic control problems and related instrumentation problems would not render the PORVs inoperable. Accident analyses assume manual operation of the PORVs and do not take credit for automatic actuation. This condition is only intended to

3/4.4.11 REACTOR COOLANT SYSTEM RELIEF VALVES (Continued)

ACTION (Continued)

ACTION statement with the time clock started at the original declaration of having three PORVs inoperable. If no PORVs are restored within the completion time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, maintaining PORV OPERABILITY may be required. See LCO 3.4.9.3.

d. If one block valve is inoperable and open, then it is necessary to either restore the block valve to OPERABLE status within the completion time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the required action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. If the block valve is inoperable, it is necessary to restore the block valve to OPERABLE status within 1 hour or close it. If block valve limit switch position indicator instrumentation ~~related to accident monitoring instrumentation identified in LCO 3.3.3.8~~ is determined to be inoperable, then the block valve shall be declared inoperable. Closing the block valve precludes the need to place the PORV in manual control since it is isolated from the system. The completion time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a completion time of 72 hours to restore the inoperable open block valve to OPERABLE status. If it cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply in order to avoid continuous operation without a redundant ability to isolate this PORV flow path. If

Provided for Information Only.

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the control room operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs) or that the Probabilistic Risk Assessment (PRA) has shown to be significant to the public health and safety.

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess the unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified by addressing the recommendations of Regulatory Guide 1.97 (Ref. 1) as required by Supplement 1 to NUREG-0737 (Ref. 2) and by evaluating the instrumentation consistent with the methodology contained in WCAP-15981 (Ref. 3). This methodology considers the use of the accident monitoring instrumentation in the design basis accident analyses, PRA , Emergency Operating Procedures (EOPs), Severe Accident Management Guidance (SAMG) procedures, and Emergency Plan (E-Plan).

The control room monitoring instrumentation Functions required to be OPERABLE by this LCO have been evaluated and selected in accordance with the screening criteria contained in WCAP-15981. The screening criteria were used to identify the PAM instrumentation important to safety (i.e., monitor plant parameters that are the basis for important operator actions to bring the unit to a safe stable state in the event of an accident). The details and results of this evaluation are documented in License Amendment Request 187 and License Amendment number [Later] and the associated NRC Safety Evaluation Report (Ref. 4).

The selected instrument Functions satisfy Criterion 3 and/or 4 of 10 CFR 50.36(c)(2)(ii), and include Regulatory Guide 1.97 monitoring instrumentation for parameters identified as important to safety in accordance with WCAP-15981. The selected PAM instrument Functions are listed in Table 3.3-11 and are discussed in more detail in the LCO section of the Bases.

APPLICABLE
SAFETY
ANALYSES

The PAM specification ensures the operability of instrumentation to monitor plant parameters necessary for safety significant operator actions so that the control room operating staff can:

- Perform the diagnosis specified in the EOPs (these variables are restricted to preplanned actions for the primary success path of DBAs), e.g., loss of coolant accident (LOCA),

- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function,

- Implement procedures or guidance that has been shown to have an important role in

Provided for Information Only.

preventing core damage or early fission product releases,

- Determine the likelihood of a gross breach of the barriers that prevent radioactivity release,
- Determine if a gross breach of a barrier has occurred, and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

The PAM instrumentation selected in accordance with WCAP-15981 is used to monitor plant parameters necessary for safety significant operator actions and satisfies Criterion 3 and/or 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for the control room monitoring instrumentation Functions important to safety (i.e., monitor plant parameters that are the basis for important operator actions to bring the plant to a safe stable state in the event of an accident).

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident. This capability is consistent with the guidance of Reference 3.

LCO 3.3.3.8 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident. Therefore, where plant design and channel availability permit, the two channels required OPERABLE by the LCO should be supplied from different trains of electrical power.

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

For some PAM Functions, Table 3.3-11 specifies one or three required channels. The following are exceptions to the two-channel requirement:

Three channels of steam generator (SG) wide range level instrumentation are required to be OPERABLE. Each SG has one installed wide range channel that assures the ability to monitor SG level during operating conditions when the level may not be in the normal range. In many accident analyses, two SGs are assumed to be available to provide the necessary heat removal capacity. The requirement for three OPERABLE channels of wide range level indication (one per SG) helps to assure adequate wide range SG level indication remains available (assuming one indication channel fails or a SG is faulted) to monitor SG level and support maintaining the necessary heat removal capacity.

One channel of high head safety injection (HHSI) flow is required to be OPERABLE. The normal SI injection flow path (automatically initiated on an SI signal) has a single installed flow instrument that indicates flow in the control room. This indicator is used to confirm automatic SI

Provided for Information Only.

flow initiation. The single HHSI flow indication is adequate considering the alternate control room indications available to confirm the operation of the SI system. An alternate method of verifying SI initiation can be provided by the High Head SI pump amperage indication, the High Head SI header pressure indication, and the SI automatic valve position indication.

Another exception to the two channel requirement is Penetration Flow Path Containment Isolation Valve (CIV) Position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each active CIV (i.e., associated penetration not isolated and designed with control room indication per the Table 3.3-11 footnotes modifying the required CIV position indication). The active CIVs addressed by this LCO only include valves designed to close on a Phase A or Phase B containment isolation signal. Valves that open on a Phase A or Phase B containment isolation signal are not required to have their position verified to confirm adequate containment isolation. Thus, the requirements of this LCO are sufficient to redundantly verify the isolation status of each isolable penetration (required to be isolated during accident conditions) either via indicated status of the active valve, or the reliability of containment isolation valves without control room indication (i.e. automatic check valves and relief valves that are not dependent on a external power source or closure signal), or prior knowledge of a passive valve, or via closed system boundary status. If a normally active CIV is known to be closed and deactivated or open under administrative controls in accordance with the provisions of the CIV technical specification, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

Table 3.3-11 provides a list of the control room indications identified as important to safety in accordance with the methodology of WCAP-15981.

The following instrument monitoring Functions are required to be OPERABLE by this LCO:

1. <u>Pressurizer Water Level</u>

   Pressurizer Level indication is used for the SI termination criteria to prevent pressurizer overfill. The termination of SI to prevent pressurizer overfill is an operator action assumed in the design basis steamline break analysis for which no automatic actuation is provided. The PRA also indicates that SI termination in the event of a steam generator tube rupture is required for long term core cooling.

   Pressurizer Level indication satisfies both Criteria 3 and 4 of 10 CFR 50.36(c)(2)(ii).

2. <u>Auxiliary Feedwater (AFW) Flow Rate</u>

   AFW Flow indication is used by the operator to confirm that the AFW System is in operation and delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level. The PRA shows that AFW Flow indication can be important to safety by providing information necessary for operator action to initiate alternate feedwater sources in the event of a failure of the AFW system.

   Two channels of AFW Flow indication are required OPERABLE for each steam generator. Due to the redundant indications required operable, the AFW Flow indications to each steam generator are treated as separate PAM Functions. Therefore,

Provided for Information Only.

consistent with General Note 2, separate ACTION statement entry is allowed for Functions 2.a, 2.b, and 2.c.

AFW Flow indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Alternate methods of determining the need for operator action in the event that the AFW flow rate indication is not available can be provided by the AFW pump amperage instrumentation (motor-driven pumps), flow control valve position indication (SG supply), and automatic turbine steam supply valve position indication.

3. Power Range Neutron Flux

The Westinghouse Power Range Neutron Flux indication is used to confirm a reactor shutdown following a design basis accident. The PRA shows that operator actions to manually shutdown the reactor in the event of a failure of the automatic actions, as determined from the Westinghouse Power Range Neutron Flux indication, can be important to safety.

The Power Range Neutron Flux indication satisfies both Criteria 3 and 4 of 10 CFR 50.36(c)(2)(ii).

The PAM Applicability is modified by a footnote that provides an exception to the PAM OPERABILITY requirement for Power Range Neutron Flux indication in MODE 3. The basis for the PAM requirement for Power Range Neutron Flux indication is that it is used to confirm an automatic reactor shutdown from power operation. Therefore, the PAM Power Range Neutron Flux indication requirements are only applicable in MODES 1 and 2 when the power range instrumentation functions to provide the necessary PAM indication.

If the power range neutron flux indications are not available, an alternate method of verifying a reactor trip is a combination of either the intermediate range or source range neutron flux indications and either the rod bottom lights or rod position indicators.

4. High Head Safety Injection (SI) Flow

High Head SI Flow indication is used to confirm automatic safety injection initiation following a design basis accident. Therefore, the required flow indicator for this PAM Function is the one installed in the automatic High Head SI flow path. The results of the PRA shows that this is a risk significant operator action. Failure to manually initiate SI flow when the automatic initiation fails can lead to a significant increase in core damage frequency. The operator action is based on the ECCS flow indication in the control room. The PRA shows that only high head safety injection is important for all accident sequences except the unlikely double-ended guillotine rupture of the largest reactor coolant pipe. Therefore, only the High Head SI Flow indication is required.

High Head SI Flow indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

If the High Head SI Flow indication is not available, an alternate method of verifying SI initiation can be provided by the High Head SI pump amperage indication, the High Head SI header pressure indication, and the SI automatic valve position indication.

Provided for Information Only.

5. Steam Generator (SG) Pressure

SG Pressure indication is provided as a target for RCS depressurization for the steam generator tube rupture accident to terminate the RCS inventory loss. In the event of a steam generator tube rupture accident, the EOPs instruct the operators to depressurize the RCS to a pressure below the secondary side pressure in the ruptured steam generator. RCS depressurization to a pressure less than the steam generator pressure terminates the RCS inventory loss and terminates the steam generator inventory gain, preventing overfill of the steam generator. The termination of the break flow is an operator action assumed in the design basis steam generator tube rupture analysis for which no automatic action is provided. The PRA shows that failure to depressurize the RCS to a pressure less than the secondary side pressure in the ruptured steam generator is a risk significant operator action.

Two channels of pressure indication are required OPERABLE for each steam generator. Due to the redundant indications required operable, the indications for each steam generator are treated as separate PAM Functions. Therefore, consistent with General Note 2, separate ACTION statement entry is allowed for Functions 5.a, 5.b, and 5.c.

SG Pressure indication satisfies both Criteria 3 and 4 of 10 CFR 50.36(c)(2)(ii).

6. Refueling Water Storage Tank (RWST) Level (Wide Range)

RWST Level provides an indication of the water inventory remaining for use by containment spray and safety injection for core cooling and containment cooling. No operator actions in the design basis accident analysis are based on the RWST Level indication. The switchover from the RWST to the containment sump is performed automatically.

The PRA shows that in the event of an accident in which the RCS inventory losses are outside of containment (e.g., steam generator tube rupture and interfacing system LOCA), the remaining RWST level is an important indication for choosing the appropriate operator actions to maintain core cooling in the EOPs. The PRA shows the importance of diagnosing the need for implementing RWST refill to maintain a sufficient inventory for long term core cooling following these events.

RWST Level indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

7. Reactor Coolant System (RCS) Pressure (Wide Range)

RCS Wide Range Pressure indication is provides the information necessary for RCS depressurization for the steam generator tube rupture accident to terminate the RCS inventory loss. In the event of a steam generator tube rupture accident, the EOPs instruct the operators to depressurize the RCS to a pressure below the secondary side pressure in the ruptured steam generator. RCS depressurization to a pressure less than the steam generator pressure terminates the RCS inventory loss and terminates the steam generator inventory gain, preventing overfill of the steam generator. The termination of the break flow is an operator action assumed in the design basis steam generator tube rupture analysis for which no automatic action is provided. RCS pressure is also used for operator action to terminate SI in the event of a steamline break to prevent pressurizer overfill for which no automatic actuation is provided. Additionally, the PRA indicates that RCS pressure is a variable important to safety for RCS cooldown and depressurization following a steam generator tube rupture.

RCS Pressure indication satisfies both Criteria 3 and 4 of 10 CFR 50.36(c)(2)(ii).

The LCO requirement for two channels can be met by using any combination of RCS Pressure Wide Range transmitter(s) or Reactor Vessel Level Indication System Pressure transmitter(s).

8. SG Water Level (Wide Range)

SG Water Level (Wide Range) indication is provided to monitor operation of decay heat removal via the SGs. SG Water Level (Wide Range) indication is used to:

- identify the faulted SG following a steam generator tube rupture,
- verify that the intact SGs are an adequate heat sink for the reactor,
- determine the nature of the accident in progress (e.g., verify a steam generator tube rupture),
- verify unit conditions for the termination of SI during secondary side HELBs outside containment, and
- verify SG tubes are covered before terminating AFW to the faulted SG to assure iodine scrubbing and design basis iodine partitioning in the event of a steam generator tube rupture.

Controlling SG level to maintain a heat sink and the diagnosis of a steam generator tube rupture based on SG level are operator actions assumed in the design basis accident analysis for which no automatic actuation is provided. In addition, the PRA shows that SG Wide Range Level indication can be important to safety by providing information for the initiation of operator actions to establish bleed and feed for a loss of heat sink event.

SG Water Level (Wide Range) indication satisfies both Criteria 3 and 4 of 10 CFR 50.36(c)(2)(ii).

If a channel of wide range SG level instrumentation is not available, an alternate method of monitoring the SG level is a combination of one channel of SG narrow range instrumentation and Auxiliary Feedwater Flow Rate indication to that SG.

9. Containment Area Radiation (High Range)

Containment Area Radiation High Range provides an indication of a loss of one or more fission product barriers. The Emergency Action Levels in the E-Plan utilize the Containment Area Radiation High Range monitor as an indication of the potential loss of one or more fission product barriers in the assessment of the declaration of a General Emergency level and the potential need for offsite radiological protection actions. The post accident Core Damage Assessment also uses the Containment Area Radiation High Range monitor as an input to the determination of core damage. The required high range monitors are designated 2RMR-RQ206 and 207.

The Containment Area Radiation High Range indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

Radiation monitor 2RMR-RQ202B or a portable radiation monitor (with appropriate multiplier if necessary) can be used as an alternate method of indication for Containment Area Radiation High Range.

10. <u>Containment Pressure (Wide Range)</u>

Containment Pressure (Wide Range) indication is provided for assessing containment cooling and containment integrity. No operator actions in the design basis accident analysis are based on the Containment Pressure indication. Containment Pressure is an indicator of the potential loss of a fission product boundary in the Emergency Action Levels in the E-Plan. Containment Pressure is a key indicator in the declaration of a General Emergency level and the potential need for offsite radiological protection actions. Containment pressure may also be used in post accident conditions to determine when to vent the containment to prevent overpressurization.

Containment Pressure (Wide Range) indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

11. <u>Core Exit Temperature</u>

Core Exit Temperature indication is provided for verification and long term surveillance of core cooling. The Core Exit Temperature indication provides information for the operators to initiate RCS depressurization following a steam generator tube rupture. The PRA shows that Core Exit Thermocouple indication is important to safety by providing information necessary to maintain subcooling for RCS cooldown and depressurization following steam generator tube rupture and other small LOCA events. It is also used as an indication for the transfer from the EOPs to the Severe Accident Management Guidance, where a greater focus is maintained on preserving the remaining fission product barriers.

Table 3.3-11 requires two OPERABLE channels of Core Exit Temperature. Footnote (c) to Table 3.3-11 requires a Core Exit Temperature channel to consist of two core exit thermocouples. Two sets of two thermocouples ensure that a single failure will not affect the ability to determine whether an inadequate core cooling condition exists.

Two OPERABLE channels of Core Exit Temperature from any core location except the three outermost rows of fuel assemblies on each side of the core are required to provide the most timely indication of the coolant temperature rise across the core exit. The three outermost rows of fuel assemblies are identified by counting straight in towards the center of the core from the outermost row consisting of three assemblies on each side of the core. The acceptable central core exit thermocouples can also be identified as being within the core area consisting of up to four fuel assemblies from the center fuel assembly (not counting the center assembly).

Provided for Information Only.

Severe accident analyses documented in WCAP-14696-A (Ref. 5) demonstrate that the coolant temperature increase at a central core location (i.e., not in the three outermost rows of fuel assemblies) provides the most rapid indication of inadequate core cooling. Therefore, in order to get the most rapid indication of coolant temperature rise in the core, the two thermocouples in each channel used to meet the LCO requirement must not be located in the three outermost rows of fuel assemblies.

Core Exit Temperature indication satisfies both Criteria 3 and 4 of 10 CFR 50.36(c)(2)(ii).

12. Penetration Flow Path Containment Isolation Valve (CIV) Position

Penetration Flow Path CIV Position indication is provided for verification of Containment Phase A and Phase B isolation. The E-Plan identifies that an elevated emergency action level should be declared following an accident in the event of a failure of the automatic containment isolation.

This requirement only applies to containment isolation valves which receive a Phase A and Phase B containment isolation closure signal. This requirement is not applicable to valves that open on receipt of a Containment Phase A or B signal. When used to verify Phase A and Phase B isolation, the important information is the isolation status of the containment penetrations. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each active CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two active valves that have control room position indication. For containment penetrations with only one active CIV having control room indication, footnote (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve with control room indication and the reliability of containment isolation valves without control room indication (i.e. automatic check valves and relief valves that are not dependent on a external power source or closure signal), or prior knowledge of a passive valve, or via closed system boundary status. If a normally active CIV is known to be closed and deactivated or open under administrative controls in accordance with the provisions of the CIV technical specification, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE. Footnote (a) to the Required Channels states that the Function is not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. Each penetration is treated separately and each penetration flow path is considered a separate function. Therefore, separate Action Statement entry is allowed for each inoperable penetration flow path.

Penetration Flow Path CIV Position indication satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3 except for the Power Range Neutron Flux indication, which is only required to be OPERABLE in MODES 1 and 2. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The

Provided for Information Only.

applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

The Applicability is modified by a footnote affecting the Power Range Neutron Flux indication requirements. The footnote provides an exception to the PAM OPERABILITY requirement for Power Range Neutron Flux indication in MODE 3. The basis for the PAM requirement for Power Range Neutron Flux indication is that it is used to confirm an automatic reactor shutdown from power operation. Therefore, the PAM Power Range Neutron Flux indication requirements are only applicable in MODES 1 and 2 when the power range instrumentation functions to provide the necessary indication.

ACTIONS

A General Note has been added in the ACTIONS to clarify the application of completion time. The ACTIONS of this Specification may be entered independently for each Function listed on Table 3.3-11. The completion time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Action was entered for that Function.

ACTIONS a.1, a.2, and a.3

ACTIONS a.1, a.2, and a.3 apply when one or more Functions have one required channel that is inoperable.

ACTION a.1 requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day completion time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a Function that has only one required channel, other control room indications available to accomplish the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments for a PAM function), and the low probability of an event requiring PAM instrumentation during this interval.

If the inoperable channel is anticipated to remain inoperable for an extended time, it is expected that action will be initiated as soon as possible to confirm the availability, functionality, and procedure impact of any applicable pre-planned alternate instrumentation required to meet ACTION a.2. This will assure the capability of performing the PAM Function is maintained and that preparations are initiated as soon as possible to meet the requirements of Action a.2.

In accordance with ACTION a.2, continuous operation with one required channel inoperable in a Function is acceptable beyond the initial 30 days provided that acceptable alternate instrumentation is available to monitor the Function(s) with an inoperable channel. If the inoperable channel is not restored to OPERABLE status in 30 days, action must be initiated immediately to establish the alternate method of monitoring the affected Function(s). This includes verification that the alternate indication is functionally capable of performing the required PAM Function and the revision of any applicable procedures necessary to implement the alternate method. In addition, ACTION a.2 requires that a report be submitted to the NRC within 14 days after ACTION a.2 becomes applicable. The report discusses the alternate method of monitoring available, the cause of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability, and given the likelihood of unit conditions that would require information provided by this instrumentation.

Provided for Information Only.

ACTION a.3 is applicable if the inoperable channel is not restored to OPERABLE status in 30 days per ACTION a.1 and an alternate method of monitoring the Function(s) is not established and a report is not submitted to the NRC within the following 14 days in accordance with ACTION a.2. In this case, the unit must be placed in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN in the following 6 hours. The allowed Completion Times to place the plant in a condition where the PAM instrumentation is no longer required OPERABLE 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.

ACTIONS b.1, b.2, and b.3

ACTIONS b.1, b.2, and b.3 apply when one or more Functions have two or more inoperable required channels (i.e., two or more channels inoperable in the same Function). When action is completed such that the affected Function(s) only have one inoperable channel, ACTION a is applicable for the remaining inoperable channel not Action b.

ACTION b.1 requires restoring the inoperable channels in the Function(s) to OPERABLE status within 7 days. The completion time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation.

In accordance with ACTION b.2, continuous operation with two or more required channels inoperable in a Function is acceptable beyond the initial 7 days provided that acceptable alternate instrumentation is available to monitor the Function(s) with the inoperable channels. If the required channel(s) (i.e. number of channels necessary to allow exit from action b) is not restored to OPERABLE status in 7 days, action must be initiated immediately to establish the alternate method of monitoring the affected Function(s). This includes verification that the alternate indication is functionally capable of performing the required PAM Function and the revision of any applicable procedures necessary to implement the alternate method. In addition, ACTION b.2 requires that a report must be submitted to the NRC within 7 days after ACTION a.2 becomes applicable. This report discusses the alternate method of monitoring available, the cause of the inoperability, and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified that provide a similar functional capability, and given the likelihood of unit conditions that would require information provided by this instrumentation.

ACTION b.3 is applicable if two or more channels in one Function remain inoperable at the end of the 7 days allowed by ACTION b.1 and an alternate method of monitoring the Function(s) is not established or a report is not submitted to the NRC within the following 7 days in accordance with ACTION b.2. In this case, the unit must be placed in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN in the following 6 hours. The allowed Completion Times to place the plant in a condition where the PAM instrumentation is no longer required OPERABLE 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.

Provided for Information Only.

SURVEILLANCE
REQUIREMENTS

A General Note has been added to the SRs to clarify the following:

- Surveillance Requirement 4.3.3.8.1 applies to each PAM instrument Function in Table 3.3-11,
- Surveillance Requirement 4.3.3.8.2 applies to each PAM instrument Function in Table 3.3-11 except for the Penetration Flow Path Containment Isolation Valve Position instrument Function, and
- Surveillance Requirement 4.3.3.8.3 applies only to the Penetration Flow Path Containment Isolation Valve Position instrument Function in Table 3.3-11.

SR 4.3.3.8.1

Performance of the CHANNEL CHECK at least once every 31 days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

In addition, it is not necessary to place a system or component in service that is not normally in service (e.g., initiate AFW flow to the SGs) in order to perform the required CHANNEL CHECK. In cases where the required instrumentation may be energized but only a single channel is available (e.g., HHSI Flow) or where there may be no flow (e.g., AFW Flow), the CHANNEL CHECK may be accomplished by comparing the indicated value to the known plant condition (e.g., zero flow). In the case of CIVs, the CHANNEL CHECK may be accomplished by comparing the indicated valve position to the known or expected valve position based on current plant conditions.

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

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels associated with the LCO required channel indications that are utilized during normal plant operation.

SR 4.3.3.8.2

A CHANNEL CALIBRATION is performed at least once every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter with

Provided for Information Only.

the necessary range and accuracy.

The Surveillance Requirements are modified by a General Note that states the CHANNEL CALIBRATION surveillance is not applicable to the Containment Isolation Valve Position Function. The Containment Isolation Valve Position Function is verified OPERABLE by a CHANNEL FUNCTIONAL TEST rather than a CHANNEL CALIBRATION.

This SR is also modified by a footnote that excludes neutron detectors from the CHANNEL CALIBRATION. The calibration method for neutron detectors is described in the Bases for 3 / 4.3.1 Reactor Trip System.

Whenever a core exit temperature thermocouple is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors is accomplished by an in-place cross calibration that compares the other sensing elements with the recently installed sensing element. The Frequency of 18-months is based on operating experience and consistency with the typical industry refueling cycle.

SR 4.3.3.8.3

A CHANNEL FUNCTIONAL TEST is only required for the Penetration Flow Path Containment Isolation Valve Position Function. This test is required to be performed at least once every 18 months, or approximately at every refueling. A CHANNEL FUNCTIONAL TEST verifies OPERABILITY of the containment isolation valve position indication instrumentation.

A general Note modifies the surveillance requirements and specifies that the CHANNEL FUNCTIONAL TEST surveillance is only applicable to the Penetration Flow Path Containment Isolation Valve Position Function. Due to the relatively simple instrument circuits involved and the lack of a conventional process sensor to adjust, the CHANNEL FUNCTIONAL TEST, rather than the CHANNEL CALIBRATION, provides the more appropriate OPERABILITY verification of these channels.

The Frequency of 18-months is consistent with the typical industry refueling cycle.

REFERENCES

1. Regulatory Guide 1.97, Rev. 2, December 1980.
2. NUREG-0737, Supplement 1, "TMI Action Items."
3. WCAP-15981, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants."
4. License Amendment number [Later] and the associated NRC Safety Evaluation Report dated [Later].
5. WCAP-14696-A, Revision 1, "Westinghouse Owners Group Core Damage Assessment Guidance."

Attachment C-1

Beaver Valley Power Station, Unit No. 1
Proposed Draft Typed Technical Specification Pages

License Amendment Request No. 314

The following Technical Specification pages are provided for information only:

IV
XVII
3/4 3-34
3/4 3-35
3/4 3-36
3/4 3-50
3/4 3-51
3/4 3-52

INDEX

Provided for Information Only.

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Provided for Information Only.

(Proposed Wording)

TABLE 3.3-6

Provided for Information Only.

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT[3]	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Deleted					
b. Containment					
i. Purge & Exhaust Isolation (RMVS 104 A & B)	2	(2)	$\leq 1.6 \times 10^3$ cpm	$10 - 10^6$ cpm	22
c. Control Room Isolation (RM-RM-218 A & B)	2	1,2,3,4 and (4)	≤ 0.47 mR/hr	$10^{-2} - 10^3$ mR/hr	41
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection (RM 215B)	1	1,2,3 & 4	N/A	$10 - 10^6$ cpm	20
ii. Particulate Activity RCS Leakage Detection (RM 215A)	1	1,2,3 & 4	N/A	$10 - 10^6$ cpm	20
b. Deleted					

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

(1) (Not used)
(2) During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.
(3) Above background.
(4) During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies.

ACTION STATEMENTS

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - This Action is not used.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 35 - This Action is not used.

ACTION 41 - a) With the number of Unit 1 OPERABLE channels one less than the Minimum Channels OPERABLE requirement:

1. Verify the respective Unit 2 control room radiation monitor train is OPERABLE within 1 hour and at least once per 31 days.

Provided for Information Only.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Deleted				
b. Containment				
i. Purge & Exhaust Isolation (RMVS 104 A & B)	S	R	M	**
c. Control Room Isolation (RM-RM-218 A & B)	S	R	M###	1,2,3,4, and ##
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection (RM 215B)	S	R#	M	1,2,3 & 4
ii. Particulate Activity RCS Leakage Detection (RM 215A)	S	R#	M	1,2,3 & 4
b. Deleted				

** During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

\# Surveillance interval may be extended to the upcoming refueling outage if the interval between refueling outages is greater than 18 months.

\## During movement of irradiated fuel assemblies and during movement of fuel assemblies over irradiated fuel assemblies.

\### Control Room intake and exhaust isolation dampers are not actuated.

Provided for Information Only.

POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The PAM instrumentation for each Function in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

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

Separate ACTION statement entry is allowed for each Function.

- -

a.1 One or more PAM Functions with one required channel inoperable, restore the required channel to OPERABLE status within 30 days, or

a.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 14 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channel of the Function to OPERABLE status, or

a.3 Be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b.1 One or more PAM Functions with two or more required channels inoperable, restore the required channel(s) to OPERABLE status within 7 days, or

b.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 7 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status, or

b.3 Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

---

* Power Range Neutron Flux PAM Function is not required in MODE 3.

SURVEILLANCE REQUIREMENTS

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

Surveillance Requirement 4.3.3.8.1 applies to each PAM Function in Table 3.3-11. Surveillance Requirement 4.3.3.8.2 applies to each PAM Function in Table 3.3-11 except for the Penetration Flow Path Containment Isolation Valve Position Function. Surveillance Requirement 4.3.3.8.3 applies only to the Penetration Flow Path Containment Isolation Valve Position Function in Table 3.3-11.

- -

4.3.3.8.1 Perform a CHANNEL CHECK at least once every 31 days.

4.3.3.8.2 **Perform a CHANNEL CALIBRATION at least once every 18 months.

4.3.3.8.3 Perform a CHANNEL FUNCTIONAL TEST at least once every 18 months.

** Neutron detectors are excluded from the CHANNEL CALIBRATION.

Provided for Information Only.

TABLE 3.3-11

POST ACCIDENT MONITORING INSTRUMENTATION

FUNCTION	REQUIRED CHANNELS
1. Pressurizer Water Level	2
2. Auxiliary Feedwater Flow Rate to Steam Generator (SG)	3 (1 per SG)
3. Power Range Neutron Flux	2
4. High Head Safety Injection Flow	1
5. SG Pressure	
a) SG "A"	2
b) SG "B"	2
c) SG "C"	2
6. Refueling Water Storage Tank Level	2
7. Reactor Coolant System Pressure (Wide Range)	2
8. SG Water Level (Wide Range)	3 (1 per SG)
9. Containment Area Radiation (High Range)	2
10. Containment Pressure (Wide Range)	2
11. Core Exit Temperature	2[c]
12. Penetration Flow Path Containment Isolation Valve Position	2 per penetration flow path[a][b]

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) A channel consists of two core exit thermocouples.

Attachment C-2

Beaver Valley Power Station, Unit No. 2
Proposed Draft Typed Technical Specification Pages

License Amendment Request No. 187

The following Technical Specification pages are provided for information only:

V
3/4 3-40
3/4 3-41
3/4 3-42
3/4 3-43
3/4 3-44
3/4 3-57
3/4 3-58
3/4 3-59

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT(3)	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Deleted					
b. Deleted					
c. Control Room Area (2RMC-RQ201 & 202)	2	1,2,3,4, and (4)	≤ 0.476 mR/hr	10^{-2} to 10^{3} mR/hr	46,47
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity (Xe-133) RCS Leakage Detection (2RMR-RQ303B)	1	1,2,3 & 4	N/A	10^{-6} to 10^{-1} µCi/cc	20
ii. Particulate Activity (I-131) RCS Leakage Detection (2RMR-RQ303A)	1	1,2,3 & 4	N/A	10^{-10} to 10^{-5} µCi/cc	20
b. Deleted					

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT(3)	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS (Continued)					
c. Noble Gas and Effluent Monitors					
i. Deleted					
ii. Containment Purge Exhaust (Xe-133) (2HVR-RQ104A & B)	2	(5)	$\leq 1.01 \times 10^{-3}$ µCi/cc	10^{-6} to 10^{-1} µCi/cc	22

Provided for Information Only.

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

(1) Not used.

(2) Not used.

(3) Above background.

(4) During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies.

(5) During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

ACTION STATEMENTS

ACTION 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 21 - This Action is not used.

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 35 - This Action is not used.

Provided for Information Only.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Deleted				
b. Deleted				
c. Control Room Area (2RMC-RQ201 & 202)	S	R	M	1, 2, 3, 4, and ##
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity RCS Leakage Detection (2RMR-RQ303B)	S	R#	M	1, 2, 3 & 4
ii. Particulate Activity RCS Leakage Detection (2RMR-RQ303A)	S	R#	M	1, 2, 3 & 4
b. Deleted				

Surveillance interval may be extended to the upcoming refueling outage if the interval between refueling outages is greater than 18 months.

During movement of recently irradiated fuel assemblies and during movement of fuel assemblies over recently irradiated fuel assemblies.

Provided for Information Only.

TABLE 4.3-3 (Continued)

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
2. PROCESS MONITORS (Continued)				
c. Noble Gas Effluent Monitors				
i. Deleted				
ii. Containment Purge Exhaust (2HVR-RQ104A & B)	S	R	M	###

During movement of recently irradiated fuel assemblies within the containment and during movement of fuel assemblies over recently irradiated fuel assemblies within the containment.

Provided for Information Only.

POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The PAM instrumentation for each Function in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

GENERAL NOTE

Separate ACTION statement entry is allowed for each Function.

a.1 One or more PAM Functions with one required channel inoperable, restore the required channel to OPERABLE status within 30 days, or

a.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 14 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channel of the Function to OPERABLE status, or

a.3 Be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b.1 One or more PAM Functions with two or more required channels inoperable, restore the required channel(s) to OPERABLE status within 7 days, or

b.2 Initiate action immediately to establish a preplanned alternate method of monitoring the PAM Function and submit a report to the NRC within the following 7 days to outline the alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status, or

b.3 Be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

* Power Range Neutron Flux PAM Function is not required in MODE 3.

SURVEILLANCE REQUIREMENTS

GENERAL NOTE

Surveillance Requirement 4.3.3.8.1 applies to each PAM Function in Table 3.3-11. Surveillance Requirement 4.3.3.8.2 applies to each PAM Function in Table 3.3-11 except for the Penetration Flow Path Containment Isolation Valve Position Function. Surveillance Requirement 4.3.3.8.3 applies only to the Penetration Flow Path Containment Isolation Valve Position Function in Table 3.3-11.

4.3.3.8.1 Perform a CHANNEL CHECK at least once every 31 days.

4.3.3.8.2 **Perform a CHANNEL CALIBRATION at least once every 18 months.

4.3.3.8.3 Perform a CHANNEL FUNCTIONAL TEST at least once every 18 months.

** Neutron detectors are excluded from the CHANNEL CALIBRATION.

Provided for Information Only.

TABLE 3.3-11

POST ACCIDENT MONITORING INSTRUMENTATION

FUNCTION	REQUIRED CHANNELS
1. Pressurizer Water Level	2
2. Auxiliary Feedwater Flow Rate to Steam Generator (SG)	
a) SG "A"	2
b) SG "B"	2
c) SG "C"	2
3. Power Range Neutron Flux	2
4. High Head Safety Injection Flow	1
5. SG Pressure	
a) SG "A"	2
b) SG "B"	2
c) SG "C"	2
6. Refueling Water Storage Tank Level (Wide Range)	2
7. Reactor Coolant System Pressure (Wide Range)	2
8. SG Water Level (Wide Range)	3 (1 per SG)
9. Containment Area Radiation (High Range)	2
10. Containment Pressure (Wide Range)	2
11. Core Exit Temperature	2[c]
12. Penetration Flow Path Containment Isolation Valve Position	2 per penetration flow path[a][b]

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) A channel consists of two core exit thermocouples.

Attachment D

Table 1

BVPS Units 1 and 2 Regulatory Guide 1.97
Instrumentation Summary Disposition

WCAP-15981 Plant Specific Implementation

License Amendment Request No. 314 (Unit 1) and No. 187 (Unit 2)

The following Table provides a summary of the WCAP-15981 evaluation and resulting disposition of all the BVPS Unit 1 and Unit 2 PAM instrumentation. The Table includes the following instrumentation:

- All BVPS instrumentation identified as monitoring Regulatory Guide 1.97 Type A variables and all BVPS instrumentation identified as monitoring Regulatory Guide Category 1 variables,
- All PAM related instrumentation included in the current Tech Specs (i.e., Accident Monitoring, and Radiation Monitoring Technical Specifications), and
- Any additional BVPS PAM instrumentation that corresponds to the instrumentation evaluated in WCAP-15981 (e.g., SI Flow).

In addition, the Table 1 Notes include references that identify the source documents (submittals and SERs) that establish BVPS compliance with Regulatory Guide 1.97.

Table 1
BVPS Units 1 and 2 Regulatory Guide 1.97 Instrumentation Summary Disposition
WCAP-15981 Plant Specific Implementation
BVPS License Amendment Requests 314 (Unit 1) and 187 (Unit 2)

BVPS Regulatory Guide 1.97 Variable Type and Category (1) (2) (6)		Contained in Current BVPS Tech Specs for PAM Related Function / Tech Spec Title and Function Number		WCAP-15981 Screening Criteria Evaluation Results	10CFR 50.36 (c)(2)(ii) Criteria Evaluation Results	Disposition Include in Tech Spec 3/4.3.3.8 Relocate from Tech Specs/ No Change	Results of WCAP-15981 Evaluation (3)
Unit 1	**Unit 2**	**Unit 1**	**Unit 2**				
RCS Pressure (Wide Range) A1, B1, C1	RCS Pressure (Wide Range) A1, B1, C1, B2, C2, D2	No	No	BVPS evaluation indicates importance in DBA analysis and PRA.	BVPS evaluation indicates Criteria 3 and 4 of 50.36 (c)(2)(ii) satisfied.	Include in Tech Spec 3/4.3.3.8-Function Number 7	A1 (See LAR Section 4.7)
RCS T Hot (Wide Range) A1, B1	RCS T Hot (Wide Range) A1, B2	No	No	BVPS evaluation indicates WCAP criteria not met.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	No change.	B3 (See LAR Section 4.10)
RCS T Cold (Wide Range) A1, B1, B3	RCS T Cold (Wide Range) A1, B2	No	No	BVPS evaluation indicates WCAP criteria not met.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	No change.	B3 (See LAR Section 4.10)
SG Level (Wide Range) D1	SG Level (Wide Range) A1, B1, B2, D2	No	No	BVPS evaluation indicates importance in DBA analysis and PRA.	BVPS evaluation indicates Criteria 3 and 4 of 50.36 (c)(2)(ii) satisfied.	Include in Tech Spec 3/4.3.3.8-Function Number 8	A1 (See LAR Section 4.7)

Table 1 *BVPS Units 1 and 2 Regulatory Guide 1.97 Instrumentation Summary Disposition* **WCAP-15981 Plant Specific Implementation** **BVPS License Amendment Requests 314 (Unit 1) and 187 (Unit 2)**							
BVPS Regulatory Guide 1.97 Variable - Type and Category(1) (2) (6)		**Contained in Current BVPS Tech Specs for PAM Related Function / Tech Spec Title and Function Number**		**WCAP-15981 Screening Criteria Evaluation Results**	**10CFR 50.36 (c)(2)(ii) Criteria Evaluation Results**	**Disposition** **Include in Tech Spec 3/4.3.3.8 Relocate from Tech Specs/ No Change**	**Results of WCAP-15981 Evaluation(3)**
Unit 1	**Unit 2**	**Unit 1**	**Unit 2**				
SG Level (Narrow Range) A1	SG Level (Narrow Range) A1, B1, D2	No	No	BVPS evaluation indicates WCAP criteria not met.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	No change.	B2 (See LAR Section 4.10)
Pressurizer Level A1, D1	Pressurizer Level A1, B1, D2	Accident Monitoring Instrumentation 3.3.3.8 - 1.	Accident Monitoring Instrumentation 3.3.3.8 -1.	BVPS evaluation indicates importance in DBA analysis and PRA.	BVPS evaluation indicates Criterion 3 and 4 of 50.36 (c)(2)(ii) satisfied.	Include Tech Spec 3/4.3.3.8- Function Number 1	A1 (See LAR Section 4.7)
Containment Pressure (Narrow Range -10 - 55 psig) A1, B1, C1	Containment Pressure (Narrow Range -5 - 55 psig) A1, B1, B2, C2, D2	No	No	BVPS evaluation indicates WCAP criteria not met.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	No change.	B3 (See LAR Section 4.10)
Containment Pressure (Wide Range 0 - 200 psia) C1	Containment Pressure (Wide Range 0 - 180 psia) C1, C2	Accident Monitoring Instrumentation 3.3.3.8 - 10.	Accident Monitoring Instrumentation 3.3.3.8 - 9.	BVPS evaluation indicates importance in PRA analysis and E-Plan.	BVPS evaluation indicates Criterion 4 of 50.36 (c)(2)(ii) satisfied.	Include Tech Spec 3/4.3.3.8- Function Number 10	C1 (See LAR Section 4.7)

Table 1 *BVPS Units 1 and 2 Regulatory Guide 1.97 Instrumentation Summary Disposition* **WCAP-15981 Plant Specific Implementation** **BVPS License Amendment Requests 314 (Unit 1) and 187 (Unit 2)**							
BVPS Regulatory Guide 1.97 Variable **Type and Category(1) (2) (6)**		**Contained in Current BVPS Tech Specs for PAM Related Function / Tech Spec Title and Function Number**		**WCAP-15981 Screening Criteria Evaluation Results**	**10CFR 50.36 (c)(2)(ii) Criteria Evaluation Results**	**Disposition** **Include in Tech Spec 3/4.3.3.8 Relocate from Tech Specs/ No Change**	**Results of WCAP-15981 Evaluation(3)**
Unit 1	**Unit 2**	**Unit 1**	**Unit 2**				
SG Pressure A1, D2	Steamline Pressure A1, B1, D2	No	No	BVPS evaluation indicates importance in DBA analysis and PRA.	BVPS evaluation indicates Criteria 3 and 4 of 50.36 (c)(2)(ii) satisfied.	Include Tech Spec 3/4.3.3.8- Function Number 5	A1 (See LAR Section 4.7)
Containment Sump Level (Wide Range) A1, B1, C1	Containment Water Level (Wide Range) A1, B1, B2, C2, D2	Accident Monitoring Instrumentation 3.3.3.8 - 9.	Accident Monitoring Instrumentation 3.3.3.8 - 8.	BVPS evaluation indicates WCAP criteria not satisfied.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	Relocate to LRM.	B2 (See LAR Section 4.9)
N/A (Not Type A or Cat. 1)	Containment Water Level (Narrow Range) A1, B1, B2, C2, D2	No	No	BVPS evaluation indicates WCAP criteria not satisfied.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	No change.	B2 (See LAR Section 4.10)
RWST Level (0 - 55 Ft.) A1, D2	RWST Level (0 - 730 in.) D2	No	No	BVPS evaluation indicates importance in PRA.	BVPS evaluation indicates Criterion 4 of 50.36 (c)(2)(ii) satisfied.	Include in Tech Spec 3/4.3.3.8- Function Number 6	D1 (See LAR Section 4.7)

Table 1

BVPS Units 1 and 2 Regulatory Guide 1.97 Instrumentation Summary Disposition

WCAP-15981 Plant Specific Implementation

BVPS License Amendment Requests 314 (Unit 1) and 187 (Unit 2)

BVPS Regulatory Guide 1.97 Variable Type and Category(1) (2) (6)		Contained in Current BVPS Tech Specs for PAM Related Function / Tech Spec Title and Function Number		WCAP-15981 Screening Criteria Evaluation Results	10CFR 50.36 (c)(2)(ii) Criteria Evaluation Results	Disposition Include in Tech Spec 3/4.3.3.8 Relocate from Tech Specs/ No Change	Results of WCAP-15981 Evaluation(3)
Unit 1	**Unit 2**	**Unit 1**	**Unit 2**				
Primary Plant DWST Level (AFW System Supply) D1	Primary Plant DWST Level (AFW System Supply) A1, D2	No	No	BVPS evaluation indicates WCAP criteria not satisfied.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	No change.	B2 (See LAR Section 4.10)
AFW Flow D2	AFW Flow A1, B1, D2	Accident Monitoring Instrumentation 3.3.3.8 - 2.	Accident Monitoring Instrumentation 3.3.3.8 - 2.	BVPS evaluation indicates importance in PRA.	BVPS evaluation indicates Criterion 4 of 50.36 (c)(2)(ii) satisfied.	Include in Tech Spec 3/4.3.3.8- Function Number 2	B1 (See LAR Section 4.7)
Core Exit Temperature B3, C1	Core Exit Temperature A1, B1, C1	Accident Monitoring Instrumentation 3.3.3.8 - 11.	Accident Monitoring Instrumentation 3.3.3.8 - 11.	BVPS evaluation indicates importance in DBA analysis and PRA.	BVPS evaluation indicates Criteria 3 and 4 of 50.36 (c)(2)(ii) satisfied.	Include in Tech Spec 3/4.3.3.8- Function Number 11	A1 (See LAR Section 4.7)
Containment Area Radiation A1, C3, E1	Containment Area Radiation (High Range) A1, B1, B2, E2	Radiation Monitoring 3.3.3.1 - 1. b. ii.	Radiation Monitoring 3.3.3.1 - 1. b.	BVPS evaluation indicates importance in E-Plan and core damage assessment.	BVPS evaluation indicates Criterion 4 of 50.36 (c)(2)(ii) satisfied.	Include in Tech Spec 3/4.3.3.8- Function Number 9	C1 (See LAR Section 4.7)

Table 1
BVPS Units 1 and 2 Regulatory Guide 1.97 Instrumentation Summary Disposition
WCAP-15981 Plant Specific Implementation
BVPS License Amendment Requests 314 (Unit 1) and 187 (Unit 2)

BVPS Regulatory Guide 1.97 Variable Type and Category(1) (2) (6)		Contained in Current BVPS Tech Specs for PAM Related Function / Tech Spec Title and Function Number		WCAP-15981 Screening Criteria Evaluation Results	10CFR 50.36 (c)(2)(ii) Criteria Evaluation Results	Disposition Include in Tech Spec 3/4.3.3.8 Relocate from Tech Specs/ No Change	Results of WCAP-15981 Evaluation(3)
Unit 1	**Unit 2**	**Unit 1**	**Unit 2**				
N.A. (Not Type A or Cat. 1)	Secondary System Radiation (1 per loop) A1, B2, E2	No	Radiation Monitoring 3.3.3.1- 2. c. iii.	BVPS evaluation indicates WCAP criteria not satisfied.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	Relocate Unit 2 Monitor to ODCM (Effluent Monitor)	C3 (See LAR Section 4.0, Change No. 2)
RCS Subcooling B2	RCS Subcooling A2, B2	Accident Monitoring Instrumentation 3.3.3.8 - 3.	Accident Monitoring Instrumentation 3.3.3.8 - 3.	BVPS evaluation indicates WCAP criteria not satisfied.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	Relocate to LRM.	B3 (See LAR Section 4.9)
Neutron Flux Source Range B1	Neutron Flux Lower Range (4) B1	No	No	BVPS evaluation indicates WCAP criteria not satisfied.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	No Change	B3 (See LAR Section 4.11)
Neutron Flux Intermediate Range B1	N/A (Not identified as a separate RG 1.97 variable) (4)	No	No	BVPS evaluation indicates WCAP criteria not satisfied.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	No Change	B3 (See LAR Section 4.11)

Table 1 *BVPS Units 1 and 2 Regulatory Guide 1.97 Instrumentation Summary Disposition* **WCAP-15981 Plant Specific Implementation** **BVPS License Amendment Requests 314 (Unit 1) and 187 (Unit 2)**							
BVPS Regulatory Guide 1.97 Variable Type and Category(1) (2) (6)		**Contained in Current BVPS Tech Specs for PAM Related Function / Tech Spec Title and Function Number**		**WCAP-15981 Screening Criteria Evaluation Results**	**10CFR 50.36 (c)(2)(ii) Criteria Evaluation Results**	**Disposition Include in Tech Spec 3/4.3.3.8 Relocate from Tech Specs/ No Change**	**Results of WCAP-15981 Evaluation(3)**
Unit 1	**Unit 2**	**Unit 1**	**Unit 2**				
Neutron Flux Power Range B1	Neutron Flux Upper Range (4) B1	No	No	BVPS evaluation indicates importance in PRA.	BVPS evaluation indicates Criterion 4 of 50.36 (c)(2)(ii) satisfied.	Include in Tech Spec 3/4.3.3.8-Function Number 3. (Westinghouse Power Range monitors only)	B1 (See LAR Section 4.7)
Reactor Vessel Level B1	Reactor Vessel Level B2, C2	Accident Monitoring Instrumentation 3.3.3.8 - 12.	Accident Monitoring Instrumentation 3.3.3.8 - 10.	BVPS evaluation indicates WCAP criteria not satisfied.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	Relocate to LRM	B3 (See LAR Section 4.9)
Containment Hydrogen Concentration A1, C1, E3	Containment Hydrogen Concentration B1, C1	No	No	N/A	N/A	Hydrogen Analyzer requirements Relocated by Amendment 259 (Unit 1) and Amendment 142 (Unit 2) in accordance with TSTF-447.(5)	N/A

Table 1
BVPS Units 1 and 2 Regulatory Guide 1.97 Instrumentation Summary Disposition
WCAP-15981 Plant Specific Implementation
BVPS License Amendment Requests 314 (Unit 1) and 187 (Unit 2)

BVPS Regulatory Guide 1.97 Variable Type and Category(1) (2) (6)		Contained in Current BVPS Tech Specs for PAM Related Function / Tech Spec Title and Function Number		WCAP-15981 Screening Criteria Evaluation Results	10CFR 50.36 (c)(2)(ii) Criteria Evaluation Results	Disposition Include in Tech Spec 3/4.3.3.8 Relocate from Tech Specs/ No Change	Results of WCAP-15981 Evaluation(3)
Unit 1	**Unit 2**	**Unit 1**	**Unit 2**				
Containment Isolation Valve Position B1	Containment Isolation Valve Position C2, D2	No	No	BVPS evaluation indicates importance in E-Plan.	BVPS evaluation indicates Criterion 4 of 50.36 (c)(2)(ii) satisfied.	Include in Tech Spec 3/4.3.3.8-Function Number 12	B1 (See LAR Section 4.7)
Radiation Level in Primary Coolant C1	N/A (Not Type A or Cat. 1)	No	No	BVPS evaluation indicates WCAP criteria not satisfied.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	No Change	C3 (See LAR Section 4.11)
High Pressure SI Flow (0-1000 gpm) D2	High Head SI Flow (0-1000 gpm) D2	No	No	BVPS evaluation indicates importance in PRA.	BVPS evaluation indicates Criterion 4 of 50.36 (c)(2)(ii) satisfied.	Include in Tech Spec 3/4.3.3.8-Function Number 4	B1 (See LAR Section 4.7)
Primary System Safety Relief Valve Position D2	Primary Safety Valve Status D2	Accident Monitoring Instrumentation 3.3.3.8 - 7.	Accident Monitoring Instrumentation 3.3.3.8 - 6.	BVPS evaluation indicates WCAP criteria not satisfied.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied.	Relocate to LRM.	D2 (See LAR Section 4.9)

Table 1 *BVPS Units 1 and 2 Regulatory Guide 1.97 Instrumentation Summary Disposition* **WCAP-15981 Plant Specific Implementation** **BVPS License Amendment Requests 314 (Unit 1) and 187 (Unit 2)**							
BVPS Regulatory Guide 1.97 Variable Type and Category(1) (2) (6)		**Contained in Current BVPS Tech Specs for PAM Related Function / Tech Spec Title and Function Number**		**WCAP-15981 Screening Criteria Evaluation Results**	**10CFR 50.36 (c)(2)(ii) Criteria Evaluation Results**	**Disposition Include in Tech Spec 3/4.3.3.8 Relocate from Tech Specs/ No Change**	**Results of WCAP-15981 Evaluation(3)**
Unit 1	**Unit 2**	**Unit 1**	**Unit 2**				
PORV Position No RG 1.97 Type or Category assigned	PORV Position D2	Accident Monitoring Instrumentation 3.3.3.8 - 5.	Accident Monitoring Instrumentation 3.3.3.8 - 4.	BVPS evaluation indicates WCAP criteria not met.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied	Relocate to LRM.	D2 (See LAR Section 4.9)
PORV Block Valve Position (No RG 1.97 Type or Category assigned)	PORV Block Valve Position (No RG 1.97 Type or Category assigned)	Accident Monitoring Instrumentation 3.3.3.8 - 6.	Accident Monitoring Instrumentation 3.3.3.8 - 5.	BVPS evaluation indicates WCAP criteria not met.	BVPS evaluation indicates that none of the four Criteria of 50.36 (c)(2)(ii) satisfied	Relocate to LRM.	D2 (See LAR Section 4.9)

TABLE 1 - NOTES

1. The list of instrumentation in Table 1 includes the following:

   a) All BVPS instrumentation identified as Regulatory Guide 1.97 Type A and/or Category 1,

   b) All instrumentation included in the current Technical Specifications for PAM related functions (i.e., Accident Monitoring, Radiation Monitoring, and Hydrogen Analyzer Technical Specifications), and

   c) Any additional BVPS instrumentation that corresponds to the instrumentation evaluated in WCAP-15981 (e.g., High Head SI Flow).

2. BVPS Unit 1 and Unit 2 reference documents for Regulatory Guide 1.97 compliance:

UNIT 1		UNIT 2	
SUBMITTAL	SERs	SUBMITTAL	SER
1. Duquesne Light Letter dated 10/13/86, Subject: Regulatory Guide 1.97, Revision 2, Supplemental Report (Complete RG 1.97 report attached) 2. Duquesne Light Letter dated 4/22/87, Subject: RG 1.97, Revision 2, Response to Interim Review Results. (Item 10, A1 classification of the Primary Plant Demineralized Water Storage Tank Level is removed). 3. Duquesne Light Letter dated 12/18/89, Subject: Response to NRC RG 1.97 Concerns. (Page 4, A1 classification of AFW Flow removed).	1. NRC Letter dated 11/20/89, Subject: Completion of Review of Regulatory Guide 1.97 Conformance (TAC No. 51071). 2. NRC Letter dated 12/30/91, Subject: Emergency Response Capability - Conformance to Regulatory Guide 1.97 (TAC No. M75944). 3. NRC Letter dated 6/15/92, Subject: Emergency Response Capability - Conformance To Regulatory Guide 1.97 (TAC No. M75944). 4. NRC Letter dated 11/17/95, Subject: Conformance to Regulatory Guide 1.97, Revision 2, Post-Accident Neutron Flux Monitoring Instrumentation for BVPS Unit 1 (TAC No. M81201).	UFSAR Table 7.5-1	NUREG-1057, Supplement No. 1, Section 7.5, May 1986. Original BVPS Unit 2 SER

3. The results of the WCAP-15981 evaluations performed in this LAR are for the sole purpose of determining the most appropriate BVPS instrumentation to be included in Technical Specification 3/4.3.3.8, "Post Accident Monitoring (PAM) Instrumentation." The proposed changes in this LAR do not include revising the current BVPS response to Regulatory Guide 1.97 documented in the references above.

4. The Unit 2 response to Regulatory Guide 1.97 refers to the Gamma-Metrics full range monitor. (The Unit 1 response refers to the Westinghouse Nuclear Instrumentation System).

5. Consistent with TSTF-447 Rev. 1, "Elimination of Hydrogen Recombiners and Change to Hydrogen and Oxygen Monitors," licensees must commit to control PAM related instrumentation removed from the Tech Specs in some document or program. For BVPS, the document for controlling items removed from the Tech Specs is the Licensing Requirements Manual (LRM).

TABLE 1 - NOTES

(continued)

6. The following is a brief description from Regulatory Guide 1.97 of the Type and Category classifications assigned to the instruments listed in Table 1.

- Type A — Provides primary information needed to permit control room operating personnel to take specified manual actions for which no automatic action is provided and that are required for safety systems to accomplish their safety functions for design basis accidents.
- Type B — Provides information to indicate whether plant safety functions are being accomplished.
- Type C — Provides information to indicate the potential for breach of fission product barriers.
- Type D — Provides information to indicate the operation of individual safety systems and other systems important to safety.
- Type E — Provides information needed to determine the magnitude of the release of radioactive materials and in continually assessing such releases.
- Category 1 — Key variables that most directly provide information on the accomplishment of a safety function.
- Category 2 — Instrumentation designated for indicating system operating status.
- Category 3 — Backup and diagnostic instrumentation.