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YOUNGBLOOD,B.J. Licensing Branch 1

SUBJECT: Confirms 800930 & 1001 meeting schedules w/Reactor Sys Branch & Structural Engineering Branch,respectively,re sacrificial shield wall at facility.

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Washington Public Power Supply System
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G02-80-210
September 22, 1980

Docket No. 50-397

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. B. J. Youngblood, Chief
Licensing Branch 1
Division of Licensing

Gentlemen:

Subject: WPPSS Nuclear Project No. 2
Confirmation of Sacrificial Shield Wall
and Reactor Systems Branch Meetings

Reference: Letter, DL Renberger, (WPPSS) to BJ Youngblood (NRC), "Request for Meeting - RSB/ICSB/Savannah River", G02-80-190, dated August 29, 1980

This letter is to confirm two meetings which have been scheduled with the staff through the WNP-2 Project Manager, Mr. David Lynch. The first meeting was requested through the referenced letter and was subsequently scheduled for September 30th with the Reactor Systems Branch (RSB) only. WPPSS assumes the RSB questions and the ICSB questions will be given to WPPSS no later than the meeting date. Of course, having the RSB questions in advance would facilitate the meeting.

The second meeting concerning the Sacrificial Shield Wall (SSW) was scheduled for October 1st with the Structural Engineering Branch as a consequence of a planned conference call on September 18th between the Branch and WPPSS and their architect engineer, Burns and Roe. WPPSS plans to respond to the five concerns raised by the Branch in the meeting to facilitate gaining NRR concurrence on the replacement weld on the SSW.

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If there are any questions, please contact our licensing engineer, Mr. O. Keener Earle.

Very truly yours,

D L Renberger

D. L. Renberger
Assistant Director
Technology

DLR:OKE:cph

cc: MD Lynch - NRC (telecopy)
TP Speis - NRC/RSB (telecopy)
FP Schauer - NRC/SEB (telecopy)
J. Ellwanger - B&R
JJ Verderber - B&R
E. Chang - GE, San Jose
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WNP-2 Files

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

50-397
MA/4

SEPTEMBER 1 9 1980

TO: All Licensees of Operating Reactor Plants and Applicants for Operating Licenses and Holders of Construction Permits

SUBJECT: ADDENDUM TO THE CLARIFICATION LETTER FOR TMI ACTION PLAN REQUIREMENTS

By letter dated September 5, 1980, we transmitted a preliminary clarification of the TMI Action Plan requirements. Attached is a set of errata sheets which amend the referenced letter (viz., missing pages, scheduling, Tech Spec considerations, etc.). Also included is a corrected table of the Implementation schedule.

It is our intention to develop and issue model technical specifications after issuance of the final requirements package.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:

OPERATING REACTOR REQUIREMENTS IMPLEMENTATION SCHEDULE
(As of September 5, 1980)

In-Place As of 9/80	Oct-Dec 1980	Jan-Mar 1981	Apr-June 1981	Jul-Sept 1981	Oct-Dec 1981	Jan-Mar 1982	Apr 1982	Jul 1982	Jan 1983
I.A.1.1(P)	I.A.1.3(P)	I.A.1.1(P)	II.B.4	II.B.1(P)	I.C.1(P)	II.B.1(P)	III.A.1.2(P)	I.A.1.3(P)	II.K.3(P)
I.A.1.2	I.A.2.1(P)	I.C.1(P)	II.E.4.1(P)	II.D.1(P)	II.F.1(P)	II.B.2(P)		II.D.1(P)	III.D.3.4(P)
I.A.2.1(P)	I.A.3.1(P)	I.C.5	II.K.2(P)	II.E.1.1(P)		II.B.3(P)			
I.A.2.3	II.K.3(P)	I.C.6	II.K.3(P)	II.E.4.2(P)		II.E.1.1(P)			
I.A.3.1(P)		II.D.3(P)	III.A.1.1(P)	II.K.3(P)		II.F.2(P)			
I.C.1(P)		II.E.1.2(P)				II.K.3(P)			
I.C.2		II.E.3.1(P)							
I.C.3		II.E.4.2(P)							
I.C.4		II.F.1(P)							
II.B.2(P)		II.F.2(P)							
II.B.3(P)		II.G.1(P)							
II.D.1(P)		II.K.1(P)							
II.D.3(P)		II.K.2(P)							
II.E.1.2(P)		II.K.3(P)							
II.E.3.1(P)		III.A.1.2(P)							
II.E.4.1(P)		III.D.1.1(P)							
II.E.4.2(P)		III.D.3.3(P)							
II.F.2(P)		III.D.3.4(P)							
II.G.1(P)									
II.K.1(P)									
II.K.2(P)									
III.A.1.1(P)									
III.A.1.2(P)									
III.D.1.1(P)									
III.D.3.3(P)									

OL REQUIREMENTS
As of September 1980

Fuel Load	Full Power	Oct-Dec 1980	Jan-Mar 1981	Apr-June 1981	July-Sept 1981	Oct-Dec 1981	Jan-Mar 1982	Apr-June 1981	Jan 1, 1983
I.A.1.1(P)	I.C.7(P)	II.B.6	I.A.1.1(P)	II.E.4.1(P)	II.D.1(P)	I.C.1(P)	II.B.1(P)	III.A.1.2(P)	II.K.3(P)
I.A.1.2	I.C.8(P)	II.K.3(P)	I.C.1(P)	II.K.3	II.E.4.2(P)	II.F.1(P)	II.B.2(P)		III.D.3.4(P)
I.A.1.3	I.G.1(P)		II.E.1.2(P)		II.K.3(P)		II.B.3(P)		
I.A.3.1(P)	II.B.1(P)		II.E.4.2(P)				II.D.1(P)		
I.B.1.2	II.B.2(P)		II.F.1(P)				II.F.2(P)		
I.C.1(P)	II.B.3(P)		II.K.3(P)				II.K.3(P)		
I.C.2	II.B.4(P)		III.A.1.2(P)						
I.C.3	II.E.1.1		III.D.3.3(P)						
I.C.4	II.E.3.1								
I.C.5	II.E.4.2								
I.C.7(P)	II.K.2(P)								
I.D.1(P)	II.K.3.3								
I.G.1(P)	III.A.1.1(P)								
II.B.4(P)	III.D.1.1								
II.D.1(P)	III.D.3.4								
II.D.3									
II.E.1.2(P)									
II.E.4.1(P)									
II.F.1(P)									
II.F.2(P)									
II.G.1									
II.K.1(P)									
II.K.3(P)									
III.A.1.1(P)									
III.A.1.2(P)									
III.D.3.3(P)									

(II.F.2)

d. An evaluation, including proposed actions, on the conformance of the inadequate core cooling instrumentation system to Regulatory Guide 1.97, Rev. 2. Any deviations should be justified.

e. A description of the computer functions associated with ICC monitoring and functional specifications for relevant software in the process computer and other pertinent calculators. The reliability of nonredundant computers used in the system should be addressed.

f. A current schedule, including contingencies, for installation, testing and calibration, and implementation of any proposed new instrumentation or information displays.

g. Guidelines for use of the additional instrumentation, and analyses used to develop these procedures.

h. A summary of key operator action instructions in the current emergency procedures for inadequate core cooling and a description of how these procedures will be modified when the final monitoring system is implemented.

i. A description and schedule commitment for any additional submittals which are needed to support the acceptability of the proposed final instrumentation system and emergency procedures for inadequate core cooling.

TECHNICAL SPECIFICATION CHANGES REQUIRED

Yes.

REFERENCES

1. NUREG-0578 (Recommendation 2.1.3.b).

2. H. Denton (NRC) letter to All Operating Nuclear Power Plants on "Discussion of Lessons Learned Short Term Requirements," dated October 30, 1979.

EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

(II.G.1)

POSITION

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss-of-offsite power, the following positions shall be implemented:

1. Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

2. Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

3. Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

4. The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

CLARIFICATION

1. While the prevalent consideration from TMI Lessons Learned is being able to close the PORV/block valves, the design should retain, to the extent practical, the capability to open these valves.

2. The motive and control power for the block valve should be supplied from an emergency power bus different from that which supplies the PORV.

3. Any changeover of the PORV and block valve motive and control power from the normal offsite power to the emergency onsite power is to be accomplished manually in the control room.

4. For those designs where instrument air is needed for operation, the electrical power supply requirements should be capable of being manually connected to the emergency power sources.

APPLICABILITY

All PWR Operating License Applicants

IMPLEMENTATION

Prior to the issuance of a fuel load license.

DOCUMENTATION REQUIRED

Each applicant shall provide sufficient documentation to support a reasonable assurance finding by the NRC that each of the positions stated above are met. The documentation should include as a minimum, supporting information including system design description, logic diagrams, electrical schematics, test procedures, and technical specifications.

REFERENCES

NUREG-0578, (Recommendation 2.1.1)
NUREG-0694, (Part 1)
NUREG-0660, (Section II.G.1)

CONTROL OF AFW INDEPENDENT OF ICS

(II.K.2.2)

POSITION

For B&W-designed reactors, provide procedures and training to initiate and control auxiliary feedwater independent of the integrated control system (ISC).

CLARIFICATION

None required

APPLICABILITY

All Operating License Applicants with B&W-designed reactors

IMPLEMENTATION

Prior to issuance of a full power license

DOCUMENTATION REQUIRED

Applicants shall provide sufficient documentation at leat four months prior to the issuance of a full power license to support a reasonable assurance finding by the NRC that the position specified above has been met.

TECHNICAL SPECIFICATIONS REQUIRED

No.

REFERENCES

NUREG-0660, (Section II.K.2, Table C.2, Item 2)
NUREG-0694, (Part II)

AUXILIARY FEEDWATER SYSTEM UPGRADING

(II.K.2.8)

POSITION

All operating Babcock and Wilcox plants were ordered to be shut down shortly after the TMI-2 accident. The Orders included both short-term and long-term actions. The NRR Bulletins and Orders Task Force reviewed the licensees compliance with the short-term actions of the Orders and issued safety evaluation reports which served as the basis for plant restart. Additional items were identified in the review of the long-term actions which requires further work by the licensees.

CLARIFICATION

The licensees were required to comply with the Commission Orders regarding certain short-term and long-term AFWS modifications. The staff evaluated the short-term actions, and safety evaluations were prepared prior to the plants being allowed to return to operation. The staff evaluation of the additional (long-term) items will be performed in conjunction with Item II.E.1.1, (Auxiliary Feedwater System Evaluation).

APPLICABILITY

All B&W Operating Reactors.

IMPLEMENTATION

No separate implementation is required for this item. All AFW system upgrade modifications for B&W plants are being reviewed as part of Item II.E.1.1.

TYPE OF REVIEW

See Item II.E.1.1.

DOCUMENTATION REQUIRED

See Item II.E.1.1.

TECHNICAL SPECIFICATION CHANGES REQUIRED

As required.

REFERENCES

NUREG-0660, (Sections II.E.1.1 and II.K.2)
NUREG-0645, Volume 1, (Section 2.4.6)

FAILURE MODE EFFECTS ANALYSIS ON ICS

(II.K.2.9)

POSITION

B&W licensees submit a failure mode and effects analysis (FMEA) of the integrated control system (ICS).

CLARIFICATION

A generic FMEA of the ICS (BAW-1564) was submitted on August 17, 1979 by the operating plant licensees. This report was reviewed by the staff and ORNL. Requests for additional information, regarding the recommendations contained in the report, were sent to the licensees on November 7, 1979. The responses to the November 7, 1979 letter have been received and are under review.

APPLICABILITY

All B&W Operating Reactors and Operating License Applicants.

IMPLEMENTATION

Operating Reactors

Open - Staff recommendations pending completion of staff review.

Operating License Applicants

Prior to issuance of a full-power license.

TYPE OF REVIEW

Postimplementation review.

DOCUMENTATION REQUIRED

Operating Reactors

To be determined following staff review.

Operating License Applicants

B&W applicants should provide the following:

1. Identify whether the previous generic submittal (BAW-1564) is applicable to your plant, and

2. Specify what actions have been taken at your facility to comply with the recommendations listed in BAW-1564.

TECHNICAL SPECIFICATION CHANGES REQUIRED

To be determined following staff review.

REFERENCES

Commission Orders on B&W Plants

"Integrated Control System Reliability Analysis," BAW-1564.

Letter from R. W. Reid (NRC) to All B&W Operating Plants, dated November 7, 1979.

NUREG-0645, (Volume 1, Section 2.4.6)
NUREG-0694, (Part 2)

SAFETY-GRADE ANTICIPATORY REACTOR TRIP

(II.K.2.10)

POSITION

Upgrade the currently installed control-grade, anticipatory reactor trip (ART) on loss-of-feedwater and turbine trip to safety-grade.

CLARIFICATION

Operating Reactors

1. IE Bulletin 79-05B, Item 5, issued on April 21, 1979, directed B&W licensees to provide a design and schedule for implementation of a safety-grade reactor trip upon:

   a. loss of feedwater;
   b. turbine trip; and
   c. significant reduction in steam generator level.

2. In accordance with IE Bulletin 79-05B, the B&W licensees submitted a conceptual design for a safety-grade, anticipatory reactor trip which would be initiated upon turbine trip and loss of feedwater only. Included in the licensees' responses was a generic evaluation prepared by B&W which proposed that the anticipatory reactor trip on low steam generator level was not necessary.

3. Staff review of these submittals resulted in a preliminary design approval for the safety-grade anticipatory reactor trip being issued to the B&W licensees on December 20, 1979. However, the approval letters also specified the additional information which would be required to be submitted prior to final staff approval of the design.

4. The staff will complete its review of the generic evaluation by B&W which indicates that the proposed anticipatory trip on low steam generator level is unnecessary. Further clarification will be provided on this matter, if required, following completion of the staff review.

Operating License Applicants

Compliance with TMI Action Plan, Item II.K.1.21, satisfies this requirement.

APPLICABILITY

All B&W Operating Reactors and Operating License Applicants.

IMPLEMENTATION DATE

Operating Reactors

Submission of final design information - October 1, 1980
Installation of safety-grade trip - June 30, 1981

Operating License Applicants

Implementation of TMI Action Plan Item II.K.1.21 prior to the issuance of the fuel load satisfies this requirement.

TYPE OF REVIEW

Preimplementation Review.

DOCUMENTATION REQUIRED

The following information was identified as required by the staff for the final design approval, as noted in item 3 above:

1. The final design submittal should include the final logic diagrams, electrical schematic diagrams, piping and instrumentation diagrams and location layout drawings.

2. For sensors located in nonseismic areas which have not previously contained RPS inputs, perform and submit an analysis which shows that the installation (including circuit routing) is designed such that the effects of credible faults (i.e., grounding, shorting, application of high voltage, or electromagnetic interference) or failures in these areas could not be propagated back to the RPS and degrade the RPS performance or operability.

3. Submit "Seismic and Environmental Qualification Summary Reports" for the equipment which has not been previously submitted. In addition, demonstrate that the environmental test conditions bound the actual worst case accident conditions expected at the installed locations.

4. Assure that the ARTs testability includes provisions to perform channel functional tests at power. Testing of this circuitry is to be included in the RPS monthly surveillance tests.

5. Include in the final design submittal the RPS check-out procedure which will demonstrate both the operability of the new trip circuitry and the continued operability of the previous RPS.

The above information should be submitted for staff review by October 1, 1980.

TECHNICAL SPECIFICATION CHANGES REQUIRED

Yes.

REFERENCES

Commission Orders on B&W Plants
IE Bulletin 79-05B, Item 5
Letter from R. W. Reid (NRC) to B&W Licensees, date December 20, 1979
Subject: Preliminary design approval for safety-grade anticipatory reactor trip and request for additional information.
NUREG-0645, (Volume 1, Section 2.4.6)
NUREG-0694, (Part 2)

CONTINUED OPERATOR TRAINING AND DRILLING

(II.K.2.11)

POSITION

Continue operator training and drilling to assure a high state of preparedness.

CLARIFICATION

In a letter from D. F. Ross, Jr. (NRC) to All B&W Operating Plants, dated August 21, 1979, each B&W licensee was requested to document the steps they had taken to insure that continued operator training and drilling incorporated the necessary lessons learned from the accident at TMI-2. This response was required to assure compliance with the long-term training requirements of the Commission Orders.

Responses to this request were received from the licensees and reviewed by the NRC staff. Based on that review, the staff concluded that the training programs had been sufficiently modified to incorporate the necessary lessons learned from TMI such that this portion of the Commission Orders was satisfied. A complete evaluation of this item is discussed in Section 2.4.6 of NUREG-0645, Volume 1.

Additional requirements, beyond the intent of the Commission Orders, are being implemented through the following items of the Action Plan: I.A.2.2, I.A.2.5, I.A.3.1, and I.G.1.

APPLICABILITY

All B&W Operating Reactors

IMPLEMENTATION DATE

COMPLETED

TYPE OF REVIEW

Postimplementation review.

DOCUMENTATION REQUIRED

No additional documentation required.

TECHNICAL SPECIFICATIONS REQUIRED

No.

REFERENCES:

Letter from D. F. Ross Jr. (NRC), to ALL BABCOCK & WILCOX OPERATING PLANTS (EXCEPT THREE MILE ISLAND, UNITS 1 & 2), dated August 21, 1979, Subject: Identification and Resolution of Long-Term Generic Issues Related to the Commission Orders of May 1979.

NUREG-0645, "Report of the Bulletins & Orders Task Force," Volume I, January 1980.

NUREG-0660, Section II.K.2, Item C.11.

THERMAL MECHANICAL REPORT-EFFECT OF HPI ON VESSEL INTEGRITY FOR SMALL BREAK LOCA WITH NO AFW

(II.K.2.13)

POSITION

Perform a detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

CLARIFICATION

The position deals with the potential for thermal shock of B&W reactor vessels resulting from cold safety injection flow. One aspect that bears heavily on the effects of safety injection flow is the mixing of safety injection water with reactor coolant in the reactor vessel. B&W has committed to provide a report by the end of July which will discuss the mixing question and the basis for a conservative analysis of the potential for thermal shock to the reactor vessel.

APPLICABILITY

All B&W Operating Reactors and Operating License Applicants.

IMPLEMENTATION DATE

Confirmatory information requested. Implementation of any modifications will be subject to the results of NRC staff review of the report.

TYPE OF REVIEW

Postimplementation Review.

DOCUMENTATION REQUIRED

Licensees shall submit results of evaluation by January 1, 1981.
Applicants shall submit results of evaluation at least four months prior to the issuance of a full power license.

TECHNICAL SPECIFICATION CHANGES REQUIRED

To be determined following staff review.

REFERENCE

NUREG-0645, (Volume 1 Section 2.4.5)
Letter from D. F. Ross Jr. (NRC) to All B&W Operating Plants, dated August 21, 1979.

EFFECTS OF SLUG FLOW ON STEAM GENERATOR TUBES

(II.K.2.15)

POSITION

While the staff believed that the potential for slug flow was not great in B&W plants, because of the venting path provided by the internal vent valves, the staff required a confirmatory evaluation of the effects of slug flow on steam generator tubes be performed by the licensees to assure that the tubes could withstand any mechanical loading which could result from slug flow.

CLARIFICATION

The request for this information was originally sent to the B&W licensees in a letter from R. W. Reid (NRC) to All B&W Operating Plants dated November 21, 1979.

The results of this analysis has been submitted by the licensees and is presently undergoing NRC staff review.

APPLICABILITY

All B&W Operating Reactors and Operating License Applicants.

IMPLEMENTATION DATE

Confirmatory information requested. Implementation of any modifications will be subject to the results of NRC staff review of the evaluation.

TYPE OF REVIEW

Postimplementation.

DOCUMENTATION REQUIRED

No additional documentation is required at this time from Licensees. Applicants must supply the requested information at least four months prior to issuance of a full power license.

TECHNICAL SPECIFICATION CHANGES REQUIRED

No.

REFERENCES

Letter from R. W. Reid (NRC) to All B&W Operating Plants, dated November 21, 1979.
NUREG-0565, (Recommendation 2.6.2.1)
NUREG-0645, (Volume 1, Section 2.4.6)
NUREG-0694, (Part 2)

REACTOR COOLANT PUMP SEAL DAMAGE

(II.K.2.16)

POSITION

Evaluate the impact of reactor coolant pump seal damage and leakage due to loss of seal cooling upon loss of offsite power. If damage cannot be precluded, licensees should provide an analysis of the limiting small-break LOCA with subsequent RCP seal damage.

CLARIFICATION

The request for this information was originally sent to the B&W licensees in a letter from R. W. Reid (NRC) to All B&W Operating Plants dated November 21, 1979.

The results of these evaluations have been submitted by the licensees and are presently undergoing NRC staff review.

APPLICABILITY

All B&W Operating Reactors and Operating License Applicants.

IMPLEMENTATION DATE

Confirmatory information requested. Implementation of any modifications will be subject to the results of NRC staff review of the evaluations.

TYPE OF REVIEW

Postimplementation.

DOCUMENTATION REQUIRED

No additional documentation is required at this time from Licensees. Applicants shall submit the requested information at least four months prior to the issuance of a full power license.

TECHNICAL SPECIFICATION CHANGES REQUIRED

No.

REFERENCES

Letter from R. W. Reid (NRC) to All B&W Operating Plants, dated November 21, 1979.
NUREG-0565, (Recommendation 2.6.2.f)
NUREG-0645, (Volume 1, Section 2.4.6)
NUREG-0694 (Part 2)

POTENTIAL FOR VOIDING IN THE RCS DURING TRANSIENTS

(II.K.2.17)

POSITION

Analyze the potential for voiding in the reactor coolant system during anticipated transients.

CLARIFICATION

The background for this concern and a request for this analysis was originally sent to the B&W licensees in a letter from R., W. Reid (NRC) to All B&W Operating Plants, dated January 9, 1980.

The results of this evaluation has been submitted by the B&W licensees and is presently undergoing staff review.

APPLICIABILITY

All B&W Operating Reactors.

IMPLEMENTATION DATE

Confirmatory information requested. Implementation of any modifications will be subject to the results of NRC staff review of the licensees' evaluation.

TYPE OF REVIEW

Postimplementation Review.

DOCUMENTATION REQUIRED

No additional documentation is required at this time.

TECHNICAL SPECIFICATION CHANGES REQUIRED

No.

REFERENCE

Letter from R. W. Reid (NRC) to All B&W Operating Plants, dated January 9, 1980.
NUREG-0660, Section II.K.2 Item C.17.

SEQUENTIAL AFW FLOW ANALYSIS

(II.K.2.19)

POSITION

Provide a benchmark analysis of sequential auxiliary feedwater flow to the steam generators following a loss of main feedwater.

CLARIFICATION

This requirement was originally sent to the B&W licensees in a letter from D. F. Ross Jr. (NRC) to All B&W Operating Plants, dated August 21, 1979.

The results of this analysis has been submitted by the B&W licensees and is presently undergoing staff review.

APPLICABILITY

All B&W Operating Reactors.

IMPLEMENTATION DATE

Confirmatory information requested. Implementation of any modifications will be subject to the results of NRC staff review of this analysis.

TYPE OF REVIEW

Postimplementation Review.

DOCUMENTATION REQUIRED

No additional documentation is required at this time.

TECHNICAL SPECIFICATION CHANGES REQUIRED

No.

REFERENCE

Letter from D. F. Ross Jr. (NRC) to All B&W Operating Plants, dated August 21, 1979.
NUREG-0645, Volume 1, Section 2.4.6.

SMALL-BREAK LOCA WHICH REPRESSURIZES THE RCS TO THE PORV SETPOINT

(II.K.2.20)

POSITION

Provide an analysis which shows the plant response to a small break loss-of-coolant accident during which the reactor coolant system is repressurized to the PORV setpoint with subsequent failure of the PORV to close.

CLARIFICATION

The requirements was originally sent to the B&W licensees in a letter from D. F. Ross Jr. (NRC) to All B&W Operating Plants, dated August 21, 1979.

The results of this analysis has been submitted by the B&W licensees and is presently undergoing staff review.

APPLICABILITY

All B&W Operating Reactors.

IMPLEMENTATION DATE

Confirmatory information requested. Implementation of any modifications will be the subject to the results of NRC staff evaluation of this analysis.

TYPE OF REVIEW

Postimplementation Review.

DOCUMENTATION REQUIRED

No additional documentation is required at this time.

TECHNICAL SPECIFICATION CHANGES REQUIRED

No.

REFERENCES

Letter from D. F. Ross Jr. (NRC) to All B&W Operating Plants, dated August 21, 1979.
NUREG-0565, Recommendation 2.6.2.c
NUREG-0645, Volume 1, Section 2.4.6