FENOC

FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
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March 28, 2001
L-01-047

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. 287 and 159

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company requests an amendment to the above licenses in the form of changes to the technical specifications (TS).

The proposed license amendment request (LAR) will revise the Beaver Valley Power Station (BVPS) Unit 1 and 2 TS to implement improvements endorsed in the NRC's Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, 58 FR 39132, July 22, 1993 (the policy statement). This LAR proposes changes that implement two enhancements from the policy statement and Improved Standard Technical Specifications (ISTS). The major change proposed in this LAR involves the application of the TS screening criteria from 10 CFR 50.36 to evaluate the content of the BVPS TS and identify those TS that do not meet the criteria. TS requirements that do not meet the criteria are proposed for relocation from the BVPS TS consistent with the guidance in the policy statement. This LAR also proposes the addition of an administrative program for TS Bases control to the BVPS TS consistent with the content of the ISTS.

This LAR represents the start of the BVPS TS conversion to the ISTS contained in NUREG-1431. BVPS currently plans to complete the conversion to the ISTS in two main phases. The first phase of the conversion involves changes to the BVPS TS that relocate TS consistent with the policy statement guidance and adopt certain administrative programs applicable to BVPS. The second phase of the BVPS conversion plan will include a single LAR that will encompass the re-organization, reformat, and expanded scope of the remaining TS consistent with the applicable content of NUREG-1431 and the licensing and design basis of the BVPS units. The second phase LAR will also include the development and submittal of the BVPS expanded bases documents associated with each TS in the ISTS. The second phase of the BVPS conversion plan is scheduled to start when Revision 2 to NUREG-1431 is approved.

A001

The BVPS plan to provide separate LAR(s) for the conversion to the ISTS was previously discussed in a meeting with the NRC staff on October 14, 1999 and documented in the NRC minutes for that meeting dated November 18, 1999. The relocation of TS from the license represents a significant portion of the work in developing, reviewing, approving and implementing an ISTS conversion LAR. By converting to the ISTS in two phases, the impact of an ISTS conversion on BVPS and NRC resources will be spread over time and effectively reduced. In addition, the submittal of this relocation LAR allows for an early start of the BVPS ISTS conversion project. During the October 14, 1999 meeting with the NRC, the changes being introduced to the standard TS in Revision 2 to NUREG-1431 were discussed. This revision to the standard TS contains a significant number of changes that will affect the BVPS TS conversion. Due to the number and scope of the changes, the NRC staff recommended that BVPS utilize Revision 2 for the conversion effort. However, the approval of Revision 2 to NUREG-1431 was delayed. Therefore, in order to begin the conversion process, BVPS is focusing first on the TS requirements that are normally relocated during a conversion. This part of the conversion process can be accomplished separately and is not affected by the revised content of NUREG-1431.

The proposed TS changes for Unit No. 1 and Unit No. 2 are presented in Attachments A-1 and A-2, respectively. The safety analysis (including the no significant hazards evaluation) is presented in Attachment B.

These changes have been reviewed by the Beaver Valley review committees. The changes were determined to be safe and do not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety analysis. Considering the number of plant procedures potentially affected by this change, an implementation period of up to 120 days is requested following the effective date of this amendment.

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

Sincerely,

Lew W. Myers

Lew W. Myers

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

Subject: **Beaver Valley Power Station, Unit No. 1 and No. 2**
BV-1 Docket No. 50-334, License No. DPR-66
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License Amendment Request Nos. 287 and 159

I, Lew W. Myers, being duly sworn, state that I am Senior Vice President of FirstEnergy Nuclear Operating Company (FENOC), that I am authorized to sign and file this submittal with the Nuclear Regulatory Commission on behalf of FENOC, and that the statements made and the matters set forth herein pertaining to FENOC are true and correct to the best of my knowledge and belief.

FirstEnergy Nuclear Operating Company

Lew W. Myers

Lew W. Myers
Senior Vice President - FENOC

COMMONWEALTH OF PENNSYLVANIA

COUNTY OF BEAVER

Subscribed and sworn to me, a Notary Public, in and for the County and State above named, this 28 th day of March, 2001.

Sheila M. Fattore

My Commission Expires:

Notarial Seal
Sheila M. Fattore, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Sept. 30, 2002
Member, Pennsylvania Association of Notaries

ATTACHMENT A-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 287
TECHNICAL SPECIFICATION RELOCATION AND BASES CONTROL

The following is a list of the affected pages:

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BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Refueling Water Storage Tank (RWST)

(proposed Wording)

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ADMINISTRATIVE CONTROLS

3/4.1.2.1 - 3/4.1.2.7 (These specification numbers are not used)

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

RELOCATE

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid storage system via a boric acid transfer pump to a charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7.a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump or a low head safety injection pump (with an open RCS vent of greater than or equal to 2.07 square inches) to the Reactor Coolant System if only the refueling water storage tank in Specification 3.1.2.7.b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

(proposed Wording)

(Next Page is 3/4 1-16)

Delete page

→ RELOCATE

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the temperature of the heat traced portion of the flow path is ≥ 65°F when a flow path from the boric acid tanks is used and the ambient air temperature of the Auxiliary Building is < 65°F.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

Delete page

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

→ RELOCATE

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and one charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via one charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTOOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the flow path from the refueling water storage tank inoperable, restore the flow path to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

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→ RELOCATE

SURVEILLANCE REQUIREMETNS (Continued)

2. Verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is ≥ 65°F when the ambient air temperature of the Auxiliary Building is < 65°F.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown by cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

Delete page

RELOCATE

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump(1) in the boron injection flow path required by Specification 3.1.2.1 or low head safety injection pump (with an open reactor coolant system vent of greater than or equal to 2.07 square inches) shall be OPERABLE and capable of being powered from an OPERABLE bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above pumps OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump or low head safety injection pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE pursuant to Specification 4.5.2.b.1.

4.1.2.3.2 When the low head safety injection pump is used in lieu of a charging pump, the low head safety injection pump shall be demonstrated OPERABLE by:

a. Verification of an OPERABLE RWST pursuant to 4.1.2.7,

b. Verification of an OPERABLE low head safety injection pump pursuant to Specification 4.5.2.b.2,

c. Verification of an OPERABLE low head safety injection flow path from the RWST to the Reactor Coolant System once per shift, and

d. Verification that the vent is open in accordance with 4.4.9.3.3.

(1) With two charging pumps OPERABLE, follow Specification 3.4.9.3.

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CHARGING PUMPS - OPERATING

→ RELOCATE

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4[1].

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 Each charging pump shall be demonstrated OPERABLE pursuant to Specification 4.5.2.b.1.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the inservice RCS cold legs is ≤ the enable temperature set forth in Specification 3:4.9.3 by verifying that the control switches are placed in the PULL-TO-LOCK position and tagged.

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(1) A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the non-isolated RCS cold legs is ≤ the enable temperature set forth in Specification 3.4.9.3.

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REACTIVITY CONTROL SYSTEMS

BORIC ACID TRANSFER PUMPS - SHUTDOWN → RELOCATE

LIMITING CONDITION FOR OPERATION

3.1.2.5 One boric acid transfer pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid transfer pump of Specification 3.1.2.1a, is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid transfer pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid transfer pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required boric acid transfer pump shall be demonstrated OPERABLE, on recirculation flow, by verifying that the pump develops a discharge pressure greater than or equal to 107 psig when tested pursuant to Specification 4.0.5.

Delete page

REACTIVITY CONTROL SYSTEMS

BORIC ACID TRANSFER PUMPS - OPERATING → RELOCATE

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least one boric acid transfer pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid transfer pump OPERABLE, restore at least one boric acid transfer pump to OPERABLE STATUS within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F; restore at least one boric acid transfer pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required boric acid transfer pump shall be demonstrated OPERABLE, on recirculation flow, by verifying that the pump develops a discharge pressure greater than or equal to 107 psig when tested pursuant to Specification 4.0.5.

Delete page

BORATED WATER SOURCES - SHUTDOWN → RELOCATE

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum contained volume of 5000 gallons,

2. Between 7000 and 7700 ppm of boron, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 175,000 gallons,

2. A minimum boron concentration of 2000 ppm, and

3. A minimum solution temperature of 45°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the water level of the tank, and

3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside ambient air temperature is <45°F.

Refueling Water Storage Tank (RWST)

BORATED WATER SOURCES - OPERATING → RELOCATE

LIMITING CONDITION FOR OPERATION

The RWST shall be OPERABLE.

3.1.2.8 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2.

a. A boric acid storage system with:

1. A minimum contained volume of 11,336 gallons,
2. Between 7000 and 7700 ppm of boron, and
3. A minimum solution temperature of 65°F.

→ RELOCATE

b. The refueling water storage tank with:

1. A contained volume between 439,050 gallons and 441,100 gallons of borated water,
2. A boron concentration between 2000 and 2100 ppm, and
3. A solution temperature of ≥ 45°F and ≤ 55°F.

→ MOVE TO SURVEILLANCES

APPLICABILITY: MODES 1, 2, 3 & 4.

ACTION: → RELOCATE

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

→ RELOCATE

The RWST shall be verified OPERABLE:

(Proposed Wording)

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by: → RELOCATE

1. Verifying the boron concentration in each water source,

2. Verifying the water level in each water source, and

3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the RWST ambient air temperature is < 45°F or > 55°F.

solution

is ≥ 45°F and ≤ 55°F

AT least once per 7 days by:

1. Verifying the boron concentration is between 2,000 and 3,100 ppm, and

2. Verifying a contained volume between 439,050 gallons and 441,100 gallons of borated water

* THIS PAGE INCLUDED FOR INFORMATION ONLY. NO CHANGES TO THIS PAGE.

POSITION INDICATION SYSTEMS-OPERATING

NOTE 1 BELOW IS REFERENCED FROM PAGE 3/4 1-20c.

LIMITING CONDITION FOR OPERATION

3.1.3.2 The shutdown and control rod position indication system shall be OPERABLE as follows:

Group Demand Counter(1), 1 per group

Individual analog rod position instrument channel, 1 per rod ±12 steps(1) accuracy(3)

(1) During the first hour following rod motion, the group demand counter is the primary indicator of precise rod position information, with the analog channels displaying general rod movement information. For power levels below 50%, a 1-hour thermal soak time is allowed before the analog channels are required to perform within the specified accuracy.

(2) For power levels below 50% a one hour thermal soak time is allowed.

(3) Malfunctions of the group demand counters or analog RPI, providing no actual rod misalignment existed during the malfunction, shall be reported in the monthly operating report.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.2.2 Each of the analog rod position indicators shall be determined to be OPERABLE by:

a. Performing a CHANNEL CHECK by intercomparison** between each analog rod position indicator and its corresponding group demand counter at least once per 12 hours.

b. Performing a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at least once per 18 months.

For Core PHYSICS TESTING in Mode 2, primary detector voltage measurements may be used to determine the position of rods in shutdown banks A and B and control banks A and B for the purpose of satisfying Specification 3.1.3.2. During Mode 2 operation, rod position indicators for shutdown banks A and B and control banks A and B may deviate from the group demand indicators by greater than ± 12 steps during reactor startup and shutdown operations, while rods are being withdrawn or inserted. If the rod position indicators for shutdown banks A and B and control banks A and B deviate by greater than ± 12 steps from the group demand indicator, rod withdrawal or insertion may continue until the desired group height is achieved. When the desired group height is achieved, a one hour soak time is allowed below 50% reactor power to permit stabilization of the rod position analog indicators. To attain thermal equilibrium during the one hour soak time, the absolute value of rod motion shall not exceed 6 steps.

** For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

Verifying that the analog rod position indicators agree with the demand position indicators within 12 steps(1) over the full-range of indicated rod travel at least once per 18 months.

Reviewers NOTE: Note (1) in above insert is referenced from page 3/4 1-20 (previous page)

(proposed Wording)

3/4.1.3.3 (This Specification number is not used)

POSITION INDICATION SYSTEM - SHUTDOWN

RELOCATE

LIMITING CONDITION FOR OPERATION

3.1.3.3 The group demand position indicators shall be OPERABLE and capable of determining within ± 2 steps the demand position for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3*, 4*, and 5*

ACTION:

With less than the above required group demand position indicators OPERABLE, open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required group demand position indicator(s) shall be determined to be OPERABLE by movement of the associated control rod at least 10 steps in any one direction at least once per 31 days when the reactor coolant system pressure is >400 psig.

* With the reactor trip system breakers in the closed position.

(proposed wording)

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

Table is duplicated in ODCM/LRM for relocated monitors

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT[3]	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
Deleted a. Fuel Storage Pool Area (RM-207) → RELOCATE	1	(1)	≤ 15 mR/hr	10^{-1} - 10^{4} mR/hr	19
b. Containment					
i. Purge & Exhaust Isolation (RMVS 104 A & B)	1	6	≤ 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	1,2,3,4,5[4],6[4] (in either unit)	≤ .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. Fuel Storage Building Gross Activity (RMVS-103 A & B)	1	(2)	≤ 4.0 x 10^{4} cpm	10 - 10^{6} cpm	21

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TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

→ RELOCATE

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT(3)	MEASUREMENT RANGE	ACTION
PROCESS MONITORS (Continued)					
2.c. Noble Gas Effluent Monitors					
i. Supplementary Leak Collection and Release System (RM-VS-110 Ch. 7 & Ch. 9)(7)	1	1,2,3,&4	$\leq 7.98 \times 10^2$cpm	10^{-2}-10^5uCi/cc(5)	35
ii. Auxiliary Building Ventilation System (RM-VS-109 Ch. 7 & Ch. 9)(7)	1	1,2,3,&4	$\leq 6.69 \times 10^2$cpm	10^{-2}-10^5uCi/cc(5)	35
iii. Process Vent System (RM-GW-109 Ch. 7 & Ch. 9)(7)	1	1,2,3,&4	$\leq 1.83 \times 10^5$cpm	10^{-2}-10^5uCi/cc(6)	35
iv. Atmospheric Steam Dump Valve and Code Safety Relief Valve Discharge (RM-MS-100 A, B, C)	1/SG	1,2,3,&4	$\leq 5.0 \times 10^1$cpm	10^{-1}-10^3uCi/cc	35
v. Auxiliary Feedwater Pump Turbine Exhaust (RM-MS-101)	1	1,2,3,&4	$\leq 6.5 \times 10^2$cpm	10^{-1}-10^3uCi/cc	35

TABLE 3.3-6 (Continued)

(Not used)

TABLE NOTATIONS

(1) With fuel in the storage pool or building. → RELOCATE

Duplicate IN ODCM LRM

(2) With Irradiated fuel in the storage pool.

(3) Above background.

(4) During movement of irradiated fuel or movement of heavy loads over spent fuel.

(5) Nominal range for Ch. 7 and Ch. 9. Alarm set on Ch. 7.

(6) Nominal range for Ch. 7 and Ch. 9. Alarm set on Ch. 9.

(7) Other SPING-4 channels not applicable to this specification.

→ RELOCATE

ACTION STATEMENTS → RELOCATE

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

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 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the applicable ACTION requirements of Specification 3.9.12 and 3.9.13.

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:

→ Duplicate in ODCM

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.

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.

(proposed Wording)

Duplicate Table ODCM/LRM

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Deleted

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area (RM 207)	S	R	M	*
b. Containment				
i. Purge & Exhaust Isolation (RMVS 104 A & B)	S	R	M	6
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###	1,2,3,4,5##,6## (in either unit)
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. Fuel Storage Building Gross Activity (RMVS-103 A & B)	S	R	M	**

RELOCATE

* With fuel in the storage pool or building. → RELOCATE
** With irradiated fuel in the storage pool.
\# 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.
\#\#\# Control Room intake and exhaust isolation dampers and CREBAPS solenoid valves are not actuated.

(proposed Wording)

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TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
PROCESS MONITORS (Continued)				
c. Noble Gas Effluent Monitors				
i. Supplementary Leak Collection and Release System (RM-VS-110 Ch. 7 & Ch. 9)	S	R	M	1, 2, 3, & 4
ii. Auxiliary Building Ventilation System (RM-VS-109 Ch. 7 & Ch. 9)	S	R	M	1, 2, 3, & 4
iii. Process Vent System (RM-GW-109 Ch. 7 & Ch. 9)	S	R	M	1, 2, 3, & 4
iv. Atmospheric Steam Dump Valve and Code Safety Relief Valve Discharge (RM-MS-100 A, B, C)	S	R	M	1, 2, 3, & 4
v. Auxiliary Feedwater Pump Turbine Exhaust (RM-MS-101)	S	R	M	1, 2, 3, & 4

RELOCATE

Delete page

LOOP ISOLATION VALVES - SHUTDOWN → RELOCATE

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 The loop isolation valves in an isolated RCS loop shall have power removed from the associated loop isolation valve operators(1).

APPLICABILITY:

Whenever an RCS loop has been isolated, MODES 5 and 6(2).

ACTION:

With the requirements of the above specification not satisfied, remove power from the isolated loop isolation valve operators(1) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 Verify at least once per 7 days that power is removed from the RCS isolated loop isolation valve operators(1).

---

(1) Power may be restored to the associated RCS isolated loop isolation valve operators provided the requirements of Surveillance Requirement 4.4.1.5.2 have been satisfied.

(2) With fuel in the vessel.

3/4.4.2 (This Specification number is not used)

3/4.4.2 SAFETY VALVES - SHUTDOWN

→ RELOCATE

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting* of 2485 PSIG +1% -3%.**

APPLICABILITY: MODES 4 and 5.

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. With a pressurizer code safety valve having discharged liquid water from a water solid pressurizer to mitigate an overpressure event, borate to a SHUTDOWN MARGIN equivalent to at least 1% ΔK/K at 200°F within the next 24 hours. Inspect that valve for potential damage, initiate corrective action to return the valve to operable status prior to increasing RCS temperature and document the inspection results in the Annual Report pursuant to Specification 6.9.1.5.b.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional requirements other than those required by Specification 4.0.5.

---

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

** Within ±1% following pressurizer code safety valve testing.

(proposed Wording)

3/4.4.3 SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting* of 2485 PSIG +1% -3%.**

APPLICABILITY: MODES 1, 2 and 3 or MODE 4 with all RCS cold leg temperatures > the enable temperature specified in 3.4.9.3.

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

b. With a pressurizer code safety valve having discharged liquid water from a water solid pressurizer to mitigate an overpressure event, be in at least HOT STANDBY within the next 6 hours, and in HOT SHUTDOWN within the following 6 hours.

with any RCS cold leg temperature ≤ the enable temperature specified in 3.4.9.3 and apply RCS overpressure protection requirements in accordance with Specification 3.4.9.3.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional requirements other than those required by Specification 4.0.5.

---

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

** Within ± 1% following pressurizer code safety valve testing.

(proposed Wording)

(3/4.4.7 (This Specification number is not used)

3/4.4.7 CHEMISTRY

RELOCATE

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the Parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one of more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At all other times

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to ≤ 500 psig, if applicable, and perform an analysis to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-10.

(Next Page is 3/4 4-18)

(Proposed Wording)

Delete page

→ RELOCATE

TABLE 3.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN	≤ 0.10 ppm*	≤ 1.00 ppm*
CHLORIDE	≤ 0.15 ppm	≤ 1.50 ppm
FLUORIDE	≤ 0.15 ppm	≤ 1.50 ppm

* Limit not applicable with T_{avg} ≤ 250°F.

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TABLE 4.4-10 → RELOCATE

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

CONTAMINANT	MINIMUM SAMPLING FREQUENCIES	MAXIMUM TIME BETWEEN SAMPLES
DISSOLVED OXYGEN	3 times per 7 days*	72 hours
CHLORIDE	3 times per 7 days	72 hours
FLUORIDE	3 times per 7 days	72 hours

* Not required with $T_{avg} \leq 250°F$

3/4.4.9.2 (This Specification number is not used)

RELOCATE

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits:

a. Restore the temperature to within the limits within 30 minutes, and

b. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer within 72 hours, and

c. Determine, from Action b above, that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.9.2.2 The normal spray water temperature differential shall be determined to be within the limit at least once per 30 minutes during system heatup or cooldown.

4.4.9.2.3 The auxiliary spray water temperature differential shall be determined to be within the limit at least once per 30 minutes during auxiliary spray operation.

(proposed Wording)

3/4.4.10 (This Specification number is not used)

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 AND 3 COMPONENTS

RELOCATE

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.10 Each ASME Code Class 1, 2, and 3 component shall be demonstrated OPERABLE in accordance with Specification 4.0.5.

(proposed Wording)

Delete page

3/4.4.12 REACTOR COOLANT SYSTEM VENTS

→ RELOCATE

LIMITING CONDITION FOR OPERATION

3.4.12 All reactor coolant system vent valves, powered from emergency buses, shall be OPERABLE* and closed** for each vent path from the following locations:

a. Reactor Vessel Head

b. Pressurizer Steam Space

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With at least one vent path from each of the above locations OPERABLE and one or more power operated vent valves inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable valve(s) is maintained closed with power removed or with the manual isolation valve closed. Power operation may continue until the next scheduled outage, at which time all reactor coolant system vent valves shall be OPERABLE prior to entry into MODE 1. The provisions of Specification 3.0.4 are not applicable.

b. With all vent paths from one of the above locations inoperable, maintain the inoperable valves closed with power removed or with the manual isolation valves closed, restore at least one of the inoperable vent paths to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With all vent paths from both of the above locations inoperable, maintain the inoperable valves closed with power removed or close the manual isolation valves, and restore at least one vent path from one of the above locations to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.12 Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by:

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SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying all manual isolation valves in each vent path are locked or sealed in the open position.

2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room.

3. Verifying flow through the reactor coolant system vent path to the Pressurizer Relief Tank.

---

* For purposes of this specification an inoperable vent valve is defined as: a valve which exhibits leakage in excess of Specification 3.4.6.2 limits, or cannot be opened and closed on demand, or does not have its normal emergency power supply OPERABLE.

** These valves may be operated for required venting operations and leak testing in Modes 3 and 4.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two separate and independent containment quench spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment quench spray subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment quench spray subsystem shall be demonstrated OPERABLE;

a. At least once per 31 days, by:

1. By verifying that each valve (manual, power-operated, or automatic) in the flow path not locked, sealed, or otherwise secured in position, is in its correct position; and

2. ~~Verifying the temperature of the borated water in the refueling water storage tank is within the limits of Specification 3.1.2.8.b.3~~

b. By verifying, at the frequency specified in the Inservice Testing Program, that each quench spray pump's developed head at the flow test point is greater than or equal to the required developed head as specified in the Inservice Testing Program and the Containment Integrity Safety Analysis.

(Proposed Wording)

3/4.7.2 (This Specification number is not used)

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 70°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to ≤ 200 psig within 30 minutes, and

b. Perform an analysis to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generator shall be determined to be < 200 psig at least once per hour when the temperature of either the primary or secondary coolant in the steam generator is < 70°F.

RELOCATE

(proposed Wording)

3/4.7.6 (This Specification number is not used)

3/4.7.6 FLOOD PROTECTION → RELOCATE

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Ohio River exceeds 695 Mean Sea Level at the intake structure.

APPLICABILITY: At all times.

ACTION:

With the water level at the intake structure above elevation 695 Mean Sea Level:

a. Be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours, and

b. Initiate and complete within 8 hours, the following flood protection measures:

   1. Install and seal the flood doors in the intake structure.

SURVEILLANCE REQUIREMENTS

4.7.6.1 The water level at the intake structure shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is below elevation 690 Mean Sea Level, and

b. Measurement at least once per 2 hours, by initiating a flood watch including communications between plant operators and upstream dam operators, when the water level is equal to or above elevation 690 Mean Sea Level.

(proposed Wording)

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3/4.7.9 SEALED SOURCE CONTAMINATION

→ RELOCATE

LIMITING CONDITION FOR OPERATION

3.7.9.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma-emitting material or 5 microcuries of alpha-emitting material shall be free of ≥ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

   a. The licensee, or

   b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.9.1.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

RELOCATE

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SURVEILLANCE REQUIREMENTS (Continued)

a. Sources in use - At least once per six months for all sealed sources containing radioactive materials.

   1. With a half-life greater than 30 days (excluding Hydrogen 3) and

   2. In any form other than gas.

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.9.1.3 Reports - A Special Report shall be prepared and submitted in accordance with 10 CFR 50.4 on an annual basis if sealed source or fission detector leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

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PLANT SYSTEMS → RELOCATE

3/4.7.12 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.12 All snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on non safety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems# required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.12.d on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.12 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.7-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment 167.

These systems are defined as those portions or subsystems required to prevent releases in excess of 10 CFR 100 limits.

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RELOCATE

SURVEILLANCE REQUIREMENTS (Continued)

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; or (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.12.e or 4.7.12.f, as applicable. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

Snubbers which have been determined to be inoperable as a result of unexpected transients, isolated damage, or other random events, and cannot be proven operable by functional testing for the same reasons, shall not be counted in determining the next visual inspection period when the provision in 4.7.12.d (that failures are subject to an engineering evaluation of component structural integrity) has been met and equipment has been restored to an operable state via repair and/or replacement as necessary.

d. Functional Tests

At least once per 18 months during shutdown, a representative sample (of at least 10 snubbers or at least 10% whichever is less) of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.12.e or 4.7.12.f an additional 10 snubbers or at least 10% whichever is less of that type of snubber shall be functionally tested.

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RELOCATE

SURVEILLANCE REQUIREMENTS (Continued)

For each large bore snubber (snubbers greater than 1500 kips) on the reactor coolant system that does not meet the functional test acceptance criteria of Specification 4.7.12.e, an engineering evaluation is required to determine the failure mode. If the failure is determined to be generic, an additional 10% (for each failure) of that type of snubber shall be functionally tested. If the failure is determined to be non-generic, an additional 10% (for each failure) of that type of snubber will be tested during the next functional test period.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle

2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.)

3. Snubbers within 10 feet of the discharge from a safety relief valve.

Snubbers that are especially difficult to remove or in high radiation zones during shutdown shall also be included in the representative sample.*

If a spare snubber has been installed in place of a failed snubber, the spare snubber shall be retested. Test results of this snubber may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

---

* Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

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For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

e. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

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RELOCATE

SURVEILLANCE REQUIREMENTS (Continued)

g. Snubber Service Life Monitoring*

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and may be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with the applicable record retention provision of the quality assurance program description referenced in the Updated Final Safety Analysis Report.

RELOCATE

* For purposes of establishing a baseline for the determination of service life monitoring, this program will be implemented over 3 successive refueling periods.

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RELOCATE

TABLE 4.7-1
SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a type of snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

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TABLE 4.7-1 (CONT'D)
SNUBBER VISUAL INSPECTION INTERVAL

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

RELOCATE

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3/4.7.13 AUXILIARY RIVER WATER SYSTEM

→ RELOCATE

LIMITING CONDITION FOR OPERATION

3.7.13.1 At least one auxiliary river water subsystem shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With less than one ARWS subsystem OPERABLE, restore at least one subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following thirty hours.

SURVEILLANCE REQUIREMENTS

4.7.13.1 At least one ARWS subsystem shall be demonstrated OPERABLE:

a. At least once per 92 days, by verifying that each pump develops at least 60 psig discharge pressure, while pumping through its test flow line.

b. At least once per 18 months during shutdown by starting an Auxiliary River Water System Pump, shutting down one Reactor Plant River Water System Pump, and verifying that the Auxiliary River Water Subsystem provides at least 8000 gpm cooling water to that portion of the Reactor Plant River Water System under test for at least 2 hours.

3/4.9.5 - 3/4.9.7 (These specification numbers are not used)

3/4.9.5 COMMUNICATIONS

RELOCATE

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

(Next page is 3/4 9-8)

(Proposed Wording)

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3/4.9.6 MANIPULATOR CRANE OPERABILITY

RELOCATE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

1. A minimum capacity of 3250 pounds, and

2. An overload cut off limit ≤ 2850 pounds.

b. The auxiliary hoist used for movement of control rods having:

1. A minimum capacity of 700 pounds, and

2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 150 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2850 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 150 hours prior to the start of such operations by performing a load test of at least 700 pounds.

3/4.9.7 (This Specification number is not used.)

DELETE Page

Containment Leakage Rate Testing Program (Continued)

b. Air Lock testing acceptance criteria and required action are as stated in Specification 3.6.1.3 titled "Containment Air Locks."

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.18 Technical Specifications (TS) Bases Control Program

INSERT Follows

(proposed Wording)

5.5 Programs and Manuals

INSERT TO Page 6-26

6.18
~~5.5.14~~ Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

TSTF-364

requires NRC approval pursuant to

1. a change in the TS incorporated in the license; or

2. a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification ~~5.5.14b~~ 6.18b.1&2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

(continued)

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

fuel cycle. The surveillance requirement for measurement of the MTC at the beginning and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (3) the reactor pressure vessel is above its minimum RT_{NDT} temperature, and (4) the protective instrumentation is within its normal operating range.

3/4.1.2.1 - 3/4.1.2.7 (These specification numbers are not used) → RELOCATE

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

With the RCS average temperature less than 200°F, a low head safety injection pump may be used in lieu of the operable charging pump with a minimum open RCS vent of 2.07 square inches. This will provide latitude for maintenance and ISI examinations on the charging system for repair or corrective action and will ensure that boration and makeup are available when the charging pumps are out-of-service. An open vent ensures that the RCS pressure will not exceed the shutoff head of the low head safety injection pumps.

MOV-1SI-890C is the low head safety injection pump discharge isolation valve to the RCS coldlegs, the valve must be closed prior to reducing RCS pressure below the RWST head pressure to prevent draining into the RCS. Emergency backup power is not required since this valve is outside containment and can be manually operated if required, this will allow the associated diesel generator to be taken out of service for maintenance and testing.

(proposed Wording)

3/4.1.2.8 Refueling Water Storage Tank (RWST)

BASES

3/4.1.2 BORATION SYSTEMS (Continued)

The minimum required volume of water for the Refueling Water Storage Tank (RWST) provides: 1) a source of water and Net Positive Suction Head (NPSH) for High Head Safety Injection and Low Head Safety Injection (LHSI), 2) adequate sump water for LHSI and Recirculation Spray Pump NPSH, and 3) water for containment Quench Spray. Specifically, the limiting case for defining the minimum RWST volume is derived from the containment analysis for subatmospheric peak pressure during a Reactor Coolant Pump suction Large Break Loss of Coolant Accident. The minimum volume corresponds to 439,050 total gallons as contained in the RWST. From this total volume, the analysis value of 430,500 gallons is considered to be delivered to the respective systems.

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analysis. → RELOCATE

The limitations for a maximum of one centrifugal charging pump to be OPERABLE and the surveillance requirement to verify all charging pumps except the required OPERABLE pump to be inoperable less than or equal to the enable temperature set forth in Specification 3.4.9.3 provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV. Substituting a low head safety injection pump for a charging pump in MODES 5 and 6 will not increase the probability of an overpressure event since the shutoff head of the low head safety injection pumps is less than or equal to the setpoint of the overpressure protection system.

Isolation of the primary grade water flow path during MODES 4, 5 and 6 precludes an unplanned boron dilution at these conditions since the sole source of unborated water to the charging pumps is isolated. This eliminates the design basis boron dilution event in MODES 4, 5 and 6. During planned boron dilution events, operator attention will be focused on the boron dilution process and any inappropriate blender operation would be readily identified through various indications which includes the output from the source range nuclear instrumentation.

3/4.1.2.9 Isolation of Unborated Water Sources – Shutdown

(Proposed Wording)

3/4.1.2.9 Isolation of Unborated Water Sources - Shutdown

3/4.1.2 BORATION SYSTEMS (Continued)

Closing either a) 1CH-90 or b) 1CH-91 and 1CH-93 will ensure that all possible flow paths are isolated from the Primary Grade Water System to the operating Reactor Coolant System flow path via the charging pumps, thus preventing any potential inadvertent boron dilution event by injection of unborated water.

The ACTION to suspend all operations involving positive reactivity changes or CORE ALTERATIONS is intended to provide assurance that no other activity will mask any potential unintentional boron dilution event. Maintaining the Primary Grade Water System isolated is necessary to ensure that the design basis boron dilution event is not credible. Thus, immediate corrective action is needed to restore positive isolation as soon as possible when not conducting planned boron dilution or makeup activities. Lack of continuous corrective action to restore the Limiting Condition for Operation (LCO) would then make a potential inadvertent boron dilution credible and require performing additional analysis to verify acceptable consequences if it should occur.

Verifying the SHUTDOWN MARGIN within one hour ensures that no unacceptable reduction of SHUTDOWN MARGIN occurred when the LCO requirements were not satisfied. The SHUTDOWN MARGIN need only be verified once since the cessation of any activities involving positive reactivity changes, CORE ALTERATIONS or use of the Primary Grade Water System with the Charging System will prevent any future potential injection of primary grade water into the Reactor Coolant System. The verification of SHUTDOWN MARGIN needs to be completed anytime that the ACTION is entered even if the LCO is subsequently satisfied before the verification is completed to ensure that no unacceptable reduction of SHUTDOWN MARGIN occurred when the LCO requirements were not satisfied.

The primary function of the surveillance is to ensure that the valve(s) used to isolate the Primary Grade Water System are locked, sealed or otherwise secured. The frequency of 31 days to ensure that the Primary Grade Water System is properly isolated is based on engineering judgment, and has proven to be acceptable. Operating experience has shown that the failure rate is so low that the 31 day frequency is justified. A time frame of 15 minutes provides a minimum reasonable time for an operator to isolate the Primary Grade Water System following a planned activity requiring its use.

RELOCATE

3/4.1.2 BORATION SYSTEMS (Continued)

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% Δk/k after xenon decay and cooldown to 200°F. The maximum boration capability requirements occur at BOL from full power peak xenon conditions and requires 11,336 gallons of 7000 ppm borated water from the boric acid storage tanks or 65,000 gallons of 2000 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boration capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 5000 gallons of 7000 ppm borated water from the boric acid storage tanks or 175,000 gallons of 2000 ppm borated water from the refueling water storage tank.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the movable control assemblies is established by observing rod motion and determining that rods are positioned within ± 12 steps (indicated position), of the respective group demand counter position. The OPERABILITY of the rod position indication system is established by appropriate periodic CHANNEL CHECKS, CHANNEL FUNCTIONAL TESTS and CHANNEL CALIBRATIONS. OPERABILITY of the control rod position indicators is required to determine control rod position and thereby ensure compliance with the control rod alignment and insertion limits. The OPERABLE condition for the analog rod position indicators is defined as being capable of indicating rod position within ± 12 steps of the associated group demand indicator. For power levels below 50 percent, the specifications of this section permit a one hour stabilization period to permit stabilization of known thermal drift in the analog rod position indicator channels. During this stabilization period, greater reliance is placed upon the group demand position indicators to determine rod position. Above 50 percent power, rod motion is not expected to induce thermal transients of sufficient magnitude to exceed the rod position indicator instrument accuracy of ± 12 steps. Limited use of rod position indication primary detector voltages is allowed as a backup method of determining control rod positions. Comparison of the group demand indicator to the calibration curve is sufficient to allow determination that a control rod is indeed misaligned from its bank when primary voltage measurements are used. Comparison of the group demand counters to the bank insertion limits with verification of rod position with the analog rod

INSERT

(proposed Wording)

Bases Insert (for Page B 3/4 1-3)

verification that the analog rod position indicators agree with the demand position indicators within 12 steps over the full range of indicated rod travel. The verification of individual rod position indicators and demand position indicators within the required 12 steps over the full range of indicated rod travel is accomplished by comparisons of the indications at specific rod positions (identified in the applicable surveillance procedure) and calibrations as necessary to ensure the required accuracy is achieved.

3/4.4.1.4 LOOP ISOLATION VALVES (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

Coolant System, and the isolated loop boron concentration is verified. Verification of the isolated loop boron concentration prior to opening the isolated loop isolation valves provides a reassurance of the adequacy of the SHUTDOWN MARGIN. This ensures that any undesirable reactivity effect from the isolated loop does not occur. The safety analyses assume a minimum SHUTDOWN MARGIN as an initial condition for Design Basis Accidents (DBAs). Violation of the LCO, combined with mixing of the isolated loop coolant into the operating loops, could result in the SHUTDOWN MARGIN being less than that assumed in the safety analyses.

LCO

LCO 3.4.1.4.1 ensures that a loop isolation valve that becomes closed in MODES 1 through 4 is fully closed and the plant placed in MODE 5.

LCO 3.4.1.4.2 ensures that power is removed from isolated loop isolation valve operators when closed to perform maintenance in MODES 5 or 6 to prevent an inadvertent loop startup.

RELOCATE

APPLICABILITY

RELOCATE

LCO 3.4.1.4.1 is applicable in MODES 1 through 4, and LCO 3.4.1.4.2 is applicable whenever an RCS loop has been isolated in MODES 5 and 6 with fuel in the reactor vessel. LCO 3.4.1.4.2 is not applicable when there is no fuel in the reactor vessel. In MODES 5 and 6, controlled startup of isolated loops is possible without significant risk of inadvertent criticality.

An RCS loop is considered isolated in MODES 5 and 6 whenever the hot and cold leg isolation valves on one RCS loop are both in a fully closed position at the same time. One isolation valve may be stroked for testing in MODES 5 and 6 and the loop will not be considered isolated when either the hot leg or cold leg loop isolation valve remains open.

(proposed wording)

3/4.4.1.4 LOOP ISOLATION VALVES (Continued)

ACTION

For LCO 3.4.1.4.1

a. Should a loop isolation valve be closed in MODES 1 through 4, the affected loop isolation valve(s) must be maintained closed and the plant placed in MODE 5 to preclude inadvertent startup of the loop and the subsequent potential inadvertent positive reactivity insertion or criticality. The completion time of the ACTIONs allow time for borating the operating loops to a shutdown boration level such that the plant can be brought to MODE 3 within 6 hours and MODE 5 within 36 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.

b. If power is inadvertently restored to one or more loop isolation valve operators, the potential exists for accidental isolation of a loop with a subsequent inadvertent startup of the isolated loop. The loop isolation valves have motor operators. Therefore, these valves will maintain their last position when power is removed from the valve operator. With power applied to the valve operators, only administrative controls prevent the valve from being operated. Although operating procedures make the occurrence of this event unlikely, the prudent action is to remove power from the loop isolation valve operators. The completion time of 30 minutes to remove power from the loop isolation valve operators is sufficient considering the complexity of the task.

For LCO 3.4.1.4.2

If power is inadvertently restored to one or more loop isolation valve operators, the potential exists for accidental isolation of a loop with a subsequent inadvertent startup of the isolated loop. The loop isolation valves have motor operators. Therefore, these valves will maintain their last position when power is removed from the valve operator. With power applied to the valve operators, only administrative controls prevent the valve from being operated. Although operating procedures make the occurrence of this event unlikely, the prudent action is to remove power from the loop isolation valve operators. The completion time of 1 hour to remove power from the loop isolation valve operators is sufficient considering the complexity of the task.

→ RELOCATE

(Proposed Wording)

3/4.4.1.4 LOOP ISOLATION VALVES (Continued)

SURVEILLANCE REQUIREMENTS (SR)

SR 4.4.1.4.1

SR 4.4.1.4.1 is performed at least once per 31 days to ensure that the RCS loop isolation valves are open, with power removed from the loop isolation valve operators. The primary function of this surveillance is to ensure that power is removed from the valve operators, since SR 4.4.1.1 ensures that the loop isolation valves are open by verifying every 12 hours that all loops are operating and circulating reactor coolant. The frequency of 31 days ensures that the required flow can be made available, is based on engineering judgment, and has proven to be acceptable. Operating experience has shown that the failure rate is so low that the 31 day frequency is justified.

SR 4.4.1.4.2

SR 4.4.1.4.2 is performed at least once per 7 days to ensure that the RCS loop isolation valves have power removed from the loop isolation valve operators. The frequency of 7 days which ensures that the power is removed from loop isolation valve operators, is based on engineering judgment, and has proven to be acceptable. Operating experience has shown that the failure rate is so low that the 7 day frequency is justified.

RELOCATE

3/4.4.1.5 ISOLATED LOOP STARTUP

BACKGROUND

The RCS may be operated with loops isolated in order to perform maintenance. While operating with a loop isolated, there is a potential for inadvertently opening the isolation valves in the isolated loop. In this event, the coolant in the isolated loop would suddenly begin to mix with the coolant in the operating loops. This situation has the potential for causing a positive reactivity addition with a corresponding reduction of SHUTDOWN MARGIN if the boron concentration in the isolated loop is less than the required SHUTDOWN MARGIN.

As discussed in the UFSAR, the startup of an isolated loop is performed in a controlled manner that virtually eliminates any inappropriate sudden positive reactivity addition from unborated water because:

a. LCO 3.4.1.5, "Isolated Loop Startup," and plant operating procedures require that the boron concentration in the isolated loop be maintained higher than the SHUTDOWN MARGIN

(proposed wording)

3/4.4.2 (This specification number is not used)

BASES

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs. per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. → RELOCATE

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

During shutdown conditions (any RCS cold leg temperature below the enable temperature specified in 3.4.9.3) RCS overpressure protection is provided by the Overpressure Protection Systems addressed in Specification 3.4.9.3.

3/4.4.2 and 3/4.4.3 SAFETY VALVES (Continued)

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

The requirement that (150)kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator in a non-isolated reactor coolant loop provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350°F, decay heat is removed by the RHR system.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant

(proposed Wording)

3/4.4.7 (This Specification number is not used)

3/4.4.7 CHEMISTRY

RELOCATE

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The primary coolant specific activity is limited in order to maintain offsite and control room operator doses associated with postulated accidents within applicable requirements. Specifically, the 0.35 µCi/gm DOSE EQUIVALENT I-131 limit ensures that the offsite dose does not exceed a small fraction of 10 CFR Part 100 guidelines and that control room operator thyroid dose does not exceed GDC-19 in the event of primary-to-secondary leakage induced by a main steam line break.

(proposed Wording)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 4.5-3 to assure compliance with the requirements of Appendix H to 10 CFR 50.

PZR Temp Limits

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

RELOCATE

Pressure-temperature limit curves shown in Figure B 3/4 4-3 were developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop.

OVERPRESSURE PROTECTION SYSTEMS

BACKGROUND

The overpressure protection system (OPPS) controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. The maximum setpoint for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup meet the 10 CFR 50, Appendix G requirements during the OPPS MODES.

(proposed wording)

3/4.4.10 (This Specification number is not used)

BASES

3/4.4.10 STRUCTURAL INTEGRITY → RELOCATE

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.11 RELIEF VALVES

BACKGROUND

The Pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are air operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permit performance of certain surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from emergency power sources in the event of a loss of offsite power. Two PORVs and their associated block valves are powered from two separate safety trains.

(proposed Wording)

BASES (Continued)

3/4.4.11 RELIEF VALVES (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

SR 4.4.11.2

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the frequency of 92 days is the ASME Code, Section XI. If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valves are closed to isolate otherwise inoperable PORVs, the maximum completion time to restore one PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the required actions fulfills the SR).

This SR is not required to be met with the block valve closed, in accordance with required ACTION b or c of this LCO.

3/4.4.12 REACTOR COOLANT SYSTEM VENTS → RELOCATE

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space, ensures the capability exists to perform this function.

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

(proposed Wording)

3/4.7.2 (This specification number is not used)

RELOCATE

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on steam generator average impact values taken at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 RIVER WATER SYSTEM

The OPERABILITY of the river water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects or accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30 day cooling water supply to safety related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants."

RELOCATE

3/4.7.6 FLOOD PROTECTION

The limitation on flood level ensures that facility operation will be terminated in the event of flood conditions. The limit of elevation 695 Mean Sea Level was selected on an arbitrary basis as an appropriate flood level at which to terminate further operation and initiate flood protection measures for safety related equipment.

3/4.7.6 (This specification number is not used)

(proposed wording)

3/4.7.7 CONTROL ROOM EMERGENCY HABITABILITY SYSTEM

The OPERABILITY of the control room emergency habitability system ensures that the control room will remain habitable for operations personnel during and following all credible accident conditions. The ambient air temperature is controlled to prevent exceeding the allowable equipment qualification temperature for the equipment and instrumentation in the control room. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS)

The OPERABILITY of the SLCRS provides for the filtering of postulated radioactive effluents resulting from a Fuel Handling Accident (FHA) and from leakage of LOSS OF COOLANT ACCIDENT (LOCA) activity from systems outside of the Reactor Containment building, such as Engineered Safeguards Features (ESF) equipment, prior to their release to the environment. This system also collects potential leakage of LOCA activity from the Reactor Containment building penetrations into the contiguous areas ventilated by the SLCRS except for the Main Steam Valve Room and Emergency Air Lock. The operation of this system was assumed in calculating the postulated offsite doses in the analysis for a FHA. System operation was also assumed in that portion of the Design Basis Accident (DBA) LOCA analysis which addressed ESF leakage following the LOCA, however, no credit for SLCRS operation was taken in the DBA LOCA analysis for collection and filtration of Reactor Containment building leakage even though an unquantifiable amount of contiguous area penetration leakage would in fact be collected and filtered. Based on the results of the analyses, the SLCRS must be OPERABLE to ensure that ESF leakage following the postulated DBA LOCA and leakage resulting from a FHA will not exceed 10 CFR 100 limits.

RELOCATE

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on sealed source contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 and 3/4.7.11 RESIDUAL HEAT REMOVAL SYSTEM (RHR) Deleted

(proposed Wording)

Delete Page

3/4.7.12 SNUBBERS

RELOCATE

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other similar event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or other similar event initiating dynamic loads. Therefore, the required inspection interval varies based upon the number of unacceptable snubbers found during the previous inspection, the total population or category size for each type of snubber, and the previous inspection interval. This criteria follows the guidance provided in NRC Generic Letter 90-09. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, or verified operable by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any-safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

Delete page

RELOCATE

SNUBBERS (Continued)

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at refueling or 18 month intervals not to exceed two (2) years. Observed failures of these sample snubbers shall require functional testing of additional units.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

DELETE PAGE

BASES

3/4.7.13 AUXILIARY RIVER WATER SYSTEM (ARWS) → RELOCATE

The operability of the ARWS ensures that sufficient cooling capacity is available to bring the reactor to a cold shutdown condition in the event that a barge explosion at the station's intake structure or any other extremely remote event would render all of the normal River Water System supply pumps inoperable. The scenario of a postulated gasoline barge impact with the intake structure and coincident explosion disabling the Reactor Plant River Water System (RPRWS) is a low probability event. Nonetheless, the ARWS provides defense in-depth in assuring shutdown cooling capability. The requirement to operate the ARWS is not coincident with a postulated Design Basis Accident, but only for the postulated gasoline barge impact event.

Although the ARWS is a manually operated non-safety system which is not required to meet single active failure criteria, the system is designed with redundant pumps and valves on a header to accommodate a single active failure on start-up. This design criteria provides a defense in-depth in order to ensure the system can adequately mitigate the consequences of the postulated event. An ARWS pump can be manually started on the emergency bus during loss of offsite power after the diesel loading sequence is complete. If there is a delay in starting the ARWS, the auxiliary feedwater system is available to remove reactor core decay heat for a short term period.

The requirements for subsystem OPERABILITY are similar to those of the RPRWS except that one subsystem is required to be OPERABLE in the MODES noted. The Limiting Condition for Operation reflects the low risk of the postulated event compared to more stringent requirements associated with safety related systems. The ACTION statement takes into account the low probability of both trains of RPRWS being disabled as a result of the postulated site scenario coincident with one of the ARWS subsystems being OPERABLE.

Revised by NRC letter dated January 24, 2000

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (Continued)

system HEPA filters and charcoal adsorbers and the resulting iodine removal capacity are consistent with the assumptions of the accident analysis.

All containment penetrations, except for the containment purge and exhaust penetrations, that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Penetration closure may be achieved by an isolation valve, blind flange, manual valve, or functional equivalent. Functional equivalent isolation ensures releases from the containment are prevented for credible accident scenarios. The isolation techniques must be approved by an engineering evaluation and may include use of a material that can provide a temporary, pressure tight seal capable of maintaining the integrity of the penetration to restrict the release of radioactive material from a fuel element rupture.

3/4.9.5 COMMUNICATIONS → RELOCATE

The requirements for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies; 2) each crane has sufficient load capacity to lift a control rod or fuel assembly; and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 (Th~~is~~se Specification number~~s~~ ~~is~~ are not used.)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE,

3/4.9.5 -

ATTACHMENT A-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 159
TECHNICAL SPECIFICATION RELOCATION AND BASES CONTROL

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BASES

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Refueling Water Storage Tank

(Proposed Wording)

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LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

(proposed Wording)

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LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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DESIGN FEATURES

ADMINISTRATIVE CONTROLS

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ADMINISTRATIVE CONTROLS

(proposed Wording)

(3/4.1.2.1 – 3/4.1.2.7 (These Specification numbers are not used)

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

RELOCATE

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid storage system via a boric acid transfer pump to a charging pump to the Reactor Coolant System if only the boric acid storage tank is OPERABLE as given in Specification 3.1.2.7.a for MODES 5 and 6 or as given in Specification 3.1.2.8.a for MODE 4; or

b. The flow path from the refueling water storage tank via a charging pump or a low head safety injection pump (with an open RCS vent of greater than or equal to 3.14 square inches) to the Reactor Coolant System if the refueling water storage tank is OPERABLE as given in Specification 3.1.2.7.b for MODES 5 and 6 or as given in Specification 3.1.2.8.b for MODE 4.

APPLICABILITY: MODES 4, 5 and 6

ACTION

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

2. Verifying that the temperature of the heat traced portion of the flow path is $\geq$ 65°F when a flow path from the boric acid tanks is used and the ambient air temperature of the Auxiliary Building is < 65°F.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

(Next Page is 3/4 1-15)

(proposed Wording)

Delete page

RELOCATE

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and one charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via one charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

b. With the flow path from the refueling water storage tank inoperable, restore the flow path to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

2. Verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is $\geq$ 65°F when the ambient air temperature of the Auxiliary Building is < 65°F.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

Delete page

 → RELOCATE

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown by cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

Delete page

CHARGING PUMP-SHUTDOWN → RELOCATE

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump[1] in the boron injection flow path required by Specification 3.1.2.1 or low head safety injection pump (with an open Reactor Coolant System vent of greater than or equal to 3.14 square inches) shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With none of the above pumps OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one charging pump or low head safety injection pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE pursuant to Specification 4.5.2.b.1.

4.1.2.3.2 When the low head safety injection pump is used in lieu of a charging pump, the low head safety injection pump shall be demonstrated OPERABLE by:

a. Verification of an OPERABLE RWST pursuant to 4.1.2.7 and 4.1.2.8,

b. Verification of an OPERABLE low head safety injection pump pursuant to Specification 4.5.2.b.2,

c. Verification of an OPERABLE low head safety injection flow path from the RWST to the Reactor Coolant System once per shift, and

d. Verification that the vent is open in accordance with 4.4.9.3.3.

(1) With two charging pumps OPERABLE, follow Specification 3.4.9.3.

Delete page

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS-OPERATING

RELOCATE

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3[1].

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1 percent $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 Each charging pump shall be demonstrated OPERABLE pursuant to Specification 4.5.2.b.1.

(1) The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 3.4.9.3 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever comes first.

RELOCATE

Delete page

REACTIVITY CONTROL SYSTEMS

BORIC ACID TRANSFER PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 One boric acid transfer pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path thru the boric acid transfer pump of Specification 3.1.2.1.a, is OPERABLE.

APPLICABILITY: MODES 4, 5 and 6.

ACTION:

With no boric acid transfer pump OPERABLE as required to complete the flow path of Specification 3.1.2.1.a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid transfer pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required boric acid transfer pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a differential pressure of $\geq$ 102 psid when tested pursuant to Specification 4.0.5.

→ RELOCATE Delete page

REACTIVITY CONTROL SYSTEMS

BORIC ACID TRANSFER PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least one boric acid transfer pump in the boron injection flow path required by Specification 3.1.2.2.a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2.a is OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With no boric acid transfer pump OPERABLE, restore at least one boric acid transfer pump to OPERABLE STATUS within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F; restore at least one boric acid transfer pump to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required boric acid pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a differential pressure of $\geq$ 102 psid when tested pursuant to Specification 4.0.5.

Delete page

RELOCATE

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A boric acid storage system with:

1. A minimum contained volume of 2315 gallons,

2. Between 7000 and 7700 ppm of boron, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained volume of 217,000 gallons,

2. A minimum boron concentration of 2000 ppm, and

3. A minimum solution temperature of 45°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the water level of the tank, and

3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water and the outside ambient air temperature is < 45°F

Refueling Water Storage Tank (RWST)

BORATED WATER SOURCES - OPERATING → RELOCATE

The RWST shall be OPERABLE

LIMITING CONDITION FOR OPERATION

3.1.2.8 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2.

a. A boric acid storage system with:

1. A minimum usable volume of 13,390 gallons,
2. Between 7000 and 7700 ppm of boron, and
3. A minimum solution temperature of 65°F.

→ RELOCATE

b. The refueling water storage tank with:

1. A minimum usable volume of 859,248 gallons,
2. A boron concentration between 2000 and 2100 ppm, and
3. A solution temperature of ≥ 45°F and ≤ 50°F.

→ MOVE TO Surveillances

APPLICABILITY: MODES 1, 2, 3 & 4.

ACTION:

→ RELOCATE

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F within the next 6 hours; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

→ RELOCATE

The RWST shall be verified OPERABLE:

(proposed Wording)

SURVEILLANCE REQUIREMENTS (Continued)

RELOCATE

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the water level in each water source, and

3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST solution temperature when the RWST ambient air temperature is > 50°F or < 45°F.

IS ≥ 45°F and ≤ 50°F

At Least once per 7 days by:

1. Verifying the boron concentration is between 3000 and 2100 ppm, and

2. Verifying a minimum usable volume of 859,248 gallons

(Proposed Wording)

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod position indicator per bank inoperable either:

   1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 (.1) Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours.

INSERT 4.1.3.2.2 From 3/4.1.3.3

3/4.1.3.3 (This Specification number is not used)

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM-SHUTDOWN → RELOCATE

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3*#, 4*# and 5*#

ACTION:

With less than the above required digital rod position indicator(s) OPERABLE, open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel (indicated) at least once per 18 months.

4.1.3.2.2

→ MOVE TO 3/4.1.3.2

*With the reactor trip system breakers in the closed position.
#See Special Test Exceptions Specification 3.10.5.

→ RELOCATE

(Proposed Wording)

Delete page

RELOCATE

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full-length shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

*This requirement is not applicable during the initial calibration of the Digital Rod Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

Duplicate Table in ODCM/LRM for relocated monitors

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT[3]	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
Deleted → RELOCATE a. Fuel Storage Pool Area (2RMF-RQ202)	1	(1)	≤75.8 mR/hr	10^{-1} to 10^{4} mR/hr	19
b. Containment Area (2RMR-RQ206 & 207)	2	1,2,3 & 4	≤2.0×10^{4} R/hr	1 to 10^{7} R/hr	35
c. Control Room Area (2RMC-RQ201 & 202)	2	1,2,3 & 4, 5[4] & 6[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. Fuel Building Vent					
i. Gaseous Activity (Xe-133) (2RMF-RQ301B)	1	(2)	≤7.82×10^{-6} μCi/cc	10^{-6} to 10^{-1} μCi/cc	21

(proposed wording)

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	SETPOINT(3)	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS (Continued)					
ii. Particulate (I-131) (2RMF-RQ301A)	1	(2)	$\leq 6.70 \times 10^{-9}$ µCi/cc	10^{-10} to 10^{-5} µCi/cc	21
c. Noble Gas and Effluent Monitors					
Deleted i. Supplementary Leak Collection and Release System					
1) Mid Range Noble Gas (Xe-133) (2HVS-RQ109C)	1	1,2,3&4	N.A.	10^{-4} to 10^{2} µCi/cc	35
2) High Range Noble Gas (Xe-133) (2HVS-RQ109D)	1	1,2,3&4	N.A.	10^{-1} to 10^{5} µCi/cc	35
ii. Containment Purge Exhaust (Xe-133) (2HVR-RQ104A & B)	1	6	$\leq 1.01 \times 10^{-3}$ 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

RELOCATE

(Proposed Wording)

(NOT USED)

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

(1) With fuel in the storage pool or building. → RELOCATE

(2) With irradiated fuel in the storage pool.

(3) Above background. → Duplicate in ODCM/LRM

(4) During movement of irradiated fuel.

ACTION STATEMENTS → RELOCATE

ACTION 19 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

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 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the applicable ACTION requirements of Specifications 3.9.12 and 3.9.13.

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:

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.

→ Duplicate in ODCM for relocated monitors

(proposed Wording)

Duplicate Table in ODCM/LRM
For relocated monitors

Deleted

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. Fuel Storage Pool Area (2RMF-RQ202)	S	R	M	*
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, 5## & 6##
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. Fuel Building Vent				
i. Gaseous Activity (2RMF-RQ301B)	S	R	M	**
ii. Particulate Activity (2RMF-RQ301A)	S	R	M	**

*With fuel in the storage pool or building → RELOCATE
**With irradiated fuel in the storage pool
#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

(Proposed Wording)

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. Supplementary Leak Collection and Release System (2HVS-RQ109C & D)	S	R	M	1, 2, 3 & 4
ii. Containment Purge Exhaust (2HVR-RQ104A & B)	S	R	M	6
iii. Main Steam Discharge (2MSS-RQ101A, B & C)	S	R	M	1, 2, 3 & 4

RELOCATE

Deleted

(proposed wording)

Delete page

LOOP ISOLATION VALVES - SHUTDOWN → RELOCATE

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 The loop isolation valves in an isolated RCS loop shall have power removed from the associated loop isolation valve operators(1).

APPLICABILITY:

Whenever an RCS loop has been isolated, MODES 5 and 6(2).

ACTION:

With the requirements of the above specification not satisfied, remove power from the isolated loop isolation valve operators(1) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 Verify at least once per 7 days that power is removed from the RCS isolated loop stop valve operators(1).

---

(1) Power may be restored to the associated RCS isolated loop isolation valve operators provided the requirements of Surveillance Requirement 4.4.1.5.2 have been satisfied.

(2) With fuel in the vessel.

3/4.4.2 (This Specification number is not used)

3/4.4.2 SAFETY VALVES - SHUTDOWN

RELOCATE

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting* of 2485 psig + 1% - 3%.**

APPLICABILITY: MODES 4 and 5

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

b. After any pressurizer code safety lift, as indicated by the safety valve position indicator, involving loop seal or water discharge; borate to a SHUTDOWN MARGIN equivalent to at least 1% delta K/K at 200 deg F within the next 24 hours. Inspect the valve for potential damage, initiate corrective action to return the valve to OPERABLE status prior to increasing RCS temperature and document the inspection results in the Annual Report pursuant to Specification 6.9.1.5.b.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional requirements other than those required by Specification 4.0.5.

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

**Within ± 1% following pressurizer code safety valve testing.

(proposed Wording)

3/4.4.3 SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting* of 2485 psig + 1% - 3%.**

APPLICABILITY: MODES 1, 2, and 3, With all RCS cold leg temperatures > the enable temperature specified in 3.4.9.3.

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

b. After any pressurizer code safety valve lift, as indicated by the safety valve position indicator, involving loop seal or water discharge; be in at least HOT STANDBY within the next 6 hours, and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional requirements other than those required by Specification 4.0.5.

with any RCS cold leg temperature ≤ the enable temperature specified in 3.4.9.3 and apply RCS overpressure protection requirements in accordance with Specification 3.4.9.3.

*The lift setting shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.
**Within ± 1% following pressurizer code safety valve testing.

(Proposed Wording)

3/4.4.7 (This Specification number is not used)

3/4.4.7 CHEMISTRY

RELOCATE

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY.. At all times.

ACTION.. MODES 1, 2, 3, and 4

a. With any one or more chemistry parameters in excess of its Steady State Limit but within its Transient Limit, restore the Parameter to within its Steady State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At all other times

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to ≤ 500 psig, if applicable, and perform an analysis to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-10.

(next page is 3/4 4-27)

(proposed Wording)

Delete page

RELOCATE

TABLE 3.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	< 0.10 ppm*	≤ 1.00 ppm*
CHLORIDE	< 0.15 ppm	≤ 1.50 ppm
FLUORIDE	≤ 0.15 ppm	≤ 1.50 ppm

*Limit not applicable with $T_{avg} \leq 250°F$.

Delete Page

TABLE 4.4-10

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

CONTAMINANT	MINIMUM SAMPLING FREQUENCIES	MAXIMUM TIME BETWEEN SAMPLES
DISSOLVED OXYGEN	3 times per 7 days*	72 hours
CHLORIDE	3 times per 7 days	72 hours
FLUORIDE	3 times per 7 days	72 hours

*Not required with $T_{avg} \leq 250°F$.

RELOCATE

3/4.4.9.2 (This Specification number is not used)

PRESSURIZER

RELOCATE

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period,

c. A maximum normal spray water temperature differential of 320°F, and

d. A maximum auxiliary spray water temperature differential of 380°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits:

a. Restore the temperature to within the limits within 30 minutes, and

b. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer within 72 hours, and

c. Determine, from Action b above, that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.9.2.2 The normal spray water temperature differential shall be determined to be within the limit at least once per 30 minutes during system heatup or cooldown.

4.4.9.2.3 The auxiliary spray water temperature differential shall be determined to be within the limit at least once per 30 minutes during auxiliary spray operation.

(proposed wording)

3/4.4.10 (This Specification number is not used)

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2, AND 3 COMPONENTS → RELOCATE

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.10 Each ASME Code Class 1, 2, and 3 component shall be demonstrated OPERABLE in accordance with Specification 4.0.5.

(proposed Wording)

Delete page

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM HEAD VENTS → RELOCATE

LIMITING CONDITION FOR OPERATION

3.4.12 All Reactor Coolant System head vent valves, powered from emergency buses shall be OPERABLE* and closed** for each of the reactor vessel head vent paths.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With at least one vent path from the above location OPERABLE and one or more power operated vent valves inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable valve(s) is maintained closed with power removed. Power operation may continue until the next scheduled outage, at which time all Reactor Coolant System head vent valves shall be OPERABLE prior to entry into MODE 1. The provisions of Specification 3.0.4 are not applicable.

b. With all vent paths from the above location inoperable maintain the inoperable valves closed with power removed or close the manual isolation valves, and restore at least one vent path from the above locations to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.12 Each Reactor Coolant System head vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying the manual isolation valve in the vent path is locked or sealed in the open position.

2. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room.

3. Verifying flow through the Reactor Coolant System Head vent path to the Pressurizer Relief Tank.

---

*For purposes of this Specification, an inoperable vent valve is defined as: a valve which exhibits leakage in excess of Specification 3.4.6.2 limits, or cannot be opened and closed on demand, or does not have its normal emergency power supply OPERABLE.

**These valves may be operated for required venting operations and leak testing in MODES 3 and 4.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two separate and independent containment quench spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one containment quench spray subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment quench spray subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days, by:

1. By verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

2. ~~Verifying the temperature of the borated water in the refueling water storage tank is within the limits of Specification 3.1.2.8.b.3.~~

b. By verifying, at the frequency specified in the Inservice Testing Program, that each quench spray pump's developed head at the flow test point is greater than or equal to the required developed head as specified in the Inservice Testing Program and the Containment Integrity Safety Analysis.

c. At least once per 18 months during shutdown, by:

1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

(proposed Wording)

3/4.7.2 (This Specification number is not used)

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

RELOCATE

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperatures of both the primary and secondary coolants in the steam generators shall be > 70°F when the pressure of either coolant in the steam generator is > 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side ≤ 200 psig within 30 minutes, and

b. Perform an analysis to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The pressure in each side of the steam generator shall be determined to be < 200 psig at least once per hour when the temperature of either the primary or secondary coolant in the steam generator is < 70°F.

(proposed Wording)

3/4.7.6 (This Specification number is not used)

3/4.7.6 FLOOD PROTECTION

RELOCATE

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the Ohio River exceeds 695 Mean Sea Level at the intake structure.

APPLICABILITY: At all times.

ACTION:

With the water level at the intake structure above elevation 695 Mean Sea Level:

a. Be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours, and

b. Initiate and complete within 8 hours, the following flood protection measures:

   1. Install and seal the flood doors in the intake structure.

SURVEILLANCE REQUIREMENTS

4.7.6.1 The water level at the intake structure shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is below elevation 690 Mean Sea Level, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation 690 Mean Sea Level.

(proposed Wording)

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3/4.7.9 SEALED SOURCE CONTAMINATION → RELOCATE

LIMITING CONDITION FOR OPERATION

3.7.9.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma-emitting material or 5 microcuries of alpha-emitting material shall be free of ≥ 0.005 microcuries of removable contamination.

APPLICABILITY: AT ALL TIMES.

ACTION:

a. With a sealed source having removable contamination in excess of the above limit, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.9.1.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive materials.

   1. With a half-life greater than 30 days (excluding Hydrogen 3) and

   2. In any form other than gas.

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 → RELOCATE

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.9.1.3 Reports - A Special Report shall be prepared and submitted in accordance with 10 CFR 50.4 on an annual basis if sealed source or fission detector leakage tests reveal the presence of ≥ 0.005 microcuries of removable contamination.

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3/4.7.10 (This specification number is not used)

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3/4.7.11 (This specification number is not used)

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3/4.7.12 SNUBBERS → RELOCATE

LIMITING CONDITION FOR OPERATION

3.7.12 All snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems* required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.12.d on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.12 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.7-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.7-1 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before amendment 49.

* These systems are defined as those portions or subsystems required to prevent releases in excess of 10 CFR 100 limits.

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SURVEILLANCE REQUIREMENTS (Continued)

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of the visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; or (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.12.e or 4.7.12.f, as applicable. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

Snubbers which have been determined to be inoperable as a result of unexpected transients, isolated damage, or other random events, and cannot be proven operable by functional testing for the same reasons, shall not be counted in determining the next visual inspection period when the provision in 4.7.12.d (that failures are subject to an engineering evaluation of component structural integrity) has been met and equipment has been restored to an operable state via repair and/or replacement as necessary.

d. Functional Tests

At least once per 18 months** during shutdown, a representative sample (of at least 10%) of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For Functional Testing type of snubber shall mean a group or combination of groups by load size and kind (i.e., hydraulic or mechanical) or any other combination of load size and kind. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.12.e or 4.7.12.f, an additional 10% shall be functionally tested.

** A one-time extension of the snubber functional test frequency for operating cycle 8 is permitted. This extension is applicable until the first re-entry into MODE 6 following defueled condition during refueling outage 2R08 or November 30, 2000, whichever is earlier.

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SURVEILLANCE REQUIREMENTS (Continued)

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle.

2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.).

3. Snubbers within 10 feet of the discharge from a safety relief valve.

Snubbers that are especially difficult to remove or in high radiation zones during shutdown shall also be included in the representative sample.*

If a spare snubber has been installed in place of a failed snubber, the spare snubber shall be retested. Test results of this snubber may not be included for the re-sampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

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RELOCATE

SURVEILLANCE REQUIREMENTS (Continued)

e. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

g. Service Life Monitoring

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and may be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with the applicable record retention provision of the quality assurance program description included in the Updated Final Safety Analysis Report. Service life will be defined to commence at plant startup subsequent to initial fuel load.

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TABLE 4.7-1
SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a type of snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

DELETE Page

TABLE 4.7-1 (CONT'D)
SNUBBER VISUAL INSPECTION INTERVAL

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

RELOCATE

DELETE Page

PLANT SYSTEMS

3/4.7.13 STANDBY SERVICE WATER SYSTEM (SWE)

→ RELOCATE

LIMITING CONDITION FOR OPERATION

3.7.13.1 At least one standby service water subsystem shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With less than one SWE subsystem OPERABLE, restore at least one subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following thirty hours.

SURVEILLANCE REQUIREMENTS

4.7.13.1 At least one SWE subsystem shall be demonstrated OPERABLE:

a. At least once per 92 days, by verifying that each pump develops at least 109 psid differential pressure, while pumping through its test flow line.

b. At least once per 18 months during shutdown by starting a Standby Service Water System Pump, shutting down one Service Water System Pump, and verifying that the Standby Service Water Subsystem provides at least 8584 gpm cooling water to that portion of the Service Water System under test for at least 2 hours.

3/4.9.5-3/4.9.7 (These Specification numbers are not used)

REFUELING OPERATIONS

COMMUNICATIONS → RELOCATE

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

(Next Page is 3/4 9-8)

(proposed Wording)

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REFUELING OPERATIONS

MANIPULATOR CRANE OPERABILITY → RELOCATE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

1. A minimum capacity of 3250 pounds, and

2. An overload cut off limit ≤ 2700 pounds.

b. The auxiliary hoist used for movement of control rods having:

1. A minimum capacity of 700 pounds, and

2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 150 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2700 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure vessel shall be demonstrated OPERABLE within 150 hours prior to the start of such operations by performing a load test of at least 700 pounds.

3/4.9.7 (This Specification number is not used.)

DELETE Page

CONTAINMENT LEAKAGE RATE TESTING PROGRAM (Continued)

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 44.7 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$ for the overall Type A leakage test and $< 0.60\ L_a$ for the Type B and Type C tests on a minimum pathway leakage rate (MNPLR) basis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ on a maximum pathway leakage rate (MXPLR)[2] basis for Type B and Type C tests and $< 0.75\ L_a$ for Type A tests.

b. Air lock testing acceptance criteria and required action are as stated in Specification 3.6.1.3 titled "Containment Air Locks."

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

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(2) For penetrations which are isolated by use of a closed valve(s), blind flange(s), or de-activated automatic valve(s), the MXPLR of the isolated penetration is assumed to be the measured leakage through the isolation device(s).

6.18 Technical Specifications (TS) Bases Control Program

INSERT Follows

(Proposed Wording)

INSERT To Page 6-26

6.18
~~5.5.14~~ Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

TSTF-364 b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

requires NRC approval pursuant to

1. a change in the TS incorporated in the license; or

2. a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification ~~5.5.14b~~ 6.18.b.1&2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 541°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the pressurizer is capable of being in an OPERABLE status with a steam bubble, 3) the reactor pressure vessel is above its minimum RT_{NDT} temperature and 4) the protective instrumentation is within its normal operating range.

3/4.1.2.1 - 3/4.1.2.7 (These Specification numbers are not used)

3/4.1.2 BORATION SYSTEMS

RELOCATE

The boron injection system ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are provided to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

With the RCS average temperature less than 200°F, Low Head Safety Injection pump may be used in lieu of the operable charging pump with a minimum open RCS vent of 3.14 square inches. This will provide latitude for maintenance and ISI examinations on the charging system for repair or corrective action and will ensure that boration and makeup are available when the charging pumps are out-of-service. An open vent insures that RCS pressure will not exceed the shutoff head of the Low Head Safety Injection pumps.

2SIS-MOV8888A and B are the Low Head Safety Injection Pump discharge isolation valves to the RCS cold legs, the valves must be closed prior to reducing RCS pressure below the RWST head pressure to prevent draining into the RCS. Emergency backup power is not required since these valves are outside containment and can be manually operated if required, this will allow the associated diesel generator to be taken out of service for maintenance and testing.

The technical specification limit on the refueling water storage tank has been established at 859,248 gallons to account for reactivity considerations and the NPSH requirements of the ECCS system and the water required for containment spray operation.

3/4.1.2.8 Refueling Water Storage Tank (RWST)

(Proposed Wording)

3/4.1.2.8 Refueling Water Storage Tank (RWST) (continued)

3/4.1.2 BORATION SYSTEMS (Continued)

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of either a LOCA or a steamline break. The limits on RWST minimum volume and boron concentration ensure that: 1) sufficient water is available within containment to permit recirculation cooling flow to the core, 2) the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1), 3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area > 3.0 ft^2) assuming complete mixing of the RWST, RCS, ECCS, chemical addition tank, containment spray system piping, and other water volumes that may eventually reside in the sump Post-LOCA with all control rods assumed to be out (ARO), 4) long term subcriticality following a steamline break assuming ARI-1 and to preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

The limitations for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV. Substituting a Low Head Safety Injection pump for a charging pump in MODES 5 and 6 will not increase the probability of an overpressure event since the shutoff head of the Low Head Safety Injection pumps is below the setpoint of the overpressure protection system.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.77% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirements occur at BOL from full power peak xenon conditions and requires 13,390 gallons of 7000 ppm borated water from the boric acid storage tanks or 100,000 gallons of 2000 ppm borated water from the refueling water storage tank.

With the RCS temperature below 350°F, one boron injection flow path is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

RELOCATE

(Proposed Wording)

3/4.1.2.9 Isolation of Unborated Water Sources - Shutdown

3/4.1.2 BORATION SYSTEMS (Continued)

RELOCATE

The boration capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 2315 gallons of 7000 ppm borated water from the boric acid storage tanks or 10,196 gallons of 2000 ppm borated water from the refueling water storage tank.

Isolation of the primary grade water flow path during MODES 4, 5 and 6 precludes an unplanned boron dilution at these conditions since the sole source of unborated water to the charging pumps is isolated. This eliminates the design basis boron dilution event in MODES 4, 5 and 6. During planned boron dilution events, operator attention will be focused on the boron dilution process and any inappropriate blender operation would be readily identified through various indications which includes the output from the source range nuclear instrumentation.

Closing either a) 2CHS-37 and 2CHS-828 or b) 2CHS-91, 2CHS-96, and 2CHS-138 will ensure that all possible flow paths are isolated from the Primary Grade Water System to the operating Reactor Coolant System flow path via the charging pumps, thus preventing any potential inadvertent boron dilution event by injection of unborated water.

The ACTION to suspend all operations involving positive reactivity changes or CORE ALTERATIONS is intended to provide assurance that no other activity will mask any potential unintentional boron dilution event. Maintaining the Primary Grade Water System isolated is necessary to ensure that the design basis boron dilution event is not credible. Thus, immediate corrective action is needed to restore positive isolation as soon as possible when not conducting planned boron dilution or makeup activities. Lack of continuous corrective action to restore the Limiting Condition for Operation (LCO) would then make a potential inadvertent boron dilution credible and require performing additional analysis to verify acceptable consequences if it should occur.

Verifying the SHUTDOWN MARGIN within one hour ensures that no unacceptable reduction of SHUTDOWN MARGIN occurred when the LCO requirements were not satisfied. The SHUTDOWN MARGIN need only be verified once since the cessation of any activities involving positive reactivity changes, CORE ALTERATIONS or use of the Primary Grade Water System with the Charging System will prevent any future potential injection of primary grade water into the Reactor Coolant System. The verification of SHUTDOWN MARGIN needs to be completed anytime that the ACTION is entered even if the LCO is subsequently

(proposed wording)

3/4.1.2.9 Isolation of Unborated Water Sources - Shutdown (continued)

3/4.1.2 BORATION SYSTEMS (Continued)

satisfied before the verification is completed to ensure that no unacceptable reduction of SHUTDOWN MARGIN occurred when the LCO requirements were not satisfied.

The primary function of the surveillance is to ensure that the valve(s) used to isolate the Primary Grade Water System are locked, sealed or otherwise secured. The frequency of 31 days to ensure that the Primary Grade Water System is properly isolated is based on engineering judgment, and has proven to be acceptable. Operating experience has shown that the failure rate is so low that the 31 day frequency is justified. A time frame of 15 minutes provides a minimum reasonable time for an operator to isolate the Primary Grade Water System following a planned activity requiring its use.

INSERT

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that 1) acceptable power distribution limits are maintained, 2) the minimum SHUTDOWN MARGIN is maintained, and 3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the movable control assemblies is established by observing rod motion and determining that rods are positioned within ± 12 steps (indicated position), of the respective group demand counter position. The OPERABILITY of the control rod position indication system is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 541°F and with all reactor coolant

(proposed Wording)

Attachment A-2
Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 159

Bases Insert (for Page B 3/4 1-5)

The verification of individual rod position indicators and demand position indicators within the required 12 steps over the full range of indicated rod travel is accomplished by comparisons of the indications at specific rod positions (identified in the applicable surveillance procedure) to ensure the required accuracy is achieved. As the individual rod position indicators for the shutdown banks do not indicate over the full range of rod travel, only points within the indicated ranges are required for comparison.

3/4.4.1.4 LOOP ISOLATION VALVES (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

not occur. The safety analyses assume a minimum SHUTDOWN MARGIN as an initial condition for Design Basis Accidents (DBAs). Violation of the LCO, combined with mixing of the isolated loop coolant into the operating loops, could result in the SHUTDOWN MARGIN being less than that assumed in the safety analyses.

LCO

LCO 3.4.1.4.1 ensures that a loop isolation valve that becomes closed in MODES 1 through 4 is fully closed and the plant placed in MODE 5.

LCO 3.4.1.4.2 ensures that power is removed from isolated loop isolation valve operators when closed to perform maintenance in MODES 5 or 6 to prevent an inadvertent loop startup.

RELOCATE

APPLICABILITY

RELOCATE

LCO 3.4.1.4.1 is applicable in MODES 1 through 4. and LCO 3.4.1.4.2 is applicable whenever an RCS loop has been isolated in MODES 5 and 6 with fuel in the reactor vessel. LCO 3.4.1.4.2 is not applicable when there is no fuel in the reactor vessel. In MODES 5 and 6, controlled startup of isolated loops is possible without significant risk of inadvertent criticality.

An RCS loop is considered isolated in MODES 5 and 6 whenever the hot and cold leg isolation valves on one RCS loop are both in a fully closed position at the same time. One isolation valve may be stroked for testing in MODES 5 and 6 and the loop will not be considered isolated when either the hot leg or cold leg loop isolation valve remains open.

ACTION

For LCO 3.4.1.4.1

a. Should a loop isolation valve be closed in MODES 1 through 4, the affected loop isolation valve(s) must be maintained closed and the plant placed in MODE 5 to preclude inadvertent startup of the loop and the subsequent potential inadvertent positive reactivity insertion or criticality. The completion time of the ACTIONs allow time

(proposed Wording)

3/4.4.1.4 LOOP ISOLATION VALVES (Continued)

ACTION (Continued)

for borating the operating loops to a shutdown boration level such that the plant can be brought to MODE 3 within 6 hours and MODE 5 within 36 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.

b. If power is inadvertently restored to one or more loop isolation valve operators, the potential exists for accidental isolation of a loop with a subsequent inadvertent startup of the isolated loop. The loop isolation valves have motor operators. Therefore, these valves will maintain their last position when power is removed from the valve operator. With power applied to the valve operators, only administrative controls prevent the valve from being operated. Although operating procedures make the occurrence of this event unlikely, the prudent action is to remove power from the loop isolation valve operators. The completion time of 30 minutes to remove power from the loop isolation valve operators is sufficient considering the complexity of the task.

For LCO 3.4.1.4.2 → RELOCATE

If power is inadvertently restored to one or more loop isolation valve operators, the potential exists for accidental isolation of a loop with a subsequent inadvertent startup of the isolated loop. The loop isolation valves have motor operators. Therefore, these valves will maintain their last position when power is removed from the valve operator. With power applied to the valve operators, only administrative controls prevent the valve from being operated. Although operating procedures make the occurrence of this event unlikely, the prudent action is to remove power from the loop isolation valve operators. The completion time of 1 hour to remove power from the loop isolation valve operators is sufficient considering the complexity of the task.

SURVEILLANCE REQUIREMENTS (SR)

SR 4.4.1.4.1

SR 4.4.1.4.1 is performed at least once per 31 days to ensure that the RCS loop isolation valves are open, with power removed

(proposed Wording)

3/4.4.1.4 LOOP ISOLATION VALVES (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

from the loop isolation valve operators. The primary function of this surveillance is to ensure that power is removed from the valve operators, since SR 4.4.1.1 ensures that the loop isolation valves are open by verifying every 12 hours that all loops are operating and circulating reactor coolant. The frequency of 31 days ensures that the required flow can be made available, is based on engineering judgment, and has proven to be acceptable. Operating experience has shown that the failure rate is so low that the 31 day frequency is justified.

SR 4.4.1.4.2 → RELOCATE

SR 4.4.1.4.2 is performed at least once per 7 days to ensure that the RCS loop isolation valves have power removed from the loop isolation valve operators. The frequency of 7 days which ensures that the power is removed from loop isolation valve operators, is based on engineering judgment, and has proven to be acceptable. Operating experience has shown that the failure rate is so low that the 7 day frequency is justified.

3/4.4.1.5 ISOLATED LOOP STARTUP

BACKGROUND

The RCS may be operated with loops isolated in order to perform maintenance. While operating with a loop isolated, there is a potential for inadvertently opening the isolation valves in the isolated loop. In this event, the coolant in the isolated loop would suddenly begin to mix with the coolant in the operating loops. This situation has the potential for causing a positive reactivity addition with a corresponding reduction of SHUTDOWN MARGIN if the boron concentration in the isolated loop is less than the required SHUTDOWN MARGIN.

As discussed in the UFSAR, the startup of an isolated loop is performed in a controlled manner that virtually eliminates any inappropriate sudden positive reactivity addition from unborated water because:

a. LCO 3.4.1.5, "Isolated Loop Startup," and plant operating procedures require that the boron concentration in the isolated loop be maintained higher than the SHUTDOWN MARGIN requirement for the operating loops, thus eliminating the potential for introducing coolant from the isolated loop that could dilute the boron concentration in the operating loops below the required SHUTDOWN MARGIN; and

(proposed Wording)

3/4.4.2 (This Specification number is not used)

During shutdown conditions (any RCS cold leg temperature below the enable temperature specified in 3.4.9.3) RCS overpressure protection is provided by the Overpressure Protection Systems addressed in Specification 3.4.9.3.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs. per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

Safety valves similar to the pressurizer code safety valves were tested under an Electric Power Research Institute (EPRI) program to determine if the valves would operate stably under feedwater line break accident conditions. The test results indicated the need for inspection and maintenance of the safety valves to determine the potential damage that may have occurred after a safety valve has lifted and either discharged the loop seal or discharged water through the valve. Additional action statements require safety valve inspection to determine the extent of the corrective actions required to ensure the valves will be capable of performing their intended function in the future.

3/4.4.4 PRESSURIZER

The requirement that 150 kw of pressurizer heaters and their associated controls and emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator in a non-isolated reactor coolant loop provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate

(proposed Wording)

3/4.4.4.7 (This Specification number is not used)

3/4.4.7 CHEMISTRY → RELOCATE

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The primary coolant specific activity is limited in order to maintain offsite and control room operator doses associated with postulated accidents within applicable requirements. Specifically, the 0.35 µCi/gm DOSE EQUIVALENT I-131 limit ensures that the offsite dose does not exceed a small fraction of 10 CFR Part 100 guidelines and that control room operator thyroid dose does not exceed GDC-19 in the event of primary-to-secondary leakage induced by a main steam line break.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 0.35 µCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.35 µCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limits shown on Figure 3.4-1 must be restricted to ensure that assumptions made in the UFSAR accident analyses are not exceeded.

Reducing T_{avg} to < 500°F minimizes the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam

(proposed wording)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 5.3-6 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

RELOCATE

PRZ Temp Limits

The limitations imposed on the pressurizer heatup and cooldown rates and auxiliary spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

Pressure-temperature limit curves shown in figure B 3/4 4-3 were developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop.

OVERPRESSURE PROTECTION SYSTEMS

BACKGROUND

The overpressure protection system (OPPS) controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. The maximum setpoint for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup meet the 10 CFR 50, Appendix G requirements during the OPPS MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures. RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

(proposed Wording)

BASES (Continued)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

The SR is required to be performed prior to entering the condition for the OPPS to be OPERABLE. This assures low temperature overpressure protection is available when the RCS cold leg temperature is less than or equal to the enable temperature. Performing the surveillance every 31 days on each required PORV permits verification and adjustment, if necessary, of its lift setpoint, and considers instrumentation reliability which has been shown through operating experience to be acceptable. The CHANNEL FUNCTIONAL TEST will verify the setpoint is within the allowed maximum limits. PORV actuation could depressurize the RCS and is not required.

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

SR 4.4.9.3.3

The RCS vent of greater than or equal to 3.14 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for an open vent or valve that cannot be locked, except

b. Once every 31 days for a valve that is locked, or provided with remote position indication, or sealed, or secured in position. A removed pressurizer safety valve fits this category.

The passive vent arrangement must only be open to be OPERABLE. This surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO.

3/4.4.10 STRUCTURAL INTEGRITY → RELOCATE

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g)(6)(i).

3/4.4.10 (This Specification number is not used)

(proposed Wording)

BASES (Continued)

3/4.4.11 REACTOR COOLANT SYSTEM RELIEF VALVES (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

SR 4.4.11.2

Block valve cycling verifies that the valve(s) can be closed if needed. The basis for the frequency of 92 days is the ASME Code, Section XI. If the block valve is closed to isolate a PORV that is capable of being manually cycled, the OPERABILITY of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valves are closed to isolate inoperable PORVs, the maximum completion time to restore one PORV and open the block valve is 72 hours, which is well within the allowable limits (25%) to extend the block valve frequency of 92 days. Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to OPERABLE status (i.e., completion of the required actions fulfills the SR).

This SR is not required to be met with the block valve closed, in accordance with required ACTIONs b or c of this LCO.

RELOCATE

3/4.4.12 REACTOR COOLANT SYSTEM HEAD VENTS

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head or the pressurizer steam space via the PORV's ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System Head vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

RELOCATE

3/4.7.2 (This Specification number is not used)

BASES

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

RELOCATE

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator average impact values taken at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the primary component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects or accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30 day cooling water supply to safety related equipment without

(Proposed Wording)

3/4.7.6 (This Specification number is not used)

BASES

3/4.7.5 ULTIMATE HEAT SINK (Continued)

exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants."

3/4.7.6 FLOOD PROTECTION

RELOCATE

The limitation on flood level ensures that facility operation will be terminated in the event of flood conditions. The limit of elevation 695 Mean Sea Level was selected on an arbitrary basis as an appropriate flood level at which to terminate further operation and initiate flood protection measures for safety related equipment.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP AND PRESSURIZATION SYSTEM

The OPERABILITY of the control room emergency air cleanup and pressurization system ensures that the control room will remain habitable with respect to potential radiation hazards for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

The control room air cleanup system includes two pressurization systems. The filtration pressurization system draws outside air through filters. The bottled air pressurization system pressurizes by discharge of air from bottles without filtration and with closure of intake and exhaust dampers. Although the bottles are shared with Unit 1, the discharge can be initiated by Unit 2 control systems in response to radiation levels. Closure of the intake and exhaust dampers can be initiated by Unit 2 control systems. However, closure of dampers in one intake and in one exhaust is dependent upon availability of Unit 1 power sources.

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS)

The OPERABILITY of the SLCRS provides for the filtering of postulated radioactive effluents resulting from a Fuel Handling Accident (FHA) and from leakage of loss of coolant accident (LOCA) activity from systems outside of the Reactor Containment building, such as Engineered Safeguards Features (ESF) equipment, prior to their release to the environment. This system also collects potential leakage of LOCA activity from the Reactor Containment building

3/4.7.8 SUPPLEMENTAL LEAK COLLECTION AND RELEASE SYSTEM (SLCRS) (Continued)

penetrations into the contiguous areas ventilated by the SLCRS except for the Emergency Air Lock. The operation of this system was assumed in calculating the postulated offsite doses in the analysis for a FHA. System operation was also assumed in that portion of the Design Basis Accident (DBA) LOCA analysis which addressed ESF leakage following the LOCA, however, no credit for SLCRS operation was taken in the DBA LOCA analysis for collection and filtration of Reactor Containment building leakage even though an unquantifiable amount of contiguous area penetration leakage would in fact be collected and filtered. Based on the results of the analyses, the SLCRS must be OPERABLE to ensure that ESF leakage following the postulated DBA LOCA and leakage resulting from a FHA will not exceed 10 CFR 100 limits.

3/4.7.9 SEALED SOURCE CONTAMINATION → RELOCATE

The limitations on sealed source contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.10 and 3/4.7.11 RESIDUAL HEAT REMOVAL SYSTEM (RHR)

Deleted

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BASES

3/4.7.12 SNUBBERS → RELOCATE

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other similar event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies based upon the number of unacceptable snubbers found during the previous inspection, the total population or category size for each type of snubber, and the previous inspection interval. This criteria follows the guidance provided in NRC Generic Letter 90-09. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, or verified OPERABLE by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at refueling or 18 month intervals not to exceed two (2) years. Observed failures of these sample snubbers shall require functional testing of additional units.

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BASES

3/4.7.12 SNUBBERS (Continued) → RELOCATE

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.13 STANDBY SERVICE WATER SYSTEM (SWE)

The OPERABILITY of the SWE ensures that sufficient cooling capacity is available to bring the reactor to a cold shutdown condition in the event that a barge explosion at the station's intake structure or any other extremely remote event would render all of the normal Service Water System (SWS) supply pumps inoperable. The scenario of a postulated gasoline barge impact with the intake structure and coincident explosion disabling the SWS is a low probability event. Nonetheless, the SWE provides defense in-depth in assuring shutdown cooling capability. The requirement to operate the SWE is not coincident with a postulated Design Basis Accident, but only for the postulated gasoline barge impact event.

Although the SWE is a non-safety system which is not required to meet single active failure criteria, the system is designed with redundant pumps and valves on a header to accommodate a single active failure on start-up. This design criteria provides a defense in-depth in order to ensure the system can adequately mitigate the consequences of the postulated event. An SWE pump can be manually started on the emergency bus during loss of offsite power after the diesel loading sequence is complete. With no loss of power signal present, the SWE is automatically started upon receipt of low service water header pressure signal. This feature is provided to prevent inadvertent plant trip on loss of running service water pump and is not required

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BASES

3/4.7.13 STANDBY SERVICE WATER SYSTEM (SWE) (Continued)

for the design basis event. If there is a delay in starting the SWE, the auxiliary feedwater system is available to remove reactor core decay heat for a short term period.

The requirements for subsystem OPERABILITY are similar to those of the SWS except that one subsystem is required to be OPERABLE in the MODES noted. The Limiting Condition for Operation reflects the low risk of the postulated event compared to more stringent requirements associated with safety related systems. The ACTION statement takes into account the low probability of both trains of SWS being disabled as a result of the postulated scenario coincident with one of the SWE subsystems being OPERABLE.

RELOCATE

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS (Continued)

requirements are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

All containment penetrations, except for the containment purge and exhaust penetrations, that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Penetration closure may be achieved by an isolation valve, blind flange, manual valve, or functional equivalent. Functional equivalent isolation ensures releases from the containment are prevented for credible accident scenarios. The isolation techniques must be approved by an engineering evaluation and may include use of a material that can provide a temporary, pressure tight seal capable of maintaining the integrity of the penetration to restrict the release of radioactive material from a fuel element rupture.

3/4.9.5 COMMUNICATIONS → RELOCATE

The requirements for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies; 2) each crane has sufficient load capacity to lift a control rod or fuel assembly; and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 (This Specification is not used.)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor

3/4.9.5-

(proposed wording)

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 and the RCS T_{avg} may fall slightly below the minimum temperature of Specification 3.1.1.5.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception is required to perform certain startup tests.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN → RELOCATE

This special test exception permits the Position Indication System to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time are derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the Position Indication Systems remain OPERABLE.

ATTACHMENT B

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request Nos. 287 and 159
TECHNICAL SPECIFICATION RELOCATION AND BASES CONTROL

A. DESCRIPTION OF AMENDMENT REQUEST

This license amendment request (LAR) proposes to revise the Beaver Valley Power Station (BVPS) Unit 1 and 2 technical specifications (TS) to implement improvements endorsed in the NRC's Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, 58 FR 39132, July 22, 1993 (the policy statement). The policy statement describes the benefits to be derived from the Improved Standard TS (ISTS) and encouraged licensees to use the ISTS as the basis for plant-specific TS amendments and for complete conversions to the ISTS. The policy statement also clarifies that "licensees may adopt portions of the improved STS without fully implementing all STS improvements."

This LAR proposes changes to the BVPS Unit 1 and 2 TS that implement two enhancements from the policy statement and ISTS. The major change proposed in this LAR involves the application of the TS screening criteria from the policy statement to evaluate the content of the BVPS TS and identify those TS that do not meet the criteria. The TS that do not meet the policy statement criteria are proposed for relocation from the BVPS license. The application of the TS criteria is discussed under the "TS Relocation" heading in the LAR. This LAR also proposes the addition of a TS Bases control program consistent with the ISTS. The addition of the ISTS Bases control program is discussed under the "Bases Control Program" heading in this LAR.

The changes proposed in this LAR represent the first phase of the BVPS conversion to the ISTS for Westinghouse Plants contained in NUREG-1431. The second phase of the BVPS conversion will include the reformat, reorganization, and expanded scope (where applicable) of the remaining BVPS TS as well as the development of the new bases consistent with the ISTS. The preparation of the LAR for the second phase of the BVPS conversion is planned to coincide with the issuance of Revision 2 to NUREG-1431.

TS RELOCATION

The key element of the TS improvements endorsed by the NRC policy statement is the criteria for evaluating the content of the TS. The criteria were utilized by the NRC and industry working groups to develop the content of the ISTS for each industry owners group. The resulting ISTS applicable to Westinghouse plants is contained in NUREG-1431.

The policy statement defines four selection criteria for determining which of the TS requirements should be retained and which could be relocated from the license. The four TS selection criteria contained in the policy statement were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36, on July 19, 1995 (60 CFR 36953). In describing the advantages of the ISTS and the application of the criteria to screen existing TS, the policy statement recognizes that the resulting clarification of the scope and purpose of the TS will help to enhance safe plant operation by focusing licensee and plant operator attention on those plant conditions most important to safety. In addition, the policy statement describes the improvement to the TS as resulting in more efficient use of NRC and industry resources.

Prior to the policy statement criteria for TS, no standard existed that could be used to evaluate or control the content of a plant's TS. Over the years, various requirements have been incorporated into TS that do not meet the current criteria for inclusion in the TS. The policy statement recognizes that the increased volume of TS has diverted both the NRC and licensee attention from the more important requirements in the TS to the extent that it has resulted in an adverse impact on safety. The policy statement encourages licensees to implement a program to upgrade TS by screening existing TS requirements using the four criteria to refocus the TS content to be more consistent with the Atomic Energy Act, 10 CFR 50.36, and previous interpretations of the regulations governing TS. As explained in the policy statement:

> By applying the four criteria contained in the Policy Statement a licensee should capture all of those specific characteristics of its facility and conditions for its operation that are required to meet the principal operative standard in Section 182a. of the Atomic Energy Act that is, that adequate protection is provided to the health and safety of the public.

The policy statement also recognized that the result of applying the criteria, and allowing certain items to be removed from the TS and requiring others to be retained, is consistent with the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in Portland General Electric Company's hearing (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the appeal board observed:

> [T]here is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

Consistent with the policy statement recommendations, this LAR identifies those BVPS Unit 1 and 2 TS requirements that do not meet the 10 CFR 50.36 criteria for retention in the TS and that are not part of the ISTS. The BVPS TS requirements (and associated Bases) that do not meet the criteria and have not been included in the ISTS are proposed for relocation from the BVPS TS.

In the following sections of this LAR, the BVPS TS that were evaluated against the criteria and identified for relocation are discussed in more detail. Unless otherwise stated, the discussions of each TS proposed for relocation apply to both Unit 1 and Unit 2. Tables 1 and 2 contain a summary listing of the BVPS TS requirements identified for relocation, along with other pertinent information associated with the relocation of these TS.

The policy statement also explains that TS requirements that do not meet any of the four criteria may be proposed for removal from the TS and relocated to licensee-controlled documents. In NRC Administrative Letter 96-04, "Efficient Adoption of Improved Standard Technical Specifications," the NRC recommended acceptable methods for relocating TS requirements that do not meet the policy statement criteria. The Administrative Letter recommended that relocated TS requirements be moved to "...licensee-controlled documents for which there is an applicable regulatory process

for future changes." One of the acceptable documents for relocated TS identified by the NRC in Administrative Letter 96-04 is a manual that is referenced in the FSAR. The Administrative Letter acknowledged that several licensees have used a manual to house relocated TS requirements.

BVPS proposes to relocate TS and associated Bases that do not meet any of the four criteria to one of two manuals, the Licensing Requirements Manual (LRM) or Offsite Dose Calculation Manual (ODCM). Both of these manuals are referenced in the BVPS Unit 1 and 2 Updated Final Safety Analysis Report (UFSAR) and the control of changes to the ODCM is also specified in the TS (6.14). In addition, the NRC has previously approved both these manuals for the relocation of TS requirements. In Generic Letter 89-01, the ODCM was specifically recommended for relocating effluent related TS and, in the BVPS specific implementation of Generic Letter 95-10, the NRC approved the LRM for the relocation of the TS requirements addressed in that Generic Letter (BVPS license amendment numbers 233 and 115 SER dated 9/7/00).

The requirements initially relocated to the LRM or ODCM will be technically the same as the current TS. However, changes in format and presentation (i.e., titles, numbering of requirements and sections, pagination, etc.) of the relocated material will be made as necessary to fit the LRM or ODCM. Future changes to the relocated material will be in accordance with 10 CFR 50.59.

In the course of screening the existing BVPS TS requirements against the policy statement criteria, several references were utilized. These references are discussed (when applicable) in the following sections of this LAR where each TS identified for relocation is evaluated. The following references were utilized:

1. WCAP 11618 "Methodically Engineered, Restructured And Improved Technical Specifications, MERITS Program – Phase II Task 5 CRITERIA APPLICATION," November 1987.

   This Westinghouse Electric Corporation report was submitted to the NRC by the Westinghouse Owners Group (WOG) in November 1987. The report contains the results of the work done by Westinghouse Electric Corporation to apply the TS criteria from the interim policy statement published in February 1987 to the previous Standard Westinghouse TS (NUREG-0452). The TS contained in NUREG-0452 are comparable to those contained in the BVPS TS. Therefore, this reference is, in most cases, directly applicable to the BVPS TS.

2. NRC Letter From T. E. Murley to W. S. Wilgus, dated May 9, 1988.

   After evaluating the application of the interim policy statement TS criteria submitted by each industry Owners Group (including reference 1 above), the NRC made their own determinations regarding which TS requirements must be retained in the ISTS and which TS requirements could be relocated. The NRC conclusions regarding the application of the TS criteria were documented in the letter identified above to the joint Owners Group Chairman at the time. The NRC also used NUREG-0452 for the reference Westinghouse TS. Similar to reference 1 above, the TS contained in NUREG-0452 are comparable to the current BVPS TS and therefore, the NRC conclusions documented in this letter are also, in most cases, directly applicable to the BVPS TS.

3. NUREG-1431 "Standard Technical Specifications Westinghouse Plants" Revision 1, April 1995.

   References 1 and 2 above comprised the initial input used in determining the content of NUREG-1431. However, NUREG-1431 is the result of the cumulative effort of the industry and NRC over several years, and it represents the final decisions made in implementing the policy statement. Therefore, the content of NUREG-1431 was the primary input utilized in this LAR to implement the policy statement criteria.

4. NUREG-0452 "Standard Technical Specifications for Westinghouse Pressurized Water Reactors" Revision 5 Draft, 1984.

   The last revision of the previous standard TS for Westinghouse plants. The interim policy statement TS criteria were applied to the TS in this NUREG to develop the ISTS (NUREG-1431). Many of the current BVPS TS are based on the TS contained within this NUREG. Therefore, the results of previous applications of the TS criteria to this NUREG are applicable to BVPS.

In order to maintain the focus of this LAR on the relocation of TS requirements, changes to retained TS were minimized. Due to the differences in format, organization, and usage rules between the current BVPS TS and the ISTS, more extensive changes to retained TS would introduce complications in the review process and require additional efforts to implement that would be best addressed in

a full ISTS conversion. In most cases, the TS selected for relocation may be removed without affecting the remaining TS. However, in some cases, this LAR includes changes to retained TS for various reasons to accommodate the relocation. The additional TS changes necessary for relocating certain TS originate from References 1-3 described above or from BVPS specific issues or are simply the necessary administrative changes needed to accommodate the removed material. Changes proposed to retained BVPS TS, utilize the NRC approved changes to NUREG-1431, Revision 1 where applicable. The following is a summary of the additional changes proposed to retained BVPS TS in order to relocate certain other TS:

1. Due to the difference in content from the standard TS in NUREG-0452, only a portion of the corresponding BVPS TS 3.1.2.8, "Borated Water Sources – Operating" has been identified for relocation. A portion of the TS containing requirements for the Refueling Water Storage Tank (RWST) must be retained. The retention of these requirements results in a proposed revision of the title, Limiting Condition for Operation (LCO), and surveillances of BVPS TS 3.1.2.8. The proposed changes to TS 3.1.2.8 result in a BVPS TS that is more consistent with the presentation of the corresponding ISTS. This BVPS specific difference is related to the TS content and the design basis of the systems involved and is described in more detail in the Design Basis section of this LAR. Due to the changes made to the RWST requirements contained in TS 3.1.2.8, additional revisions to TS 3.6.2.1, "Quench Spray System" are necessary. The Quench Spray System TS surveillance requirement to verify the RWST temperature limits specified in 3.1.2.8.b.3 is no longer a correct reference. Consistent with the content of the ISTS, the affected Quench Spray System surveillance was deleted. The elimination of the requirement to verify RWST temperature in the Quench Spray System surveillance is discussed in more detail in the Justification section of this LAR.

2. The relocation of the BVPS TS 3.1.3.3 Rod Position Indication System - Shutdown requires that an 18 month surveillance be retained in the TS for Rod Position Indication – Operating. The surveillance is necessary to confirm operability of the position indication system and is required to be retained in the TS by Reference 2. The details of this change to retained TS 3.1.3.2 are discussed in the Justification section of this LAR.

3. BVPS Radiation Monitors are identified for relocation from TS 3.3.3.1 consistent with the policy statement criteria and the content of the TS in Reference 3. The radiation monitors identified for relocation are listed in Table 2. Not all radiation monitors addressed by TS 3.3.3.1 are being relocated. Therefore, TS 3.3.3.1 remains in the BVPS TS, but is revised to accommodate the removal of the requirements associated with the radiation monitors listed in Table 2. The revisions to TS 3.3.3.1 consist of the removal of the affected monitors and associated Actions and Surveillance Requirements from the TS. Specific revisions to TS 3.3.3.1 are discussed in more detail in the Justification section of this LAR.

4. The relocation of BVPS TS 3.4.2 RCS Safety Valves Shutdown requires that the applicable Modes and Actions of the TS for safety valves operating be revised consistent with the TS in Reference 3. The Mode of applicability for the operating safety valve TS must be expanded from Modes 1-3 to include a reference to the enable temperature for the RCS low temperature overpressure protection system. The Actions must be revised to account for the change to the applicability. The proposed changes match the applicable modes for the safety valves operating TS and the low temperature overpressure protection TS to provide complete pressure protection for the RCS. The proposed change allows the safety valve – shutdown TS to be relocated. The specific changes to the safety valve operating TS are discussed in more detail in the Justification section of this LAR.

5. This LAR also includes administrative changes to the TS and Bases to accommodate the removal of the TS and associated Bases proposed for relocation. The TS and Bases index are revised including TS titles where applicable. Footers are also revised to reflect pagination changes made to the remaining TS and Bases pages to accommodate the removed material. In addition, the TS Bases have been revised to incorporate text necessary to support the changes made to the retained TS.

The following Tables contain a summary of the BVPS TS requirements that were identified for relocation. In addition to listing the BVPS TS identified for relocation, Table 1 shows the relationship of each BVPS TS to the comparable previous standard TS (Reference 4) and the current ISTS (Reference 3). The comparison of the BVPS TS to the previous standard TS is useful as the previous standard TS were used by Westinghouse and the NRC (in References 1 and 2 above) to initially apply the TS criteria during development of the ISTS. In most cases, Table 1 shows a BVPS TS

that directly corresponds to a previous standard TS that has already been reviewed against the criteria by both Westinghouse and the NRC. If Table 1 does not show the TS as retained in NUREG-1431, it was found not to meet the criteria and therefore not included in the new standard TS. Table 1 also shows the proposed destination for each relocated requirement and summarizes additional information pertinent to the TS identified for relocation. Table 2 contains the details regarding TS requirements for individual radiation monitors identified for relocation.

BASES CONTROL PROGRAM

In addition to the relocation of TS that do not meet the criteria, this LAR proposes to add the "Technical Specifications Bases Control Program" identified in Section 5.5.14 of NUREG-1431 to the Unit 1 and 2 BVPS TS. This program provides controls for processing changes to the TS Bases. The program will be incorporated into the Administrative Control section of the BVPS Unit 1 and 2 TS just as it is in the ISTS including any NRC approved changes. No BVPS specific changes, other than those necessary to accommodate the current BVPS Administrative Controls format and numbering, are proposed to the ISTS program. The program contains the following provisions:

1) A requirement to ensure changes to the bases are made under appropriate administrative controls,

2) Criteria to establish when prior approval by the NRC is required for a Bases change,

3) The frequency for sending Bases changes to the NRC, and

4) A requirement to ensure the Bases are maintained consistent with the UFSAR.

The addition of this program to the Unit 1 and 2 BVPS TS represents an administrative enhancement to the current TS that is consistent with the ISTS. The program includes the guidance for establishing appropriate controls for TS bases changes including a provision to assure NRC review and approval is obtained when required.

TABLE 1

SUMMARY OF TECHNICAL SPECIFICATIONS PROPOSED FOR RELOCATION

Unit	BVPS Technical Specification	Contained In Previous Standard Technical Specifications NUREG 0452	Contained In Current Standard Technical Specifications NUREG 1431	Destination Document For Relocated Requirements
1&2	3/4.1.2.1 Boration Systems Flow Paths - Shutdown	Yes	No	LRM(1)
1&2	3/4.1.2.2 Boration Systems Flow Paths - Operating	Yes	No	LRM(1)
1&2	3/4.1.2.3 Boration Systems Charging Pump - Shutdown	Yes	No	LRM(1)
1&2	3/4.1.2.4 Boration Systems Charging Pumps - Operating	Yes	No	LRM(1)
1&2	3/4.1.2.5 Boration Systems Boric Acid Transfer Pumps - Shutdown	Yes(2)	No	LRM(1)
1&2	3/4.1.2.6 Boration Systems Boric Acid Transfer Pumps - Operating	Yes(2)	No	LRM(1)
1&2	3/4.1.2.7 Boration Systems Borated Water Sources Shutdown	Yes	No	LRM(1)
1&2	3/4.1.2.8 Boration Systems Borated Water Sources Operating (Partial)(3)	Yes	No	LRM(1)
1&2	3/4.1.3.3 Reactivity Control Systems Position Indication System – Shutdown	Yes	No(4)	LRM(1)
2	3/4.10.5 Special Test Exceptions Position Indication System Shutdown	Yes	No	LRM(1)
1&2	3/4.3.3.1 Monitoring Instrumentation Radiation Monitoring (Partial)(5)	Yes	No	ODCM(6) LRM(1)
1&2	3/4.4.1.4.2 RCS Loop Isolation Valves Shutdown	Yes(7)	No	LRM(1)
1&2	3/4.4.2 RCS Safety Valves Shutdown	Yes	No(8)	LRM(1)

TABLE 1 (Continued)

SUMMARY OF TECHNICAL SPECIFICATIONS PROPOSED FOR RELOCATION

Unit	BVPS Technical Specification	Contained In Previous Standard Technical Specifications NUREG 0452	Contained In Current Standard Technical Specifications NUREG 1431	Destination Document For Relocated Requirements
1&2	3/4.4.7 RCS Chemistry	Yes	No	LRM(1)
1&2	3/4.4.9.2 RCS Pressurizer Temperature Limits	Yes	No	LRM(1)
1&2	3/4.4.10 RCS Structural Integrity	Yes	No	LRM(1)
1&2	3/4.4.12 RCS Vents	Yes	No	LRM(1)
1&2	3/4.7.2.1 Steam Generator Pressure/ Temperature Limitation	Yes	No	LRM(1)
1&2	3/4.7.6.1 Flood Protection	Yes	No	LRM(1)
1&2	3/4.7.9.1 Sealed Source Contamination	Yes	No	LRM(1)
1&2	3/4.7.12 Snubbers	Yes	No	LRM(1)
1	3/4.7.13.1 Auxiliary River Water System	No	No	LRM(1)
2	3/4.7.13.1 Standby Service Water System	No	No	LRM(1)
1&2	3/4.9.5 Refueling Operations Communications	Yes	No	LRM(1)
1&2	3/4.9.6 Refueling Operations Manipulator Crane Operability	Yes	No	LRM(1)

NOTES

(1) For relocated TS, the LRM or Licensing Requirements Manual is an appendix of the Unit 1 and 2 BVPS UFSAR. Changes to the LRM are controlled via the 10 CFR 50.59 process.

(2) This BVPS TS was based on an early version of NUREG-0452. In the later revisions of NUREG-0452, the TS for the Boric Acid Transfer Pumps was included in the TS for Boration Systems Flow Paths. The transfer pump requirements were generically evaluated for relocation by Westinghouse and the NRC in References 1 and 2 as part of the flow path TS.

TABLE 1 NOTES (Continued)

(3) This BVPS TS contains requirements for the Boric Acid Storage System and the RWST. In NUREG-0452, the RWST requirements applicable during operating Modes were contained in two TS. In addition to Boration Systems, the RWST requirements were contained in a separate ECCS TS. The ECCS RWST requirements were retained in NUREG-1431 and the Boration System TS was approved for relocation. The Beaver Valley TS only include one set of RWST requirements for operating Modes located in the Boration Systems TS. Therefore, only the Boric Acid Storage System portion of this Beaver Valley TS is proposed for relocation. The RWST requirements contained within this TS are not being proposed for relocation and will be retained within the TS consistent with the content of NUREG-1431.

(4) In Reference 2, it was specified that surveillances necessary to ensure system operability be retained within the TS. To satisfy this requirement, the 18 month surveillance from the Position Indication System – Shutdown TS in NUREG-0452 was retained in the NUREG-1431 TS for Position Indication System – Operating. In order to relocate the Position Indication System – Shutdown TS and conform to the ISTS and Reference 2, a surveillance corresponding to the NUREG-0452 surveillance will also be retained in the BVPS Position Indication System – Operating TS for both units.

(5) Consistent with the development of NUREG-1431, each radiation monitor addressed in the TS is evaluated separately for relocation. Those monitors found to meet the criteria of 10CFR 50.36 are retained within technical specification 3.3.3.1. Those monitors found not to meet the criteria are proposed to be relocated to the ODCM. The ODCM was chosen for radiation monitors to be consistent with the location recommended by GL 89-01 for the radiation monitors relocated from the TS by that GL. Table 2 contains a list of radiation monitors selected for relocation.

(6) The ODCM or Offsite Dose Calculation Manual is referenced in the BVPS Unit 1 and 2 UFSAR and in the TS (6.14). The ODCM is appropriate location for effluent related radiation monitors consistent with the recommendations of Generic Letter 89-01. The ODCM currently contains requirements for radiation monitors relocated from the TS by Generic Letter 89-01. Radiation monitors that are not related to effluent requirements will be relocated to the LRM.

(7) This Beaver Valley TS was based on the NUREG-0452 TS 3.4.1.5 "RCS Isolated Loop." The Beaver Valley version of this TS addresses the same design basis concern (boron dilution) as the NUREG-0452 Isolated Loop requirements for Modes 5 and 6. However, no corresponding Mode 5 and 6 TS requirement was retained in NUREG-1431.

(8) Relocation of the RCS Safety Valves - Shutdown TS requires a revision to the RCS Safety Valves – Operating TS consistent with the ISTS. The Mode of applicability of the operating safety valve TS must be expanded to include a reference to the enable temperature specified in the RCS low temperature overpressure protection system TS. This revision is consistent with Reference 3 and ensures the applicable modes for the safety valves operating TS and the low temperature overpressure protection system TS are matched to provide continuous RCS overpressure protection without the need for the Safety Valves – Shutdown TS.

TABLE 2
RADIATION MONITORS FROM TECHNICAL SPECIFICATION 3.3.3.1
PROPOSED FOR RELOCATION

UNIT	TYPE	NUMBER	DESCRIPTION	RELOCATED TO
1	Area	RM-207	Fuel Storage Pool Area	LRM
1	Noble Gas Effluent	RM-VS-110	Supplementary Leak Collection and Release System	ODCM
1	Noble Gas Effluent	RM-VS-109	Auxiliary Building Ventilation System	ODCM
1	Noble Gas Effluent	RM-GW-109	Process Vent System	ODCM
1	Noble Gas Effluent	RM-MS-100 (A,B,C)	Atmospheric Steam Dump Valve and Code Safety Relief Valve Discharge	ODCM
1	Noble Gas Effluent	RM-MS-101	Auxiliary Feedwater Pump Turbine Exhaust	ODCM
2	Area	2RMF-RQ202	Fuel Storage Pool Area	LRM
2	Noble Gas Effluent	2HVS-RQ109C	Supplementary Leak Collection and Release System - Mid Range Noble Gas	ODCM
2	Noble Gas Effluent	2HVS-RQ109D	Supplementary Leak Collection and Release System - High Range Noble Gas	ODCM

B. DESIGN BASES

TS RELOCATION

The design bases of the TS requirements identified for relocation are summarized below. The TS have been grouped together where applicable to facilitate the discussion of a common design basis.

Boration Systems TS

3/4.1.2.1 Flow Paths – Shutdown

3/4.1.2.2 Flow Paths – Operating

3/4.1.2.3 Charging Pump – Shutdown

3/4.1.2.4 Charging Pumps – Operating

3/4.1.2.5 Boric Acid Transfer Pumps – Shutdown

3/4.1.2.6 Boric Acid Transfer Pumps – Operating

3/4.1.2.7 Borated Water Sources – Shutdown

3/4.1.2.8 Borated Water Sources – Operating

These TS have been addressed as a group because each requirement represents an element of the same TS section (Boration Systems) with common functional requirements and design bases.

The boration systems TS listed above address the systems that provide the means to control the chemical neutron absorber (boron) concentration in the RCS to help maintain the Shutdown Margin (SDM) in various Modes of operation. A sufficient SDM ensures that 1) the reactor can be made subcritical from all operating conditions, 2) The reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in shutdown conditions. In order to support adequate SDM, the boration systems TS require sources of borated water, one or more flow paths to inject this borated water into the RCS, boric acid transfer pumps, and charging or low head safety injection pumps to provide the necessary head. The operating TS applicable to this

equipment require redundant flow paths or components operable to ensure the boration capability is maintained in the event of a single failure.

The BVPS boration system TS contain some format and content differences from the standard TS reviewed for relocation in References 1 and 2.

The TS reviewed in References 1 and 2 from NUREG-0452 contained redundant requirements for some components in the boration system section and the Emergency Core Cooling System (ECCS) section of the TS. Specifically the requirements for the charging pumps (both operating and shutdown) and Refueling Water Storage Tank (RWST) (operating only) were contained in two TS sections (Boration Systems and ECCS). The TS operability requirements for the affected equipment were similar in each TS section but were separated due to the different functional requirements described in each TS section (Boration and ECCS). In the ISTS (NUREG-1431), the redundant requirements for this equipment were eliminated. The boration system requirements for the charging pumps and the RWST (both operating and shutdown) were removed from the TS and the corresponding ECCS requirements were retained in the ISTS ECCS section consistent with the application of the policy statement criteria. However, the BVPS TS do not contain redundant requirements for the RWST – Operating in the ECCS section as did NUREG-0452. The requirements for the RWST are only contained in the BVPS boration systems TS 3.1.2.8 "Borated Water Sources – Operating." This BVPS TS contains the requirements for the Boric Acid Storage System and the RWST applicable in Modes 1-4. As this is the only BVPS TS to contain requirements for the RWST in Modes 1-4, TS 3.1.2.8 has an additional design basis (ECCS) that is separate from the boration systems design basis described above. The ECCS design basis of the RWST meets the policy statement criterion 3 for retention in the TS. Therefore, only the boration systems requirements for the Boric Acid Storage System contained in TS 3.1.2.8 have been identified for relocation (see markup of TS 3.1.2.8 in Attachment A of this LAR). The RWST portion of TS 3.1.2.8 is not proposed for relocation and will be retained in the TS due to the ECCS design basis applicable to these requirements. The RWST requirements retained in BVPS TS 3.1.2.8 effectively address the ECCS function of the RWST similar to the ISTS requirements for the RWST. In order to better address the applicable design basis and remaining RWST content of TS 3.1.2.8, the Limiting Condition for Operation (LCO) and TS title are revised similar to the corresponding ISTS LCO for the RWST (see markup of TS 3.1.2.8 in Attachment A of this LAR). A later LAR that will reorganize the BVPS TS

consistent with the ISTS will include reformatting and moving the RWST requirements to the ECCS section of the TS making the remaining BVPS RWST TS requirements even more consistent with the ISTS.

The BVPS TS for the Boric Acid Transfer Pumps Operating and Shutdown (3.1.2.5 and 3.1.2.6) are based on an older version of NUREG-0452 than the corresponding TS reviewed in References 1 and 2. Later versions of NUREG-0452 combined the boric acid transfer pump TS requirements into the boron injection flow path TS. This was a reasonable change to the standard TS as the transfer pump is an inherent part of the associated flow path and the operability of these separate elements of the boration system are closely related. However, the presentation of these TS requirements, combined or separate, does not alter the design basis or functional requirements of the boration system. Therefore, the design basis of the boron injection flow path TS reviewed in References 1 and 2 is the same as the BVPS design basis for the separate flow path and transfer pump TS.

Reactivity Control Systems

3/4.1.3.3 Position Indication System – Shutdown

Special Test Exceptions

3/4.10.5 Position Indication System – Shutdown (Unit 2 only)

The rod position indication system consists of individual rod position indicators and group demand indicators. The individual rod position indication system in Unit 1 is based on an analog design and Unit 2 utilizes a digital system. Both types of individual rod position indication systems are required to meet the same TS functional and design basis requirements (i.e., indication accuracy). In Modes 1 and 2, both the individual and demand position indicators are required operable. In shutdown Modes (3, 4, and 5, with the reactor trip breakers closed) only one of the two types of indication is required operable. The control rod position indicating system provides indication of rod position to the operator. This indication is used by the operator to verify that the rods are correctly positioned. In operating Modes (1 and 2), this indication is used during reactor startup and operation to monitor rod position to verify insertion and alignment limits are met (initial conditions of DBAs) and to verify that the rods are fully inserted into the core immediately following a reactor trip. However, in the shutdown Modes addressed by TS 3.1.3.3

Position Indication System – Shutdown, the position indicating system provides information only and is not relied on by the operators to verify insertion or alignment limits (which are required only in Modes 1 and 2), or reactor trip.

BVPS Special Test Exception 3.10.5 (Unit 2 only) provides an exception to the operability requirements specified in TS 3.1.3.3 for the digital rod position indication system during shutdown conditions. The exception is required during rod drop time measurements. The rod drop measurements require the use of the digital position indication system to obtain the measurement data. This special test exception is consistent with the standard TS contained in Reference 4 and is applicable to Westinghouse plants using a digital rod position indication system. As stated in the LCO of test exception 3.10.5 it is only applicable to Unit 2 TS 3.1.3.3, Position Indication System – Shutdown. Unit 1 utilizes an analog rod position indication system and has no corresponding test exception.

3/4.3.3.1 Monitoring Instrumentation Radiation Monitoring

The specific monitors proposed for relocation from the BVPS TS are listed in Table 2. The selected radiation monitors consist of area and noble gas effluent monitor types. The following descriptions apply to each type of monitor listed on Table 2.

Area radiation monitors provide continuous surveillance of radiation levels in selected areas in the plant. The areas monitored include locations where personnel may be present and where significant radiation levels may occur. The alarms associated with the area monitors provide sufficient warning of high radiation levels and/or abnormal conditions to operating personnel.

Noble gas effluent radiation monitors analyze effluent gases from various ventilation exhaust systems and steam discharge piping. The monitors provide information to trend and control plant effluents to protect the health of plant operating personnel and to limit plant effluent releases to within the limits of 10 CFR 20. The alarms associated with the effluent monitors also warn operating personnel of abnormal releases. Additionally, some of the effluent 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.

3/4.4.1.4.2 RCS Loop Isolation Valves - Shutdown

This BVPS TS provides requirements to remove the power from loop isolation valve operators whenever an RCS loop has been isolated in Modes 5 and 6 (with fuel in the vessel). The requirements of the TS provide assurance that an RCS isolation valve is not inadvertently opened. Due to the possible differences in boron concentration between the isolated loop and the RCS, the opening of an RCS isolation valve has the potential to cause a positive reactivity addition due to boron dilution. Dilution of the RCS boron concentration would result in a reduction in shutdown margin. The TS requirements place administrative controls on the power to the RCS isolation valves to preclude the possibility of inadvertent valve operation that could result in a reduction in the shutdown margin.

This BVPS TS functions with two other TS to accomplish the requirement to prevent a positive reactivity addition that could result in a reduction in shutdown margin. The RCS Loop Isolation Valves – Operating TS contains requirements that are similar to the RCS Loop Isolation Valve – Shutdown TS. Only the Operating TS is applicable in Modes 1-4. In addition, the BVPS TS "Isolated Loop Startup" contains requirements controlling the opening of RCS isolation valves in loops that have been isolated for more than 4 hours. The "Isolated Loop Startup" TS requirements include the verification of boron concentration in the isolated loop prior to opening an RCS isolation valve. These additional isolated loop TS requirements are not identified for relocation and will be retained in the BVPS TS consistent with similar TS in the ISTS.

The corresponding TS to the BVPS "RCS Loop Isolation Valves – Shutdown" TS in Reference 4 (NUREG-0452), "Isolated Loop" addressed the boron concentration of the isolated loop. This previous standard TS required that the boron concentration of the isolated loop be maintained greater than or equal to the RCS boron concentration and required daily surveillance testing to confirm this requirement was met. The basis for this TS was to prevent a positive reactivity addition that could result in a reduction of shutdown margin. The NUREG-0452 TS was applicable in Modes 1-5. The TS was reviewed for inclusion in the ISTS and only the Mode 1-4 requirement was incorporated into Reference 3 (NUREG-1431). The requirements for Modes 5 and 6 were not retained in the ISTS. The resulting Mode 1-4 ISTS requirement was further revised to address power being removed from the RCS valve operators instead of boron concentration.

The original BVPS Unit 1 TS contained the Reference 4 TS requirements for boron concentration. However, the BVPS TS was later revised to eliminate the boron concentration requirements and replace them with the current requirements to remove power from the RCS isolation valves. The revision of the BVPS TS eliminated the need for periodic isolated loop sampling to verify boron concentration thus reducing the dose received by personnel performing the sampling. However, the design basis of the BVPS TS remained unchanged, i.e., to prevent a positive reactivity addition that could result in a reduction in shutdown margin (the same as the NUREG-0452 TS requirements (for Modes 5 and 6) that were not included in NUREG-1431).

3/4.4.2 RCS Safety Valves - Shutdown

This TS requires at least one pressurizer code safety valve to be operable with a lift setting of 2485 psig +1%, -3% in Modes 4 and 5. The pressurizer code safety valves operate to prevent the RCS from being pressurized above the safety limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs. per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition that could occur during shutdown.

Automatic RCS pressure protection is addressed by TS 3.4.3, "Safety Valves – Operating" (in Modes 1-3), TS 3.4.2, "Safety Valves – Shutdown" (Modes 4 and 5) and TS 3.4.9.3, "Overpressure Protection Systems" (in Mode 4 when any RCS cold leg temperature is less than or equal to the enable temperature for the overpressure protection system, Mode 5 and Mode 6 when the reactor vessel head is on). The safety valve shutdown and overpressure protection system TS requirements provide overlapping protection in the shutdown Modes. However, the setpoint associated with the code safety valve (2485 psig) required operable by the safety valve shutdown TS does not provide complete pressure protection for shutdown conditions. The setpoint is too high to protect the 10 CFR 50 Appendix G pressure limits applicable at low RCS temperatures. The overpressure protection system (addressed in TS 3.4.9.3) functions to provide the appropriate pressure protection for low RCS temperature conditions.

3/4.4.7 RCS Chemistry

The RCS Chemistry TS places limits on the oxygen, chloride and fluoride content in the RCS to minimize corrosion. Maintaining the RCS chemistry within the specified limits will ensure that long-term degradation of the RCS pressure

boundary is not exacerbated by poor chemistry. Minimizing the effects of long-term degradation will help reduce the potential for RCS leakage or failure, and ensure the structural integrity of the RCS for the life of the plant.

3/4.4.9.2 RCS Pressurizer Temperature Limits

Although the pressurizer is normally operated in temperature ranges above those for which there is a concern of nonductile failure, the TS limits imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements. Operation of the pressurizer within the specified limits reduces the potential for component or piping failure that could result in a loss of coolant accident.

3/4.4.10 RCS Structural Integrity

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. The ASME programs provide activities to prevent and detect system degradation rather than actions to mitigate unplanned events or transients. The TS requirements provide for continued long-term maintenance of RCS structural integrity. Implementation of the ASME programs is also required by 10 CFR50.55a (g) and TS 4.0.5.

3/4.4.12 RCS Vents

The RCS vents are provided to exhaust noncondensible gases and/or steam from the primary system that may inhibit natural circulation core cooling following any event involving a loss of offsite power and for which long term cooling is required, such as a loss of coolant accident. The functional capabilities and testing requirements for the RCS vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements."

3/4.7.2.1 Steam Generator Pressure Temperature Limitation

The steam generator pressure and temperature limits specified in this TS ensure that the pressure induced stresses are within the maximum allowable fracture toughness stress limits. The values of the specified limits are based on maintaining

a steam generator reference transition nil ductility temperature sufficient to prevent brittle fracture. Brittle fracture of the steam generator could result in a loss of steam generator integrity and cause a loss of coolant accident.

3/4.7.6.1 Flood Protection

The limit on the Ohio River flood level specified in this TS ensures that facility operation will be terminated in the event that flood conditions threaten safety related equipment. The specified TS limit was selected as an appropriate water level at which to terminate plant operation and initiate flood protection measures for safety related equipment. The TS limit provides for equipment protection and requires a precautionary plant shutdown in the event of a high water level.

3/4.7.9.1 Sealed Source Contamination

The TS limits on sealed source contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limits on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. The requirements of this TS provide additional measures to help meet the goal of the lowest achievable dose to plant personnel.

3/4.7.12 Snubbers

The Snubber TS requires that all snubbers be operable. Snubbers are passive devices that are designed to prevent unrestrained pipe motion under dynamic loads and allow normal thermal expansion of piping and nozzles to eliminate excessive thermal stresses during heatup or cooldown. The TS requirements include visual inspections, functional tests, and service life monitoring to verify snubber operability. Inoperable snubbers are required to be repaired or replaced. Snubber inspections are part of the inservice inspection program. Requirements for the inservice inspection program are contained in 10 CFR 50.55a. The TS requirements assure the long-term maintenance of snubbers.

3/4.7.13.1 Auxiliary River Water System (Unit 1)

3/4.7.13.1 Standby Service Water System (Unit 2)

These TS require the operability of the backup cooling water systems, "Auxiliary River Water System" (ARWS) for Unit 1 and the "Standby Service Water System" (SSWS) for Unit 2. Each of these systems consist of two 100% capacity subsystems (pumps and valves) on a common header. The pumps associated with these systems are housed in an intake structure that is separate from the normal safety-related cooling water system intake structure. These backup systems are classified as non-safety systems. The associated TS requires only one of the two backup cooling water subsystems operable.

The normal safety-related Service and River Water Systems consist of two 100% redundant trains. Each unit has three pumps available in the common seismic Category I intake structure. As each train only requires one operable pump there is an additional 100% capacity pump available for each unit. The normal Service and River Water System pumps are housed in flood protected concrete cubicles in the intake structure. In addition, each unit also has redundant safety-related auxiliary feedwater systems that have sufficient short-term capacity to meet the shutdown cooling requirements for the reactor if there was a delay in starting a Service or River Water System pump.

The operability of ARWS and SSWS ensure that sufficient cooling capacity is available to bring the reactor to a cold shutdown condition in the event that an explosion at the station's normal (safety-related) cooling water system intake structure would render all of the normal cooling water supply pumps inoperable. The postulated scenario of a gasoline barge impact with the normal intake structure and coincident explosion powerful enough to completely disable the safety-related Service or River Water Systems is beyond the scope of the design basis accidents and anticipated operational occurrences described in Chapters 6 and 14/15 of the BVPS Unit 1/2 UFSARs.

3/4.9.5 Communications

The TS requires that direct communications be maintained between the control room and personnel at the refueling station. The requirement for communication ensures that personnel at the refueling station can be promptly informed of

significant changes in the facility status or core reactivity conditions during core alterations. The TS requirements are consistent with basic safe work practices.

3/4.9.6 Manipulator Crane Operability

The TS addresses the operability requirements for the manipulator crane and auxiliary hoist. The TS requirements ensure that: 1) the cranes will be used for the movement of control rods and fuel assemblies; 2) each crane has sufficient load capacity to lift a control rod or fuel assembly; and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations. The TS contains requirements designed to prevent damage to control rods, fuel assemblies, core internals, and the reactor vessel.

BASES CONTROL PROGRAM

The addition of the Bases Control Program to the BVPS TS does not involve changes to any plant system nor does it involve plant design considerations. The proposed program addition is administrative in nature and serves to provide guidance for making changes to the TS Bases documentation. The addition of the program also serves to make the BVPS TS more consistent with the TS in Reference 3 and the BVPS TS Bases control methods more consistent with standard industry practice. As such, the addition of this program is not associated with the design basis of any plant system or component.

C. JUSTIFICATION

The NRC issued a Final Policy Statement dated July 22, 1993, to provide guidance for the scope and purpose of TS for nuclear power plants. The policy statement encouraged licensees to implement a voluntary program to update their TS to be consistent with improved vendor-specific ISTS issued by the NRC in September 1992. The current ISTS for Westinghouse plants was published in NUREG 1431, Rev. 1 (Reference 3). As stated in the policy statement, experience in the development of the improved standard TS and in the review of license amendment requests has led the Commission to conclude that safety benefits can be realized from adopting portions of the ISTS without fully implementing all ISTS improvements. This LAR proposes changes to the BVPS TS that are consistent with the guidance of the NRC policy statement and with the content of the ISTS for Westinghouse plants.

TS RELOCATION

The proposed change to relocate selected TS consistent with the policy statement criteria and the content of Reference 3, serves to clarify the scope and purpose of the BVPS TS and will result in TS that are more consistent with the Atomic Energy Act, 10 CFR 50.36, and previous interpretations of the regulations governing TS. The resulting BVPS TS will help to enhance safe plant operation by focusing BVPS and NRC personnel attention on those plant conditions most important to safety. In addition, the revised BVPS TS will result in more efficient use of NRC and BVPS resources.

Consistent with the guidance of NRC Administrative Letter 96-04, "Efficient Adoption of Improved Standard Technical Specifications," BVPS proposes to relocate TS and associated Bases that do not meet any of the four policy statement criteria to one of two manuals, the LRM or ODCM. Both of these manuals are referenced in the BVPS Unit 1 and 2 UFSAR and the control of changes to the ODCM is also specified in the TS (Section 6.14). The destination document for each TS proposed for relocation is identified in Table 1. Relocation of TS requirements to the LRM or 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 for material removed from the TS, including record retention and reporting requirements. The provisions of 10 CFR 50.59 assure future changes to the relocated material will be consistent with safe plant operation.

The requirements relocated to the LRM or ODCM will be technically the same as the current TS. However, administrative changes in format and presentation (i.e., titles, numbering of requirements and sections, pagination, etc.) of the relocated material will be made as necessary to fit the LRM or ODCM. All future changes to the relocated material will be in accordance with 10 CFR 50.59.

The four policy statement criteria contained in 10 CFR 50.36(c)(2)(ii) for determining which regulatory requirements and operating restrictions should be included in the TS 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.

Specific justifications for the revisions proposed in this LAR are discussed below. Unless otherwise specified, each of the following discussions is applicable to the TS of both units. Although a discussion of each TS, or group of TS proposed for relocation follows, Criterion 4 is addressed once for all TS requirements proposed for relocation.

The TS requirements proposed for relocation have been evaluated to identify requirements that are addressed by Probabilistic Risk Assessment (PRA), and if addressed, to determine if the requirement was identified by risk analysis as significant to public health and safety. The documents utilized to evaluate the risk insights relevant to the BVPS TS proposed for relocation include the generic evaluations performed by Westinghouse in Reference 1, the BVPS plant specific risk assessments documented in the Individual Plant Evaluations (IPEs) for Units 1 and 2 and the associated PRA Update Reports for both units.

In Reference 1, Westinghouse established risk-related measures to evaluate the standard Westinghouse Plant TS to determine if a structure, system, or component addressed by a TS requirement met Criterion 4. Consistent with the guidance in the policy statement, Westinghouse evaluated the standard TS proposed for relocation to verify that none of the requirements to be relocated contained constraints of prime importance in limiting the likelihood or severity of accident sequences that are commonly found to dominate risk. The risk measures used in the Westinghouse evaluations were core melt frequency and off-site health effects. The measures used to assess risk in Reference 1 are based on information that can

be found in a plant's IPE and PRA Update Report and, therefore, can be used for plant-specific evaluations. The BVPS TS with corresponding standard TS addressed in Reference 1 were evaluated to ensure the Westinghouse conclusions contained in Reference 1 were applicable to BVPS. The BVPS TS proposed for relocation that did not have a corresponding standard TS assessed by Westinghouse in Reference 1, were evaluated utilizing guidance established in Reference 1. None of BVPS TS requirements proposed for relocation were found to pertain to structures, systems, or components identified as significant to public health and safety consistent with the measures established in Reference 1. Therefore, none of the BVPS TS proposed for relocation meet Criterion 4. The results of the Criterion 4 evaluations performed for those BVPS TS that correspond to standard Westinghouse TS are consistent with the findings of the NRC documented in References 2 and 3.

Approximately 12 plants, with unit(s) of Westinghouse design, have implemented the ISTS to date (NRC Web Page STS Conversion Summary Information). Each of these plants has relocated TS similar to the TS proposed for relocation by BVPS and evaluated by Westinghouse in Reference 1 and the NRC in Reference 2. In addition to the TS relocations approved for ISTS conversions, the NRC has approved numerous TS requirement relocations in accordance with Generic Letters such as 86-12, 89-01, 91-08, 93-08, and 95-10. The large number of TS requirements previously approved for relocation by the NRC has resulted in a significant amount of industry experience relevant to relocated TS. This industry experience includes plant operation with numerous relocated TS that address standard Westinghouse plant structures, systems, and components similar in design and function to the structures, systems, and components addressed by the BVPS TS proposed for relocation. The substantial industry operating experience with similar relocated TS provides additional support for the conclusion that the relocation of the BVPS TS proposed in this LAR is acceptable and has not been shown by operating experience to be significant to the public health and safety. The industry experience also provides additional justification that the relocated requirements can be safely managed by licensees in accordance with the provisions of 10 CFR 50.59.

The following discussions provide more detailed information regarding each of the TS proposed for relocation.

Boration Systems Technical Specifications

3/4.1.2.1 Flow Paths – Shutdown

3/4.1.2.2 Flow Paths – Operating

3/4.1.2.3 Charging Pump – Shutdown

3/4.1.2.4 Charging Pumps – Operating

3/4.1.2.5 Boric Acid Transfer Pumps – Shutdown

3/4.1.2.6 Boric Acid Transfer Pumps – Operating

3/4.1.2.7 Borated Water Sources – Shutdown

3/4.1.2.8 Borated Water Sources – Operating

These TS have been addressed as a group because each TS represents an element of the same TS section (Boration Systems) with common functional requirements and design basis.

As discussed in the design basis section of this LAR, the boration systems TS listed above address the systems that provide the means to control the chemical neutron absorber (boron) concentration in the RCS to help maintain the SDM in various Modes of operation. Adequate SDM is an assumption of the safety analyses. However, the retained TS for Rod Insertion Limits, SDM, and boron concentration provide adequate assurance in all required Modes that SDM is maintained within the applicable design and analyses limits. In addition, the operability of the boration system components that are also part of the emergency core cooling system (ECCS) and that are required to mitigate a DBA or transient as part of the ECCS are being retained in the TS (RWST) or are addressed in the existing ECCS TS (e.g., charging pumps).

In the case of the uncontrolled boron dilution event, the safety analyses rely on administrative control of unborated water source isolation valves to preclude the event, or the event is responded to by an automatic function (reactor trip) or adequate time is available for operator action to mitigate the event (isolate the dilution source). The functional requirements addressed by the boration systems TS listed above are not specifically assumed to be available or credited in any safety analysis to mitigate the consequences of a design basis accident or transient.

Based on the discussions above, the boration systems TS proposed for relocation were found not to meet any policy statement criteria. The BVPS specific application of the policy statement selection criteria to these TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

As previously discussed in the Design Basis section of this LAR, TS 3/4.1.2.8, "Borated Water Sources – Operating" contains requirements for the RWST that must remain in the TS. Therefore, TS 3/4.1.2.8 has been edited to reflect the retention of the RWST requirements and relocation of the requirements associated with the Boric Acid Storage System. In addition, the presentation of the retained RWST requirements in TS 3/4.1.2.8 is revised to be more consistent with the corresponding TS in the ISTS. The TS title, LCO statement, and surveillances are revised to reflect the proposed content of the TS (RWST only) and to be more similar to the ISTS presentation of the RWST requirements. For example, the values for boron concentration, RWST volume and RWST temperature have been moved from the LCO to the surveillances consistent with the presentation of these values in the ISTS RWST TS. The proposed revisions to the retained RWST requirements are not intended to introduce any technical changes. These changes affect only the presentation of the RWST requirements. The proposed revisions are made to more accurately reflect the new content of the TS and be more consistent with the ISTS. In addition, changes have been made to the associated bases to reflect the removal of the boration system TS and the new title of TS 3/4.1.2.8, "RWST".

As a result of the revision proposed to the RWST requirements in TS 3/4.1.2.8, the reference to the RWST temperature limits in TS 3/4.6.2.1, "Quench Spray System" is no longer correct. The Quench Spray System TS surveillance 4.6.2.1.a.2 requires that the temperature of the borated water contained in the RWST be verified within the limits of Specification 3.1.2.8.b.3 at least once per 31 days. The proposed changes to the RWST requirements result in the elimination of 3.1.2.8.b.3 (due to format changes) which causes a disconnect between these TS. Although the RWST temperature is related to the performance of the Quench Spray System, TS 3/4.1.2.8, not TS 3/4.6.2.1, contains the requirements applicable to the RWST. TS 3/4.1.2.8 specifies the limits applicable to the RWST and the surveillances necessary to ensure the RWST is maintained within those limits. TS 3/4.1.2.8 also contains appropriate Actions applicable to the RWST when limits are not met that ensure the RWST is restored to within the required limits or the plant is put in an operational mode where the RWST is no longer required.

Surveillance 4.1.2.8 requires that the RWST temperature be verified within limits every 24 hours whenever the ambient air temperature exceeds the upper or lower temperature limit. As such, the Quench Spray System surveillance requirement to verify RWST temperature every 31 days is not necessary to assure the RWST temperature is maintained within the required limits and provides no additional safety benefit. In addition, the Quench Spray System TS does not specify the limits being verified by this surveillance nor does it contain Actions that are appropriate for the RWST if this surveillance is not met. The inclusion of an RWST surveillance in the Quench Spray System TS introduces confusion as to which TS is applicable when the RWST temperature is found not within limits. Consistent with the corresponding ISTS requirements, the redundant and potentially confusing Quench Spray surveillance is deleted. Based on the above discussion, the deletion of this unnecessary surveillance is considered an administrative change made to improve the internal consistency and clarity of the TS and to conform more closely to the content of the corresponding ISTS requirements.

Reactivity Control Systems

3/4.1.3.3 Position Indication System – Shutdown

Special Test Exceptions

3/4.10.5 Position Indication System – Shutdown (Unit 2 only)

Rod position indication requirements during startup and operation are addressed by the retained LCO, "Rod Position Indication" which satisfies criterion 2 of the NRC Policy Statement (verification of initial conditions of a design basis accident).

The TS requirement for rod position indication during shutdown, Modes (3, 4, and 5, with the reactor trip breakers closed) requires only one of the two types of indicators operable. In the shutdown Modes, the position indicating system provides information only and is not relied on by the operators to verify insertion or alignment limits (which are required only in Modes 1 and 2). Therefore, in the shutdown Modes the rod position indication system is not used to verify the initial conditions of a DBA. Additionally, during shutdown Modes, the rod position indication system is not used to verify a reactor trip, or assist in the mitigation of any other DBA or transient.

BVPS Special Test Exception 3.10.5 (Unit 2 only) provides an exception to the operability requirements specified in TS 3/4.1.3.3 for the digital position indication system during shutdown conditions. It does not contain requirements that are assumed in a design basis accident or transient. This special test exception is associated only with Unit 2 digital rod position indication system in TS 3/4.1.3.3, Position Indication System – Shutdown and is, therefore, proposed for relocation with TS 3/4.1.3.3. Unit 1 has no corresponding test exception.

Based on the discussions above, the BVPS TS 3/4.1.3.3 and Special Test Exception 3/4.10.5 were found not to meet any of the policy statement criteria. The BVPS specific application of the policy statement selection criteria to these TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

The bases text associated with the Movable Control Assemblies section of the TS does not contain a specific reference to the TS for Position Indication System – Shutdown. Therefore, no bases text from this section is proposed for relocation to the LRM.

In order to relocate the TS for Position Indication System – Shutdown, changes are required to the retained Position Indication System – Operating TS consistent with the guidance in Reference 2 and the content of Reference 3. Reference 2 identified the need to retain the Position Indication System – Shutdown surveillance that verified the agreement of the individual rod position indicators with the demand position indicators within 12 steps over the full indicated range of rod travel every 18 months. This surveillance was determined necessary to verify a key functional requirement of the rod position indication system and was retained in the ISTS TS 3.1.8, "Rod Position Indication" as SR 3.1.8.1.

The 18-month surveillance retained in the ISTS Rod Position Indication TS corresponds directly to the BVPS Unit 2 surveillance 4.1.3.3 from TS 3/4.1.3.3, "Position Indication System – Shutdown". The BVPS Unit 2 surveillance is also necessary to confirm the operability of the position indication system. Therefore, this BVPS shutdown TS surveillance will be retained as an additional surveillance in the Position Indication System – Operating TS consistent with the guidance in Reference 2 and the content of Reference 3. The wording of the Unit 2 surveillance will be revised to more closely agree with the ISTS wording while still conforming to the current TS format and presentation. The revised wording is considered editorial in nature and is not intended to introduce a technical change to the Unit 2

surveillance. The associated Unit 2 TS bases are also revised to reflect the proposed change to this surveillance requirement and to describe how it is met.

The BVPS Unit 1 surveillance from the TS for Position Indication System – Shutdown does not correspond to the surveillance requirement identified as necessary in Reference 2 and retained in Reference 3. The Unit 1 shutdown surveillance requirement does not address individual and demand position agreement over the full range of rod motion. The Unit 1 surveillance only addresses demand position indication accuracy over a limited range of rod movement. This surveillance is specific to the Unit 1 TS for Position Indication System – Shutdown and is not required to confirm a key functional requirement of the TS for Position Indication System – Operating.

The Unit 1 TS for Position Indication System – Operating does contain an 18-month surveillance that specifies a Channel Functional Test and Channel Calibration be performed on the individual rod position indication systems (4.1.3.2.2.b). This Unit 1 surveillance effectively verifies the individual and demand rod position indications agree within the required accuracy by calibration. However, the Unit 1 surveillance does not state the functional requirement of the system (i.e., demand and individual rod position agreement within 12 steps over the full range of rod travel) as clearly as the corresponding ISTS surveillance. Although the Unit 1 individual rod position indication system is based on an analog design, the functional requirement verified by the 18-month surveillance (demand and individual position indicator accuracy within 12 steps) should be the same as the digital system. Additionally, the word "digital" in the ISTS TS for rod position indication is bracketed indicating optional and that the TS is intended to be applicable to analog designs as well. Also, the acceptability of the ISTS 18-month surveillance for analog system designs has been previously determined and approved by the NRC for the Watts Bar initial license.

In order to more clearly state the functional requirement of the system in the Unit 1 TS for Position Indication System – Operating, and to make the Unit 1 18-month surveillance more consistent with the ISTS, the Unit 1 surveillance 4.1.3.2.2.b is revised. The proposed change replaces the 18-month Channel Functional Test and Channel Calibration surveillance with the same surveillance proposed for the Unit 2 Position Indication System – Operating TS (i.e., to verify the agreement of the individual rod position indicators with the demand position indicators within 12 steps over the full range of indicated rod travel every 18 months). Rod position is an assumption of the accident analysis and is the reason this TS has been retained.

The accuracy of this rod position instrumentation is relied upon to determine the plant is being operated within the assumptions of the applicable safety analysis. Therefore, the proposed surveillance specifies more directly that the key functional requirement of the system be verified instead of specifying that a calibration be performed. The calibrations specified in the current surveillance are more appropriate for instrumentation with specific setpoints that must be verified. The TS surveillances, with the proposed change, would require system accuracy be verified within the specified limit by a position check once every 12 hours and a more extensive verification over the full range of indicated motion every 18 months. The proposed surveillances are more consistent with the ISTS surveillances for the rod position indication system and provide adequate assurance of the position indication system operability.

As the proposed Unit 1 surveillance now specifies an accuracy requirement, Unit 1 TS 3/4.3.1.2 LCO footnote 1 allowing a one hour thermal soak time below 50% power before the specified accuracy must be met is applicable and is identified as such in the proposed surveillance. The allowance provided by the footnote currently modifies the position indication accuracy specified in the LCO. The addition of this footnote to the accuracy specified in the proposed surveillance makes the reference to the required accuracy consistent within the Unit 1 TS and avoids the potential for confusion regarding system operability and whether the proposed surveillance is met below 50% power.

The associated Unit 1 TS bases are also revised to reflect the proposed change to this surveillance requirement and to describe how it is met. The proposed change to the 18-month surveillance in TS 3/4.3.1.2 is consistent with the intent of the guidance provided in Reference 2 and the content of Reference 3. In addition, the proposed change makes the Unit 1 and 2 TS requirements more consistent and provides a Unit 1 surveillance with clear acceptance criteria based on meeting the assumptions of the safety analysis and key functional requirement of the system.

3/4.3.3.1 Monitoring Instrumentation Radiation Monitoring

The specific monitors proposed for relocation from the BVPS TS are listed in Table 2. The radiation monitors selected for relocation consist of area and noble gas effluent monitor types described in the Design Basis section of this LAR. The monitors and all associated TS requirements (i.e., LCO, Actions, and Surveillances) will be relocated together to form a complete radiation monitor specification in the ODCM or the LRM with all the applicable requirements from

the TS. However, as some of the radiation monitor TS requirements are common to retained and relocated monitors, the markup of TS 3/4.3.3.1 in Attachment A of this LAR only shows the requirements specific to the relocated Monitors as being removed from the TS. The TS requirements applicable to retained monitors, that will be reproduced in the ODCM or LRM, are marked in Attachment A as being duplicated in the ODCM/LRM.

The effluent related monitors will be placed in the ODCM and the area monitors will be relocated to the LRM. The effluent related radiation monitors are being relocated to the ODCM instead of the LRM to be consistent with the location selected for effluent monitors previously removed from the TS in accordance with Generic Letter 89-01.

As the content of TS 3/4.3.3.1 varies from plant to plant and depends somewhat on the time the plant was initially licensed, generic evaluations of TS 3/4.3.3.1 will not be appropriate for all plants. The Standard TS in Reference 3 provide guidance as to the types and bases for the radiation monitors required to remain within the TS. The radiation monitors retained in Reference 3 perform one the following functional requirements that have been determined to meet the policy statement criteria for inclusion in the TS. The radiation monitors retained in the TS must be:

1. Assumed in the safety analysis to provide automatic initiation of a system or component that is part of the primary success path and which functions 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 3),

2. Assumed in the safety analysis to provide an indication or alarm that is relied on by operators to initiate manual action that is part of the primary success path and which functions 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 3),

3. Installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary (Criterion 1 - RCS leak detection monitors), or

4. Instrumentation that monitors Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs

Conditions During and Following an Accident," Type A or Category I variables. Regulatory Guide 1.97 Type A variables are those indications that provide the primary information required for the control room operators to take specific 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 (Criterion 3). In general, Regulatory Guide 1.97 Category I (non-Type A) variables provide indications intended to assist operators in minimizing the consequences of accidents. As such, Category I (non-Type A) instrumentation has been identified as important for reducing risk to the public (Criterion 4).

The specific monitors proposed for relocation from the BVPS TS are listed in Table 2. These 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 the monitors proposed for relocation are inoperable require only that area surveys be performed on a daily basis or that explanations of inoperability be provided in an annual effluent report. The TS actions do not impact or reference the operability of other systems or require that plant operation be terminated at any time. Additionally, the radiation monitors proposed for relocation do not:

1. 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.

2. 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.

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

4. Monitor variables that have been identified as Regulatory Guide 1.97 Type A or Category I variables in the BVPS Unit 1 and Unit 2 responses to Regulatory Guide 1.97. The BVPS Unit 2 Regulatory Guide 1.97 variable Type and Category are contained in UFSAR Table 7.5-1. The BVPS Unit 1 Regulatory Guide 1.97 variable Type and Category are identified in the Unit 1 response to Generic Letter 82-33, Regulatory Guide 1.97, Revision 2,

Supplemental Report, transmitted to the NRC by letter dated October 13, 1986.

Some of the effluent monitors proposed for relocation provide indications used to assess selected plant parameters following an accident consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements." However, as stated in item 4 above, the monitors proposed for relocation do not provide indication for post accident variables that have been identified as Regulatory Guide 1.97 Type A or Category I. Therefore, the effluent monitors proposed for relocation do not meet the current post accident monitoring instrumentation criteria of Reference 3 for inclusion in the TS.

Radiation monitors in TS 3/4.3.3.1 that perform functional requirements identified in Reference 3 as meeting the policy statement criteria for inclusion in the TS have been retained within the BVPS TS. Therefore, based on the above discussions, the relocation of the radiation monitors listed in Table 2 is acceptable and results in the BVPS TS content being more consistent with Reference 3.

3/4.4.1.4.2 RCS Loop Isolation Valves - Shutdown

This BVPS TS provides requirements to remove the power from loop isolation valve operators whenever an RCS loop has been isolated in Modes 5 and 6 (with fuel in the vessel). The requirements of the TS provide assurance that an RCS isolation valve is not opened in an isolated loop. Due to the possible differences in boron concentration between the isolated loop and the RCS, the uncontrolled opening of an RCS isolation valve has the potential to cause a positive reactivity addition due to boron dilution. However, BVPS TS 3/4.4.1.5, "Isolated Loop Startup" also contains requirements to prevent a boron dilution event in Modes 5 and 6. The requirements of TS 3/4.4.1.5 not only require that a loop remain isolated but provide the conditions under which an isolation valve may be opened in an isolated loop. The Bases for TS 3/4.4.1.5 is also to prevent a boron dilution event. Additionally, the BVPS TS 3/4.4.1.4.2, "RCS Loop Isolation Valves – Shutdown" references the isolated loop startup TS to take exception to its requirement that power be removed from the isolation valves. A footnote in TS 3/4.4.1.4.2 provides an exception to the LCO requirement when the conditions in TS 3/4.4.1.5 are met for opening loop isolation valves.

Both BVPS TS provide loop isolation requirements to prevent a boron dilution event in Modes 5 and 6. The duplicate requirements in these Modes are similar to

the previous standard TS requirements in Reference 4 (see the Design Basis Discussion for more information regarding Reference 4). The current standard TS in Reference 3 eliminated this duplication of TS requirements. Reference 3 contains only one loop isolation TS applicable in Modes 5 and 6, TS 3.4.18, "RCS Isolated Loop Startup." The Reference 3 standard isolated loop startup TS and the BVPS TS 3/4.4.1.5, "Isolated Loop Startup" provide similar precautions to protect against a boron dilution event.

The operating restrictions associated with these TS have been identified as meeting criterion 2 of the policy statement. However, per Reference 3, two TS are not required to perform this function in Modes 5 and 6. BVPS TS 3/4.4.1.4.2, "RCS Loop Isolation Valves – Shutdown" contains requirements that are reasonable precautions to prevent inadvertently opening an isolation valve. However, TS 3/4.4.1.5, "Isolated Loop Startup" contains a more complete set of requirements in that it specifies both the requirement for isolated loops to remain isolated, and the conditions under which the loop isolation valves may be opened. BVPS TS 3/4.4.1.5 "Isolated Loop Startup" contains requirements to ensure: 1) That the isolated loop boron concentration is greater than or equal to the minimum required for shutdown margin in Modes 5 or 6, 2) That the isolated loop is drained and refilled from the RWST or RCS prior to opening the isolation valve and, 3) that the isolation valve is opened within 4 hours following refill of the loop. The requirements of TS 3/4.4.1.5 address a potential single failure of the Chemical and Volume Control System blender that could result in unborated water being used to fill the isolated loop and serve to minimize the potential for boron concentration stratification in the refilled loop.

TS 3/4.4.1.5, "Isolated Loop Startup" provides requirements that are adequate to address the accident of concern, a boron dilution event, and is more consistent with the standard TS in Reference 3. Therefore, TS 3/4.4.1.4.2 is not relied on to satisfy policy statement criteria and is proposed for relocation consistent with the elimination of this requirement from the standard TS in Reference 4.

How the RCS loops are maintained isolated (the subject of TS 3/4.4.1.4.2) is more of a procedural concern than material for a separate TS. The relocated requirements of TS 3/4.4.1.4.2, "RCS Loop Isolation Valves – Shutdown will be more appropriate as administrative controls in the LRM that support the TS 3/4.4.1.5, "Isolated Loop Startup." The relocation of TS 3/4.4.1.4.2 leaves the remaining BVPS TS more consistent with the content of Reference 3 and is

acceptable based on the more complete set of requirements provided by the retained TS 3/4.4.1.5, "Isolated Loop Startup."

TS 3/4.4.2 RCS Safety Valves - Shutdown

The pressurizer safety valves provide automatic overpressure protection for the RCS. The pressurizer safety valves prevent the RCS from being pressurized above the safety limit. There are currently two TS that address the RCS safety valves. TS 3/4.4.2 requires at least one safety valve operable in Modes 4 and 5 and TS 3/4.4.3 requires all the safety valves operable in Modes 1-3. However, the setpoint (2485 psig) associated with the code safety valve required operable by TS 3/4.4.2 does not provide complete pressure protection for shutdown conditions. The setpoint is too high to protect the 10 CFR 50 Appendix G pressure limits applicable at low RCS temperatures. The overpressure protection system (addressed in TS 3.4.9.3) provides the appropriate pressure protection for low RCS temperature conditions and is the system relied on in the safety analyses for RCS low temperature overpressure protection.

The standard TS of Reference 3 specify two TS for automatic RCS overpressure protection credited in the applicable safety analyses (the Pressurizer Safety Valve TS 3.4.10 and the Low Temperature Overpressure Protection System TS 3.4.12). There is no separate TS for RCS safety valves - shutdown in the standard. The applicability of the standard TS assures one method of automatic pressure protection is always operable when required. Standard TS 3.4.10 requires the pressurizer safety valves operable to provide overpressure protection in Modes 1-3 and includes Mode 4 with all RCS cold leg temperatures greater than the "enable temperature" of standard TS 3.4.12. The enable temperature is that temperature at which the low temperature overpressure protection system is required operable. Once the enable temperature specified in standard TS 3.4.12 is reached in any RCS cold leg, the low temperature overpressure protection TS becomes applicable and the associated overpressure protection system is required to be operable. In this manner, the standard TS provide continuous RCS overpressure protection utilizing the two TS discussed above.

In order to make the BVPS TS for RCS pressure control more consistent with the standard TS, the Mode of applicability and Actions for BVPS TS 3/4.4.3 Safety Valves – Operating must be revised. In addition to Modes 1-3, the proposed change requires that the pressurizer safety valves be operable in Mode 4 with all cold leg temperatures above the enable temperature specified in TS 3/4.4.9.3. This

change effectively incorporates the Mode requirement for safety valves specified in the standard TS into the BVPS TS. The BVPS TS for the low temperature overpressure protection system already contains the same applicability as the standard so no change is required to that TS. The proposed revision to the applicability of the safety valve TS only references the enable temperature specified in 3.4.9.3 consistent with how this temperature is referenced in other BVPS TS. The actual enable temperature for the low temperature overpressure protection function is contained only within one BVPS TS (3/4.4.9.3). Referencing the enable temperature in other TS instead of specifying the temperature in each TS serves to minimize TS changes when the enable temperature is revised. The BVPS Unit 1 enable temperature is 329°F which is well within Mode 4 so the standard applicability is acceptable and is proposed for Unit 1. However, the BVPS Unit 2 enable temperature is 350°F which corresponds to the defined limit of Mode 3 operation (≥ 350°F). For Unit 2, the standard applicability of all cold leg temperatures greater than 350°F would be in Mode 3, not Mode 4. Mode 3 is already a part of the Unit 2 applicability for TS 3/4.4.3 and the standard TS applicability that references Mode 4 would not be correct. However, in order to maintain some consistency between the Unit TS and to help identify the requirement for the safety valves to be operable at the point where the low pressure protection system is no longer required, a revision is also proposed for the Unit 2 TS 3/4.4.3 applicability. The revision to the Unit 2 TS applicability is slightly different than the standard as it omits reference to Mode 4 and simply specifies "With all RCS cold leg temperatures above the enable temperature specified in 3.4.9.3." Currently this applicability would be equivalent to Mode 3. However, if the Unit 2 enable temperature is ever reduced, the proposed change would correctly identify the new Unit 2 applicability based on the revised enable temperature. The Actions associated with TS 3/4.4.3, Safety Valves - Operating are also revised to incorporate the change in applicable Modes and give clear direction that the requirements of Specification 3.4.9.3 are applicable. The Actions are revised to ensure the plant is placed in a mode where the TS is no longer required (i.e., any cold leg temperature ≤ the enable temperature specified in 3.4.9.3) if the safety valve is not restored within the allowed time.

In addition to the revisions described above, the title of BVPS TS 3/4.4.3 is revised to delete the reference to "operating". As this change removes the TS containing the shutdown requirements for safety valves, the distinction between operating and shutdown requirements in the TS is no longer required. This change is consistent with the title of the corresponding standard TS in Reference 3. The TS Bases

associated with the safety valves is also revised to remove reference to TS 3/4.4.2 (Safety Valves - Shutdown) and to reference TS 3/4.4.9.3 (Overpressure Protection Systems) as providing overpressure protection during shutdown conditions. The bases change is consistent with the changes made to the TS and more correctly identifies the overpressure protection system credited in the safety analyses for shutdown conditions.

As revised the requirements of BVPS TS 3/4.4.3, "Safety Valves" and BVPS TS 3/4.4.9.3, "Overpressure Protection Systems" ensure continuous RCS overpressure protection. As such, the BVPS TS 3/4.4.2, Safety Valves - Shutdown, requirement for a single pressurizer safety valve to be operable during all of Modes 4 and 5 is not required for RCS overpressure protection. In addition, the operability of a single safety valve in Modes 4 and 5 is not an assumption of any safety analysis for the mitigation of a design basis accident or transient in Modes 4 and 5. Therefore, the requirements contained in this TS do not meet any policy statement criteria. The BVPS specific application of the policy statement selection criteria to this TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

3/4.4.7 RCS Chemistry

The RCS Chemistry TS provides limits on particular chemical properties of the RCS and surveillance requirements to monitor those properties. The TS limits function to help control degradation of the RCS pressure boundary by minimizing corrosion. However, the degradation of the RCS pressure boundary is a long-term process, for which other more direct methods of monitoring and correction are available. The inservice inspections required by regulations (10 CFR 50.55a) and the RCS leakage limits in TS are examples of existing requirements to monitor and prevent long-term degradation of the RCS boundary and that provide for long-term structural maintenance of acceptable RCS conditions. The RCS Chemistry TS requirements are not of immediate importance to the operator and are not required to ensure operability of the RCS pressure boundary. Therefore, the requirements contained in this TS do not meet any policy statement criteria. The BVPS specific application of the policy statement selection criteria to this TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

3/4.4.9.2 RCS Pressurizer Temperature Limits

Temperature limits are placed on the pressurizer to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements to maintain pressurizer integrity. A failure of pressurizer integrity would result in an analyzed event (loss of coolant accident) for which numerous systems and components are required and retained in the TS. While the pressurizer limits represent operating restrictions and Criterion 2 includes operating restrictions, Criterion 2, as discussed in the policy statement, applies to those operating restrictions required to preclude unanalyzed accidents and transients (e.g., fracture of the reactor vessel). As such, the pressurizer temperature limits are not relied on to prevent or mitigate a DBA or transient, nor are these limits an operating restriction that is required to preclude an unanalyzed accident or transient. Therefore, the requirements contained in this TS do not meet any policy statement criteria. The BVPS specific application of the policy statement selection criteria to this TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

3/4.4.10 RCS Structural Integrity

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. The requirements of this TS are directed toward the prevention of degradation and continued long term maintenance of acceptable structural and component conditions. Although important to the general maintenance of systems and structures the ASME inspection and testing programs are also required by regulations (10 CFR 50.55a). Current BVPS TS 4.0.5 also requires the application of surveillances to implement these ASME programs. Additionally, there are specific system related TS (e.g., RCS leakage limits and RCS Loops) that require the systems and components important to safety to be operable. The system related TS provide the specific requirements important to plant operators to ensure the safe operation of the plant including appropriate remedial measures and other actions for inoperable components or systems and excessive leakage. Given the regulations and existing TS 4.0.5 require the implementation of ASME programs and that specific TS address the operability of required systems and components (including RCS Loops and RCS leakage), a separate TS (3/4.4.10) is not required to highlight or support the ASME programs. As such TS 3/4.4.10 is not required to ensure the immediate operability of safety systems. Therefore, TS 3/4.4.10 does

not meet any of the policy statement criteria. The BVPS specific application of the policy statement selection criteria to this TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

3/4.4.12 RCS Vents

The RCS Vents are provided to exhaust non-condensable gases and/or steam from the RCS which may inhibit natural circulation core cooling following any event involving a loss of offsite power and for which long term cooling is required, such as a loss of coolant accident. The functional capabilities, and testing requirements specified for the RCS vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements." However, the operation of the RCS Vents is not an automatic action and is not assumed in any safety analysis described in UFSAR to mitigate a design basis accident. As such, the operation of the RCS vents is not part of the primary success path for any design basis event. Rather, the operation of these vents is the result of operator action taken for long term cooling concerns after an event has occurred, and is required only when there is indication that natural circulation is not occurring and no other means of core cooling is available. Therefore, TS 3/4.4.12 does not meet any of the policy statement criteria that pertain to equipment or systems assumed to operate in the safety analyses to mitigate or prevent a design basis accident or transient. The BVPS specific application of the policy statement selection criteria to this TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

3/4.7.2.1 Steam Generator Pressure/Temperature Limitation

The steam generator pressure and temperature limits specified in this TS ensure that the pressure induced stresses are within the maximum allowable fracture toughness stress limits. The values of the specified limits are based on maintaining a steam generator reference transition nil ductility temperature sufficient to prevent brittle fracture. The TS places limits on variables consistent with the structural analysis. However, these TS limits are not initial condition assumptions of a design basis accident or transient described in the UFSAR, but represent operating restrictions. Although Criterion 2 includes operating restrictions, Criterion 2, as discussed in the policy statement, applies to those operating restrictions required to preclude unanalyzed accidents and transients (e.g., fracture of the reactor vessel). The failure of steam generator integrity results in analyzed events (e.g., steam generator tube rupture or other loss of coolant accident) for which adequate

mitigation systems and components are provided and required operable by the TS. The steam generator pressure/temperature limitation is not an initial condition of a design basis accident or transient, nor is this limitation an operating restriction that is required to preclude an unanalyzed accident or transient. Therefore, TS 3/4.7.2.1 does not meet any of the policy statement criteria. The BVPS specific application of the policy statement selection criteria to this TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

3/4.7.6.1 Flood Protection

The specified TS limit was selected as an appropriate river water level at which to terminate plant operation and initiate flood protection measures for safety related equipment. The TS requirements provide for equipment protection and specify a precautionary plant shutdown in the event of a high water level. The TS water level limit selected for flood protection is not a process variable or operating restriction that is an initial condition of a design basis accident or transient analysis described in the UFSAR. The river water level is not a "controlled" variable or operating restriction directly related to power operation or any design basis accident or transient. Additionally, there are retained TS requirements for the required safety systems that control the operability and provide appropriate remedial actions for the equipment that may be affected by river water level. The water level limit is primarily intended for equipment protection and although the TS does require a precautionary plant shutdown, river water level changes are relatively slow and do not have an immediate or direct effect on reactor operation or the health and safety of the public. Therefore, TS 3/4.7.6.1 does not meet any of the policy statement criteria. The BVPS specific application of the policy statement selection criteria to this TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

3/4.7.9.1 Sealed Source Contamination

The TS limits on sealed source contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limits on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. The requirements of this TS provide additional measures to help meet the goal of the lowest achievable dose to plant personnel. The requirements of this TS do not affect reactor operation or pertain to plant equipment, process variables, or operating restrictions that are required for

safe operation of the plant. Nor are the requirements of this TS related to any design basis accident or transient. The primary purpose of the TS is to control radioactive sources to minimize personnel exposure. Therefore, TS 3/4.7.9.1 does not meet any of the policy statement criteria. The BVPS specific application of the policy statement selection criteria to this TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

3/4.7.12 Snubbers

The snubber TS requires all snubbers to be operable. Snubbers function to prevent unrestrained pipe motion under dynamic loads and allow normal thermal expansion of piping and nozzles to eliminate excessive thermal stresses during heatup or cooldown. The TS requirements assure the long-term maintenance of snubbers and that defective snubbers are repaired or replaced or that the operability of affected systems are evaluated. However, the operability of required systems or components designed with snubbers is primarily determined and controlled by the TS requirements associated with the affected system or component. The snubber TS is not required to assure the operability of other systems or components specified in the TS. Additionally, the snubber TS surveillance requirements are implemented in the BVPS inservice inspection program and the inservice inspection program is required by 10 CFR 50.55a. As such, controls other than the snubber TS exist to assure required systems and components are maintained operable and appropriate snubber inspections are performed. The requirements for snubber operability in TS 3/4.7.12 do not directly impact reactor operation, do not identify a parameter that is an initial condition of a design basis accident or transient, and are not considered part of the primary success path which functions or actuates to mitigate a design basis accident or transient. Therefore, TS 3/4.7.12 does not meet any of the policy statement criteria. The BVPS specific application of the policy statement selection criteria to this TS is consistent with the findings of Reference 2 and the content of Reference 3.

3/4.7.13.1 Auxiliary River Water System (Unit 1)

3/4.7.13.1 Standby Service Water System (Unit 2)

These TS require the operability of the backup cooling water systems, "Auxiliary River Water System" (ARWS) for Unit 1 and the "Standby Service Water System" (SSWS) for Unit 2. The TS requirements address the operability of the required cooling water system equipment and components. The ARWS and the SSWS TS

contain requirements related to the cooling water systems they address and not instrumentation that is used to detect and indicate a significant abnormal degradation of the reactor coolant pressure boundary. Nor do these TS contain requirements for process variables, design features, or operating restrictions that are an initial condition of a design basis accident or transient analysis. As such, the requirements of these TS do not meet Criteria 1 and 2 of the policy statement.

The BVPS Unit 1 River Water System and the BVPS Unit 2 Service Water System are the redundant-train safety-grade safety-related systems which provide the cooling water from the Ohio River credited in the design basis accidents and design basis anticipated operational occurrence transients described in Chapters 6/14 and 6/15 of the BVPS Unit 1/Unit 2 UFSARs, respectively. The River Water System and Service Water System provide the design basis heat transfer cooling from the Ohio River to remove heat from various power plant auxiliary systems during all modes of operation, in accordance with 10 CFR 50 Appendix A General Design Criteria. The River Water System and the Service Water System are designed to withstand any single active or passive failure, natural phenomena and design basis conditions such as pipe breaks and missiles. The River Water System at Unit 1 and Service Water System at Unit 2 are the primary success path of the UFSARs' safety analyses when crediting the use of the Ohio River for component or system cooling. The Unit 1 River Water System will remain in the BVPS Unit 1 TS (3/4.7.4.1) and the Unit 2 Service Water System will remain in the BVSP Unit 2 TS (3/4.7.4.1).

The ARWS and SSWS were designed to provide a backup means (from the Unit 1 River Water and Unit 2 Service Water Systems) to supply cooling water to the plant in the event of a total loss of the normal River or Service Water Systems due to an explosion caused by a gasoline barge impacting the main intake structure. The ARWS or SSWS are not required to function coincident with a postulated design basis accident or anticipated operational occurrence typically described in Chapters 6 and 15 of an UFSAR. The policy statement, in discussing Criterion 3, clearly identifies the design basis events of concern as being described in Chapters 6 and 15 (or equivalent) of a plant's UFSAR. The gasoline barge impact and explosion falls within the scope of Chapter 2 in both Units' UFSARs which addresses various postulated events at nearby locations. Section 9.16.1 in the Unit 1 UFSAR and Section 9.2.1.2 in the Unit 2 UFSAR describe the gasoline barge impact and explosion as being the site related basis for maintaining the ARWS and SSWS as a backup means for providing Ohio River cooling water.

Therefore, the gasoline barge explosion is a licensing basis requirement addressing a postulated event at a nearby location as described in Chapters 2 and 9 of the UFSARs. However, this event is beyond the scope of the design basis accidents and anticipated operational occurrences described in Chapters 6 and 14/15 of the BVPS Unit 1/2 UFSAR. Therefore, the ARWS and SSWS are not part of the primary success path or credited in any safety analysis to function or actuate to mitigate a design basis accident or design basis anticipated operational occurrence as described in Chapters 6 and 14/15 of the BVPS Unit 1/2 UFSAR. Therefore, the requirements of these TS do not meet criterion 3 of the policy statement.

3/4.9.5 Communications

The purpose of this TS is to ensure that personnel at the refueling station can be promptly informed of significant changes in the facility status or core reactivity conditions during core alterations. The requirement also allows for the efficient coordination of more routine activities that require interaction between the control room and personnel in containment. Although this requirement represents good operational practice, and is designed to ensure safe refueling operations, the communication requirement of this TS is not credited in any design bases accidents or transients described in the UFSAR. Therefore, TS 3/4.9.5 does not meet any of the policy statement criteria. The BVPS specific application of the policy statement selection criteria to this TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

3/4.9.6 Manipulator Crane Operability

This TS ensures that the manipulator crane and auxiliary hoist will have sufficient load capacity, correct overload cut off limit (manipulator crane), and load indicators (auxiliary hoist) for their intended purpose and will be used correctly during refueling. The requirements of this specification ensure that the core internals and reactor vessel are protected from excessive lifting force during refueling operations in the event they are inadvertently engaged during lifting operations. Although this specification contains requirements designed to prevent damage to fuel assemblies, core internals, and reactor vessel, these requirements are not relied upon in any safety analysis described in the UFSAR to prevent or mitigate the consequences of a DBA. The limitations of this specification apply only to the design requirements of the cranes. Plant design requirements are adequately governed by regulations and the required quality assurance plan and are not required to be part of the TS. Therefore, TS 3/4.9.6 does not meet any of the

policy statement criteria. The BVPS specific application of the policy statement selection criteria to this TS is consistent with the findings of References 1 and 2 and the content of Reference 3.

BASES CONTROL PROGRAM

The proposed program addition is administrative in nature and serves to provide guidance for making changes to the TS bases documentation. The addition of the program also serves to make the BVPS TS more consistent with the TS in Reference 3 and the BVPS bases control methods more consistent with standard industry practice. The adoption of this ISTS program by BVPS is reasonable considering the relatively small size of the BVPS TS bases compared to the ISTS bases. The ISTS bases control program has been determined to be acceptable for controlling the ISTS bases which are significantly larger and contain much more detail regarding the operability requirements for the associated TS than do the current BVPS bases.

The ISTS bases control program provides adequate regulatory control to ensure that prior NRC approval of bases changes will be requested when required by 10 CFR 50.59. The program also requires appropriate plant administrative controls be established for the review and approval of TS bases changes and that the administrative controls include provisions to ensure the bases are maintained consistent with the UFSAR. Additionally, the bases control program contains provisions to ensure that bases changes implemented without NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Based on the assurance provided by these program requirements, changes to the BVPS TS bases will be adequately controlled and the proposed addition of the ISTS program is acceptable. Additionally the proposed change is consistent with the TS improvements described in the policy statement, in that it will result in more efficient use of BVPS and NRC resources and enhance safety by focusing the attention of those resources on requirements more significant to plant safety.

The proposed addition of the ISTS bases control program includes the NRC approved change to NUREG-1431, Revision 1 contained in change number TSFT-364. This change updates a reference to 10 CFR 50.59 in the bases control program to be consistent with proposed changes to that regulation. The change is shown in the TS markups included in Attachment 1.

D. SAFETY ANALYSIS

TS RELOCATION

The evaluations discussed above for the TS proposed for relocation confirm that the requirements in these TS are not the specific assumptions of a BVPS safety analysis and that they do not meet the 10 CFR 50.36 criteria for inclusion in the TS; i.e., they do not pertain to:

1. Instrumentation used to detect a significant abnormal degradation of the RCS pressure boundary,

2. Process variables, design features or operating restrictions that are an initial condition of a design basis accident or transient,

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, or

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

Based on the above, it can be concluded that the TS proposed for relocation do not meet the subjective statement of the purpose of TS cited in the policy statement where TS were interpreted "as being reserved for those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Additionally, the proposed relocation of TS does not reduce the effectiveness of the TS requirements being relocated. Rather, the relocation of the TS result in a change in the regulatory control required for future changes made to the TS requirements. The requirements contained within the affected TS will continue to be implemented by the appropriate plant procedures (e.g., operating and maintenance procedures) in the same manner as before. However, future changes to the relocated requirements will be controlled in accordance with 10 CFR 50.59 instead of requiring a license amendment per 10 CFR 50.90.

Consistent with the guidance of NRC Administrative Letter 96-04, the proposed destination documents (LRM and ODCM) for the relocated TS are referenced in the BVPS Unit 1 and 2 UFSAR. In addition, changes to the ODCM are controlled

by the TS (Section 6.14). Both the LRM and ODCM have been previously determined by the NRC to be acceptable for relocated TS. The ODCM was identified as an acceptable location for relocated TS in Generic Letter 89-01 and the LRM was previously approved in BVPS license amendment numbers 233 and 115 (SER dated 9/7/00) for the relocation of TS. Therefore, adequate regulatory control of the destination documents exists to ensure that future changes to these requirements are controlled by the provisions of 10 CFR 50.59 and that prior NRC review and approval of changes will be requested when required by 10 CFR 50.59. The provisions of 10 CFR 50.59 establish adequate controls for material removed from the TS, including record retention and reporting requirements. The provisions of 10 CFR 50.59 assure future changes to the relocated material will be consistent with safe plant operation.

Based on the design basis and justification discussions in Sections B and C of this LAR, and the conclusions of Westinghouse and the NRC in References 1 and 2, and the control of future changes to the relocated TS provided by 10 CFR 50.59, the relocation of the affected TS will not: 1) affect the applicable accident analyses, 2) adversely affect the safe operation of the BVPS units, or 3) reduce the margin of safety derived from the affected TS.

In order to support the relocation of selected TS, this LAR proposes additional changes to the BVPS TS. The additional changes include:

1. The revision of retained TS 3/4.1.2.8, Borated Water Sources Operating, to an ISTS like RWST TS and removal of the Boric Acid Storage System portion of the TS,

2. The addition of the 18 month surveillance in retained TS 3/4.1.3.2, Position Indication Systems-Operating and revision of the corresponding Unit 1 surveillance,

3. The revision of retained TS 3/4.3.3.1, Radiation Monitoring, to remove only those requirements selected for relocation,

4. The revision of the applicability and Actions of retained TS 3/4.4.1.4.1, Safety Valves-Operating to match the applicability of the low temperature overpressure protection system TS, and

5. Various administrative changes made to the pagination, table of contents, bases text and TS titles to accommodate the removal of the TS selected for relocation and the changes listed above.

The individual changes listed above refine the requirements of the affected TS consistent with the policy statement and associated References 1-3.

The revision of retained TS 3/4.1.2.8, Borated Water Sources Operating is made consistent with the application of the policy statement criteria, the content of Reference 3, and the applicable BVPS safety analyses. The revisions made to the TS after the relocation of the Boric Acid Storage System per the policy statement criteria were made to adjust the format and presentation of the remaining TS requirements to be more consistent with the corresponding ISTS. The retained RWST parameters assumed in the BVPS safety analyses and included in the TS were not changed. Similarly, the elimination of the Quench Spray System TS surveillance to verify the RWST temperature every 31 days does not reduce the ability of the plant staff or the requirements of the TS to assure that the RWST is operated within the specified limits. The surveillance for RWST temperature retained in TS 3/4.1.2.8 requires that RWST temperature be verified every 24 hours when the ambient air temperature is outside the TS limits. As such, the TS continue to require that the RWST parameters be verified within the limits assumed in the BVPS safety analyses. Therefore, the revision of the retained RWST portion of TS 3/4.1.2.8 and the elimination of the Quench Spray System surveillance are considered administrative changes made only to more closely conform to the corresponding standard TS and improve the consistency and clarity of the TS. Therefore, these changes do not adversely affect the safe operation of the plant.

The addition of the 18-month surveillance in retained TS 3/4.1.3.2, Position Indication Systems-Operating and revision of the corresponding Unit 1 surveillance were also made consistent with the application of the policy statement criteria, content of Reference 3, and requirements stated in Reference 2. The revision of the retained TS for Position Indication Systems – Operating was made to retain a surveillance requirement determined necessary to verify a key operability requirement of the system. The revision of the Unit 1 surveillance to be more consistent with the corresponding Unit 2 and ISTS surveillance provides an alternate but equivalent requirement for the operability of the system. The revised Unit 1 surveillance specifies that the key functional requirement of individual and demand indication accuracy of the system be met with a clear statement of the

relevant acceptance criteria. The accuracy of this indication is required to provide assurance that rod position is within the assumptions of the safety analysis. The bases revisions associated with this change were made to reflect the revised surveillance and provide additional descriptive detail. As such, the revised surveillance does not diminish the operability requirements for the system but improves the clarity of the surveillance and brings it more in line with the associated LCO requirements and safety analysis assumptions. Therefore, this change does not adversely affect the safe operation of the plant.

The revision of retained TS 3/4.3.3.1, Radiation Monitoring, to remove only those requirements selected for relocation represents an administrative change to accommodate the relocations previously discussed. The TS requirements applicable to the retained monitors are not altered and, therefore, remain consistent with the assumptions in the setpoint calculation or safety analysis associated with each monitor. As such, this change does not adversely affect the safe operation of the plant.

The revision of the applicability and Actions of retained TS 3/4.4.1.4.1 for Safety Valves-Operating to match the applicability of the low temperature overpressure protection system TS provides a logical improvement to make the BVPS RCS pressure protection TS provide complete coverage (operating and shutdown) consistent with the ISTS. The revision of the TS applicability also provides more accurate guidance regarding the operability of the RCS pressure protection systems that takes into account the applicable safety analyses and eliminates unnecessary overlapping operability requirements (safety valves and low temperature systems). When all RCS cold legs are above the low temperature overpressure protection enable temperature the RCS safety valves are required operable. With one cold leg below this temperature the low temperature overpressure protection system is required to be operable. This division of TS applicabilities results in the correct overpressure protection system required operable consistent with the safety analysis and without requiring safety valves operable at low temperature conditions where they do not provide the appropriate protection. As such, this change continues to provide assurance that adequate overpressure protection for the RCS is maintained and that the appropriate systems are required operable consistent with the safety analyses. Therefore, this change does not adversely affect the safe operation of the plant.

Additional changes made to the index, bases, format, presentation, pagination, or titles of retained TS are only those changes necessary to accommodate the

relocation of the TS addressed in this LAR. These changes are considered administrative in nature and necessary to fully incorporate changes previously discussed in this LAR. The administrative changes do not introduce any new or different technical changes than previously discussed. Therefore, these changes have no effect on the safe operation of the plant.

BASES CONTROL PROGRAM

The proposed program addition is administrative in nature and serves to provide guidance for making changes to the TS bases documentation. The addition of the program also serves to make the BVPS TS more consistent with the TS in Reference 3 and the BVPS bases control methods more consistent with standard industry practice. The addition of this program to the TS provides adequate regulatory control to assure bases changes are performed in accordance with 10 CFR 50.59 and that the changes are performed under appropriate administrative controls. The 10 CFR 50.59 process provides assurance that prior NRC review and approval of changes will be requested when appropriate. As such, the addition of this program enhances the administrative control of changes to bases documents and conforms to an industry standard method that provides control of bases changes in accordance with the 10 CFR 50.59 process. Additionally, the proposed change does not affect the operation of the plant or any analyzed accident or transient described in the UFSAR. Therefore, this change does not adversely affect the safe operation of the plant.

E. NO SIGNIFICANT HAZARDS EVALUATION

This license amendment request (LAR) revises the Beaver Valley Power Station (BVPS) Unit 1 and 2 technical specifications (TS) to implement improvements endorsed in the NRC's Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, 58 FR 39132, July 22, 1993 (the policy statement). The major change proposed in this LAR involves the application of the TS screening criteria from the policy statement (codified in 10 CFR 50.36) to evaluate the content of the BVPS TS. The BVPS technical specifications proposed for relocation in this license amendment request do not meet the 10 CFR 50.36 criteria for retention in the technical specifications; i.e., the technical specifications proposed for relocation do not contain requirements that pertain to: 1) Installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary, 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, 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 or, 4) A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Consistent with the policy statement guidance, the TS that do not meet the criteria are proposed for relocation to documents controlled by BVPS. The proposed locations for the relocated TS requirements are the Licensing Requirements Manual (LRM) and Offsite Dose Calculation Manual (ODCM). The LRM and ODCM are referenced in the Unit 1 and Unit 2 Updated Final Safety Analysis Report (UFSAR). Changes to documents referenced in the UFSAR are required to be made in accordance with 10 CFR 50.59. As such, changes to the relocated TS requirements will be in accordance with the provisions of 10 CFR 50.59 and prior NRC review and approval of changes will be requested if required by 10 CFR 50.59.

In order to support the relocation of certain TS, this LAR also proposes changes to retained TS and bases. In addition, this LAR also proposes the addition of a TS Bases control program consistent with the ISTS. The proposed changes in this LAR that affect retained TS and bases are administrative in nature and are made to support the relocation of TS and provide clarifications and enhancements that serve to make the existing BVPS TS more consistent with the content of the Improved Standard Technical Specifications (ISTS) for Westinghouse Plants contained in NUREG-1431.

Consistent with the guidance in the NRC policy statement, the relocation of TS requirements and changes to retained TS proposed in this LAR will also result in TS that are more consistent with the Atomic Energy Act, 10 CFR 50.36, and previous interpretations of the regulations governing TS content.

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

The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility

licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

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

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

   The proposed amendment does not involve a significant increase in the probability of an accident previously evaluated because no changes are being made to any event initiator. Nor is any analyzed accident scenario being revised. The initiating conditions and assumptions for accidents described in the UFSAR remain as previously analyzed.

   The proposed amendment also does not involve a significant increase in the consequences of an accident previously evaluated. The amendment does not reduce the current operability requirements contained in the TS proposed for relocation. The proposed relocation of TS requirements only affects the level of regulatory control involved in future changes to the requirements. Additionally, the TS proposed for relocation do not meet the 10 CFR 50.36 criteria for retention in the TS.

   The additional changes proposed to retained TS in this LAR, including the addition of the TS Bases Control Program, are either enhancements, clarifications, or administrative in nature, and are made to support the relocation of TS and to be more consistent with the ISTS and plant specific safety analyses. The changes to retained TS have no adverse effect on the safety analyses for design basis accidents described in the UFSAR. The initiating conditions and assumptions for accidents described in the UFSAR remain as previously analyzed.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The proposed amendment does not involve any physical changes to the plant or the modes of plant operation defined in the TS. The proposed amendment does not involve the addition or modification of plant equipment nor does it alter the design or operation of any plant systems. No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes.

There are no changes in this amendment that would cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new mode of failure has been created and no new equipment performance requirements are imposed. The proposed amendment has no effect on any previously evaluated accident.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

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

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

The relocation of TS does not reduce the effectiveness of the requirements being relocated. Rather, the relocation of the TS results in a change in the regulatory control required for future changes made to the requirements. Additionally, the technical specifications proposed for relocation do not meet the 10 CFR 50.36 criteria for retention in the technical specifications.

The requirements contained within the affected TS will continue to be implemented by the appropriate plant procedures (e.g., operating and maintenance procedures) in the same manner as before. However, future changes to the relocated requirements will be controlled in accordance with 10 CFR 50.59 instead of a license amendment pursuant to 10 CFR 50.90. The provisions of 10 CFR 50.59 establish adequate controls over requirements removed from the TS and assure future changes to these requirements will be consistent with safe plant operation.

The additional changes proposed to retained TS in this LAR, including the addition of the TS Bases Control Program, are either enhancements, clarifications, or administrative in nature, and are made to support the relocation of TS and to be more consistent with the ISTS and plant specific safety analyses. These changes do not alter any operating parameters or design requirements assumed in a safety analysis for systems or components important to the mitigation and control of design bases accident conditions within the facility. Nor do these changes alter safety limits or safety system settings required for safe operation of the plant or the assumptions of any safety analysis.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

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

G. ENVIRONMENTAL CONSIDERATION

This license amendment request changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. It has been determined that this license amendment request involves no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. This license amendment request may change requirements with respect to installation or use of a facility component located within the restricted area or change an inspection or surveillance requirement; however, the category of this licensing action does not individually or cumulatively have a significant effect on the human environment. Accordingly, this license amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this license amendment request.

H. REFERENCES

1. WCAP 11618, "Methodically Engineered, Restructured And Improved Technical Specifications, MERITS Program – Phase II Task 5 CRITERIA APPLICATION," November 1987.

2. NRC Letter From T. E. Murley to W. S. Wilgus (Industry Owners Groups Chairman), dated May 9, 1988.

3. NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 1, April 1995.

4. NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," Revision 5 Draft, 1984.

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

6. NRC Administrative Letter 96-04, "Efficient Adoption of Improved Standard Technical Specifications," October 1996.

7. Generic Letter 89-01, "Implementation Of Programmatic Controls For Radiological Effluent Technical Specifications In The Administrative Controls Section Of The Technical Specifications And The Relocation Of Procedural Details Of Rets To The Offsite Dose Calculation Manual Or To The Process Control Program," January 1989.