NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

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April 19, 1984

Docket No. 50-423

B11111

Director of Nuclear Reactor Regulation
Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Reference: (1) B. J. Youngblood to W. G. Counsil, Draft Safety Evaluation Report (DSER) for Millstone Nuclear Power Station, Unit No. 3, dated December 20, 1983.

Dear Mr. Youngblood:

Millstone Nuclear Power Station, Unit No. 3
NRC Procedures and Systems Review Branch
Transmittal of Responses to Draft Safety Evaluation Report Open Items

Reference (1) transmitted to us the open items for Millstone Unit No. 3, including items under the responsibility of the NRC's Procedures and Systems Review Branch (PSRB). On February 21, 1983, a meeting was held in Bethesda, Maryland between the NRC PSRB and Northeast Utilities to discuss each of these open items. Attachment I provides the status of each PSRB open item as a result of this meeting. The status of each item is defined by one of the following three categories:

Closed- No further NNECO input or action required for resolution of item.

Confirmatory- NNECO to provide requested information on the Millstone Unit No. 3 docket at a later date.

Open - No resolution at this time; NNECO to address.

Attachment II formally transmits our response to each DSER PSRB open item. The responses in Attachment II are being provided as they will appear in an upcoming FSAR amendment.

B001
1/40

If you have any concerns related to the information contained herein or any questions related to our responses, please contact our licensing representative, Ms. P. Capello-Bandzes at (203) 665-3714.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL

By Northeast Nuclear Energy Company, Their Agent

W G Counsil

W. G. Counsil
Senior Vice President

cc: Mr. F. J. Liederbach
NRC Procedures and Systems Review Branch

Mr. R. A. Becker
NRC Procedures and Systems Review Branch

Mr. R. Gruel
Pacific Northwest Laboratory--Battelle

STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

Then personally appeared before me W. G. Counsil, who being duly sworn, did state that he is Senior Vice President of Northeast Nuclear Energy Company, an Applicant herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Applicants herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Lorraine J. D'Amico

Notary Public

My Commission Expires March 31, 1988

ATTACHMENT I

Status of Each DSER PSRB Open Item

Open Item	Subject	Status
PSRB-01	TMI Action Item I.C.1	Open**
PSRB-02	ANSI/ANS 3.2-1981 (Draft 7) or 1982, Section 5.3	Closed
PSRB-03	Procedures Generation Package Familiarization	Closed
PSRB-04	Alarm Response Procedures	Closed
PSRB-05	Procedures that Include Immediate Actions to be Memorized	Closed
PSRB-06	Commitment Concerning Plant Operations	Closed
PSRB-07	Procedures for Abnormal Release of Radioactivity	Closed
PSRB-08	Temporary Operating and Maintenance Procedures	Closed
PSRB-09	TMI Action Items I.C.7 and I.C.8	Closed
PSRB-10	ATWS Procedures	Closed
PSRB-11	Tests of Failed Fuel Monitors (Q640.2)	Confirmatory*
PSRB-12	Automatic Closure of Main Steam Isolation Valves (Q640.3)	Confirmatory*

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* NNECO considers these items to be closed, however, the NRC Reviewer has requested that we categorize these items as confirmatory, until the follow-up FSAR changes appear in a FSAR amendment.

** NNECO considers this item to be confirmatory.

Open Item	Subject	Status
PSRB-13	Conformance to Regulatory Guide 1.52 (Q640.5)	Confirmatory*
PSRB-14	Conformance to Regulatory Guide 1.95 (Q640.4)	Confirmatory*
PSRB-15	Test Abstract Descriptions (Q640.7)	Confirmatory
PSRB-16	Loss of Instrument Air Test (Q640.13)	Confirmatory*
PSRB-17	Solid State Protection System (Q640.17)	Confirmatory*
PSRB-18	Regulatory Guide 1.68, Rev. 2, Appendix A.1.2 and A.5.T (Q640.19)	Open
PSRB-19	Real or Dummy Fuel Assemblies for Vibration Test (Q640.20(2))	Confirmatory*
PSRB-20	NUREG-0694, Item I.G.1 (Q640.22)	Confirmatory*
PSRB-21	Regulatory Guide 1.62, Rev. 2, Appendix A (Q640.26)	Confirmatory*
PSRB-22	Preoperational Tests 76-84 (Q640.27)	Confirmatory*
PSRB-23	Swing Load Test (Q640.28)	Confirmatory*
Q640.15	BTP PSB-1	Confirmatory*
Q640.16	Preoperational Test Number 51 (Diesel Generator)	Confirmatory*

* NNECO considers these items to be closed, however, the NRC Reviewer has requested that we categorize these items as confirmatory, until the follow-up FSAR changes appear in a FSAR amendment.

ATTACHMENT II

Response to DSER PSRB Open Items

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-01 TMI Action Item I.C.1, "Short Term Accident and Procedures Review" (Draft SER Section 13.5.2.2 and 13.4.2.3)

The applicant has committed to implement Supplement 1 to NUREG-0737 and to submit his procedures generation package (PGP) on October 1, 1984, which is 3 months before the start of operator training on the Millstone Unit 3 simulator. The PGP should be sumitted as an FSAR amendment because it provides the basis for developing the plant's EOPs. The staff's review of the PGP must be completed before issuance of the operating license and will be addressed in a supplement to the SER. Until completion of the staff review of the PGP, Task Action Plan Item I.C.1 will remain an open item.

Response:

NNECO considers this to be a confirmatory item. In our April 15, 1983[1] submittal to the NRC, we agreed to provide the NRC with a procedures generation package (PGP) by October 1, 1984 which will include a writer's guide, Millstone Unit No. 3 specific changes from the Westinghouse Owner's Group generic guidelines, and a description of our verification, validation, and training programs. PGPs have been submitted for our Millstone Unit No. 1 [2,3] Millstone Unit No. 2[4], and Haddam Neck Plant[5] which are available for NRC review. The Millstone Unit No. 3 PGP will be similar to these in general content and format. Reviewing these documents would give the NRC a general idea of what to expect for the Millstone Unit No. 3 PGP.

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(1) W. G. Counsil letter to D. G. Eisenhut, A02959, dated April 15, 1983.

(2) W. G. Counsil letter to D. M. Crutchfield, A02959, dated May 13, 1983.

(3) W. G. Counsil letter to D. M. Crutchfield, A03666, dated March 9, 1984.

(4),(5) W. G. Counsil letter to J. R. Miller and D. M. Crutchfield, A02959, dated September 1, 1983.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-02 ANSI/ANS 3.2-1981 (Draft 7), Section 5.3 (Draft SER Section 13.5.2.2)

The applicant should modify Sections 13.5 and 13.5.1.1 of the FSAR to commit to Section 5.3 of ANSI/ANS 3.2-1981 (Draft 7) instead of Section 5.3 of ANSI N18.7-1976/ANS 3.2, or provide justification for this deviation from the SRP. A commitment to conform to Section 5.3 of ANSI/ANS 3.2-1982 would also be acceptable, because Section 5.3 ANSI/ANS 3.2-1982 is the same as Section 5.3 of ANSI/ANS 3.2-1981 (Draft 7). This is an open item.

Response:

Refer to revised FSAR Tables 1.9-1 and 1.9-2.

The following justification is provided for the use of ANSI N18.7-1976/ANS 3.2:

1. Millstone Station has one common set of Administrative Control Procedures to control procedure preparation, approval, format and use. These administrative procedures are used for all three nuclear units at Millstone. The above procedures have been developed in accordance with the Northeast Utilities Topical QA report as approved by the NRC for Millstone Units 1, 2, and 3, and Connecticut Yankee. These documents reference ANSI N18.7-1976/ANS 3.2 as endorsed by Regulatory Guide 1.33.

2. The NRC endorses the 1976 version of ANSI/ANS 3.2, Section 5.3 in Regulatory Guide 1.33, Revision 2 which is in accordance with Regulatory Guide 1.70, Revision 3.

3. Emergency Operating Procedures are developed based on the Westinghouse Owners' Group Emergency Response Guidelines as approved by the NRC. The Emergency Operating Procedures are functional-based as described in the FSAR. Since the requirement for functional-based EOPs is already explicitly addressed and the approach is to comply with NRC approved procedure guidelines, little will be gained by committing to a partial standard which is not addressed within existing regulatory guides.

PSRB-02

TABLE 1.9-1 (Cont)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
11.5 (Rev. 3)	Table 1, item 6 - Fuel storage area ventilation system.	No automatic termination of effluents.	11.5.2.2.9
	Table 2, item 5 - Spent fuel pool treating system.	No automatic termination of effluents.	9.3.2 9.4.2
	Table 2, items 16 and 17 - Steam generator blowdown system.	No automatic termination of effluents.	11.5.2.3.3
471.26 12.2 (Rev. 2)	I.2 - Tabulation of concentrations of airborne radioactive materials	Only normal operation and anticipated operational occurences are addressed	12.2.2 471.26
14.2 (Rev. 2)	II.4 - Categories of reportable occurrences that are repeatedly being experienced at other facilities.	FSAR does not provide categories of occurrences.	14.2
15.4.6.	Entire SRP	FSAR does not address this accident scenario.	15.4.6
15.4.8 (Rev. 1)	III - Stresses should be evaluated to emergency conditions for these accidents.	Westinghouse considers this a faulted condition as stated in ANSI N18.2.	15.4.8
15.6.5 (Rev. 2)	II.3 - TMI Action Plan, II.K.3.30 and II.K.3.31.	No modifications have been made to the small break LOCA model.	15.6.5.3
15.7.3 (Rev. 2)	III.1.a - Radionuclide inventory in failed components.	FSAR analyzed postulated tank failure using 1% fuel defects.	2.4.13.3 15.7.3.2

INSERT A

FSAR Table 1.9-1
Insert A

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
13.5.2	C.2 - ANSI/ANS 3.2-1981, Section 5.3	FSAR uses ANSI N18.7 - 1976/ANS 3.2, Section 5.3	13.5.2

PSRB-02

TABLE 1.9-2 (Cont)

3. SRP 11.5, Table 2, items 16 and 17 require an automatic control feature, which automatically terminates effluents of the steam generator blowdown system.

B. Justification for differences from SRP

1. During fuel handling activities, the fuel building ventilation is processed by the fuel building filtration units. Accident analysis indicates that the filters prevent the release of excessive amounts of radioactive effluent.

2. The spent fuel pool cooling and purification is a closed system, therefore, termination of effluents is unnecessary. Monitoring is accomplished using the reactor plant sampling system radiation monitor, 3SSR-RE08, and area radiation monitors surveying the fuel pool. Safety evaluations described in FSAR Section 9.1.3 show this to be adequate.

3. Monitoring of the steam generator blowdown system is provided by the reactor plant sampling system radiation monitor, 3SSR-RE08. An evaluation of the accident scenario for a steam generator tube rupture shows that such an event would be identified by the air ejector system monitor, 3ARC-RE21 or the main steam line monitors, 3MSS*RE75-78. The main steamline monitors would identify which steam generator is affected and operator action would close valves to prevent release of steam generator blowdown effluents.

SRP 12.2

SRP TITLE: RADIATION SOURCES 471.26

A. Actual differences between FSAR and SRP

SRP 12.2, Paragraph I.2 requires tabulation of the calculated concentrations of radioactive material, by nuclide, expected during normal operation, anticipated operational occurrences, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. FSAR Section 12.2 does not tabulate the calculated concentrations of radioactive material expected during accident conditions.

B. Justification for differences from SRP

During accident conditions, local surveys and measurements will be performed as required and exposures will be limited to the requirements of NUREG-0737.

INSERT B

PSRB-02

FSAR Table 1.9-2
Insert B

SRP 13.5.2

SRP TITLE: OPERATING AND MAINTENANCE PROCEDURES

A. Actual differences between FSAR and SRP

The SRP references Section 5.3 of ANSI/ANS 3.2 - 1981 (Draft 7). The FSAR is written based on Section 5.3 of ANSI N18.7 - 1976/ANS 3.2 which is endorsed in Regulatory Guide 1.33.

B. Justification for differences from SRP.

1. The NRC endorses the 1976 version of ANSI/ANS 3.2, Section 5.3 in Regulatory Guide 1.33, Revision 2 which is in accordance with Regulatory Guide 1.70, Revision 3.

2. Emergency Operating Procedures are developed based on the Westinghouse Owner's Group Emergency Response Guidelines as approved by the NRC. The Emergency Operating Procedures are functional-based as described in the FSAR. Since the requirement for functional-based EOPs is already explicitly addressed, and the approach is to comply with NRC approved procedure guidelines, little will be gained by committing to a partial standard which is not addressed within existing regulatory guides.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-03 Procedures Generation Package (Draft SER Section 13.5.2.2)

The applicant has stated that (a) the PGP will be submitted to NRC 3 months before start of operator training, (b) all proposed operating and maintenance procedures will be completed at least 3 months before fuel loading, and (c) procedures will be available for review in advanced draft form at least 6 months before fuel loading. It is the staff's position that procedures must be completed in sufficient time to ensure operator and appropriate plant staff familiarization. The FSAR should describe how adequate operator and plant staff familiarization will be ensured. This is an open item.

Response:

Refer to FSAR sections 14.2.1.1 and 14.2.9. Millstone Unit No. 3 will approve and utilize its operating procedures to the extent possible to support the testing program. This ensures that the operating staff is knowledgeable about the plant and its procedures.

In addition, all operators undergo an extensive on-the-job systems training program (see FSAR Section 13.2) which requires an understanding of related system procedures.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-04 Alarm Response Procedures (Draft SER Section 13.5.2.2)

The applicant should describe the system to classify or subclassify alarm responses and the methods used by operators to retrieve or refer to alarm response procedures. This is an open item.

Response:

A control room annunciator response procedure is prepared to provide the operator with an immediate reference document for plant alarms. This reference document lists each control room annunciator alarm identifying its procedure reference for operator actions. Annunciator response forms are used to organize alarm lists by main board annunciator grouping, row, and column designation. Each system procedure has a designated alarm response section.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-05 Procedures That Incluoe Immediate
Actions to be Memorized (Draft SER Section 13.5.2.2)

The applicant should identify procedures that include immediate actions that must be memorized by the plant operators. This is an open item.

Response:

Millstone Unit No. 3 is proceeding with the development of its Emergency Operating Procedures based upon the Westinghouse Owner's Group Emergency Response Guidelines, Revision 1, pending the approval of Revision 1 by the NRC. In the event that Revision 1 is not approved by the NRC, NNECO will provide justification for the deviations from Revision 0 of the Westinghouse Guidelines.

The procedures that include immediate actions to be memorized are:

1. Reactor Trip or Safety Injection
2. Loss of all A.C. Power

Immediate actions to be memorized are indicated by asterisks(*) in the Westinghouse Guidelines.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-06 Commitment Concerning Plant Operations (Draft SER Section 13.5.2.2)

Although implied in Section 13.5.2 of the FSAR, a clear commitment is needed that plant operations will be performed in accordance with written and approved procedures. This is an open item.

Response:

Refer to revised FSAR Section 13.5.2.

These procedures are reviewed and improved, if necessary to ensure operability of safety systems prior to taking credit for the system(s) to satisfy Technical Specification requirements.

Special Procedures

Special procedures are prepared as necessary to support infrequent operations. The requirements for review, approval, and changes are the same as station procedures.

13.5.2 Operating and Maintenance Procedures

Operating and maintenance procedures are divided into several categories which are described in the following subsections. Table 13.5-1 lists the appropriate procedures which will initially be prepared or currently exist.

Operating and maintenance procedures preparation is the responsibility of the appropriate department head. When a procedure is written, the department supervisor will forward the procedure for review and approval in accordance with Technical Specifications. Unit specific procedures are approved by the Unit Superintendent and common station procedures are approved by the Station Superintendent.

INSERT A

Independent position verification of safety related components/systems (valves, breakers, and control switches) with no indication in the control room will be performed prior to the return-to-service of the component/system.

All proposed operating and maintenance procedures will be completed at least 3 months prior to fuel loading. Procedures will be available for review in advance draft form at least 6 months prior to fuel loading.

13.5.2.1 Control Room Operating Procedures

13.5.2.1.1 General Operating Procedures

These procedures cover major plant evolutions, and an initial list is included in Table 13.5-1. Step-by-step instructions are provided for the function or task with the appropriate cross reference to system operating procedures for details of specific system operation. Appropriate precautions and limitations are included.

13.5.2.1.2 System Operating Procedures

These procedures provide step-by-step details for system operations with appropriate prerequisites, precautions, limitations, and alarm responses. Each procedure covers the expected modes of operation of the system as well as startup, shutdown, filling and venting, and standby operation as applicable. Table 13.5-1 includes an initial list of system operating procedures which may be modified as experience dictates.

PSRB-06

(FSAR pg. 13.5-4)

Insert A

Plant operations will be performed in accordance with written and approved station and administrative control procedures.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-07 Procedures for Abnormal Release of Radioactivity
(Draft SER Section 13.5.2.2)

A procedure or procedures covering abnormal releases of radioactivity are not evident. This is an open item.

Response:

Abnormal releases of radioactivity are addressed by administrative control procedures with respect to reporting requirements. Emergency Plan Implementing Procedures (EPIP) address required actions such as radiological dose assessment (refer to FSAR section 13.3, draft Emergency Plan, Appendix D, Index of EPIPs). System operating procedures provide step-by-step details for system operations with appropriate prerequisities, precautions, limitations, and alarm responses as described in FSAR sections 13.5.2.1.2 and 13.5.2.2.5.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-08 Temporary Operating and Maintenance Procedures
(Draft SER Section 13.5.2.2)

Section 13.5.2 should be expanded to address temporary operating and maintenance procedures. This is an open item.

Response:

Refer to revised FSAR sections 13.5.1.3, Special Procedures and 13.5.2.2.10.

These procedures are reviewed and improved, if necessary to ensure operability of safety systems prior to taking credit for the system(s) to satisfy Technical Specification requirements.

Special Procedures

INSERT A

~~Special procedures are prepared as necessary to support infrequent operations. The requirements for review, approval, and changes are the same as station procedures.~~

13.5.2 Operating and Maintenance Procedures

Operating and maintenance procedures are divided into several categories which are described in the following subsections. Table 13.5-1 lists the appropriate procedures which will initially be prepared or currently exist.

Operating and maintenance procedures preparation is the responsibility of the appropriate department head. When a procedure is written, the department supervisor will forward the procedure for review and approval in accordance with Technical Specifications. Unit specific procedures are approved by the Unit Superintendent and common station procedures are approved by the Station Superintendent.

Independent position verification of safety related components/systems (valves, breakers, and control switches) with no indication in the control room will be performed prior to the return-to-service of the component/system.

All proposed operating and maintenance procedures will be completed at least 3 months prior to fuel loading. Procedures will be available for review in advance draft form at least 6 months prior to fuel loading.

13.5.2.1 Control Room Operating Procedures

13.5.2.1.1 General Operating Procedures

These procedures cover major plant evolutions, and an initial list is included in Table 13.5-1. Step-by-step instructions are provided for the function or task with the appropriate cross reference to system operating procedures for details of specific system operation. Appropriate precautions and limitations are included.

13.5.2.1.2 System Operating Procedures

These procedures provide step-by-step details for system operations with appropriate prerequisites, precautions, limitations, and alarm responses. Each procedure covers the expected modes of operation of the system as well as startup, shutdown, filling and venting, and standby operation as applicable. Table 13.5-1 includes an initial list of system operating procedures which may be modified as experience dictates.

P5RB-08

(FSAR pg. 13.5-4)

Insert A

Special procedures are prepared as necessary to support infrequently performed evolutions which will not be included in the permanent list of station procedures. A special procedure can be written for any type of station procedure (i.e. maintenance, operating). The form of a special procedure will be the same as the applicable type of station procedure. All requirements for review, approval, revisions, and changes are the same as for permanent station procedures.

(FSAR pg. 13.5-6)

Insert B

13.5.2.2.10 Special Procedures

This topic is covered by administrative procedures. (Refer to FSAR Section 13.5.1.3, Special Procedures)

PSRB-08

13.5.2.2.3 Instrument Maintenance Instructions

Instrument maintenance instructions are prepared for the performance of periodic calibration, testing, and channel checking of safety related plant instrumentation and all instruments used to satisfy technical specification requirements. These instructions will ensure measurement accuracies adequate to maintain plant safety parameters within operational and safety limits. In addition, instrument maintenance instructions outline the periodic calibration and accuracy requirements of test equipment necessary to support the calibration of safety related instrumentation.

13.5.2.2.4 Chemistry Procedures

Chemistry procedures are prepared covering the routine analysis and sampling methods to ensure compliance with plant chemistry and discharge limits.

13.5.2.2.5 Radioactive Waste Procedures

Procedures for operation of radwaste systems are included in system operating procedures.

13.5.2.2.6 Plant Security Instructions

This topic is discussed in Section 13.6.

13.5.2.2.7 Material Control Procedures

This topic is covered by administrative procedures in Section 13.5.1.3.

13.5.2.2.8 Maintenance and Modification Procedures

Maintenance procedures are prepared to cover safety related work which requires a specific technique or sequence not normally part of an individual's routine skill.

The procedures support the requirements and programs of Section 13.5.1.3 which covers administrative control of maintenance and modification.

13.5.2.2.9 Fire Protection Procedures

The Fire Protection Program is described in Section 9.5.1. Procedures for fire protection are included under System Operating Procedures in Table 13.5-1.

INSERT B →

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-09 TMI Action Items I.C.7 (NSSS Vendor Review of Procedures) and I.C.8 (Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants) in FSAR Table 1.10-1 (Draft SER Section 13.5.2.2)

FSAR Table 1.10-1 should be revised to provide a brief explanation of how Task Action Plan Items I.C.7 and I.C.8 have been resolved, as described in Section 13.5.2.3. Suitable cross-reference between Table 1.10-1 and Section 13.5.2 of the FSAR should be provided. This is an open item.

Response:

Refer to revised FSAR Table 1.10-1 and Section 13.5.2.1.4.

The Millstone Unit No. 3 Emergency Operating Procedures are being prepared based upon Revision 1 to the Westinghouse Owners' Group Emergency Response Guidelines which have not been approved by the NRC at this time. In the event that Revision 1 is not approved by the NRC, NNECO will provide justification for the deviations from Revision 0 of the Westinghouse Guidelines.

Item and Title	Position	FSAR Reference
I.C.5 Procedures for Feedback of Operating Experience	MNPS-3 meets the requirements of this item.	13.1.1
I.C.6 Procedures for Verification of Correct Performance of Operating Activities	MNPS-3 meets the requirements of this item.	13.5.1.3 13.5.2
I.C.7 NSSS Vendor Review of Procedures	~~MNPS-3 meets the requirements of this item.~~ ← INSERT A	13.5.2.1.4 ~~14.2.3.2~~
I.C.8 Pilot Monitoring of Selected Emergency Procedures for NTOLS	~~Northeast Nuclear Energy Company (NNECo.) remains available to correct any deficiencies found in its emergency procedures if the NRC opts to conduct a pilot monitoring program. The PMP may not be necessary considering our response to I.C.1.~~ ← INSERT B	← 13.5.2.1.4
I.D.1 Control Room Design Review	A control room design review will be performed for MNPS-3 to meet the requirements of this item.	18.0
I.D.2 Plant Safety Parameter Display Console	A position has not been taken for MNPS-3 due to the lack of clear definition of NRC requirements for this item.	*
I.G.1 Training during Low-Power Testing	MNPS-3 will address this item following the issuance of NRC finalized criteria for this item.	*
II.B.1 Reactor Coolant System Vents	Safety grade reactor vessel and pressurizer venting capability is provided in the MNPS-3 design.	5.4.15 7.5
II.B.2 Plant Shielding	The MNPS-3 plant shielding design is outlined in Chapter 12.	3.11 12.3.2
II.B.3 Post-Accident Sampling	MNPS-3 has a post-accident sampling system which meets the requirements of this item.	9.3.2.6
II.B.4 Training for Mitigating Core Damage	MNPS-3 will develop and implement a training program utilizing the INPO guidelines for "Recognizing and Mitigating the Consequences of Severe Core Damage" as the basis for the program.	13.2.1 13.2.2

PSRB-09

(FSAR Table 1.10-1)

Insert A

Commitment to implement emergency operating procedures based on NRC-approved Westinghouse Emergency Response Guidelines eliminates the requirements for additional NSSS vendor review of emergency operating procedures.

(FSAR Table 1.10-1)

Insert B

Commitment to implement emergency operating procedures based on NRC approved Westinghouse Emergency Response Guidelines eliminates the requirement for pilot monitoring of selected emergency Procedures for near-term operating license applicants.

PSRB-09

13.5.2.1.3 Abnormal Operating Procedures

Operating procedures are prepared for abnormal operation of the unit. Abnormal operation is a condition that could degrade into an emergency or could violate Technical Specifications if proper action were not taken. These procedures identify the symptoms of the abnormal condition, automatic actions that may occur, and the appropriate immediate and subsequent operator actions. Table 13.5-1 includes a list of abnormal operating instructions.

13.5.2.1.4 Emergency Operating Procedures

Emergency operating procedures are prepared for conditions which might possibly lead to injury of plant personnel or the public if the release of radioactivity in excess of established limits occurs. These procedures include symptoms of the emergency conditions, automatic actions that may or should occur, and immediate and subsequent operator actions. All immediate actions are required to be memorized by the operator since the primary responsibility for detection of an emergency and initiation of corrective action rests upon the operator. ~~Emergency operating procedures will be prepared using the concept and general program prepared by Westinghouse Owners Group when accepted by the NRC~~. Table 13.5-1 includes an initial list of emergency operating procedures which may be modified as experience dictates. INSERT C

13.5.2.2 Station Services Procedures

Station services procedures are written by the chemistry, health physics, security, quality assurance, production test, building services, training, stores, nuclear records, computer operations, station services engineering and any other station services group. These procedures control the specific activities of these departments in support of unit or station operation (may be common site or unit specific). Station calibration procedures written by the maintenance or instrument departments are also station services procedures.

Station services procedures are approved as outlined in Section 13.5.1.2. These procedures support Section 12.5 requirements. Station services procedures will be updated to reflect Millstone 3 at least 6 months prior to fuel load. These procedures meet the requirements of Regulatory Guide 1.33, are implemented as required on two operating units, and are updated periodically.

13.5.2.2.1 Health Physics Procedures

Health physics procedures support Section 12.5 and 10CFR20 requirements.

13.5.2.2.2 Emergency Preparedness Procedures

Emergency preparedness procedures are covered under Section 13.3.

(FSAR pg. 13.5-5)

Insert C

Emergency Operating Procedures will be prepared based upon Revision 1 to the Westinghouse Owner's Group Emergency Response Guidelines, pending its approval by the NRC.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

<u>PSRB-10 ATWS Procedures (Draft SER Section 13.5.2.2)</u>

As discussed in Section 15.8, "Anticipated Transients Without Scram," the applicant should modify Section 15.8 (or provide sufficient cross-reference) to reflect the applicant's commitment to develop procedures for anticipated transients without scram based on the NRC-approved Westinghouse Emergency Response Guidelines. This is an open item.

Response:

Refer to revised FSAR Section 15.8.

PSRB-10

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

A discussion of anticipated transients without SCRAM (ATWS) is presented in WCAP-8330, 1974. The information provided in WCAP-8330, 1974 is applicable to Millstone 3.

INSERT A

15.8.1 Reference for Section 15.8

WCAP-8330, 1974. Westinghouse Anticipated Transients Without Trip Analysis.

PSRB-10

(FSAR pg. 15.8-1)

Insert A

Northeast Nuclear Energy Company (NNECO) has committed to develop procedures for anticipated transients without scram based on NRC-approved Westinghouse Owner's Group Emergency Response Guidelines (refer to section 13.5.2).

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-11 Tests of Failed Fuel Monitors - Question 640.2 (Draft SER Section 14.2.7)

Question Q640.2:

FSAR Subsection 14.2.7.7, exception 1 to Regulatory Guide 1.68 (Initial Test Programs for Water-Cooled Nuclear Power Plants), Appendix A, Section 5q states that Millstone 3 does not have a failed fuel detection system. FSAR Subsection 11.5.2.3.7 describes a Failed Fuel Monitor used to continuously monitor the reactor coolant system for failed fuel. Delete the exception in FSAR Subsection 14.2.7.7 and add an appropriate test description to FSAR Subsection 14.2.12.

Response:

Refer to revised FSAR Table 14.2-1 for a description of this test.

This test remains as an exception because it shall be tested for proper operation prior to power ascension (instead of at power as stated in Regulatory Guide 1.68). Refer to revised FSAR Section 14.2.7.7.

Additional Concerns Identified in Draft SER:

Tests of the failed fuel monitor should be conducted during startup testing (at 25% and 100% power) in accordance with Regulatory Guide 1.68, Appendix A, Section 5.g.

Response:

Refer to revised FSAR Table 14.2-2, Startup Test 29 for a description of this test.

Testing of the failed fuel detection system will be conducted at 25 and 100 percent power in accordance with Startup Test 29.

Refer to revised FSAR Section 14.2.7.7 for deletion of exception to Regulatory Guide 1.68, Appendix A, Section 5.g.

PSRB-11

TABLE 14.2-2 (Cont)

29. STARTUP TEST - PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM

Prerequisites for Testing

INSERT A ~~The plant is at approximately 50 percent power.~~ DELETE | 640.28

Test Objective and Summary

This test will verify the operability of process and effluent radiation monitors. INSERT B Samples will be taken at monitored points and analyzed. The results of the analysis will be compared to the readings of the monitor.

Acceptance Criteria

The process and effluent monitor responses are consistent with sample results.

INSERT C

FSAR Table 14.2-2
Startup Test No. 29

Insert A

The plant is at approximately 50 percent power for testing of process and effluent radiation monitors.

The plant is at approximately 25 and 100 percent power for testing of the failed fuel detection system.

Insert B

Testing will include the failed fuel detection system.

Insert C

The failed fuel detection system response is consistent with sample results.

14.2.7.4 Regulatory Guide 1.37, Revision 0 - Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants 640.1

The Millstone 3 initial test program will conform to the intent of Regulatory Guide 1.37.

14.2.7.5 Regulatory Guide 1.41, Revision 0 - Preoperational Testing of Redundant Onsite Electrical Power Systems to Verify Proper Load Group Assignments 640.1

For position on Regulatory Guide 1.41, see FSAR Section 1.8.

14.2.7.6 Regulatory Guide 1.52, Revision 2 - Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

For position on Regulatory Guide 1.52, see FSAR Section 1.8.

14.2.7.7 Regulatory Guide 1.68, Revision 2 - Initial Test Programs for Water-Cooled Nuclear Power Plants

The Millstone 3 initial test program will conform to Regulatory Guide 1.68, except as specified in this section:

* → ~~1. The failed fuel detection system (Appendix A, Section 5) shall be tested during a preoperational test.~~ 640.2 Delete

1. ~~2.~~ During power escalation, testing will be conducted at the 30-percent power level instead of at the 25-percent power level. Westinghouse supplied plants have generic data for the 30-percent level which they do not have at the 25-percent level (Section C.8; Appendix A, Section 5).

Note: there is a change to this section addressed in SER Question PSRB 23 that inserts a new item '2'

3 ~~3.~~ The MSIV closure test will be performed at less than 20-percent power to demonstrate the proper dynamic response of the plant and to verify proper integrated operation of plant equipment. Plant response to a full power trip will be verified by the generator trip at 100-percent power. Closure of the MSIVs at 100-percent power would not provide any additional information significant enough to warrant subjecting the plant to such a severe thermal transient (Appendix A, Section 5.m.m).

4 ~~4.~~ The loss of feedwater heaters test will not be performed. Since plant response to load swings and large load reductions is demonstrated in other tests, there is no need to subject the plant to this additional transient (Appendix A, Section 5.k.k).

5 ~~5.~~ Millstone 3 does not have a partial scram feature (Appendix A, Section 5.j).

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-12 Automatic Closure of Main Steam Isolation Valves - Question 640.3 (Draft SER Section 14.2.7)

Question Q640.3:

Your exception to testing the automatic closure of all main steam isolation valves (FSAR Subsection 14.2.7.7(3)) at 100 percent power does not supply adequate technical justification for conducting the test at a low power level. Provide adequate technical justification or revise the FSAR to indicate that the test will be conducted at full power.

Response:

FSAR Sections 15.2.3.1 and 15.2.4 indicate that the dynamic response of the plant to a MSIV closure is bounded by the response of the plant to the turbine trip event because closure time for turbine stop valves is faster than MSIV's. FSAR Section 15.2.3.2 describes the LOFTRAN code used to model the turbine trip event. This program does not take credit for steam dump. Plant response to a turbine trip from 100% power will be demonstrated per Startup Test 39 (FSAR Table 14.2-2). The combination of an MSIV trip at 20% power, the LOFTRAN model, and a 100% power turbine trip (with normal steam dump operation) should provide adequate verification of plant response to this transient.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-13 Conformance to Regulatory Guide 1.52,
Paragraphs C.2.1, C.3.1, and C.3.p. - Question 640.5 (Draft SER Section 14.2.7)

Question Q640.5:

Certain exceptions to Regulatory Guide 1.140 (Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants) listed in FSAR Section 1.8 need to be deleted or modified as described below to be acceptable.

1. Modify exception to Paragraph C.2.f to delineate how the ductwork leak tests performed using the methods of the Associated Air Balance Council differ from the requirements given in Section 6 of ANSI N510:1975, and provide technical justification for any testing that does not address those differences.

2. Modify exception 1 to Paragraph C.3.i to provide assurance that the data provided in the certified fan performance curves will most closely represent the manner in which the fan will be installed in the appropriate system.

3. Modify exception 2 to Paragraph C.3.i to either reference the displacement criteria that will be used, or agree to meet the criteria given in the 1980 revision to ANSI N509 Section 5.7.3.

4. The exception to Paragraph C.3.1 states that an exception is taken to the following: "Class B leakage rates shall be determined for one damper of each type instead of every damper." If the intent is to not test each damper's leak rate, expanded technical justification will be required and the exception rewritten to clarify what is actually intended.

5. Modify FSAR Subsection 14.2.7.15 to either include the exceptions listed in FSAR Section 1.8 or to reference FSAR Section 1.8.

Response:

Refer to revised FSAR Section 14.2.7.15 and revised Table 1.8-1 for the response to this question.

Additional Concerns Identified in Draft SER:

The exceptions noted in this item should also be addressed in the statement of conformance to RG 1.52, Paragraphs C.2.1, C.3.1, and C.3.p.

Response:

Refer to revised FSAR Table 1.8-1.

P5RB-13

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		filter components on a cell by cell basis. Demisters, heaters, fans and casings will be decontaminated by wash down process; wash down liquid will drain to an aerated drain system. Paragraph C.2.1 Housing leak tests are performed in accordance with the provisions of Section 6 of ANSI N510-1975 as recommended in this paragraph. However, ductwork tests are performed using acceptable methods of the Associated Air Balance Council. INSERT A Paragraph C.3.d (Clarification) All HEPA filters are shipped to an NRC Quality Assurance Station for testing. However, if data confirm that HEPA filters are damaged by the additional transportation, and/or the handling at the NRC facility, the decision to send all HEPA filters for additional testing will be reconsidered. If HEPA filters are not sent to the NRC Quality Assurance Station, sufficient additional testing remains to ensure HEPA filter reliability. The HEPA filter cell testing is conducted initially at the manufacturer's facilities and again after installation at the plant site. All HEPA filters furnished are equipped with face guards in accordance with MIL-F-51068. When installed in the filter housing, the HEPA filters and housing are inspected for defects and tested for leak tightness in accordance with ANSI N510-1975. Paragraph C.3.e (Clarification) Filter and adsorber mounting frames are constructed and designed in accordance with the recommendations of Section 4.3 of ERDA 76-21, except for the frame tolerance guidelines in Table 4.2. The tolerances selected for HEPA and adsorber mountings are sufficient to satisfy the bank leak test criteria of Paragraphs C.5.c and C.5.d of Regu-	3

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		latory Guide 1.52, Rev. 2. Paragraph C.3.g Millstone 3 is in accordance with ANSI N509, except access to the control building filter units is not provided with hinged doors or inspection windows. Access is via 20-inch by 40-inch bolted panels. Other units are provided with hinged doors or bolted panels with inspection windows. There is no internal lighting. Paragraph C.3.h Exception is taken to the recommendations of Section 4.5.8 of ERDA 76-21 relative to drain sizes and arrangement. Normally closed manual valves, instead of water seals and traps, will be provided to control the discharge of the fire sprinkler flow. Sprinkler flow will be a timed discharge, and the water will be contained within the housing until it is removed to the liquid radwaste system at a controlled rate. Paragraph C.3.i The dwell time for the minimum 2 inches of the carbon adsorber unit is 0.25 sec. For bed depths greater than 2 inches, where the dwell time is less than 0.2 sec per 2 inches of total bed depth, experimental verification of filter assembly will be provided. Paragraph C.3.k When conservative calculations show that the maximum decay heat generation from collected radioiodines is insufficient to raise the carbon bed temperature above 250°F with no system overflow, small capacity ESF atmosphere cleanup systems may be designed without an air bleed cooling mechanism. Exception is taken to the requirement of any cooling mechanism satisfying single-failure criteria because a backup mechanism is provided.	

TABLE 1.8-1 (Cont)

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R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		In addition, exception is taken to provide humdity control for the decay heat removal system cooling air flow which uses room air of less than 70 percent relative humidity.	
		Paragraph C.3.l	
		System resistances will be determined in accordance with Section 5.7.1 of ANSI N509-1976 except that fan inlet and outlet losses will not be calculated in accordance with AMCA 201. Fan blast area data necessary to calculate inlet and outlet losses, per AMCA 201, are the responsibility of fan manufacturers, and are not available from them.	
		~~Exception is taken to Section 5.7.2 of ANSI N509-1976: copies of fan ratings or test reports are not necessary when certified fan performance curves are furnished.~~	Delete
		~~Exception is taken to balancing techniques defined in Section 5.7.3 of ANSI N509-1975. Displacement criteria following normal industry practice will be used when maximum vibration velocity method imposes unrealistic requirements at certain operating speeds.~~	
		Documentation will not be furnished in accordance with Section 5.7.5 where AMCA certification ratings are submitted.	
		Paragraph C.3.n	
		Exception is taken to Section 5.10.3.5 of ANSI N509-1976: ductwork, as a structure, will have a resonant frequency above 25 Hz, but this may not be true for the unsupported plate or sheet sections. The design provides for specification of the resonant frequency range of the support hangers. Specifying the resonant frequency of the unsupported plate or sheet has no meaning in the design.	
		Paragraph C.3.p	
		Exception is taken to the provisions in Section 5.9 of ANSI N509-1976 of designing dampers to ANSI B31.1 and to using butter-	

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TABLE 1.8-1 (Cont)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		fly valves. Class B dampers may be designed and tested to meet the verification of strength and leaktightness necessary for use in a containment air stream. (Note: This exception does not pertain to containment penetrations.) In addition, exception is taken to the following: Class B leakage rates shall be determined for one damper of each type instead of every damper. INSERT B → Paragraph C.4.a Exception is taken to full compliance with Section 2.3.8 of ERDA 76-21, i.e., the plant does not use any communications system, floor drains are as noted in Paragraph C.3.h above, decontamination areas and showers are not "nearby," filters are not used at duct inlets, and duct inspection hatches are not provided. Paragraph C.4.b Partial compliance, with a minimum spacing between filter frame of 2 ft-6 in. instead of a minimum of 3 feet. This is deemed adequate since replacement of filter elements would be minimal due to system function, use, and location. Paragraph C.4.d (Clarification) ESF atmosphere cleanup systems are run a minimum of 10 hours per month. However, if the field data confirms that it is unnecessary to run the trains 10 hours per month to reduce the amount of moisture present on the filters, this decision will be reconsidered.	3
1.53*	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 0, June 1973)	Comply, with the following clarifications: 1. Regulatory Position C.1 Due to the trial-use status of the source document, IEEE 379-1972, departure from certain provisions may occur. The phrase "any and all combinations of	3.1.1

FSAR Table 1.8-1

Insert A

Ductwork leak testing is performed using the direct measurement method. Measurement apparatus included a blower, calibrated orifice, and manometer. Since the only ductwork utilized on ESF air cleaning systems is classified as Leakage Class II, this method provides equivalent accuracy and ANSI N510 methods.

Insert B

Damper leakage will not impact on the air cleaning effectiveness of ESF systems.

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Open Items

Procedures and Systems Review Branch

PSRB-14 Conformance to Regulatory Guide 1.95, Position C.5 - Question 640.4 (Draft SER Section 14.2.7)

Question Q640.4:

If you intend to conduct an initial control room gross leakage rate test as part of the preoperational test program, delete FSAR Paragraph 14.2.7.11. FSAR Section 14.2.7 should be limited to discussion of Regulatory Guide exceptions relating to the initial test program.

Response:

Refer to revised FSAR Section 14.2.7.11 for the response to this question.

Additional Concerns Identified in Draft SER:

The statement of conformance to RG 1.95, Position C.5, should be included in FSAR Subsection 14.2.7 and should only discuss those exceptions relating to the initial test program, if any.

Response:

Refer to revised FSAR Section 14.2.7.12.

INSERT 14.2.7.12

conditions will be verified by analysis based on as-built HPSI pump and system head-capacity curves; however, the operability of the check valves will be demonstrated by testing. Power system response to a safety injection signal will be verified during other testing (Section C.1.a.(2)).

640.4 14.2.7.~~12~~ 13 Regulatory Guide 1.108, Revision 1 - Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants

For position on Regulatory Guide 1.108, see FSAR Section 1.8.

640.4 14.2.7.~~13~~ 14 Regulatory Guide 1.116, Revision 0-R - Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems

The Millstone 3 initial test program will conform to the intent of Regulatory Guide 1.116.

640.4 14.2.7.~~14~~ 15 Regulatory Guide 1.128, Revision 1 - Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants

The Millstone 3 initial test program will conform to the intent of Regulatory Guide 1.128.

640.4 14.2.7.~~15~~ 16 Regulatory Guide 1.140, Revision 1 - Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units for Light-Water-Cooled Nuclear Power Plants

640.5 For position on Regulatory Guide 1.140, see FSAR Section 1.8.

14.2.8 Utilization of Reactor Operating and Testing Experience in Development of Test Program

The Millstone 3 test program will utilize information gained from operating and testing experience at similar nuclear plants to provide guidance in developing test procedures and schedules and to alert personnel to potential problem areas.

The Millstone 3 Superintendent will designate individuals on the plant staff to review pertinent industry literature, such as NRC IE bulletins, circulars and information letters, vendor information notices and applicable event reports from other facilities. Commitments resulting from this review will be tracked to ensure incorporation into plant procedures or design.

(FSAR pg. 14.2-20)

INSERT

14.2.7.12 Regulatory Guide 1.95, Revision 1 - Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release.

For position on Regulatory Guide 1.95, see FSAR Section 1.8.

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Open Items

Procedures and Systems Review Branch

PSRB-15 Test Abstract Descriptions - Question 640.7 (Draft SER Section 14.2.12)

Question Q640.7:

Regulatory Guide 1.70 Paragraph 14.2.12 states that test descriptions should include a "summary of acceptance criteria." To comply, you should include, for all tests listed below, acceptance criteria or a discussion of the sources for the acceptance criteria to be used when test procedures are prepared. This information is necessary for the NRC inspectors who review test procedures and evaluate test results. The test description should provide "traceability" to acceptance criteria sources such as: specific FSAR Subsections, Technical Specifications, topical reports, vendor-furnished test specifications, and/or accident analysis assumptions.

1. Preoperational Test Numbers 1-11, 13-14, 16-29, 31-60, 62-68, 71, and 73-75.

2. Startup Test Numbers 1-2, 7-8, 11-13, 17-19, 22-24, 26, 28-35, and 38.

Response:

The sources for acceptance criteria will be provided in the individual test procedures when they are prepared. Traceability to specific FSAR subsections, Technical Specifications, topical reports, vendor-furnished test specifications, and/or accident analysis assumptions will be provided.

Approved test procedures for satisfying FSAR test commitments will be made available to the NRC staff personnel from the office of Inspection and Enforcement approximately 60 days prior to their intended use as required by Regulatory Guide 1.68 or approximately 60 days prior to fuel load, whichever is sooner.

Additional Concerns Identified in Draft SER:

Test abstract descriptions should be expanded to indicate the sources of acceptance criteria.

Response:

Refer to FSAR Table 14.2-3.

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TABLE 14.2-3

PREOPERATIONAL/ACCEPTANCE/STARTUP TESTS

ACCEPTANCE CRITERIA SOURCES

Test Number	Title	Sources
P1	RCS Cold Hydro	FSAR Table 5.4-15
P2	Control Rod Drive	FSAR 4.6.3; Vendor Specification 001 (Westinghouse)
P3	Fuel Transfer	Vendor Specification 001 (Westinghouse)
P4	Polar Crane	Vendor Specification 014 (Harnischfeger)
P5	Volume Control (Charging and Letdown)	Westinghouse (W) NSSS SU Manual (NEU-SU-2.2.3); W Precautions, Limitations, and Setpoints (PLS) Vendor Specifications 001 (W) and 459 (Combustion Engineering)
P6	Volume Control (Boric Acid)	NEU-SU-2.2.3; IEB 81-02
P7	Volume Control (BTRS)	NEU-SU-2.2.3; W PLS
P8	Fuel Pool Cooling	FSAR 9.1.3
P9	Containment Recirculation	FSAR 6.2.2.3
P10	Residual Heat Removal	FSAR 5.4.7, 6.3
P11	LP Safety Injection	FSAR 6.3; R.G. 1.79, 1.108
P12	HP Safety Injection	FSAR 6.3; R.G. 1.79, 1.108
P13	Quench Spray	FSAR 6.2.2; R.G. 1.1, 1.26, 1.29, 1.97
P14	Reactor Plant Sampling	FSAR 9.3.2, 9.3.4
P15	Containment Local Leak Rate Testing	FSAR 6.2.4, 6.2.6; Table 6.2-4; 10CFR50 Appendix J
P16	Containment Ventilation	FSAR 6.2.5.4, 9.4.7, 9.5.10.4
P17	Auxiliary Bldg. Ventilation	FSAR 9.4.3.1
P18	Waste Building Vent	FSAR 9.4.2, 9.4.9.1
P19	Fuel Building HVAC	FSAR 9.4.2, 9.4.9.1
P20	ESF Building HVAC	FSAR 9.4.5
P21	Control Building HVAC	FSAR 6.4.3, 6.4.5, 9.4.1; R.G. 1.95

Test Number	Title	Sources
P22	Screen House HVAC	FSAR 9.4.8.1.1
P23	EGE Vent	FSAR 9.4.6.1.3, 9.4.6.5
P24	Supplementary Leak Detection and Release	FSAR 6.2.3.3
P25	Main Steam	FSAR 10.3.3; NEU-SU-2.8.3, 2.8.5
P26	Steam Dump Control	NEU-SU-2.8.3, 2.8.5
P27	Steam Generator Blowdown	FSAR 10.4.8
P28	Main Feedwater	FSAR 10.4.7; Vendor Specification 021 (General Electric)
P29	Steam Generator Water Level Control	FSAR 10.4.7.2
P30	Auxiliary Feedwater	FSAR 10.4.9
P31	Service Water	FSAR Table 9.2-1
P32	Reactor Plant Component Cooling	FSAR Table 9.2-5
P33	Reactor Plant Chilled Water	FSAR 9.2.2.2.1, Table 9.2-7
P34	Charging Pump Cooling	FSAR 9.2.2.4.2, Table 9.2-10
P35	SI Pump Cooling	FSAR 9.2.2.5.2, Table 9.2-12
P36	NST Cooling	FSAR 9.2.2.3.2
P37	Reactor Plant Gaseous Drains	FSAR 9.3.3; R.G. 1.70
P38	Instrument Air and Containment Instrument Air	FSAR 9.3.1.1.4.1; R.G. 1.68.3
P39	Rad. Liquid Waste	FSAR 9.3.3, 11.2, 11.5; R.G. 1.70
P40	Boron Recovery	FSAR 9.3.5.1
P41	Rad. Gaseous Waste	FSAR 11.3
P42	Rad. Solid Waste	FSAR 11.4
P43	Steam Generator Chemical Feed	FSAR 10.4.7; Vendor Specification 053 (Yarway)
P44	Fire Protection - Water	FSAR 9.5.1

Test Number	Title	Sources
P45	Fire Protection - CO_2 and HALON	FSAR 9.5.1
P46	4KV Normal and Emergency Distribution	FSAR 8.3.1.1, Table 8.1-2
A/P47	480V Normal and Emergency Distribution	FSAR 8.3.1.1
P48	120 VAC Instrumentation Non-Vital Distribution	Vendor Specification E261 (Solidstate Controls)
P49	120 VAC Instrumentation Vital Distribution	Vendor Specification E622 (Elgar)
P50	125 VDC Distribution	FSAR 8.3.2.1, Table 8.3-5; Vendor Specification E262 (GE)
P51	Diesel Generator	FSAR 8.1.7, 9.5.6.1; R.G. 1.79, 1.108
P52	Diesel Generator Fuel	FSAR 9.5.4
P53	RSST	FSAR 8.3.1.1, Table 8.1-2
P54	Communications	FSAR 9.5.2; IEB 79-18
P55	Nuclear Instruments	Westinghouse PLS
P56	Incore Nuclear Instrumentation	Vendor Specification 001 (W)
P57	Process and Area Rad. Monitoring	FSAR Tables 11.5-1,2; 12.3.4
P58	ESF Actuation (Diesel Sequencer)	FSAR 8.3
P59	Reactor Trip (Solid State Protection)	FSAR Table 15.0-4; W PLS
P60	Process Protection and Control Instrumentation Racks	Vendor Specification 001 (W)
P61	Protection/Safeguards System Response Time	FSAR Chapter 15
P62	DRPI	Vendor Specification 001 (W)
P63	Loose Parts Monitor	FSAR 4.4.6.4
P64	Seismic Monitor	Vendor Specification 319 (Terra Technology)

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Test Number	Title	Sources
P65	Emergency Lighting	FSAR 9.5.3
P66	ESF Integrated Test w/o Loss of Normal Power	FSAR 7.3; R.G. 1.79
P67	ESF Test with Loss of Normal Power	FSAR 8.3.1.1.2.4; R.G. 1.108
P68	Leak Detection	Technical Specifications; R.G. 1.79, 1.10
P69	Containment Isolation	FSAR 6.2.4
P70	Containment Integrated Leak Rate	FSAR 6.2.6; ANSI N45.4; 10CFR50 Appendix J
P71	Integrated Precore Hot Functional Testing	R.G. 1.68, 1.79
P72	Reactor Coolant and Associated Systems Expansion and Restraint	*
P73	Reactor Coolant and Selected Systems Piping Vibration	NETM-50
P74	Thermal Expansion of Piping and Components of Secondary Systems	NETM-50
P75	Control System Test for Turbine Runback	NEU-SU-2.74, 3.13
P76	RCIV	Vendor Specification 001 (W)
P77	Condensate and Condensate Storage	*
A78	Turbine Plant Sampling	FSAR 10.4.7.4
A79	Turbine Plant Component Cooling	*
A/P80	Heat Tracing	IEN 79-24
P81	RWST Cooling	*
P82	Reactor Vessel Head Vent	*
A83	Condenser Air Removal	FSAR 10.4.2.1
A84	Leak Test of SFP Gates and Transfer Tube	*

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Test Number	Title	Sources
S1	Initial Core Load	W Nuclear Design Report
S2	Post-Core Hot Functional	See individual tests
S3	CRDM	NEU-SU-2.5.1; R.G. 1.68
S4	RPI	NEU-SU-2.5.4
S5	Rod Drop Times	NEU-SU-2.5.3; Technical Specifications; R.G. 1.68
S6	Rod Control System	NEU-SU-2.5.2
S7	Pressurizer Spray and Heater Capacity	NEU-SU-2.1.5
S8	RTD Bypass Loop Flow	NEU-SU-2.1.9
S9	Reactor Coolant System Flow	FSAR Table 4.4-1, Technical Specifications
S10	Reactor Coolant Flow Coastdown	FSAR 10.3
S11	Movable Incore Detectors	NEU-SU-2.9.3
S12	Operational Alignment of Process Temperature Inst.	NEU-SU-2.9.6; R.G. 1.68
S13	Computer Programs	Baseline data aquisition
S14	Vibration and Loose Parts Monitoring	*
S15	Water Chemistry Control	*
S16	Radiation Survey	FSAR 12.3.1
S17	Initial Criticality	Technical Specifications
S18	Low Power Physics Test	W Nuclear Design Report
S19	Boron Reactivity Worth	W Nuclear Design Report
S20	Pseudo Rod Ejection	FSAR 15.4
S21	Natural Circulation	FSAR 14.2.12.2, 15.2.6; R.G. 1.68
S22	Power Ascension	R.G. 1.68
S23	Dynamic Automatic Steam Dump Control	NEU-SU-2.8.5

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Test Number	Title	Sources
S24	Auto Steam Generator Level Control	NEU-SU-2.8.2
S25	Shutdown from Outside Control Room	R.G. 1.68.2
S26	Station Blackout	R.G. 1.68
S27	MSIV Closure	FSAR 10.3.3
S28	Operational Alignment of Nuclear Instrumentation	W PLS
S29	Process and Effluent Monitoring	FSAR 12.3.4, Table 11.5-1,2
S30	Core Performance	Technical Specifications; R.G. 1.68
S31	Power Coefficient	NEU-SU-2.9.11
S32	Axial Flux Difference Instrumentation Calibration	Technical Specifications
S33	Ventilation Systems Operability	FSAR 9.4, Table 9.4-1; R.G. 1.68
S34	Turbine Generator and Feedwater Turbine Operability	Baseline data acquisition
S35	Calibration of Steam and Feedwater Flow Inst.	NEU-SU-2.9.4
S36	Auto Reactor Control	NEU-SU-2.8.1; W PLS
S37	Load Swing	NEU-SU-3.4.7, 3.4.8
S38	Auxiliary Coolant Systems Performance	FSAR 9.2.2, 9.2.7
S39	Unit Trip From 100% Power	FSAR 15.2.3; R.G. 1.68
S40	Warranty Run	NEU-SU-3.5.1
S41	Secondary Plant Performance	*
S42	Containment Penetration Temperature Monitoring	*

Note: This listing is only a partial summary of the acceptance criteria sources used to prepare the indicated test procedures. A detailed listing will be available in each test.

* The sources of acceptance criteria for these tests can be found in the test abstract descriptions.

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Open Items

Procedures and Systems Review Branch

PSRB-16 Loss of Instrument Air Test -
Question 640.13 (Draft SER Section 14.2.12)

Question Q640.13:

FSAR Subsection 9.3.1.1.4.1 states that while the instrument air system is not safety related, it does have an interface with components that are part of safety related systems. Modify Preoperational Test Number 38 (Instrument Air and Containment Instrument Air), or the preoperational test objectives in FSAR Table 14.2-1 for all safety related systems that interface with instrument air, to include individual valve testing in accordance with Section C.8 of Regulatory Guide 1.68.3 (Preoperational Testing of Instrument and Control Air Systems), or revise the current exception to Regulatory Guide 1.68.3 in FSAR Subsection 14.2.7.9 to include a listing of the applicable safety related systems.

Response:

Refer to revised FSAR Table 14.2-1 for the response to this question.

Additional Concerns Identified in Draft SER:

The loss of instrument air test should be conducted to simulate both a gradual loss of pressure as well as a sudden loss of pressure.

Response:

Refer to revised FSAR Table 14.2-1, Preoperational Test 38.

Air-operated valves will be tested to verify failure position on a sudden loss of air pressure on an individual basis. In addition, during hot functional testing, a gradual loss of instrument air test will be performed as Preoperational Test 38 in accordance with FSAR Section 9.3.1.1.4.1 and Regulatory Guide 1.68.3.

TABLE 14.2-1 (Cont)

38. PREOPERATIONAL TEST - INSTRUMENT AIR AND CONTAINMENT INSTRUMENT AIR

Prerequisites for Testing

General prerequisites have been met. The system has been pressure tested using instrument air quality gas.

Test Objective and Summary

Testing will be performed to provide assurance that the instrument air system will provide clean dry air at the proper pressure to end use equipment.

All air operated valves are individually tested to ensure proper operation. This testing includes proper response to loss of air. 640.13

Compressors will be tested for manual and automatic starting, quality and volume of air delivered and verification of instrument readings. Cooling water flows to the compressors will be verified. Instrument air dryers will be coupled to the compressor and full flow air tests will be conducted. Dryers will be operated full cycle with automatic switching of dryer towers verified. Instruments and alarm settings will be verified. Total air demand at normal steady state conditions, including leakage from the system, will be verified to be in accordance with design. Quality of air will be evaluated at the dryer outlet. Further verification of cleanliness shall be verified by blowdown of instrument air lines through a filter cloth. A loss of instrument air test shall be conducted at near normal operating conditions to verify acceptability of emergency response procedures and system response. A test shall be conducted to demonstrate that plant equipment designed to be supplied by the instrument air system is not supplied by other air supplies having less restrictive air quality requirements. Plant components requiring large quantities of instrument air shall be operated simultaneously while the system is at near normal steady state conditions to verify that pressure transients in the distribution system do not exceed acceptable values. Functional testing shall be performed to verify that failures resulting in an increase in the supply system pressure will not cause peak transient pressures above the design pressure of the system components.

* gradual

Acceptance Criteria

All equipment in the instrument air system will perform in an acceptable manner in accordance with design requirements.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-17 Solid State Protection System -
Question 640.17 (Draft SER Section 14.2.12)

Question Q640.17:

Modify Preoperational Test Number 59 (Solid State Protection System) to provide assurance that a manual reactor trip will both remove voltage from the under-voltage trip coil and energize the shunt trip coil (see I&E Bulletin 83-01, February 25, 1983).

Response

Refer to revised FSAR Table 14.2-1, Preoperational Test 59.

TABLE 14.2-1 (Cont)

59. PREOPERATIONAL TEST - SOLID STATE PROTECTION SYSTEM

Prerequisites for Testing

General prerequisites have been met.

Test Objective and Summary

Testing will demonstrate proper operation of the reactor trip and engineered safeguards actuation logic and output signals of the solid state protection system in response to simulated input signals on each channel. Each design logic condition will be tested and proper coincidence logic verified. Fail safe operation on loss of power will be verified. The manual reactor trip up to the tripping of the reactor trip breakers will also be tested. INSERT

Acceptance Criteria

The solid state protection system produces proper logic response for specified input signals.

PSRB-17

FSAR Table 14.2-1
Preoperational Test #59

INSERT

This will include testing to individually test that a manual trip will remove power from the reactor trip breaker undervoltage coil and energize the shunt trip coil.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-18 Regulatory Guide 1.68, Rev. 2, Appendix A.1.2 and A.5.t - Question 640.19 (Draft SER Section 14.2.12)

Question Q640.19:

Regulatory Guide 1.68, Revision 2, Appendix A.1.2 and A.5.t prescribe testing for various valves. Modify Preoperational Test Number 71 (Integrated Precore Hot Functional Testing) to provide for a more complete demonstration of the operability of pressurizer power operated relief valves; main steam line relief valves; atmospheric steam dump valves; main steam bypass valves; and main steam control valves. Such a demonstration should include response times, relieving capacities, setpoints, and reset pressures. Open and reclosure setpoints for all relief valves should be checked at temperature. Where valves are not tested in-situ with the process fluid, testing should be conducted to verify that discharge piping is clear and will not choke or produce back-pressure affecting set-reset pressures of the valves. When referencing bench tests instead of performing installed capacity checks, technical justification should be provided.

(NOTE: This item is not applicable to ASME Code safety valves subject to ASME Section XI preservice tests.)

Response:

Refer to revised FSAR Table 14.2-1, Preoperational Test 71.

These valves will be tested at temperature, in place, with the process fluid. The relief capacity of the atmospheric dump values and main steam bypass valves is demonstrated in tests described in FSAR Table 14.2-2 test numbers 23, 37, and 39. The capacity of the PORV's is addressed in FSAR Section 5.4.13.2. ASME Code Safety Valves, will be subject to ASME Section XI testing.

Additional Concerns Identified at March 21, 1984 Meeting:

Relief capacity of the PORVs and atmospheric dump valves has not been adequately demonstrated. FSAR Subsection 5.4.13.2 addresses an evaluation program whose results will be reported to the NRC prior to fuel load, and startup tests 23, 37, and 39 do not specifically address determination of valve relief capacity. Provide reference to where specific testing is accomplished which ensures that the relief capacity of the PORVs and atmospheric dump valves is less than the value assumed in the safety analysis (FSAR Subsections 15.1.4 and 15.6.1).

Response:

Refer to revised FSAR Table 14.2-1, Preoperational Test 71.

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Procedures and Systems Review Branch

PSRB-18 (Cont.)

Although specific testing is not performed to ensure that PORV and atmospheric dump valve relief capacity is less than the value assumed in the safety analysis, the following system design limits apply which limit the effect of excessive relief capacity:

1. As stated in FSAR Section 15.6.1.1, a PORV is sized to relieve approximately 50% of what a pressurizer safety valve would relieve. Bench test results provided by EPRI indicate that PORV relief capacity is 372,600 lbm/hr, while safety valve design relief capacity is 420,006 lbm/hr (refer to EPRI NP-2628-LD, dated September, 1982). Since the RCS is analyzed for inadvertent safety valve opening, inadvertent PORV opening is, therefore, bounded by the former event.

2. Inadvertent atmospheric dump valve opening is bounded by the analyzed main steam line rupture event due to relative pipe sizing. FSAR Section 15.1.5.2 states that the Main Steam Line Rupture event is analyzed for an equivalent 1.4 ft^2 break. Since failure of an Atmospheric Steam Dump would result in a maximum of a 0.35 ft^2 break, this accident is bounded.

71. PREOPERATIONAL TEST - INTEGRATED PRECORE HOT FUNCTIONAL TESTING

Prerequisites for Testing

General prerequisites have been met. The reactor coolant system cold hydrostatic test has been completed. All preoperational testing of systems required to support hot plant operations has been completed and reviewed for adequacy for the joint test groups with all test deficiencies corrected or specifically waived.

Test Objective and Summary

Testing will demonstrate the satisfactory performance of systems and components during the heatup of the reactor coolant system (RCS), operation at normal temperature, pressure, and cooldown. Specific testing will include:

1. Heatup of the RCS to normal operating temperature and pressure utilizing the reactor coolant pumps and pressurizer heaters. This test will include demonstration of solid system pressure control and the capability to add hydrazine to the RCS

2. Perform periodic vibration measurements of reactor coolant pumps

3. Demonstrate that the operation of pressurizer pressure and level control systems including heater and spray operation. Perform preliminary spray flow adjustments

4. Demonstrate that the operation of the steam generator atmospheric and condenser steam dump valves is acceptable within specific limits

INSERT A

5. Demonstrate the capability of the chemical and volume control system to provide charging water at rated flow against normal RCS pressure, verify letdown flow rate for various operating modes and verify the excess letdown and seal water flows

6. Perform RCS incore thermocouple and RTD isothermal calibration

7. Verify ability to maintain steam generator levels and proper operation of feedwater control systems, steam dumps and level instrumentation

8. Demonstrate proper functioning of the main steam isolation valves at normal operating temperature and pressure

9. Demonstrate the proper operation of steam generator safety valves, verifying setpoints with a pressure-assist device and verifying proper reseating and leakage within specified limits

TABLE 14.2-1 (Cont)

10. Demonstrate the proper operation of pressurizer safety and relief valves, and the capability of the pressurizer relief tank to condense a steam discharge from the pressurizer

    a. Proper actuation, operation, RESET and response time of the power operated relief valves (PORV) will be demonstrated by simulating a high pressure signal to each valve INSERT B

    b. The PORV will be operated manually to confirm valve operability and the ability of the pressurizer relief tank (PRT) to condense a discharge. Leakage following operation will be verified within acceptable limits. Discharge header leakage detection instrumentation will be verified operable in accordance with design requirements.

    c. Operability of PORV and PRT instrumentation, controls, interlocks, and alarms will be verified.

    d. Safety valve leakage at RCS normal pressure will be verified within specified limits. Actual safety valve operation will be demonstrated by hydrostatic bench test to verify set points.

11. Operate the reactor coolant pumps for a minimum of 240 hours at full flow to achieve approximately 1 million vibration cycles on reactor internals. Following hot functional testing, the internals are removed and inspected for vibration effects

12. Demonstrate proper operation of reactor coolant pump trips and alarms

13. Demonstrate the operability of remote shutdown controls

14. Perform or complete those portions of the following system tests (see individual descriptions), which require the RCS to be at or near normal operating temperature and pressure:

    a. Reactor coolant system expansion and restraint

    b. Chemical and volume control

    c. Boron thermal regeneration

    d. Residual heat removal

    e. Low pressure safety injection

    f. High pressure safety injection

    g. Reactor plant sampling

    h. Containment ventilation

TABLE 14.2-1 (Cont)

i. Auxiliary building ventilation

j. Engineered safety features building HVAC

k. Main steam

l. Steam dump control

m. Steam generator blowdown

n. Main feedwater

o. Steam generator water level control

p. Auxiliary feedwater

q. Service water

r. Reactor plant component cooling

s. Reactor plant chilled water

t. Charging pump cooling

u. Safety injection pump cooling

v. Neutron shield tank cooling

w. Steam generator chemical feed

x. Reserve station service transformers

y. Loose parts monitor system

z. Reactor coolant and associated system piping vibration

aa. Thermal expansion of piping and components of secondary systems

15. Perform or complete tests as necessary to ensure the operability of the following systems:

a. Condensate system

b. Extraction steam system

c. Feedwater heater drains and vents system

d. Turbine plant component cooling system

e. Turbine plant sampling system

TABLE 14.2-1 (Cont)

f. Normal AC power distribution system

16. Perform a controlled plant cooldown, using the steam dump and residual heat removal systems. Demonstrate the capability to de-gas and add hydrogen to the RCS

INSERT G

Acceptance Criteria

Systems and components tested will meet specified design, safety analysis, and Technical Specification requirements.

FSAR Table 14.2-1
Preoperational Test #71

INSERT A

a. Proper actuation, operation, reset and response time of the valves will be demonstrated. The actuation setpoint and reset pressures of these valves are a function of instrument calibration.

b. Operability of instrumentation, controls, interlocks and alarms will be verified.

INSERT B

The actuation setpoint and reset pressures of these valves are a function of instrument calibration.

INSERT C

17. Demonstrate that the operation of the main steam control valves is acceptable within specific limits. Proper actuation and response time of these valves will be demonstrated. The actuation setpoint pressures of these valves are a function of instrument calibration.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-19 Real or Dummy Fuel Assemblies
For Vibration Test - Question 640.20(2) (Draft SER Section 14.2.12)

Question Q640.20:

In FSAR Section 1.8 (Table 1.8N-1, page 6 of 39) the degree of compliance to Regulatory Guide 1.20 (Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing) states that testing and test inspections will be conducted during hot functional testing.

1. Modify Preoperational Test Number 71 (Integrated Precore Hot Functional Testing) Item 11 in FSAR Table 14.2-1 to include a cross-reference to FSAR Section 3.9.2 for additional information on vibration testing.

2. Modify or provide a new startup test description in FSAR Table 14.2-2 that describes the post-core load vibration assessment testing and inspection intended to be accomplished. (Appropriate reference may be used for description.)

Response:

Refer to revised FSAR Table 14.2-1 for the response to this question.

There is no post-core load vibration assessment testing and inspection intended to be accomplished. Westinghouse stated in WCAP-8780, Verification of Neutron Pad and 17 x 17 Guide Tube Designs by Preoperational Tests on the Trojan I Power Plant, that vibration levels were lower than Indian Point II (prototype for Westinghouse 4 loop plant) and were in agreement with predicted results. Also, it was concluded that the internals are free from harmful vibrations.

Refer to FSAR Section 3.9N.2.3 for additional information.

Additional Concerns Identified in Draft SER:

Technical justification should be provided for not utilizing real or dummy fuel assemblies in the vibration test in accordance with Regulatory Guide 1.20, Position C.2.2.2.c.

Response:

Refer to FSAR Sections 3.9N.2.3 and 3.9N.2.4 for technical justification for not using real or dummy fuel assemblies in the vibration test. This position has been found to be acceptable by the NRC's Mechanical Engineering Branch in their review of the FSAR as stated in Section 3.9.2.3 of the Millstone Unit No. 3 Draft Safety Evaluation Report.

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Open Items

Procedures and Systems Review Branch

PSRB-20 NUREG-0694, Item I.G.1 - Question 640.22 (Draft SER Section 14.2.12)

Question Q640.22:

NUREG-0694, "TMI Related Requirements for New Operating Licenses," Item I.G.1, requires Applicants to perform "a special low power testing program approved by NRC to be conducted at power levels of greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training." To comply with this requirement, modify Startup Test Number 21 (Natural Circulation) to ensure accomplishment of the following objectives:

Testing - The test should demonstrate the following plant characteristics: length of time required to stabilize natural circulation, core flow distribution, ability to establish and maintain natural circulation with or without onsite and offsite power, the ability to uniformly borate and cool down to hot shutdown conditions using natural circulation, and subcooling monitor performance.

Training - Each licensed reactor operator (RO or SRO who performs RO or SRO duties, respectively) should participate in the initiation, maintenance, and recovery from natural circulation mode. Operators should be able to recognize when natural circulation has been stabilized and should be able to control saturation margin, RCS pressure, and heat removal rate without exceeding specified operating limits.

If these tests have been performed at a comparable prototype plant, they need be repeated only to the extent necessary to accomplish the above training objectives. Test data should be used as feedback for simulator verification and update. Attachment 4 to a letter from E. P. Rahe (Westinghouse) to H. R. Denton (NRC) dated July 8, 1981, contains an acceptable approach for accomplishing the testing objectives listed above.

Response:

Refer to revised FSAR Table 14.2-2, Startup Test 21 and FSAR Section 14.2.10.2.

PSRB-20

TABLE 14.2-2 (Cont)

21. STARTUP TEST - NATURAL CIRCULATION

Prerequisites for Testing

The low power physics test has been completed. Nuclear steam supply systems and all necessary plant secondary and auxiliary systems are operational. Plant operating procedure prerequisites are met except where special conditions required by this test state otherwise.

Test Objective and Summary

The purpose of this test is to demonstrate the plant's capability to remove core heat by natural circulation. The test will be initiated by tripping all reactor coolant pumps and monitoring the establishment of natural circulation. ~~, including observing the length of time for the plant to stabilize flow and temperature distribution, and the ability to maintain the cooling mode.~~ INSERT

Acceptance Criteria

Natural circulation cooling can be established and maintained.

FSAR Table 14.2-2
Startup Test #21 Insert

This test will determine the length of time necessary to stabilize natural circulation and will demonstrate the reactor coolant flow distribution by use of incore thermocouples. Effects of changes in charging flow and steam flow on subcooling margin will be determined and subcooling margin monitor performance shall be verified.

This test shall be performed with available licensed reactor operators (RO and SRO) in the control room who will participate in the initiation, maintenance and recovery from natural circulation mode. Data shall also be taken for feedback for the Millstone Unit 3 simulator response to natural circulation. Operators not directly performing the test shall receive training in natural circulation on the Millstone Unit 3 specific simulator with specific instruction in those areas where simulator response may differ from actual plant performance.

Specific concerns of Attachment 4 to the July 8, 1981 letter from E. P. Rahe to H. R. Denton are addressed as follows:

1. Manual operation of TDAFW Pump will be performed during Preoperational Test 30, Auxiliary Feedwater. Pre-core hot functional testing will verify Auxiliary Feedwater System capability to maintain SG levels. Since all TDAFW Pump controls are supplied from DC power sources, a loss of AC power verification test will not be performed.

2. Pressurizer spray and heater as well as charging and steam flow effects on margin to saturation temperature will be tested during Startup Test 2 - Post-core Hot Functional.

3,4,5. Natural Circulation Test and Station Blackout Test will be performed as Startup Tests 21 and 26, respectively.

PSRB-20

In addition to the normal source range instrumentation, special submersible neutron detectors are used to monitor flux changes throughout the loading of the core. Data from these instruments will be used to determine, directly or through calculations (i.e., inverse count rate ratio), if an abnormal situation exists. Personnel involved in the monitoring, calculating, or evaluation of data will be briefed on their responsibilities prior to the test.

The procedure starts with the insertion of the temporary nuclear monitor detectors, and those fuel assemblies which contain neutron sources, into the vessel. This is followed by insertion of the remaining fuel assemblies in a sequence to be determined in conjunction with the NSSS Vendor. Throughout the loading sequence, the following is performed: RCS boron concentration and coolant temperature is recorded; high flux alarms are set at appropriate limits; temporary and source range detectors are monitored visually and at least one channel is monitored audibly; an inverse count rate ratio (ICCR) is calculated. A status board is used to record fuel assembly/detector locations for each step of the procedure. At completion of core loading, a final core configuration is recorded.

Core loading operations are suspended should any of the following conditions occur:

1. An unexpected increase in count rate above a specified level

2. An unexpected change in RCS boron concentration or water temperature

3. An unexpected containment radiation monitoring alarm occurs.

4. An insufficient number of neutron detector channels becomes available for monitoring.

5. ICCR data indicates that an abnormal condition exists.

If core loading has been suspended for any reason, required surveillances (i.e., boron concentration, water temperature, neutron count rate, etc.) shall continue at the required frequency. Loading operations will not resume until the reason for the suspension has been understood and corrected, or has been evaluated and found acceptable.

14.2.10.2 Post-Core Hot Functional

After completion of fuel load, the technical specification shutdown margin for a fully loaded core will be verified. Steps are then taken to align and check the operability of instruments, equipment, and control systems necessary for plant heatup. Since the PCHF proceeds in steps from cold conditions to operating temperature and pressure, some prerequisites are not required until just before the appropriate condition for testing. Such prerequisites are identified within the test itself, and met before proceeding. Furthermore, several systems may be tested as appendices of the PCHF. Any

prerequisites necessary for these tests will be stated in the applicable test appendix; failure to meet them affects only the test appendix, not the remaining portions of the PCHF. Plant procedures will be used to the maximum extent possible for conducting the PCHF.

Along with general precautions associated with the plant operating procedures, some important precautions for PCHF include: the requirement that reactivity changes be made under the direct supervision of a senior reactor operator, and vigilance to assure any boron dilution does not lower reactor coolant system (RCS) concentration below that required for fueling shutdown.

The PCHF initially prepares the plant for heatup. Upper core internals are installed; the reactor vessel head is placed and the studs are tensioned; cables, ductwork, and insulation are connected; and the missile shield is put in place. While at ambient temperature, the rod control system will be checked out and rod drop times are measured. After the prerequisites have been met for plant heatup (RCS filled and vented, reactor coolant pumps (RCP) operable, etc.), the RCS is heated to normal operating temperature and pressure using RCP heat. At selected points in the heatup, RCS leak tests will be performed, operation of instrumentation will be checked and compared, and plant systems will be tested in accordance with the PCHF appendices.

When normal operating temperature and pressure are reached, the following tests will be performed:

1. Pressurizer spray and heater effectiveness will be checked.

2. RCS design flow will be verified.

3. Rod drop times under hot conditions will be checked.

4. Flow coastdown will be conducted.

INSERT

In addition, items encountered during the pre-core hot functional which were unsatisfactory and systems not previously checked under hot conditions will be tested. This includes a checkout of incore movable detectors, auxiliary feedwater performance verification, and steam dump controls testing.

14.2.10.3 Initial Criticality

Upon completion of the PCHF, the primary system is at hot shutdown with reactor coolant pumps operating, RCS temperature controlled using the steam bypass/dump system, and RCS boron concentration equal to or greater than the value for core loading. Remaining deficiencies are reviewed by the JTG and resolution obtained prior to authorization of performance of the initial criticality procedure. In addition to the regular plant systems necessary for initial criticality, special equipment, such as a reactivity computer and recorders for monitoring/plotting data, are checked out and verified as operational.

(FSAR pg. 14.2-23)

Insert

5. Effect of pressurizer heaters and normal spray and changes in charging and steam flow on margin to saturation temperature will be checked. Auxiliary spray will not be tested due to cyclic limitations based on differential temperature.

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Open Items

Procedures and Systems Review Branch

PSRB-21 Regulatory Guide 1.68, Rev. 2, Appendix A - Question 640.26 (Draft SER Section 14.2.12)

Question Q640.26:

Our review of your test program description concludes that the operability of several of the systems and components listed in Regulatory Guide 1.68 (Revision 2) Appendix A may not be adequately demonstrated by your initial test program. Expand FSAR 14.2.12 to address the following items:

NOTE: Although some of these systems are designated for testing in Preoperational Test Number 71 (Integrated Precore Hot Functional Testing) Part 15, individual test descriptions for these systems should be included in FSAR Chapter 14 to adequately describe what testing will be done. Inclusion of a test description in FSAR Chapter 14 does not necessarily imply that the test becomes subject to FSAR Chapter 17 Quality Assurance Program controls. Certain tests to be performed prior to fuel loading to verify system operability may be referred to as "acceptance tests" to distinguish them from "preoperational tests" subject to FSAR Chapter 17 test control.

Preoperational Testing

R.G. 1.68 Appendix A	FSAR Section	Description
1.a(2)(f)	5.4.12	Loop stop valves
1.a(2)(h)	5.4.15	Reactor vessel head vent system
1.d(9)	10.4.7	Condensate storage system
1.e(5)	10.4.7	Steam extraction system
1.e(8)	10.4.7	Condensate system
1.e(10)	10.4.7	Feedwater heater and drain systems
1.3(12)	10.4.2	Condenser air evacuation system
1.g(1)	8.3.1.1.1	Normal ac power distribution system
1.h(5)	7.6.6	Reactor coolant system loop isolation valve interlocks

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Open Items

Procedures and Systems Review Branch

PSRB-21 Continued

R.G. 1.68 Appendix A	FSAR Section	Description
1.h(8)	6.3.5	Refueling water storage tank level and temperature indication
1.h(10)	9.2.5	Ultimate heat sink
1.j(7)	6.3.2.5	Leak detection systems used to detect failures in ECCS and containment recirculation spray systems located outside containment
1.j(16)	10.4.7	Hotwell level control systems
1.j(17)	10.4.7	Feedwater heater temperature, course of postulated accidents:
		a) containment wide range pressure indicators
		b) containment sump level monitors
		c) containment radiation monitors
		d) humidity monitors
1.j(24)	7.1.1.5	Reactor control and ESF annunciators
1.k(2)	12.5	Personnel monitors and radiation survey instrument tests
1.k(3)	12.5	Laboratory equipment used to analyze or measure radiation levels and radioactivity concentration
1.k(4)	6.5.1.4	HEPA filter and charcoal adsorber efficiency and inplace leak tests. Modify the appropriate test abstracts to ensure that testing in accordance with Regulatory Guide 1.52 (Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants), Positions C.5.a - C.5.d, and Regulatory Guide 1.140 (Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air

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Open Items

Procedures and Systems Review Branch

PSRB-21 Continued

R.G. 1.68 Appendix A	FSAR Section	Description
		Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants), Positions C.5.a - C.5.d, is accomplished.
1.1(8)	9.3.2	Turbine plant sampling system
1.m(3)	9.1.4	Operability and leak tests of sectionalizing devices and drains, and leak tests of gaskets or bellows in the refueling canal and fuel storage pool
1.m(4)	9.1.4	Dynamic (100%) and static (125%) tests of cranes, hoists, and associated fuel storage and handling systems
1.n(3)	9.2.7	Turbine plant component cooling system
1.n(16)	6.3.2.2.2	Cooling and heating systems for tank
1.n(18)		Heat tracing and freeze protection systems
1.o(1)	9.1.5	Polar crane dynamic (100%) and static (125%) loading tests

Power Ascension Tests

5.w		Containment penetration coolers. Provide a preoperational test description or, on those penetrations where coolers are not used, provide a startup test description that will demonstrate that concrete temperatures surrounding hot penetrations do not exceed design limits.
5.ii	15.3.2	Demonstrate that the dynamic response of the plant is in accordance with design for limiting reactor coolant pump trips. The method for initiating pump trip should result in the fastest credible coastdown in flow.

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Open Items

Procedures and Systems Review Branch

PSRB-21 Continued

Response:

Refer to the discussion below and to revised FSAR Tables 14.2-1 and 14.2-2 for the response to this question.

Instrumentation testing is performed within the phase 1/2 testing of the system to which the instruments belong. Instrument loop continuity and calibrations are performed during this testing. Indications, alarm and comptuer readings are verified where appropriate. This includes annunciation.

Refer to FSAR Table 1.8-1 for conformance to Regulatory Guides 1.52 and 1.140.

Tests and calibrations for personnel monitors, radiation survey instruments and laboratory equipment are performed in accordance with station procedures which are currently in use on site for Units 1 and 2.

Table Q640.26-1 indicates where each test abstract may be found.

Additional Concerns Identified in Draft SER:

1.g(1) - Testing should be provided to verify equipment operability at maximum and minimum design voltage (BTP PSB-1, paragraph B.4).

1.m(4), 1.0(1) - Documentation should be provided which ensures that the construction load testing of the polar crane and cranes, hoists, and associated fuel storage and handling system is accomplished at 125% load (static test) and 100% load (dynamic test).

5.w - Hot penetrations which are not serviced by reactor plant component cooling should also be monitored during startup testing.

Response:

1.g(1)

Testing will be performed as specified in response to NRC Question Q430.11.

1.m(4), 1.0(1)

Refer to revised FSAR Table 14.2-1, Preoperational Test 4.

Crane load testing of the polar crane and cranes, hoists, and associated fuel storage and handling system is accomplished at 125% load (static test) and 100%

PSRB-21 Continued

load (dynamic test) by our architect-engineer, Stone and Webster Engineering Corporation during construction. Crane test documentation is obtained during equipment turnover to the NNECO organization and is attached to the applicable Phase I test procedure document. Subsequent testing will be in accordance with Technical Specification requirements.

5.w

Refer to revised FSAR Table 14.2-2, Startup Test 42.

PSRB-21

TABLE 14.2-2 (Cont)

42. STARTUP TEST - CONTAINMENT PENETRATION TEMPERATURE MONITORING

Prerequisites for Testing

The plant is at approximately 30, 50, 75, 90, and 100 percent power.

Test Objective and Summary

640.26 Testing will monitor the temperature of hot penetrations serviced by reactor plant component cooling.

Insert A

A

Acceptance Criteria

Reactor plant component cooling can maintain the penetrations within design temperature limits.

(FSAR Table 14.2-2)

Startup Test #42

Insert A

Additionally, testing will monitor the temperature of other penetrations determined to be hot penetrations but not serviced by reactor plant component cooling.

TABLE 14.2-1 (Cont)

4. PREOPERATIONAL TEST - POLAR CRANE

Prerequisites for Testing

INSERT B

General prerequisites have been met. ~~All component testing including the construction load test has been completed.~~

Test Objective and Summary

This test will verify operability of polar crane control circuits and ability to handle the reactor vessel head and various internals components. | 640.27

Acceptance Criteria

The crane control circuits and interlocks function in accordance with design. The crane is capable of installation and removal of the reactor vessel head and those internal components placed during cold hydrostatic and hot functional testing.

(FSAR Table 14.2-1)

Preoperational Test #4

Insert B

All component testing including the construction 125% static and 100% dynamic load tests have been completed.

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Open Items

Procedures and Systems Review Branch

PSRB-22 Preoperational Tests 76-84 - Question 640.27 (Draft SER Section 14.2.12)

Question Q640.27

Modify FSAR Figure 14.2-5 to include the following preoperational tests listed in FSAR Table 14.2-1.

4 - Polar Crane

10 - Residual Heat Removal

14 - Reactor Plant Sampling

26 - Steam Dump Control

27 - Steam Generator Blowdown

39 - Radioactive Liquid Waste

53 - Reserve Station Service Transformers

54 - Communications

65 - Emergency Lighting

72 - Reactor Coolant and Associate System Expansion and Restraint

73 - Reactor Coolant and Selected Systems Piping Vibration

74 - Thermal Expansion of Piping and Components of Secondary Systems

75 - Control System Test for Turbine Runback Operation

Response:

Refer to revised FSAR Figure 14.2-5 for the response to this question.

Additional Concerns Identified in Draft SER:

Preoperational tests 76 through 84 should be included in FSAR Figure 14.2-5.

Response:

Refer to revised FSAR Figure 14.2-5.

PSRB-22

TESTING TIME FOR PREOPERATIONAL OR ACCEPTANCE TESTS

16 MONTHS OR MORE BEFORE FUEL LOAD	BETWEEN 12 & 16 MONTHS BEFORE FUEL LOAD	BETWEEN 12 & 8 MONTHS BEFORE FUEL LOAD	BETWEEN 8 & 5 MONTHS BEFORE FUEL LOAD	WITHIN 5 MONTHS BEFORE FUEL LOAD
PRIMARY GRADE WATER	INSTRUMENT AIR	AUXILIARY FEEDWATER	RADIOACTIVE GASEOUS WASTE	CONTAINMENT RECIRCULATING VENT
CARBON DIOXIDE FIRE PROTECTION	WATER FIRE PROTECTION	CONTAINMENT VACUUM	FEEDWATER	MAIN STEAM
HALON FIRE PROTECTION	CHILLED WATER	QUENCH SPRAY	REACTOR PLANT AND GASEOUS DRAINS	EXTRACTING STREAM
SCREENHOUSE VENTILATION	SERVICE WATER	REACTOR PLANT COMPONENT COOLING WATER	STEAM GENERATOR LEVEL CONTROL	N_2 SHIELD TANK COOLING
CHLORINE	CHARGING PUMPS COOLING	DIESEL GENERATOR AND FUEL OIL	RADIOACTIVE SOLID WASTE	REACTOR COOLANT
STATION ELECTRICAL SERVICE 4.16KV	SAFETY INJECTION PUMP COOLING	CONTAINMENT RECIRCULATING SPRAY	SPENT FUEL POOL COOLING PURIFICATION	FUEL HANDLING
STATION ELECTRICAL SERVICE 480V	Turbine Plant Component Cooling	PROCESS PROTECTION AND CONTROL	CONTAINMENT	CONTROL ROD DRIVE
STATION ELECTRICAL SERVICE 125VDC AND 120VAC		CONTAINMENT ATMOSPHERIC MONITORING	LOW PRESSURE SAFETY INJECTION	POTENTIALLY CONTAMINATED AND VITAL AREA
~~EMERGENCY LIGHTING~~ Delete		Condensate and Condensate Storage	HIGH PRESSURE SAFETY INJECTION	HEATING, VENTILATING AND AIR CONDITIONING
Reserve Station Service Transformers		Reactor Vessel Head Vent	SOLID STATE PROTECTION	NUCLEAR INSTRUMENTS
			SAFEGUARDS ACTUATION	DIGITAL INCORE
			CONTAINMENT LEAK MONITORING	PROCESS AND AREA RADIATION MONITORS
			LOOSE PARTS MONITOR	CHEMICAL VOLUME CONTROL
			SAFETY PARAMETER DISPLAY SYSTEM	DIGITAL ROD POSITION
			CONDENSATE INCORE THERMOCOUPLES	SEISMIC MONITOR
			H_2 RECOMBINER	BORON RECOVERY
			REACTOR PLANT AERATED DRAINS	SECURITY
			RADIOACTIVE LIQUID WASTE	RESIDUAL HEAT REMOVAL
			POLAR CRANE	REACTOR PLANT SAMPLING
			Emergency Lighting	STEAM DUMP CONTROL
			Refueling Water Storage Tank Cooling	RESERVE STATION SERVICE TRANSFORMERS
			Heat Tracing	COMMUNICATIONS
			Spent Fuel Pool Gates and Transfer Tube Leak Test	REACTOR COOLANT AND ASSOCIATED SYSTEM EXPANSION AND RESTRAIN
				REACTOR COOLANT AND SELECTED SYSTEMS PIPING VIBRATION
				THERMAL EXPANSION OF PIPING AND COMPONENTS OF SECONDARY SYSTEMS
				CONTROL SYSTEM TEST FOR TURBINE RUNBACK OPERATION
				STEAM GENERATOR BLOWDOWN
				Reactor Coolant Loop Isolation Valves
				Turbine Plant Sampling
				Condenser Air Removal

TESTING TIME FOR MAJOR TESTS (MILESTONES)

COLD HYDRO — ILRT — ESF TESTING — HOT FUNCTIONAL

18 15 14 13 12 11 10 9 8 7 6 5 4 3 2 1

MONTHS BEFORE FUEL LOADING

FIGURE 14.2-5
PREOPERATIONAL TEST PHASE
MILLSTONE NUCLEAR POWER STATION
UNIT 3
FINAL SAFETY ANALYSIS REPORT

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-23 Swing Load Test - Question 640.28 (Draft SER Section 14.2.12)

Certain startup test listed below do not specify the power level at which the test will be conducted, instead stating that testing will be conducted at selected or various power levels. Modify the individual test abstracts to include the specific power level values at which each of the tests will be conducted. Modify FSAR Figure 14.2-6 to indicate which tests will be conducted during each power plateau during the startup program, or provide a clarification stating that these tests will be conducted at power levels consistent with Regulatory Guide 1.68, Revision 2.

14 - Loose Parts Monitoring System

15 - Water Chemistry Control

16 - Radiation Survey

28 - Operational Alignment of Nuclear Instrumentation

29 - Process and Effluent Radiation Monitoring System

30 - Core Performance

31 - Power Coefficient Measurements

33 - Ventilation System Operability

34 - Turbine Generator and Feedwater Turbine Operability Test

35 - Calibration of Steam and Feedwater Flow Instrumentation at Power

37 - Load Swing Test

Response:

Refer to revised FSAR Table 14.2-2 for the response to this question.

Additional Concerns Identified in Draft SER:

The load swing test (startup test 37) should include testing at 50% power in accordance with Regulatory Guide 1.68, Appendix A, Section 5.h.h.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

PSRB-23 (Cont.)

Response:

Load Swing Test (Startup Test 37) will not be performed at 50% power for the following reasons:

1. Load Swing Test will be performed at 30%, 75%, and 100% plateaus.

2. A 4% Load Swing Test will be performed at 50% as part of the power coefficient test (Startup Test 31).

3. We will conduct a full reactor trip at 50% which is significantly more limiting than the 10% load swing test.

Refer to revised FSAR Section 14.2.7.7 for Regulatory Guide 1.68, Revision 2 conformance statement Appendix A, Section 5.h.h.

PSRB-23

14.2.7.4 Regulatory Guide 1.37, Revision 0 - Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants | 640.1

The Millstone 3 initial test program will conform to the intent of Regulatory Guide 1.37.

14.2.7.5 Regulatory Guide 1.41, Revision 0 - Preoperational Testing of Redundant Onsite Electrical Power Systems to Verify Proper Load Group Assignments | 640.1

For position on Regulatory Guide 1.41, see FSAR Section 1.8.

14.2.7.6 Regulatory Guide 1.52, Revision 2 - Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

For position on Regulatory Guide 1.52, see FSAR Section 1.8.

14.2.7.7 Regulatory Guide 1.68, Revision 2 - Initial Test Programs for Water-Cooled Nuclear Power Plants

The Millstone 3 initial test program will conform to Regulatory Guide 1.68, except as specified in this section:

~~1. The failed fuel detection system (Appendix A, Section 5) shall be tested during a preoperational test.~~ | ~~640.2~~ Delete

1 ~~2.~~ During power escalation, testing will be conducted at the 30-percent power level instead of at the 25-percent power level. Westinghouse supplied plants have generic data for the 30-percent level which they do not have at the 25-percent level (Section C.8; Appendix A, Section 5).

Insert B

3 ~~3.~~ The MSIV closure test will be performed at less than 20-percent power to demonstrate the proper dynamic response of the plant and to verify proper integrated operation of plant equipment. Plant response to a full power trip will be verified by the generator trip at 100-percent power. Closure of the MSIVs at 100-percent power would not provide any additional information significant enough to warrant subjecting the plant to such a severe thermal transient (Appendix A, Section 5.m.m).

4 ~~4.~~ The loss of feedwater heaters test will not be performed. Since plant response to load swings and large load reductions is demonstrated in other tests, there is no need to subject the plant to this additional transient (Appendix A, Section 5.k.k).

5 ~~5.~~ Millstone 3 does not have a partial scram feature (Appendix A, Section 5.j).

(FSAR pg. 14.2-17)

Insert B

2. Load swing testing will be conducted at the 30%, 75%, and 100% plateaus. A 4% load swing test will be conducted at the 50% plateau as a part of the power coefficient test (see FSAR Table 14.2-2 Startup Test 31). Additionally, a full reactor trip at the 50% plateau will be conducted.

6 ~~6.~~ Demonstration of the design capability of reactor residual or decay heat removal systems will be done during power ascension testing only if it is not done during hot functional or low power tests (Appendix A, Section 5.1).

7 ~~7.~~ The following systems will be tested during the startup test phase only if they are not completed during the preoperational test phase:

<u>Reg. Guide Section</u>	<u>Component Tested</u>
Appendix A, Section 4p	Pressurizer and main steam relief valves
Appendix A, Section 4r	Reactor coolant purification and cleanup system
Appendix A, Section 5.c.c	Gaseous and liquid waste radioactive waste systems

8 ~~8.~~ Portions of Appendix A, Section 5.s, will not be conducted. Millstone 3 does not have an integrated control system or a reactor coolant flow control system.

9 ~~9.~~ The auxiliary (startup) and emergency feedwater control systems and the steam pressure control systems will be tested before the power ascension test phase since these systems are not used at power levels above the low power operation modes (Appendix A, Section 5.s).

10 ~~10.~~ The individual rod position indication system is the primary means for determining control rod misalignments. The design of the nuclear instrumentation is not intended to detect a misaligned control rod but rather to detect anomalous core conditions. Therefore, tests will not be conducted in accordance with Appendix A, Section 5.i. However, data on nuclear instrumentation characteristics will be obtained during the core performance test.

11 ~~11.~~ When several emergency loads are identical in type, size, and manufacturer, only one of these loads will be started and operated with the maximum and minimum design voltage available. Testing every emergency load would not provide any additional assurances significant enough to warrant the required increase in the scope of the test program (Appendix A, Section 1g).

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

Question Q640.15 (Section 14.2.12)

In accordance with the test requirements listed in Regulatory Guide 1.41 (Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments), Position C.2:

1. Modify Preoperational Test Number 50 (125 V dc Distribution) to incorporate testing to verify that at the minimum and maximum design battery voltages, required Class IE systems can be started and operated. At minimum battery voltage, with charges deenergized demonstrate capability to start all IE loads. Then, with the chargers energized verify ability of the chargers to supply loads and charge batteries. For more information on problems with maximum battery voltage conditions, see IE Information Notice 83-08, March 9, 1983.

2. Modify Preoperational Test Number 53 (Reserve Station Service Transformers) to demonstrate the proper operation of transformer cooling under design load or describe how data from testing under available load will be extrapolated to verify cooling capability under design loading.

Response:

Refer to revised FSAR Table 14.2-1 for the response to this question.

Preoperational Test Number 50, 125 V dc distribution, tests the design capabilities of the 125 V dc system. The concerns of Regulatory Guide 1.41, Position C.2, are met in Preoperational Test Number 67 - Engineered Safety Feature Test with Loss of Normal Power.

By design, the components powered from the 125 V dc system will function between the maximum and minimum bus voltages. Refer to the response to Question 430.41. Since the dc components are designed for minimum expected voltage, the testing of all IE loads for minimum voltage starting is not necessary. The cooling capabilities of the reserve station service transformers (RSST) have been demonstrated by vendor test. With the RSST under rated load, equilibrium temperatures were below design limits.

Additional Concerns Identified at March 21, 1984 Meeting:

Response should reference testing accomplished to address the concerns of Item 430.11 (conformance with Branch Technical Position PSB-1, NUREG-0800, Appendix 8A). Additionally, the response to Item 640.15 should be modified accordingly.

Response:

Refer to revised FSAR Table 14.2-2, Startup Test 26.

Q640.15

TABLE 14.2-2 (Cont)

26. STARTUP TEST - STATION BLACKOUT

Prerequisites for Testing

The plant is in the 10 to 20 percent power range with all plant loads being supplied by the Millstone 3 generator.

Test Objective and Summary

This test will demonstrate that the plant responds as designed following a plant trip with no offsite power. The reactor will be tripped. The diesel start, load sequencing, and plant response including natural circulation will be monitored. The turbine-driven auxiliary feedwater pump shall be run for a minimum of 2 hours with motor-driven auxiliary feedwater pumps and turbine-driven auxiliary feedwater pump cubicle
640.12 ventilation secured.

INSERT

Acceptance Criteria

640.12 The plant responds in accordance with design. The turbine-driven auxiliary feedwater pump will remain within design limits and pump room ambient conditions do not exceed environmental qualification limits for safety related equipment in the room.

Q640.15

FSAR Table 14.2-2
Startup Test 26

INSERT

AC power to the inverters and battery chargers will be removed for a period of two hours to force battery operation.

Millstone Nuclear Power Station, Unit No. 3

Open Items

Procedures and Systems Review Branch

Question Q640.16 (Section 14.2.12)

1. In accordance with Regulatory Guide 1.108 (Periodic Testing of Diesel Generator Units used as Onsite Electric Power Systems at Nuclear Power Plants), Position C.2.a.4, modify Preoperational Test Number 51 (Diesel Generator) or Number 67 (Engineered Safety Features Test With Loss of Normal Power) to demonstrate proper operation during diesel generator load shedding, including a test of the loss of the largest single load and complete loss of load and verify that the voltage requirements are met and that the overspeed limits are not exceeded. Your testing should, in addition, provide assurance that any time delays in the diesel generator's restart circuitry will not cause the supply of starting air to be consumed in the presence of a safety injection signal (see I&E Information Notice Number 83-17, March 31, 1983).

2. Modify Preoperational Test Number 51 (Diesel Generator) to include testing to ensure the satisfactory operability of all check valves in the flow path of cooling water for the diesel generators from the intake to the discharge (see I&E Bulletin No. 83-03: Check Valve Failures in Raw Water Cooling Systems of Diesel Generators).

Response

Refer to revised FSAR Table 14.2-1 for the response to this question.

There are no check valves in the service water lines that provide cooling water to the diesels. As such, the concerns of I&E Bulletin 83-03 do not apply to the Millstone Unit No. 3 design.

Additional Concerns Identified at March 21, 1984 Meeting:

Modify Preoperational Test Number 51 (Diesel Generator), Test Objective and Summary Item 7, in accordance with the revised clarification in FSAR Table 1.8-1 to Regulatory Guide 1.108 (Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants), Position C.2.a.4, as revised by the response to Item 430.16. The test of the loss of the single largest load and complete loss of load should be conducted with the diesel initially at its 2000-hour rating.

Response

Refer to revised FSAR Table 1.8-1, Regulatory Guide 1.108.

TABLE 1.8-1 (Cont)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference	
1.108	Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants (Rev. 1, August 1977)	Comply, with the following clarifications and exceptions: Section C.2(a)2: Proper operation for design-accident-loading-sequence will be demonstrated under conditions as close to design as possible. ~~Section C.2(a)3: The full-load-carrying capability will be demonstrated for the 22-hour period at the 2000-hour rating.~~ ~~Section C.2(a)4: The test of the loss of the largest single load and of complete loss of load will be demonstrated with the unit operating at the 2000-hour rating.~~ Section C.2(a)9: Comply as stated in the ERRATA dated September 1977.	8.3.1	430.15 430.16
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I (Rev. 1, October 1977)	Comply	13.3.1	
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (Rev. 0, March 1976)	Comply		
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors (Rev. 1, July 1977)	Comply	2.3.5.2.3	

Q640.16