Docket No.: 50-423 SEP 25 1985

Mr. John F. Opeka
Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Opeka:

Subject: Preliminary Safety Evaluation to be Included in SER Supplement 3 for Millstone Nuclear Power Station, Unit 3

Enclosure 1 contains safety evaluations which are proposed for inclusion in SER Supplement 3 for Millstone 3. These evaluations are being transmitted to you for your information.

Enclosure 2 lists 21 SER open items on which the staff has been unable to continue its review due to NNECo's failure to provide requested additional information in a timely manner. In some instances more than a month has passed since the original NNECo submittal date. It is imperative that all of the requested information be provided to the staff promptly so that we can continue our review.

It is our understanding that all of the requested information necessary to resume our review will be docketed by September 30, 1985. Because resumption of our review requires receipt of the requested information, the completion of our review and subsequent issuance of your operating license could be delayed if NNECo does not submit the information by September 30, 1985.

Any comments or concerns should be directed to the Licensing Project Manager, Elizabeth L. Doolittle at (301) 492-4911.

Sincerely,

ORIGINAL SIGNED BY:
Paul W. O'Connor
for B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosures: As stated

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See next page

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Mr. J. F. Opeka
Northeast Nuclear Energy Company

Millstone Nuclear Power Station
Unit No. 3

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ENCLOSURE 1

PRELIMINARY SAFETY EVALUATION

MILLSTONE NUCLEAR POWER STATION, UNIT 3

DOCKET NO. 50-423

NUREG-1031
Supplement No ~~2~~ 3

Safety Evaluation Report

related to the operation of Millstone Nuclear Power Station, Unit No. 3

Docket No. 50-423

Northeast Nuclear Energy Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

~~June~~ October 1985

NUREG-1031, Supp. No. ~~2~~ 3 SER RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3 ~~JUNE~~ October 1985

ABSTRACT

and Supplement 2 issued in September 1985

This report supplements the Safety Evaluation Report (NUREG-1031) issued in July 1984, ~~and~~ Supplement 1 issued in March 1985 by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission with respect to the application filed by Northeast Nuclear Energy Company (applicant and agent for the owners) for a license to operate Millstone Nuclear Power Station, Unit No. 3 (Docket 50-423). The facility is located in the Town of Waterford, New London County, Connecticut, on the north shore of Long Island Sound.

This supplement provides more recent information regarding resolution or updating of some of the open and confirmatory items and license conditions identified in the Safety Evaluation Report.

1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

In July 1984 the U.S. Nuclear Regulatory Commission (NRC) staff issued its Safety Evaluation Report (SER)(NUREG-1031) on the application filed by Northeast Nuclear Energy Company (applicant), acting as agent and representative for the owners for a license to operate Millstone Nuclear Power Station, Unit No. 3, Docket No. 50-423. The SER was supplemented in March 1985 by Supplement 1 (SSER 1), and in September 1985 by Supplement 2 (SSER 2) which documented the resolution of several outstanding and confirmatory items and license conditions in further support of the licensing activities. This report is Supplement ~~2~~ 3 to that SER (SSER ~~2~~ 3).

SSER ~~2~~ 3 provides more recent information regarding the resolution or updating of some of the outstanding and confirmatory items and license conditions identified in the SER and its supplement.

Each of the following sections or appendices is numbered the same as the corresponding SER section or appendix that is being updated. Each section is supplementary to and not in lieu of the discussion in the SER, unless otherwise noted. Appendix A continues the chronology of the staff's actions related to the processing of the Millstone 3 application. Correspondence between the applicant and the NRC staff is listed chronologically in this appendix. Appendix B lists references cited in this report.* Appendix D contains abbreviations used in this supplement, and Appendix F lists principal staff members who contributed to this supplement. Appendix H contains errata to the SER. ~~Appendix J reproduces a report prepared for NRC by EG&G Idaho, Inc., "Control of Heavy Loads at Nuclear Power Plants, Millstone Unit 3."~~

App. H?

Copies of this SER supplement are available for inspection at the NRC Public Document Room at 1717 H Street, N.W., Washington, D.C., and at the local Public Document Room of the Waterford Public Library, Rope Ferry Road, Route 156, Waterford, Connecticut.

The NRC Project Manager for Millstone 3 is Ms. Elizabeth L. Doolittle. Ms. Doolittle may be contacted by writing to her at the Division of Licensing, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

1.5 Outstanding Items

The staff identified certain outstanding items in the SER that had not been resolved with the applicant. The status of these items is listed Table 1.1 (an updated version of SER Table 1.3) and is discussed further in the sections of this report as indicated. If the staff has completed its review of an item, the notation "closed" so indicates. The staff will complete its review of unresolved items before the operating license is issued. Resolution of each of these unresolved items will be discussed in future supplements to the SER.

*Availability of all material cited is described on the inside front cover of this report.

1.6 Confirmatory Items

The staff identified confirmatory items in its SER that required additional information to confirm preliminary conclusions. The status of these items is listed in Table 1.2 (an updated version of SER Table 1.4) and is discussed further in the sections of this report as indicated.

If the staff has completed its review of an item, the notation "closed" so indicates.

1.7 License Condition Items

In Section 1.7 of the SER, the staff identified seven license conditions. These include several issues that must be resolved by the applicant as a condition for issuance of an operating license, and other longer term resolution issues that will be cited in the operating license issued, to ensure that NRC requirements are met during plant operation.

The current status of license conditions is in Table 1.3 (an updated version of SER Table 1.5).

1.10 Summary of Principal Review Matters

Table 1.4 (an updated version of SER Table 1.6) lists completed and estimated licensing, construction, and operation milestones.

Table 1.1 Listing of outstanding items (revised from SER Table 1.3)

Item	Status	Section*
(1) Internally generated missiles	Closed (SSER 1)	
(2) Diesel generators	Under review	3.5.2
(3) Protection against postulated pipe breaks outside containment	Under review	
(4) Loading combinations	Closed (SSER 1)	
(5) Design and construction of component supports	Closed (SSER 1)	
(6) Inservice testing of pumps and valves	Under review	
(7) Equipment qualification	Under review	3.10.1, 3.10.2
(8) Flow measurement capability	Under review	4.4.4.2
(9) Loose parts detection program	Closed (SSER 3) ✓ ~~Awaiting information~~	4.4.5
(10) Subcompartment analysis	Awaiting information	6.2.1.2
(11) Mass and energy release analysis	Changed to confirmatory item 71 (SSER 2)	6.2.1.4
(12) Volumetric inspection of Class 2 components	Closed (SSER 2)	6.6.3
(13) Power-operated relief valve and block valve, compliance with NUREG-0737	Closed (SSER 3) ✓ ~~Under review~~	8.3.3.4
(14) Fire protection	Under review	9.5.1
(15) Functional capability of ac and dc emergency lighting	Closed (SSER 3) ✓ ~~Under review~~	9.5.3
(16) Shift technical advisor training program and operating experience for startup	Under review	13.1.2
(17) Emergency Plan	Under review ✓ ~~Awaiting information~~	
(18) Limitation on overtime	Closed (SSER 2)	13.5.1.2
(19) Q list	Closed (SSER 3) ✓ ~~Awaiting information~~	17
(20) Detailed Control Room Design Review	Awaiting information	—

*Section of this supplement where item is discussed.

Table 1.2 Listing of confirmatory items (revised from SER Table 1.4)

Item	Status	Section*
(1) Plant's seismic capability beyond design basis	Closed (SSER 3) ~~Under review~~	2.5.2.7
(2) Dynamic loading	Closed (SSER 3) ~~Under review~~	2.5A.3.2
(3) Liquefaction potential	Closed (SSER 3) ~~Under review~~	2.5.4.4
(4) Shoreline slope	Closed (SSER 3) ~~Under review~~	2.5.5.1
(5) Turbine maintenance program	Awaiting information	
(6) Barrier design procedures	Closed (SSER 1)	
(7) Inservice examination of all pipe welds in break exclusion area	Awaiting information	
(8) Jet impingement effects	Awaiting information	
(9) Ultimate capacity of containment	Closed (SSER 1)	
(10) Design of spent fuel racks	Closed (SSER 3) ~~Under review~~	3.8.4
(11) Program evaluation related to TMI Item II.D.1	Awaiting information	
(12) Predicted cladding collapse time	Deleted (SSER 1, Appendix H)	
(13) Fuel assembly mechanical response	Closed (SSER 3) ~~Under review~~	4.2.3.3
(14) Margins itemized in WCAP-8691	Closed (SSER 3) ~~Under review~~	4.4.4.1
(15) Thermal-hydraulic analyses to support N-1 loop operation	Under review	
(16) Control rod drive structural materials	Under review	
(17) ASME Code cases for Section III, Class I, components	Closed (SSER 2)	5.2.1.2
(18) Yield strength of austenitic stainless steels in reactor coolant pressure boundary	Under review	—

*Section of this supplement where item is discussed.

Table 1.2 (Continued)

Item	Status	Section*
(19) Onsite demonstration of ultrasonic inspection	Closed (SSER 3) ~~Under review~~	5.2.4
(20) Preservice inspection program review and relief requests	Awaiting Information ~~Under review~~	
(21) Preservice and inservice inspection of steam generators	Closed (SSER 3) ~~Under review~~	5.4.2.2
(22) Containment liner weld channel venting	Closed (SSER 2)	6.2.6
(23) Maximum external differential pressure on containment	Awaiting information	
(24) Minimum containment pressure for emergency core cooling system	Closed (SSER 2)	6.2.1.5
(25) Procedures for actuating hydrogen recombiner	Awaiting information	
(26) Secondary enclosure building	Awaiting information	
(27) Sump flow approach velocity	Under review	
(28) Compliance with GDC 51	Closed (SSER 3) ~~Under review~~	6.2.7
(29) Cable separation in nuclear steam supply system process cabinets	Closed (SSER 1)	
(30) Design modification for automatic reactor trip using shunt coil trip attachment	Closed (SSER 2)	7.2.2.4
(31) Reactor coolant pump underspeed trip	Under review	
(32) Conformance with Branch Technical Position ICSB-26	Closed (SSER 1)	
(33) Test of engineered safeguard P-4 interlock	Closed (SSER 1)	
(34) Steam generator level control and protection	Closed (SSER 2)	7.3.3.4

*Section of this supplement where item is discussed.

Table 1.2 (Continued)

Item	Status	Section*
(35) Confirmatory test related to IE Bulletin 80-06	Closed (SSER 2)	7.3.3.5
(36) Control building isolation reset	Closed (SSER 2)	7.3.3.8
(37) Power lockout feature for motor-operated valves	Closed (SSER 1)	
(38) Failure mode and effects analyses of engineered safety features actuation system	Closed (SSER 1)	
(39) Non-Class 1E control signals to Class 1E control circuits	Closed (SSER 2)	7.3.3.11
(40) Sequencer deficiency report	Closed (SSER 2)	7.3.3.13
(41) Balance-of-plant instrumentation and control system testing capability	Closed (SSER 2)	7.3.3.14
(42) Instrument accuracy related to Positions [Attachments] 4, 5, and 6, NUREG-0737, Item II.F.1	Closed (SSER 2)	7.5.2.4
(43) Description and analysis demonstrating compliance with GDC 5	Closed (SSER 1)	
(44) Physical separation of offsite circuits within a common right of way	Closed (SSER 3) ~~Under review~~	8.2.2.1
(45) Physical separation of offsite circuits between switchyard and Class 1E system	Closed (SSER 3) ~~Under review~~	8.2.2.2
(46) Generation rejection scheme	Under review	
(47) Description and analysis demonstrating compliance with GDC 17	Closed (SSER 1)	
(48) Description and analysis demonstrating compliance with GDC 18	Closed (SSER 1)	—

*Section of this supplement where item is discussed.

Table 1.2 (Continued)

Item	Status	Section*
(49) Positive statement of compliance with BTP PSB-1	Closed (SSER 1)	
(50) Compliance with Position 1 of BTP PSB-1	Closed (SSER 3) ✓ ~~Under review~~	8.3.1.3
(51) Adequacy of station electric distribution system voltage	~~Closed (SSER 3)~~ ✓ ~~Under review~~	8.3.1.5
(52) Routing of power cables in the cable spreading area	Deleted (SSER 1, Appendix H)	
(53) Battery charger and transformer used as isolation devices	Closed (SSER 3) ~~Under review~~	8.3.3.3.10
(54) Design criteria of associated circuits from isolation device to load	Deleted (SSER 1, Appendix H)	
(55) Core damage procedure (II.B.3, Criterion 2)	Closed (SSER 1)	
(56) Control of concrete dust	Closed (SSER 3) ✓ ~~Under review~~	
(57) Qualification of engine-mounted control panels	✓ Closed (SSER 3) ~~Under review~~ ([illegible] by RS)	
(58) 7-day fuel oil of storage of each diesel generator	Under review	
(59) Airborne radioactivity monitoring	Under review	
(60) Process control program for solidification of wet wastes	Closed (SSER 3) ✓ ~~Under review~~	11.4
(61) TMI Action Plan Item II.F.1.1	Closed (SSER 3) ✓ ~~Under review~~	11.5
(62) TMI Action Plan Item I.C.1 - procedures generation package nuclear steam supply system	Under review	
(63) Physical Security Plan	~~Closed (SSER 3)~~ ✓ ~~Under review~~	13.6
(64) Initial test program	Awaiting information	

*Section of this supplement where item is discussed.

Table 1.2 (Continued)

Item	Status	Section*
(65) Reactor coolant pump trip during loss-of-coolant accident	Under review	
(66) TMI Action Plan Item III.D.1.1	Awaiting information	
(67) Analysis of dropped control rod	Closed SSER3 ✓ ~~Under review~~	15.4.3
(68) Steam generator tube rupture	Under review	
(69) No failure in emergency core cooling system (ECCS) is not most limiting case in evaluating ECCS	Deleted (SSER 2)	Appendix H
(70) QA program commitments	Awaiting Information ~~Under review~~	
(71) Mass and energy release analysis	Under review	6.2.1.4

*Section of this supplement where item is discussed.

Table 1.3 Listing of license conditions (revised from SER Table 1.5)

Item	Status	Section*
(1) Instrumentation for monitoring post-accident conditions, RG 1.97, Rev. 2, requirements	Under review	
(2) Compliance with NUREG-0612 ("Heavy Load Handling")	Closed (SSER 2)	9.1.5
(3) Installation of postaccident sampling system	Unchanged	
(4) Sediment control during fuel oil storage tank refill	Closed (SSER 3) ✓ ~~Under review~~	9.5.4.2
(5) Moisture in air start system	~~Un~~changed	9.5.6
(6) Preheating of rocker arm lubrication oil system	Closed (SSER 3) ✓ ~~Under review~~	9.5.7
(7) Blockage of access hatch in diesel generator exhaust system	Closed (SSER 3) ✓ ~~Under review~~	9.5.8
(8) Plant-specific analyses utilizing NOTRUMP (TMI Item II.K.3.31)	Added (SSER 2)	15.9.13

*Section of this supplement where item is discussed.

Table 1.4 Major licensing, construction, and operation milestones (revised from SER Table 1.6)

Milestone	Date
Limited work authorization (LWA) issued	May 1, 1974
Site work commenced	June 5, 1974
Construction permit issued	August 9, 1974
Estimated commercial operation date changed from November 1979 to 1982 (applicant)*	December 1975
Estimated commercial operation date changed from 1982 to May 1, 1986 (applicant)*	October 1977
Safety Analysis Report docketed	February 2, 1983
Safety Evaluation Report issued	July 1984
ACRS full committee meeting	September 1984
Safety hearings	None
Ready for fuel loading (applicant)	November 1985*

*Announced delays were a result of applicant's inability to raise the necessary capital in time to maintain the construction schedule.

ATTE E| Held Control, 29489

ATTACHMENT 1

NORTHEAST NUCLEAR ENERGY CORPORATION
MILLSTONE UNIT NO. 3
SUPPLEMENTAL SAFETY EVALUATION REPORT

INSERVICE INSPECTION SECTION
MATERIALS ENGINEERING BRANCH

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

This evaluation resolves the part of Confirmatory Item No. 19, NUREG-1031, concerning the demonstration of the adequacy of ultrasonic inspection techniques for cast stainless steel pipe. The applicant has not yet submitted his request for relief from the ASME Code preservice inspection (PSI) requirements; therefore, the PSI review is still incomplete.

5.2.4.3 Evaluation of Millstone Nuclear Power Station Unit 3 Compliance with 10 CFR 50.55a(g)

The NRC staff and the nondestructive testing industry are in general agreement that performing ultrasonic testing of cast stainless steel is extremely difficult because of the poor acoustical properties of the materials of construction. The staff has therefore adopted the practice of requesting confirmatory demonstrations to determine if the particular material of construction at a given plant has adequate acoustical properties to permit a valid examination with state-of-the-art instrumentation.

5.2.4.4 Ultrasonic Examination Demonstrations

On November 9, 1984, members of the Headquarters and Region I staff and their consultants from Pacific Northwest Laboratories and Oak Ridge National Laboratory observed a demonstration at the plant site of the capability of the applicant's ultrasonic examination procedure and instrumentation to detect actual flaws and artificial reflectors.

After the staff and its consultants observed the examination of four welds during the plant site demonstration on November 9, 1984, the staff reached the preliminary conclusion that the ultrasonic transducer was not penetrating the weld from the elbow side and the examination of the pipe was marginal because of the high electronic noise relative to the signal that must be interpreted. This was based on the observation that there was lack of a strong and consistent back-wall reflection around the circumference with the 0° transducer.

The applicant disagreed that a persistent 0° back-wall signal is necessary to ensure that an adequate angle beam examination could be made or that this should be the determinant that a valid inspection is possible. The applicant also pointed out that during the same in-plant demonstration, they had shown the ability of their ultrasonic procedure and equipment to detect significant flaws in fatigue cracked specimens of cast stainless steel. This was observed by the staff. The applicant has also provided evidence of penetration of the ultrasound during the actual preservice examination by submitting data sheets in which signals from counterbore and ID geometry are noted.

As a result of the November 1984 demonstration, a detailed request for additional information was submitted to the applicant. This information was supplied by the applicant in letters of May 7, 1985, July 1, 1985, and July 2, 1985. These responses were reviewed by the Headquarters and the Regional staff. To resolve these matters, a Regional Inspector was sent to the site for a further demonstration on June 6, 1985.

The Region I Inspection Report, No. 50-423/85-22, details that three representative samples of Millstone centrifugally cast piping containing 15% throughwall depth fatigue cracks were examined with the defects verified. In the in-plant demonstration, one of the same welds tested in the November 1984 demonstration was again ultrasonically examined. It was

reported that an ID reflector was detected, but that transducer wobble, beam spread and redirection of the beam in the cast stainless grain structure caused its location to be misreported. This could perhaps explain the disagreement as to if there was ultrasonic penetration during the November 1984 demonstration. The inspector concluded that the ultrasonic examination was adequate, in that, a significant indication could have been detected using the applicant's method of ultrasonic examination, if such a defect were present.

5.2.4.5 Calibration Block

The staff also questioned whether the one basic calibration block used by the applicant was adequate to establish the ultrasonic parameters for the inspection of both the statically cast stainless steel elbows, the centrifugally cast stainless steel pipe and also the ferritic steam generator nozzle to elbow welds. The applicant stated that the calibration block used for the PSI examinations meets the material and product form requirements of Section XI in that it was of the same specification (SA-351 Grade CF8A) of those materials in the main loop piping, which is one of the materials being joined by the weld (i.e., elbow to steam generator nozzle). With respect to the thickness of the calibration block, the wall thicknesses in the examination volumes of the pipes examined were within ± 1 inch of the calibration block thickness. The Regional Inspector determined that the calibration block was nominally 6 dB more attenuative than the cast pipe and 1 dB to 2 dB more attenuative than the cast elbow. This means that the use of the block was generally twice as conservative in the scanning sensitivity as that required for the pipe and slightly more conservative as required for the elbow. This should adequately compensate for any nominal difference in thickness between the calibration block and the pipes examined.

5.2.4.6 Future Inservice Inspections

Northeast Utilities, as a member of the Westinghouse Owners Group, is participating in a development program designed to improve the current defect detection and characterization reliability for inspection of main coolant loop piping systems. The prime emphasis will be on field usable inspection techniques and data processing systems. The program places emphasis on carefully prepared test samples which are designed to represent actual field conditions. The applicant has stated he will perform examinations on main coolant loop welds using the best available technology. In addition, in preparing the Inservice Inspection Program, he will select welds which show the best acoustical properties and have the best access for ultrasonic examination of the weld and required volume in accordance with staff positions.

5.2.4.7 Conclusions

Based on the above, the staff has reached the following conclusions regarding the ability to perform preservice and inservice inspections of the cast stainless steel pipe at Millstone, Unit 3:

1. The examination procedure, instrumentation and calibration standard meet the requirements of the ASME Boiler and Pressure Vessel Code, Section XI.

2. The examination results were meaningful, i.e., significant defects, if present, could have been detected in the volume required to be examined by the Code.

3. The applicant's participation in the Westinghouse Owners Group primary coolant piping examination research and development programs and his commitment to use the best available technology to perform main coolant loop weld inspections provides assurance that future inservice inspections will be adequate.

4. The basic objective of inservice inspections of the piping welds in the reactor coolant boundary is to perform a repetitive examination of a representative sample of welds in order to detect generic service-induced degradation. In addition to providing a baseline, the preservice examinations will identify those welds which will optimize the effectiveness of future inservice examinations. This will be implemented by the applicant's commitment to select welds which show the best acoustical properties and have the best access for ultrasonic examination of the weld and required volume for future Inservice Inspection Programs.

The staff has determined that the cast stainless steel pipe and elbow welds at Millstone 3, as demonstrated, have sufficiently good acoustical properties to permit a valid ultrasonic examination with state-of-the-art instrumentation. Since the applicant has committed to perform future examinations using the best available technology and to select welds with the best (19) acoustical properties and access, we consider ~~the~~ confirmatory items regarding preservice examinations of welds in cast stainless piping to be resolved.

5.4.2.2 Steam Generator Tube Inservice Inspection

5.4.2.2.1 Compliance with the Standard Review Plan (SRP)

The July 1981 edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (NUREG-0800) includes acceptance criteria recommending that the applicant perform inspections based on Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes" and the applicable Standard Technical Specifications, NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors."

When we issued our Safety Evaluation, NUREG-1031, July 1984, the applicant had not yet submitted his proposed Plant Technical Specification. Therefore, the review of compliance with our criteria was identified as Confirmatory Item No. 21. We have now completed this review. We find that the applicant's inservice inspection program complies with the recommendations in Regulatory Guide 1.83 and the applicable Standard Technical Specification concerning provisions for a baseline examination, selection, and sampling of the tubes, inspection intervals, actions to be taken in the event defects are identified and reporting requirements. In addition, by letter of June 25, 1985, the applicant stated that to supplement the requirements for eddy current testing of steam generator tubes contained within the technical specifications, he would inspect the full length of the tubes for the initial three percent sample of an inservice inspection.

The staff concludes that the inservice inspection program of steam generator tubes, confirmatory item (21) is acceptable and meets the inspection and testing requirements of General Design Criterion 32. This conclusion is based on the applicant following the recommendations in Regulatory Guide 1.83, and the Standard Technical Specifications, NUREG-0452, as reviewed by the staff and determined to be appropriate for this application.

QAB / Bill Beine X24512

Enclosure 1

Millstone 3 - SSER 2
QA Input

17 QUALITY ASSURANCE

17.1 General

FSAR Amendment 13 updated FSAR Section 17.2 so that it now references Revision 6 of the "Northeast Utilities Quality Assurance Program Topical Report" (NU-QA-1) instead of an earlier revision. Revision 6 of NU-QA-1 has been reviewed and found acceptable by the staff.

17.2 Organization

The applicant's organization for QA is basically that reported in Chapter 17 of the Millstone 3 SER. However, in Figure 17.1, (a) the block shown as Vice President Purchasing and Materials Management is now Systems Director Purchasing and Materials Management, (b) the block shown as Vice President System Transmission & Distribution Engineering & Operations is now Vice President Transmission & Distribution Engineering & Operations, and (c) the Betterment Construction QA block is now under the Supervisor Construction QA block instead of the Supervisor Design and Operations QA block.

The staff finds the applicant's organization for QA continues to be acceptable.

17.3 Quality Assurance Program

This section of the Millstone 3 SER is unchanged.

17.4 Conclusion

The staff's conclusion reported in the Millstone 3 SER -- that Northeast Utilities description of the QA program for operations is in compliance with applicable NRC regulations, meets the requirements of Appendix B to 10CFR50, and is acceptable -- is unchanged.

The applicant has acceptably revised either the FSAR or its topical report on QA to include the applicable responses to staff questions on QA and, as noted in 17.1 above, has updated the FSAR QA commitment to Rev. 6. This could close confirmatory item 70, "QA Program Commitments," discussed on page 17-4 of the SER. However, FSAR Amendment 12 added the following alternative to the applicant's commitment to comply with R. G. 1.123:

> Certain standard catalogue or non-engineered items may be procured without seller qualification as described in Section 7 of the Millstone 3 Quality Assurance Program Manual referenced in FSAR Section 7.1.2.

The staff believes it is the applicant's intent to use this alternative during the operations phase of Millstone 3. However, FSAR Section 17.1.2 ~~(and 17.1.2, if there has been a misprint)~~ does not address QA for operations. Further, the

quoted alternative references the Millstone 3 Quality Assurance Program Manual which is not reviewed by the staff ~~Licensing Section of the QA Branch~~ and which can be changed without NRC notification. Therefore, the applicant should describe its controls for the procurement of "certain standard catalog or non-engineered items" in either the Millstone 3 FSAR or in the Northeast Utilities Quality Assurance Program Topical Report (NU-QA-1) which is referenced in FSAR Section 17.2. The controls should address item 7B4 on page 17.1-16 of the Standard Review Plan (NUREG-0800) which states: "For commercial 'off-the-shelf' items where specific quality assurance controls appropriate for nuclear applications cannot be imposed in a practicable manner, special quality verification requirements shall be established and described to provide the necessary assurance of an acceptable item by the purchaser." This question of quality verification requirements for procurement of "certain standard catalog or non-engineered items" requires clarification and is one of two unresolved parts of confirmatory item 70. The second part is discussed below.

Outstanding item 19, "Q list," is closed, since differences between the staff and the applicant regarding the Q list have eliminated. However, as the second of two unresolved parts of confirmatory item 70, the staff will require that the applicant show the major components of the supplementary leak collection and release system and of the engineered safety features filter systems with a comparable level of detail as found in FSARs for other recently licensed nuclear power plants.

Sectio J N. Cnoton x28967

ENCLOSURE

SAFETY EVALUATION REPORT SUPPLEMENT

MILLSTONE NUCLEAR POWER STATION UNIT 3

DOCKET NO. 50-423

STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH

STRUCTURAL ENGINEERING SECTION A

3.8.4 Other Seismic Category I Structures

In the Safety Evaluation Report (SER), it was noted that the applicant had informed the staff that the design of the spent fuel pool racks complied with the current staff acceptance criteria (SRP Section 3.8.4, Appendix D). The staff's intention to review the information which will confirm such compliance was also indicated in the SER.

By a letter dated May 20, 1985, the applicant has now provided details of its analysis/design of the spent fuel racks. In particular, these details include: (1) description of the fuel rack assembly; (2) models and procedures for seismic analysis; (3) loads and load combinations for structural analysis; (4) sliding and overturning analysis; and (5) structural acceptance criteria.

The staff review of this information indicates that the applicant's seismic analysis is consistent with the staff acceptance criteria accounting for the nonlinearities resulting from the gap between the fuel cell and the fuel assembly, the boundary conditions of the fuel rack support locations and energy losses at the support locations. Thus, the nonlinear model accounts for fuel to rack impact loading, support pad lift off, hydrodynamic forces, and the nonlinearity of sliding friction interfaces. The sliding and overturning analysis indicates that the impact between adjacent rack modules, and rack module and pool wall is prevented. The factor of safety against overturning is much greater than the staff acceptance criteria. The load combinations and acceptance criteria are in accordance with the earlier staff position paper "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," and they also meet the requirements of SRP Section 3.8.4, Appendix D using the 1980 edition, winter 82 addendum of subsection NF of the ASME Boiler and Pressure Vessel Code, Section III, Division 1.

Based on the above findings, the staff concludes that the design of the spent fuel pool racks at Millstone 3 complies with the intent of the staff acceptance criteria and, therefore, ~~this issue~~ is resolved.
confirmatory item (10)

SGEB | J. CHEN 128153

ENCLOSURE

MILLSTONE UNIT NO 3
DOCKET NO: 50-423

SUPPLEMENTAL SER INPUT -- GEOTECHNICAL ENGINEERING
PREPARED BY: J. CHEN, GES, SGEB, DE

The following paragraphs summarize the staff's further evaluation of the confirmatory issues noted in our SER (Sections 2.5.4 and 2.5.5 of NUREG - 1031, July 1984). These issues pertain to (1) The dynamic analysis for the emergency generator enclosure building, (2) the impact of the beach sand deposits on the function of the pump house under SSE condition, and (3) the design adequacy of the west retaining wall. The evaluation is based on the review of additional information submitted by the applicant after the issuance of the SER (NUREG-1031, July 1984). The staff's evaluation of these items is in accordance with the criteria outlined in the current standard Review Plan (SPR) (NUREG - 0800) and current licensing policy.

SUPPLEMENTAL SAFETY EVALUATION INPUT

2.5.4.3.2 Dynamic Loading

As stated in our SER, in a letter dated June 26, 1984 the applicant ~~promised in June 13, 1984 that~~ committed to perform additional analysis incorporating the as built soil/foundation condition ~~would be performed~~ to confirm the design of the emergency generator enclosure (EGE) building.

During the meeting of May 14, 1985, the applicant indicated ~~revealed~~ that it had re-examined the locations of field density tests and discovered errors in locating some of the test samples during construction. The applicant stated ~~promised~~ that corrections to the FSAR will be made in a future amendment to reflect the relocation ~~this discovery~~. The re-location of those test samples confirms that the majority of the strip footings of the EGE building were founded on basal till with one strip footing founded on a few feet of structural fill over till. The presence of this small amount of structural fill is judged to have insignificant effect on the seismic response of the EGE building.

The staff accepts the applicant's explanation that errors were probably made during construction in locating some of the test samples and concurs with the applicant's assessment that the presence of small amount of structural fill would have no impact on EGE building. The soil-structure-interaction analysis performed for EGE building presented in the FSAR is ~~therefore~~ acceptable to the staff. and therefore confirmatory item (2) is resolved.

2.5.4.4 Liquefaction Potential

As stated in our SER, the applicant's analyses performed in May 1984, indicated that the beach sand deposits might be liquefiable under seismic condition and the effects of liquefied beach sands on the functionality of the pumphouse need to be assessed.

The applicant has performed additional analysis assuming that liquefaction which
The staff has reviewed this information ~~and~~
of beach sand deposits would take place under SSE condition. ~~The analysis~~ concludes that
shows that the effect of the liquefied beach sand deposits would not impair the functionality of the pumphouse.

The staff concurs with the applicant's assessment that a liquefaction - induced flow slide of beach sand deposits will not adversely affect the supply of emergency cooling water. Confirmatory item (3) is resolved.

2.5.5.1 Shoreline Slope

As stated in our SER, additional information is required from the applicant to justify the adequacy of the retaining wall design.

(to justify adequacy of the retaining well design)

The applicant has provided additional information in (Ref. 1 & 2) ~~which confirms~~
The staff has reviewed this information and concludes
~~to the staff's satisfaction~~ that the design of the retaining wall is adequate.
Confirmatory item (4) is resolved.

CONCLUSIONS:

Based on the information submitted (Ref. 1 & 2), the staff concludes that (1) the liquefaction of beach sand deposits under SSE condition would not affect the safety function of the pumphouse, (2) the foundation model used for the seismic analysis of the EGE building is acceptable, and (3) the west retaining wall design is acceptable.

REFERENCES: 1. Letter from W. Counsil of NUSCO to B. Youngblood of NRC, dated October 18, 1984, SUBJECT: RESPONSES TO SAFETY EVALUATION REPORT CONFIRMATORY ITEMS

2. Stability and Design Evaluation of West Retaining Wall, Stone & Webster Engineering Corp. Transmitted May 15, 1985

MCTB | Jay Lee x27633

METEOROLOGY AND EFFLUENT TREATMENT BRANCH INPUT TO
SUPPLEMENTAL SAFETY EVALUATION REPORT NO. 2
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

11.4 Solid Radioactive Waste Management System ~~(SER Confirmatory Item No. 60)~~

Process Control Program

In the SER, the staff concluded that the proposed solid radwaste system at Millstone, Unit No. 3, is acceptable provided that the applicant provide an acceptable process control program (PCP) complete with a compliance program to meet the requirements set forth in 10 CFR Part 61.

The applicant submitted a "Process Control Program" dated June 1985 for Millstone Units 1, 2, and 3. The PCP has not been revised from that submitted for Millstone Unit Nos. 1 and 2 except to indicate that it is also applicable to Millstone Unit No. 3.

The applicant states in the PCP that all solidified radioactive wastes will meet the requirements set forth in 10 CFR Parts 20 and 61. We find therefore, the Millstone Unit Nos. 1, 2, and 3 PCP to be acceptable on an interim basis. The acceptability of this process control program is based on currently available guidelines, but a future revision should address full compliance with 10 CFR Part 61 when revised process control program guidance becomes available from NRC.
Confirmatory item (60) is resolved.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

Item II.F.1 Attachment 1, Noble Gas Effluent Monitor ~~(SER Confirmatory Item No. 61)~~

In the SER, we concluded that the high range noble gas monitoring systems to be installed at Millstone Unit 3 meet the requirements of Items (1), (2), (3), and (4) and Table in II.F.1-1 in NUREG-0737 and meet the intent of the guidelines in RG 1.97, Revision 3.

We further concluded however, that the applicant should provide additional information on the monitor calibration method, detector energy response characteristics, and calculational method to be used for converting instrument readings to release rate as a function of time after an accident.

The applicant has provided this additional information in its letter dated June 6, 1985. In it, the applicant states that the monitors will be calibrated using four check sources in the actual detector geometry during each refueling outage or every 18 months.

We have determined that the applicant has the capability to develop time dependent correction factors which could be used in the Millstone 3

off-site dose calculational procedure. The correction factors will take into account (1) an expected isotopic composition of noble gases as a function of time after an accident, (2) detector counting efficiencies for each noble gas isotope, and (3) noble gas isotopic gamma energies. These correction factors can be used to convert the monitor readouts to actual radioactive gaseous releases.

We find that the applicant's method of caliberation and calculational method to be used for converting instrument readings to release rates as a function of time after an accident to be acceptable. Therefore, the staff concludes the noble gas monitors installed at Millstone Unit No. 3 meets the requirements of TMI Action Plan Item II.F.1, Attachment 1 and the guidlines provided in Regulatory Guide 1.97, Rev. 3. Confirmatory item (61) is resolved.

SSEB / N Chokshi x28967

ENCLOSURE
SAFETY EVALUATION REPORT SUPPLEMENT
MILLSTONE NUCLEAR POWER STATION UNIT 1
DOCKET NO. 50-423
DIVISION OF ENGINEERING

2.5.2.7 Safe Shutdown Earthquake

2.5.2.7.2 Estimates of Seismic Capacity at Millstone 3 based on Probabilistic Study

In the SER, the staff reaffirmed the seismic design basis (Newmark type response spectrum anchored at 0.17g) approved for Millstone 3 at the CP stage. However, there was sufficient uncertainty associated with the causes of the 1982, body wave magnitude 5.75, New Brunswick earthquake that the staff made a limited evaluation of the ground motion resulting from reoccurrence of that size earthquake in the vicinity of Millstone 3. In order to address this uncertainty, the staff utilized the preliminary insights gained from the Millstone 3 Probabilistic Safety Study (PSS) to conclude in the SER that the contributions to core melt from the seismic hazard for peak accelerations less than 0.30g are small and that differences of 40% to 50% in ground motion at accelerations less than or equal to about 0.17g (SSE level) to 0.25g (assumed ground motion level for nearby magnitude 5.75 event) are not significant when viewed from the perspective of risk. However, the staff also stated in the SER that as confirmation of the conclusion drawn, the staff will require the

applicant to utilize the results of the PSS to document the seismic capability, at acceleration up to 0.25g, with high confidence of low probability of failure for individual controlling failure modes of structures and equipment. In addition, the applicant was required to assess plant fragilities for various acceleration levels considering those risk scenarios that include the majority of seismic risk to the plant.

By a letter dated December 6, 1984, the applicant submitted a report titled "A Program to Determine the Capability of the Millstone 3 Nuclear Power Plant to Withstand Seismic Excitation Above the Design SSE" (Ref. 1) for the staff review. The report contained information on the following aspects:

1. Identification of dominant contributors to seismic risk.

2. Evaluation of the high confidence, low frequency of failure accelerations for critical structures and equipment.

3. Evaluation of the high confidence, low frequency of failure accelerations for the dominant plant damage states.

4. Evaluation of the frequencies of occurrence of significant plant damage states from seismic events.

5. Evaluation of the contributions of various acceleration ranges to the frequencies of occurrence of significant plant damage states.

6. Investigation of the sensitivity of the results of the Millstone Unit 3 PSS to variation in the assumptions or analytical models employed for the seismic fragility and hazard development.

The plant damage states for which the seismic initiated accident sequences have been shown to be important contributors are: (Ref.1) TE-Transient (caused by loss of offsite power) with failure of onsite emergency power or RCS heat removal; AE-large LOCA with failure of safety injection and containment quench sprays; SE-small LOCA or seismic ATWS with failure of safety injection and containment quench sprays; and V3-LOCA with containment bypass. The staff's Boolean expressions (relating the plant damage states to the component failures) discussed in the staff evaluation of Millstone 3 PSS (Ref. 2) are almost identical to those presented in the applicant's report. (The major difference is that the staff assigns the reactor coolant pump seal failure on station black out to plant damage state SE, not TE).

The components which dominate the seismic risk for each damage state are shown in Table 1 along with their median acceleration capacities, uncertainties and high confidence-low probability failure (characterized as failure frequency less than 5% with 95% confidence) capacities. It should be noted that the Boolean expressions for the plant damage states contain some components whose capacities may be lower than those shown in Table 1; however, failures of these components by themselves are not critical. Design and construction errors and relay chatter are not explicitly considered in the capacity determination or plant system analysis.

Table 1 Critical Component for Plant Damage States and their Fragilities

Plant Damage State	Component/Structure Failure Mode	(50%-50%) Median Acceleration in g's	Uncertainty Random	Uncertainty Modeling	(95%-5%) High Confidence, Low Frequency, Failure Level (g's)
TE	1. Loss of offsite power	0.20	0.20	0.25	0.10
	2. Emergency generator enclosure building, wall footing failure	0.88	0.20	0.46	0.30
	3. Diesel generator oil cooler-anchor bolt failure	0.91	0.24	0.43	0.30
	4. Control building diaphragm	1.00	0.24	0.43	0.39
	5. Service water pumphouse sliding	1.30	0.24	0.49	0.39
	6. Engineered safe guard feature building. Failure of shear wall near basemat	1.70	0.23	0.43	0.58
AE	1. Reactor Coolant System Piping	1.59	0.48	0.51	0.31
SE	1. Reactor Vessel core geometry distortion	0.99	0.21	0.33	0.35
	2. Control Rod drive system (Failure to Scram)	1.00	0.30	0.38	0.33
V3	1. Containment cranewall failure	2.20	0.39	0.38	0.62

Based on the above discussed structure/component fragilities, the applicant calculated plant damage state fragilities/margin as shown in Table 2

Table 2

FRAGILITIES OF DIFFERENT PLANT DAMAGE STATES

Plant Damage State	(50%-50%) Median Acceleration in g's	High Confidence Low Frequency of Failure Level (g's) (95%-5%)
V3 LOCA w/containment bypass	2.05	0.60
AE Large LOCA with Early Core Melt	1.22	0.45
SE Small LOCA or ATWS with Early Core Melt	0.77	0.40
TE Transient (loss of offsite power) with Early Core Melt	0.61	0.26

The Staff review of the structural/component capacities and plant damage state fragilities indicated that, except for the loss of offsite power, critical structures and components identified in the report have high confidence, low frequency of failure accelerations of at least 0.3g. However, the staff noted that the applicant's analysis did not explicitly address the consequences of seismically induced liquefaction of beach sands nor the stability of the beach sand slope under events greater than SSE which may prevent the intake structure from conveying the cooling water for the safe shut down. In addition, the staff requested the applicant to evaluate the fragility analyses for the emergency generator enclosure (EGE) building and buried service water piping system considering as-built foundation support conditions and variations in the assumed ratio of peak ground velocity to peak ground acceleration.

See Section 2.5.4

The staff met with the applicant on May 14, 1985 to discuss the above issues and by a letter dated July 12, 1985, the applicant provided results of its analyses for the staff review. To address the liquefaction issue, the applicant examined a seismically induced flow slide for three sections through the slope and dredged channel. A final post-flow slope was assumed based on observations from the 1964 Alaska earthquake. The examination of the post-flow channel elevation and the service water pump inlet elevation indicates that a seismically induced flow slide into the intake channel will not adversely affect the supply of water required for cooling of safety-related systems. Although the applicant's analyses did not address specific seismic events above SSE, based on the staff's past review experience and the applicant's study, the staff concludes that the flow slide of beach sand slope due to seismic events up to 0.3g would not be a significant seismically induced failure mode.

The results of revised fragility analysis of EGE building and service water piping system indicate that changes in the median acceleration capacities and high confidence, low frequency of failure levels are minor and, therefore, impacts on the plant damage state frequencies are minimal.

The applicant also presented the plant damage state frequencies obtained by convolving plant damage state fragility curves with the seismic hazard curves (See staff evaluation of the Millstone 3 PSS, Ref. 2, for further discussion on plant damage state frequencies). From these calculations, the applicant also extracted information as to what ranges of acceleration contribute most

significantly to the overall frequency of occurrence of the damage state. For the plant damage state, TE, which is the predominant contributor to the coremelt frequency (in the staff analysis, Ref. 2, SE is the predominant contributor), it was observed that the contribution to 95% confidence frequency of occurrence and the median frequency of occurrence from the acceleration ranges below 0.3g is small.

The applicant conducted a number of sensitivity studies to examine the influence of assumptions made in the component fragility analysis and different seismic hazard models (i.e. Lawrence Livermore study presented in Ref. 3) on the plant damage state frequencies. These studies, in general, indicate that variations considered do not significantly impact the plant's capability to withstand seismic events greater than SSE.

Based on the review of the above information and discussions with the applicant, the staff finds the following:

°Given the use of the same set of hazard curves, the applicant's analysis is in general agreement with that done by the staff in its review of the PSS. The applicant's results appear somewhat more conservative than the staff's for a given set of hazard curves due to neglect of overlap between different plant damage states in the applicant's work (Ref.2). While neglect of overlap increases the overall core damage frequency, it diminishes the percent contribution of the lower acceleration range since plant damage state overlap is most significant at higher accelerations.

°The applicant's plant damage state Boolean expressions (derived from the PSS fault trees) are almost identical to those developed by the staff. The applicant's Boolean expressions are slightly more conservative than the staff's and the staff would place some failures in different plant damage states. However, the differences are minor.

°In general, the critical structures and components at Millstone 3 have high confidence, low frequency of failure accelerations of at least 0.3g (Relay chatter and design/construction errors are not considered in this evaluation. See Ref. 2 for further discussion on relay chatter issue).

°The dominant plant damage state, TE, at Millstone 3 has high confidence, low frequency of failure acceleration greater than 0.25g.

°The analysis of contributions of different peak ground accelerations ranges to the plant damage state frequencies confirms that the contribution of earthquakes of up to 0.3g is not significant.

Based on the above findings, the staff concludes that Millstone 3 and its critical structures and components possess significant margins beyond the design basis SSE and there is a high confidence that the frequency of the plant damage is significantly low at acceleration levels less than 0.25g. Therefore, ~~the~~ confirmatiory issue (1) regarding the plant's seismic capability beyond design basis, i.e. reoccurrence of earthquake the size of 1982 New Brunswick event in the vicinity of the plant, is considered resolved.

References:

1. M. K. Ravindra and others, Structural Mechanics Associates. "A program to determine the capability of the Millstone 3 Nuclear Power Plant to Withstand Seismic Excitation above the design SSE,"NTS/SMA 20601.01-R2, November 1984.

2. "Millstone 3 - Risk Evaluation Report," NUREG-1152, to be published (Draft August 1985).

3. D. L. Bernreuter and others, Lawrence Livermore National Laboratory," Seismic Hazard Characterization of the Eastern United States: Methodology and Interim Results for Ten Sites," NUREG/CR-3756, April 1984.

NTEB / Barry Elliot x27742

ATTACHMENT 1

NORTHEAST UTILITIES
MILLSTONE NUCLEAR POWER STATION, UNIT 3
DOCKET NO. 50-423

MATERIALS APPLICATION SECTION
MATERIALS ENGINEERING BRANCH

6.2.7 Fracture Prevention of Containment Pressure Boundary

In a previous SER input we indicated that ferritic materials that are used in the containment pressure boundary will be reviewed to the fracture toughness criteria for Class 2 components identified in the Summer 1977 Addenda of Section III of the ASME Code. For Class 2 components, the fracture toughness criteria in the Summer 1977 Addenda of Section III of the ASME Code permits the materials to be either Charpy V-notch tested at or below the Lowest Service Temperature, evaluated to the nil-ductility transition temperature requirements of Table NC-2311(a)-1 of the ASME Code, or evaluated using the fracture mechanics methods contained in Appendix G of the ASME Code.

Ferritic materials that are in the Millstone-3 containment pressure-boundary were procured to earlier fracture toughness criteria than those in the Summer 1977 Addenda of the ASME Code. Hence, many materials were not Charpy V-notch tested at or below the Lowest Service Temperature. To demonstrate that these materials meet the review criteria, the applicant used the fracture toughness data presented in NUREG-0577, "Potential for

Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," USNRC, October 1979 and ASME Code Section III, Summer 1977 Addenda, Subsection NC. This data indicates that all materials meet the nil-ductility transition temperature criteria of Table NC-2311(a)-1 except for ferritic materials in the feedwater line.

The ferritic materials in the feedwater line were evaluated using the fracture mechanics methods in Appendix G of the ASME Code. The licensee used a lower bound reference stress intensity factor (26.78 ksi $\sqrt{in.}$) for determining the allowable material fracture toughness. According to Appendix G, the reference stress intensity value used in the analysis would be applicable for ferritic material at 180°F below the materials nil-ductility transition temperature. Additional fracture toughness data for materials with similar composition and heat treatment as the Millstone 3 feedwater materials is reported in a text by Rolfe and Barsom titled, "Fracture and Fatigue Control in Structures, Applications of Fracture Mechanics" (Prentice-Hall, 1977). This data indicates that the reference stress intensity value assumed in the Appendix G fracture mechanics analysis is conservative. The crack sizes assumed in the evaluation were greater than that permitted during the preservice examination of the component and allowed for flaw growth in service. The fracture mechanics analysis

indicates that the ferritic materials in the feedwater line would meet the safety margins recommended in Appendix G of the ASME Code. Additional fracture mechanics analysis performed by the licensee indicates that the critical crack size for brittle fracture would be greater than twice the depth used in the Appendix G analysis.

Based on our review of the available fracture data and material fabrication histories, the use of correlations between metallurgical characteristics and material fracture toughness, and fracture mechanics analysis performed by the licensee, we conclude that the ferritic components in the Millstone 3 containment pressure boundary meet the fracture toughness requirements that are specified for Class 2 components by the 1977 Addenda of Section III of the ASME Code. Compliance with these Code requirements provides reasonable assurance that the Millstone 3 reactor containment pressure boundary will behave in a nonbrittle manner, that the probability of rapidly propagating fracture will be minimized and that the requirements of GDC 51 are satisfied.
Confirmatory item (28) is resolved.

CPB/Amira Gill 27091

ENCLOSURE 1

SUPPLEMENT #2 FOR MILLSTONE UNIT 3 SAFETY EVALUATION REPORT

4.2.3.3(4) Structural Damage from External Forces

In SSER 1, we agreed with the applicant that the Millstone 3 deviation from the approved bounding Westinghouse seismic response curve appeared to be a secondary effect. However, we stated that we would complete the review pending the applicant's further analysis of seismic and LOCA loads on fuel assemblies.

By a letter dated February 1, 1985 from W. G. Counsil (NNECO) to B. J. Youngblood (NRC), the applicant provided the results of a combined seismic-and-LOCA loads analysis including asymmetric blowdown load using the approved methodology described in WCAP-9401. The results show that the combined loads on grids and non-grid components were less than the allowable strengths for Millstone Unit 3.

We, therefore, conclude that the applicant has demonstrated acceptable results for fuel assemblies under combined seismic-and-LOCA conditions, and confirmatory item (13) is resolved.

4.4.4.1 Fuel Rod Bowing

A significant parameter which affects the thermal hydraulic design of the core is rod-to-rod bowing within fuel assemblies. The Westinghouse methods for predicting the effects of rod bow on DNB, WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," have been approved by the staff.

The FSAR stated that there is a 9.1 percent margin to accommodate full and low flow DNBR penalties due to fuel rod bowing. In a previous SER we stated that the applicant should verify that (1) the breakdown of this margin into individual factors is consistent with WCAP-8691, and (2) that this margin (in whole or part) was not used in any other analysis.

Per our request, the applicant inserted into the Bases of the Technical Specifications the markdown of the generic margins that were used to offset the reduction in DNBR due to rod bowing. Also, in a letter dated May 2, 1985 the applicant stated that: "the DNBR margin used to offset the worst case rod bow penalty is not used in any other analysis," therefore, the applicant's use of available margins to offset rod bow penalties is acceptable, and confirmatory item (14) is resolved.

4.4.4.2 Crud Deposition

In response to question Q492.4 and subsequently Q492.7, the applicant stated that the Reactor Coolant System (RCS) flow measurement is based upon performing a precision heat balance flow measurement at the beginning of each fuel cycle and using the result to calibrate the RCS elbow tap flow indicators. In a letter dated September 19, 1984 the applicant described the inspection of the venturis prior to start up of each cycle via ports located upstream and downstream of the venturis. The applicant stated that cleaning will be done by hydrolasing when required. These inspection ports will be installed during the first refueling outage. The applicant stated that if the venturis are not inspected, an additional 0.1% will be added to the total RCS flow measurement uncertainty. We find this acceptable.

In a letter dated July 15, 1985, the applicant stated that the method used in determining error of an instrument loop is the statistical combination of the groups of components in an instrument which are statistically independent. Errors which are not statistically independent are combined arithmetically. The applicant stated that vendor technical manuals and drawings were used as sources of uncertainty. The applicant's results were as follows:

	Four Loops	Three Loops
The total uncertainty in determining core power is	±0.44%	±0.50%
Total RCS flow uncertainty based on a precision heat balance is	±2.31%	±2.32%

	Four Loops	Three Loops
The accuracy of the RCS elbow tap flow indicators in determining total flow is	±.54%	±.54%
Combining the uncertainty of the above gives total RCS flow measurement uncertainty equals	±2.37%	±2.37%
If venturis are not verified clean then total RCS flow measurement uncertainty equals	±2.47%	±2.47%

However, the applicant has not provided enough detail for us to determine that this analysis is valid. For example, the three loop value should be higher than the two loop value. The applicant should provide a breakdown of the total flow uncertainty into its components similar to the analysis provided in the applicant's February 16, 1984 letter on the same subject.

4.4.5 Loose Parts Monitoring System

The applicant has provided a description of the Loose Parts Monitoring System (LPMS) which will be used by the Millstone Unit 3. The design will consist of twelve active instrumentation channels, each comprising a piezoelectric accelerometer (sensor) and signal conditioning equipment. Sensors are fastened mechanically to the reactor coolant system (RCS) at each of the following potential loose parts collection regions:

1. Reactor pressure vessel-upper head region.

2. Reactor pressure vessel-lower head region.

3. Each steam generator-reactor coolant inlet region.

The system will be capable of detecting a metallic loose part that weighs from 0.25 to 0.30 pounds impacting within 3 feet of a sensor and having a kinetic energy of 0.5 foot pounds on the inside surface of the RCS pressure boundary.

In a letter dated August 26, 1985, the applicant submitted, in response to our question 492.5, a report describing operation of the system hardware and implementation of the loose part detection program.

In that report, the applicant agreed to install a second sensor on each steam generator by the end of the first refueling outage. We find that acceptable. We also find the applicant's discussion regarding the LPMS qualification for an operating basis earthquake (OBE) acceptable. The discussion of the calibration of the LPMS every refueling or every 18 months, whichever is greater, and the frequency of the operability checks are also acceptable. Open ~~Confirmatory~~ item (9) is resolved.

However, the staff will ~~still~~ require the applicant to include a technical specification on the operability of the LPMS similar to the generic Westinghouse Technical Specification as shown in Section 3/4.3.3.9.

PSLB / C Gaskin X74752

50-423

ENCLOSURE 2

13.6 Physical Security Plan

13.6.1 ~~0~~ Introduction

The Northeast Nuclear Energy Company acting as agent for the Northeast Utilities has filed with the Nuclear Regulatory Commission for the Millstone Nuclear Power Station Unit 3 the following security plans which have since been amended:

"Millstone Nuclear Power Station Physical Security Plan,"
"Millstone Nuclear Power Station Contingency Plan,"
"Millstone Nuclear Power Station Guard Training and Qualification Plan."

This Safety Evaluation Report (SER) summarizes how the applicant has provided for meeting the requirements of 10 CFR Part 73. The SER is composed of a basic analysis that is available for public review, a protected Appendix, and a protected response force size worksheet.

Based on a review of the subject documents and visits to the site, the staff has concluded that the protection provided by the Northeast Utilities against radiological sabotage at the Millstone Nuclear Power Station meets the requirements of 10 CFR Part 73. Accordingly, the protection provided will ensure that the health and safety of the public will not be endangered.

2.0 Physical Security Organization

To satisfy the requirements of 10 CFR 73.55(b) Northeast Utilities has provided a physical security organization that includes a Security Shift Supervisor who is onsite at all times with the authority to direct the physical protection activities. To implement the commitments made in the physical security, guard training and qualification plan, and the safeguards contingency plan, written security procedures specifying the duties of the security organization members are available for inspection. The training program and critical security tasks and duties for the security organization personnel are defined in the "Millstone Nuclear Power Station Guard Training and Qualification Plan" which meets the requirements of 10 CFR Part 73, Appendix B for the training, equipping and qualification of the security organization members. The physical security plan and the training program provide commitments that preclude the assignment of any individual to a security related duty or task prior to the individual being trained, equipped and qualified to perform the assigned duty in accordance with the approved guard training and qualification plan.

3.0 Physical Barriers

In meeting the requirements of 10 CFR 73.55(c) the licensee has provided a protected area barrier which meets the definition of 10 CFR 73.2(f)(1). An isolation zone, to permit observation of activities along the barrier, of at least 20 feet is provided on both sides of the barrier with the exception of

the locations listed in the Appendix. The staff has reviewed those locations and determined that the security measures in place are satisfactory and continue to meet the requirements of 10 CFR 73.55(c).

Illumination of 0.2 foot-candles is maintained for the isolation zones, protected area barrier, and external portions of the protected area. In areas where illumination of 0.2 foot-candles cannot be maintained, special procedures are applied as described in the Appendix.

Patrols of the protected area are performed at random intervals to detect the presence of unauthorized persons, vehicles and materials.

3.1 Identification of Vital Areas

The Appendix contains a discussion of the applicant's vital area program and identifies those areas and items of equipment determined to be vital for protection purposes. Vital equipment is located within vital areas which are located within the protected area and which require passage through at least two barriers, as defined in 10 CFR 73.2(f)(1) and(2), with certain exceptions, to gain access to vital equipment. The staff has reviewed those exceptions and has determined that the barriers are sufficiently substantial to meet the intent of the two barrier requirement.

Except for the exceptions noted in the Appendix, vital area barriers are separated from the protected area barrier. The control room and central alarm station are provided with bullet-resistant walls, doors, ceilings, floors, and windows. Based on these findings and the analysis set forth in the Appendix, the staff has concluded that the applicant's program for identification and protection of vital equipment satisfies the regulatory intent. However, this program is subject to on site validation by the staff in the future, and to subsequent changes if found to be necessary.

4.0 Access Requirements

In accordance with 10 CFR 73.55(d) all points of personnel and vehicle access to the protected area are controlled. The individual responsible for controlling the final point of access into the protected area is located in a bullet-resistant structure. As part of the access control program, vehicles (except under emergency conditions), personnel, packages, and materials entering the protected area are searched for explosives, firearms and incendiary devices by electronic search equipment and/or physical search.

Vehicles admitted to the protected area, except licensee designated vehicles, are controlled by escorts. Licensee designated vehicles are limited to onsite station functions and remain in the protected area except for operational maintenance, repair, security, and emergency purposes. Positive control over these vehicles is maintained by personnel authorized to use the vehicles or by the escort personnel.

A picture badge/key card system, utilizing encoded information, identifies individuals that are authorized unescorted access to protected and vital areas and is used to control access to these areas. Individuals not authorized

unescorted access are issued non-picture badges that indicate an escort is required. Access authorizations are limited to those individuals who have a need for access to perform their duties.

Unoccupied vital areas are locked and alarmed. Access to the reactor containment is positively controlled to assure that only authorized individuals are permitted to enter. In addition, all doors and personnel/equipment hatches into the reactor containment are locked and alarmed. Keys, locks, combinations, and related equipment are changed on an annual basis. In addition, when an individual's access authorization has been terminated due to the lack of reliability or trustworthiness, or for poor work performance, the keys, locks, combinations, and related equipment to which that person had access are changed.

5.0 Detection Aids

In satisfying the requirements of 10 CFR 73.55(e) the licensee has installed intrusion detection systems at the protected area barrier, at entrances to vital areas, and at all emergency exits. Alarms from the intrusion detection system annunciate within the continuously manned central alarm station located in the protected area and within a secondary alarm station also located in the protected area. In addition, the central alarm station is constructed so that the walls, floors, ceilings, doors, and windows are bullet-resistant. The alarm stations are located and designed in such a manner so that a single act cannot interdict the capability of calling for assistance or responding to alarms. The central alarm station contains no other functions or duties that would interfere with its alarm response function.

The intrusion detection systems transmission lines and associated alarms annunciation hardware are line-supervised and tamper-indicating. Alarm annunciators indicate the type of alarm and its location when activated. An automatic indication of when the alarm system is on standby power is provided in the central alarm station.

6.0 Communications

As required in 10 CFR 73.55(f) the licensee has provided for the capability of continuous communications between the central and secondary alarm station operators, guards, watchmen, and armed response personnel through the use of a conventional telephone system, and a security radio system. In addition, direct communication with the local law enforcement authorities is maintained through the use of a conventional telephone system and a two-way FM radio link.

All non-portable communication links, except the conventional telephone system, are provided with an uninterruptible emergency power source backed up by diesel generators.

7.0 Test and Maintenance Requirements

In meeting the requirements of 10 CFR 73.55(g) the licensee has established a program for the testing and maintenance of all intrusion alarms, emergency alarms, communication equipment, physical barriers, and other security related devices or equipment. Equipment or devices that do not meet the design performance criteria or have failed to otherwise operate will be compensated for

by appropriate compensatory measures as defined in the "Millstone Nuclear Power Station Physical Security Plan" and in site procedures. The compensatory measures defined in these plans will assure that the effectiveness of the security system is not reduced by failures or other contingencies affecting the operation of the security related equipment or structures.

Intrusion detection systems are tested for proper performance at the beginning and end of any period that they are used for security. Such testing will be conducted at least once every seven days.

Communication systems for onsite communications are tested at the beginning of each security shift. Off site communications are tested at least once each day.

Audits of the security program are conducted once every 12 months by the Northeast Utilities Services Company (NUSCO) System Security Staff, which is independent of site security management and supervision. The audits, focusing on the effectiveness of the physical protection provided by the onsite security organization in implementing the approved security program plans, include, but are not limited to: a review of the security procedures and practices; system testing and maintenance programs; and local law enforcement assistance agreements. The NUSCO System Security Staff prepares a report documenting their findings and recommendations and submits it to the Northeast Nuclear Energy Company (NNECO) for review and necessary action.

8.0 Response Requirements

In meeting the requirements of 10 CFR 73.55(h) the licensee has provided for armed responders immediately available for response duties on all shifts consistent with the requirements of the regulations (see Appendix). Considerations used in support of this number are attached. In addition, liaison with local law enforcement authorities to provide additional response support in the event of security events has been established and documented.

The applicant's safeguards contingency plan for dealing with thefts, threats, and radiological sabotage events satisfies the requirements of 10 CFR Part 73, Appendix C. The plan identifies appropriate securi[illegible] events which could initiate a radiological sabotage event and identifies the applicant's preplanning, response resources, safeguards contingency participants and coordination activities for each identified event. Through this plan, upon the detection of abnormal presence or activities within the protected or vital areas, response activities using the available resources would be initiated. The response activities and objectives include the neutralization of the existing threat by requiring the response force members to interpose themselves between the adversary and their objective, instructions to use force commensurate with that used by the adversary, and authority to request sufficient assistance from the local law enforcement authorities to maintain control over the situation.

9.0 Employee Screening Program

In meeting the requirements of 10 CFR 73.55(a) to protect against the design basis threat as stated in 10 CFR 73.1(a)(1)(ii), Northeast Utilities has provided for an employee screening program. Personnel who successfully complete

the employee screening program or its equivalent may be granted unescorted access to protected and vital areas at the Millstone site. All other personnel requiring access to the site are escorted by persons authorized and trained for escort duties and who have successfully completed the employee screening program. The employee screening program, except as noted below, is based upon accepted industry standards and includes a background investigation, psychological evaluation, and a continuing observation program. The proposed screening program deviates from the excepted industry standard in that a psychological evaluation is not conducted on all employees. The utility has been directed to justify this deviation. This is considered an open item which must be resolved prior to licensing.

The plan also provides for a "grandfather clause" exclusion which allows recognition of a certain period of trustworthy service with the utility or contractor as being equivalent to the overall employee screening program. The staff has reviewed the applicant's screening program against the accepted industry standards (ANSI N18.17 1973) and has determined that the Northeast Utilities program is acceptable with the exception of the psychological evaluation matter.

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ENCLOSURE 2

SUPPLEMENTAL SAFETY EVALUATION REPORT
POWER SYSTEMS BRANCH
MILLSTONE NUCLEAR POWER STATION
UNIT 3
DOCKET NUMBER 50-423

8.0 ELECTRIC POWER SYSTEMS

8.2 Offsite Electric Power Systems

8.2.2.1 Physical Separation of Offsite Circuits Within a Common Right-of-Way

In a letter dated April 1, 1985, the applicant provided FSAR revisions as they will appear in Amendment 13 of the FSAR. In this letter, the applicant stated that his revision incorporates the response to NRC question number 430.4. This resolves confirmatory item 44 dealing with the inclusion of the information in the FSAR.

8.2.2.2 Physical Separation of Offsite Circuits Between Switchyard and Class 1E System

In a letter dated April 1, 1985, the applicant provided FSAR revisions as they will appear in Amendment 13 of the FSAR. In this letter the applicant stated that this revision incorporates the response to NRC question number 430.5. This resolves confirmatory item 45 dealing with the inclusion of this information in the FSAR. During a site visit held on April 10 and 11, 1985, the staff observed that the A division cables from the normal station service transformer are routed in cable trays when they pass through the B division cable tunnel. This contradicts sheet 2 of the FSAR Figure 8.3-7 which shows this cable in

embedded conduit. The applicant has indicated this figure will be corrected to resolve this discrepancy. This is acceptable.

8.2.2.5 Generator Rejection Scheme

The staff will pursue with the applicant concerns regarding stability of the offsite power system at Millstone, therefore confirmatory item (46) remains open.

8.3.1.3 Description of Compliance with Position 1 of BTP PSB-1

In a letter dated April 1, 1985, the applicant provided FSAR revisions as they will appear in Amendment 13 of the FSAR. One of these revisions include the response to NRC question number 430.9 which the staff has previously reviewed and found acceptable. This resolves confirmatory item 50. As part of its review of the Millstone 3 Technical Specifications, the staff will ensure that the second level undervoltage protection setpoints are acceptable and the description of the logic matches that in the FSAR revision.

8.3.1.5 Adequacy of Station Electric Distribution System Voltage

As part of the site visit on April 10 and 11, 1985, the staff reviewed the results of the Millstone Unit 3 voltage drop analysis and found the results to be acceptable but noted that the grid voltage limits necessary to maintain adequate plant voltages may not be outside of the normal grid voltage extremes. This could result in the grid being incapable of supplying adequate voltages to safety loads during periods when the grid is operating at its normal voltage extremes. The staff will pursue this item with the applicant and report its resolution in a future supplement.

The NRC regional office will verify the test results which substantiate the Millstone Unit 3 voltage analysis. This item remains confirmatory pending completion of the Region verification.

8.3.1.7 Diesel Generator Protective Relaying

This item was identified as confirmatory in Section 8.3.1.7 of the SER but was not assigned a confirmatory item number in Table 1.4 of the SER. The staff reviewed with the applicant S&W drawings numbered: 12179-ESK-8KK (Revision 2), 12179-ESK-8KF (Revision 6), 12179-ESK-5DS (Revision 11), and 12179-ESK-8KG (Revision 6). The staff confirmed that the design for bypassing diesel generator protective relaying under accident conditions meets the staff position. This item is, therefore, considered complete.

8.3.1.11 Diesel Generator Load Acceptance Test After Operation at No Load

In section 8.3.1.11 of the staff's SER it was stated that the method by which the diesel generator's no load capability is considered in the load acceptance tests would be pursued with the applicant. In a letter dated June 7, 1985, the applicant provided information on how the deleterious effects of extended no load operation will be minimized on the diesel generators. The staff evaluation of this response is addressed in section 9.5.4.1 of the supplement.

8.3.3.3.2 Frequency of Cable Identification Markings

This item was identified as confirmatory in Section 8.3.3.3.2 of the SER but was not assigned a confirmatory item number in Table 1.4 of the SER. The staff reviewed the cable's color code identification to determine that the 15 ft. marking interval is sufficient to facilitate visual verification that the cables are installed in conformance with separation criteria. Because the majority of cables were continuously marked (solid color cable) such that they contrasted with black cables marked at approximately 15 foot intervals, the staff found that visual verification was not a problem. This item is, therefore, considered complete.

8.3.3.3.8 Adequacy of Protection Provided Class 1E Circuits From the Effects of Non Class 1E Circuits

The applicant has performed tests and analysis to justify less than the minimum separation specified in IEEE 384-1974 between Class 1E and non Class 1E circuits. This is in accordance with section 5.1.1.2 of IEEE 384 which allows the test and analysis approach. The tests and analyses that were performed are presented in Wyle Test Report No. 47506-02 dated February 25, 1985. The following configurations were tested:

1. A test in free air consisting of a fault cable inside a SWEC protective wrap (SILTEMP 188CH fabric), with three target cables in contact with outside of wrap. The purpose of the test was to demonstrate that a faulted cable enclosed within SWEC protective wrap does not affect external cables with one-inch separation, which represents field installations of free air drops for cables going from:

    a. tray to tray
    b. tray to conduit
    c. conduit to conduit
    d. tray/conduit to equipment

2. A test in free air consisting of a fault cable, in contact with a target cable which was wrapped in the SWEC protective wrap. The purpose of the test was to demonstrate that a faulted cable external to the SWEC protective wrap does not affect the protected cable with one-inch separation, which represents field installations of free air cable drops the same as in item 1 above.

3. A test consisting of a dropout fault cable in the upper tray of a horizontal four tray stack. The cable drops out of the upper tray, over the top of the covered tray below it and proceeds down past the lower three trays. The purpose of the test was to:

a. Demonstrate the acceptability of a single, solid, nonventilated cable tray cover as a barrier with one-inch separation from the tray cover.

b. Demonstrate that a faulted dropout cable from a tray does not affect cables in trays below it.

4. A test consisting of a dropout target cable in a tray just below the upper tray of a horizontal four tray stack. The cable drops out of the tray, runs along the ventilated cover of the covered tray below it, and proceeds down past the lower two trays. The fault cable is located just below the cover of the covered tray and other target cables are located in the trays above and immediately below the covered tray. The purpose of the test was to:

   a. Demonstrate that a faulted cable in a tray with a single, ventilated cover does not affect cables in trays above it.

   b. Demonstrate that a faulted cable in a tray with a single, ventilated cover does not affect a drop out cable from a tray above it with one-inch separation.

5. A test consisting of two tests between a horizontal, four-tray stack and one vertical tray with one inch separation. The fault cable was placed in the vertical tray with four target cables in the horizontal trays during the first test. The positions of the fault cable and one of the target cables were interchanged for the second test. The purpose of the test was to demonstrate that, at a perpendicular crossing of a horizontal and vertical tray with a single cover on the vertical tray or a cover on the top and bottom on the horizontal tray (ventilated or nonventilated), a faulted cable in either tray does not affect the cable in the other tray.

6. A test consisting of two tests between a horizontal cable tray and a parallel conduit mounted one inch above the fault cable at the top of the tray centerline. The fault cable was placed in the tray with three target

cables in the conduit during the first test. The positions of the fault cable and target cables were interchanged for the second test. The purpose of the test was to:

a. Demonstrate that a faulted cable enclosed in a conduit does not affect external cables with one-inch separation.

b. Demonstrate that a faulted cable external to a conduit and with one-inch separation does not affect cables internal to the conduit.

7. A test between two horizontal conduits with zero-inch separation. The fault cable was placed in the lower conduit with two target cables in the upper conduit. The purpose of the test was to demonstrate that a faulted cable enclosed in a conduit does not affect cables in another conduit with 1/8-inch separation between conduits for low energy power, control, and instrumentation circuits (applicant's "K", "C". and "X" service).

The staff has reviewed the test results applicable to the above test configurations and finds them acceptable. The field installations at Millstone 3 identified above are, therefore, also acceptable. These separations are applicable only between Class 1E and non Class 1E circuits.

8.3.3.3.10 Transformer Used as an Isolation Device

The staff evaluation (of the applicant's test results and design provisions to ensure that non Class 1E circuits are sufficiently isolated and will not cause unacceptable influence on any Class 1E circuits) was inadvertently left out of the staff's SER. SER Section 8.3.3.3.10 should be corrected as follows:

By Section 8.3.1.1.2 (Item 3) and Figure 8.3-3 of the FSAR, the applicant indicated that non Class 1E nuclear steam supply system loads are connected to the Class 1E 120V vital ac buses through transformers that are qualified as

~~isolation devices. The staff disagrees that the transformers are qualified isolation devices.~~

In its SER the staff stated that

~~By~~ letters dated August 29, 1983 and June 12, 1984, the applicant provided results of tests and design provisions to ensure that non Class 1E circuits are sufficiently isolated and will not cause unacceptable influence on any Class 1E circuits, and that the results of the staff review would be reported in a supplement to the SER. The staff has reviewed this information and finds that

~~These~~ results of tests and design provisions included the following items:

1. The transformer is Class 1E and is protected by a fuse and a circuit breaker that are physically separated.

2. The circuit from the transformer to the loads are protected by transformer output fuses and feeder circuit fuses.

3. The output circuit of the transformer is run in dedicated conduit to the 120 volt non Class 1E distribution panel.

4. The loads are limited to control and instrument loads.

5. The circuits from the 120 volt non Class 1E distribution panel are routed in raceways designated nonsafety; thus, circuits associated with redundant safety division are intermixed. The staff found this aspect of the design to be unacceptable.

6. The test report, with respect to a bolted short on the output of the transformer demonstrated that associated Class 1E circuits and power supplies were not adversely affected.

7. The test report, with respect to hot short, indicated electrical transients may adversely affect Class 1E circuits. The staff found this aspect of the design to be unacceptable.

Subsequently, by letter dated July 18, 1984 the applicant committed to perform additional testing to demonstrate the hot short capability of the isolation transformer. Given the reverse assumption that the isolation transformer passes the additional testing, the staff concludes that the above design provisions and the isolation capability of the transformer meet the guidelines of RG 1.75 and is, therefore, acceptable. Given the assumption that the isolation transformer fails to pass the additional testing, the applicant has committed to either route the associated cables independently so that redundant associated cables are not intermixed or to remove the subject non Class 1E circuits from Class 1E power sources. The staff concludes that either of these commitments would provide adequate protection for and independence between Class 1E circuits and are, therefore, acceptable.

During the site visit performed by the staff on April 10 and 11, 1985, the applicant provided Appendix B of Test No. T3345BP002. The NRC staff found the results to be satisfactory. This item is, therefore, considered complete.

8.3.3.3.15 Coordination of Breakers

The staff's original evaluation stated that for those series-connected circuit breakers used as isolation devices periodic testing and calibration will be included in the Millstone 3 Technical Specifications. The applicant, however, committed by letter dated June 12, 1984 to periodically test and calibrate these devices to ensure that proper breaker coordination is maintained. This commitment on routine testing of the breakers acceptably resolve the staff's concern on breaker coordination and in accordance with staff practice on recent operating license reviews for similar situations, a technical specification covering testing of these breakers will not be required.

8.3.3.3.16 Design Criteria of Associated Circuits From the Isolation Device to Load

The staff confirmed that information presented by letter dated June 12, 1984 or

by proposed amendment 8 to the FSAR was included in amendment 8 to the FSAR dated May 1984. This item is, therefore, considered complete.

8.3.3.4 Compliance With the Guidelines of NUREG-0737

II.G.1 Emergency Power for Pressurizer Equipment

The power supplies to the pressurizer power operated relief valves (PORVs) and their associated block valves was originally taken from opposite power trains. In its original evaluation, the staff indicated that this met the objective of TMI Action Plan Item II.G.1 Clarification 2, but did not meet the recommendations of BTP RSB 5-2 for overpressurization protection while operating at low temperatures. In a letter dated April 11, 1985 the applicant committed to change this arrangement such that both series valves (PORV and block valve) will be powered from the same electrical division but from different power supplies. One PORV will receive power from 125V vital dc, and its associated block valve will receive power from the same train 480V ac emergency bus. The other PORV and block valve will have the same arrangement but will be powered from the opposite division. This meets the requirements of both the TMI Action Plan and BTP RSB 5-2. This resolves open item 13.

9.0 AUXILIARY SYSTEMS

9.5.3 Lighting System

The Millstone Unit 3 SER (NUREG-1031 dated July, 1984) required the applicant to do the following:

1. For the auxiliary shutdown panel/purple switchgear room:

    a. "The illumination level provided by the purple emergency ac lighting system should be increased to a minimum of 10 ft-candles over the work area."

    b. "Adequate ac lighting (a minimum of 10 ft-candles) should be provided in the auxiliary shutdown area from the other train of the ac lighting system."

    c. "The illumination level provided by the dc lighting system should be increased to a minimum of 10 ft-candles in those areas of the purple switchgear room where work may be performed to restore ac power."

2. "Since the control room, the orange switchgear room, and the diesel generator room may require access during certain events so that ac power can be restored, the dc emergency lighting system illumination intensity shall be increased to a minimum of 10 ft-candles at those work stations where work may be performed to restore ac power."

3. "The ac emergency lighting system illumination intensity shall be increased to a minimum of 10 ft-candles at the work station instead of an average of 10 ft-candles."

4. "For the other safety-related areas, the illumination intensity shall be increased to a minimum of 10 ft-candles at the panel surfaces and at the work stations and 2 to 5 ft-candles on the basis of the activity level for access and egress to safety-related plant areas."

In letters dated June 29, 1984 and July 18, 1985, the applicant addressed these concerns. He committed to items 1a, 1b, 1c and 2. With regards to Items 3 and 4, the applicant committed to storing portable battery powered lighting onsite to supplement the emergency lighting, and provide the following for access/egress lighting:

- o Access/egress routes between manned work stations will be illuminated to .5 FC average maintained.

- o Slight hazards will be illuminated to .5 FC minimum at the center point of the hazard.

- o High hazards will be illuminated to 2 FC at the center point of the hazard.

The staff has evaluated the information and finds it acceptable, therefore ~~open~~ outstanding item (15) is resolved.

On the basis of its review, the staff concludes that the various lighting systems provided at Millstone Unit 3 are in conformance with the standards, criteria, and design basis can perform their design function, and, therefore, are acceptable.

9.5.4.1 Emergency Diesel Engine Auxiliary Support Systems (General)

(1) Concrete Dust Control

The Millstone Unit 3 SER (NUREG-1031 dated July, 1984) stated the following on concrete dust control:

"It is the staff's position that, before initial startup, the concrete floor and walls shall be painted with an appropriate paint or treated to minimize the generation of concrete dust. In a letter dated May 17, 1984, the applicant has committed to treat the floor slab with an appropriate sealant to preclude generation of concrete dust. The staff requires that sealant be applied before initial startup."

In a letter dated January 24, 1985, the applicant committed to complete the above by plant startup. T~~his issue~~ Confirmatory item (56) is closed.

(2) Vibration of Instruments and Controls

In a letter dated January 24, 1985, the applicant modified a commitment made in a letter dated May 8, 1984 and found acceptable in the Millstone Unit 3 SER. The applicant has now committed to the following for qualifying the engine mounted instrumentation and controls for vibration:

"(a) Actual vibrational levels of the equipment will be measured to confirm that they are within the tolerances specified as acceptable by the equipment manufacturers. The vibration levels will be measured during preoperational or qualification testing of the diesel generator units.

(b) Equipment within the panels will undergo in-house vibration testing to ensure that it will remain operable, under actual equipment vibration levels, throughout the 18-month calibration period.

(c) Equipment which can not be qualified by one of the above methods will be replaced by items that can be qualified.

(d) The engine skid-mounted panels will be removed from the engine skid and mounted as freestanding floor panels.

The applicant shall keep the staff advised as to the qualification method being pursued. Vibration measurements and the complete qualification package will be submitted for staff approval. The program is to be completed by the end of the first refueling outage."

The above additional information expands on the original commitment, therefore, the staff continues to find the program acceptable. ~~The results of the above tests will be reviewed by the Region II staff.~~ I Confirmatory item (57) is closed pending review of the results of the above tests by Region I.

9.5.4.2 - 9.5.8 Emergency Diesel Engine Auxiliary Systems Piping Classification

The applicant was requested to define and provide the industry standards to which the engine mounted auxiliary systems (fuel oil, cooling water, air starting, lube oil, and combustion air intake and exhaust) piping and components were designed. In a letter dated June 29, 1984, the applicant provided the standards to which the engine mounted piping was designed. He stated that this piping and the associated components, such as valves, fabricated headers, fabricated special fittings, and the like are designed, manufactured, and inspected in accordance with the manufacturer's standards which are equivalent to ASME Section III Class 2 requirements as well as the guidelines and requirements of ANSI Standard N45.2 "Quality Assurance Program Requirements for Nuclear Facilitates" and 10 CFR 50 Appendix B. The engine mounted auxiliary system piping and associated components are intentionally overdesigned (subjected to low working stresses) for the application, and thereby resulting in

high operational reliability. We find the design of the engine mounted auxiliary system piping and components as stated acceptable.

In addition, commensurate with the safety function performed by the air starting system, the Millstone Unit 3 SER (NUREG-1031) required the following:

" (1) All air starting engine-mounted piping and components that are pressurized to high-energy pressures (275 psig or greater) during standby, starting, and/or operation will be designed seismic Category I, ASME Code, Section III, Class 3 (Quality Group C).

(2) All high energy air starting piping will be adequately restrained to prevent damage to other diesel generator piping, components, and equipment from pipe whip. Note: Seismic restraints and seismic supports may not be adequate as pipe whip restraints."

In a letter dated April 11, 1985, the applicant provided his justification for not performing a high energy line analysis of the air starting system or classifying the entire air starting system as ASME Section III Class 3 (Quality Group C). The staff reviewed the applicant's justification for not performing a high energy line analysis for the diesel generator air starting system. Based on the available criteria and guidelines ~~provided by the Auxiliary Systems Branch~~ on high energy line failure, we find the air starting system for Millstone Unit 3 as designed acceptable.

On the basis of its review, the staff concludes that the emergency diesel engine auxiliary systems (fuel oil, cooling water, lube oil air starting and combustion air intake and exhaust) with regards to piping design meet the requirements of GDC 2, 4, 5, and 17 and meet the recommendations of NUREG/CR-0660, the guidance of the cited RGs, SRP Sections and industry codes and standards; they can perform their design safety function and, therefore, are acceptable.

9.5.4.2 Emergency Diesel Fuel Oil Storage and Transfer System

With regards to sediment control, the SER (NUREG-1031) stated the following license condition:

"the design of the system as described above allows the diesel generator day tank to be filled from either fuel oil storage tank. Thus, fuel oil can be drawn from one fuel oil storage tank while the other tank is being filled and then allowed to stand for 24 hours while the sediment settles out. This is an acceptable procedure for sediment control in the fuel oil system. Therefore, the staff requires that the plant operating procedures be modified to incorporate this filling procedure."

In a letter dated January 24, 1985, the applicant provided the following with regards to sediment control in his fuel oil storage tanks:

"Correspondence with diesel fuel oil suppliers has indicated that turbulence caused by incoming fuel would not be sufficient to disturb an existing sediment bed if the initial fuel oil level is greater than 4 feet. This level is based on anticipated sediment levels, pumping rate of fuel delivery trucks, and existing tank design. The fill line is such that it is terminated upon penetration of the storage tank top. Therefore, any disruption of the fuel oil present in the tank will occur at the fluid surface and not in the area of the sediment layer.

The applicant will require in the plant operating procedures that refilling operations are started prior to the tank dropping below the 50% (5.2 feet) level. This requirement will ensure that unacceptable sediment concentrations are not realized.

In the event that filling does not commence prior to reaching this 50% point, an adequate settling period will be provided for the recently filled tank, with transfer being accomplished form the alternate tank. Information provided by fuel oil suppliers indicates that a 1-hour settling period per foot of final product height is generally utilized. Therefore, the allowed settling time will be based on the final tank height at the conclusion of the filling operation. The settling period will be provided in the plant operating procedures."

The staff has evaluated the sediment control justification and associated operating procedures and finds them acceptable, and hence removes the license condition.

On the basis of its review, the staff concludes that the emergency diesel engine fuel oil storage and transfer system meets the requirements of GDC 2, 4, 5, and 17 and meets the recommendations of NUREG/CR-0660, the guidance of the cited RGs and SRP Section 9.5.4, and industry codes and standards. Thus, it can perform its design safety function and, therefore, is acceptable.

9.5.6 Emergency Diesel Air Starting System

In the Millstone SER (NUREG-1031) the staff accepted the applicant's justification for delaying the installation of the air dryers on the diesel generators until first refueling with the following license condition:

1. "The air dryers shall be installed at the first opportunity but no later than before startup of the first refueling."

With regard to the second part of the license condition, namely blowdown of the air receivers and inspection of inline filters, the applicant by letters dated May 4, 1984 and January 7, 1985 provided additional information. The staff reviewed the information and found it acceptable and hence removes this portion of the license condition, (5) from SER Table 1.5.

~~The above license condition shall be removed upon installation of the air dryers. This issue was closed.~~

The Millstone SER (NUREG-1031) stated the following:

> "Operating experience at two nuclear power plants has shown that during periodic surveillance testing of a standby diesel generator, initiation of an emergency start signal (LOCA or LOP) resulted in the failure of the diesel to start and perform its function because of depletion of the starting air supply from repeated activation of the starting relay. This event occurred as the result of inadequate procedures and from a failure in engine starting and control circuit logic to address a built-in time delay relay to ensure the engine comes to a complete stop before attempting a restart. During the period that the relay was timing out, fuel to the engine was blocked while the starting air was uninhibited. This condition with repeated start attempts depleted starting air and rendered the diesel generator unavailable until the air system could be repressurized. This is an unacceptable operating condition. The applicant was asked to review his procedures and/or control system logic to ensure that this event will not occur at Millstone Unit 3."

In a letter dated June 29, 1985. The applicant stated that he reviewed the diesel control scheme and concluded that the event described above was not applicable to Millstone.

On the basis of its review, the staff concludes that the emergency diesel engine air starting system meets the requirements of GDC 2, 4, 5, and 17 and meets the guidance of the cited RGs and SRP Section 9.5.6, the recommendations of NUREG/CR-0660, and industry codes and standards. Thus, it can perform its design safety function and is, therefore, acceptable.

9.5.7 Emergency Diesel Engine Lubricating Oil System

In the Millstone Unit 3 SER (NUREG-1031) the staff found the justification provided by the applicant for not preheating the rocker arm lubricating oil system acceptable. However, the staff imposed the following license condition based on the applicant's justification:

> "Upon actuation of the diesel generator low room temperature alarm, the room air temperature shall be increased to 50°F or greater, or this may result in diesel generator being placed in a limiting condition for operation."

In a letter dated January 24, 1985, the applicant provided additional information. The applicant stated plant operating procedures will include the actions that would be taken to increase D/G room temperature upon actuation of low room temperature alarm. The staff finds this acceptable, and removes ~~the above~~ license condition (6) from SER Table 1.5. This issue is closed.

The Millstone SER (NUREG-1031) stated the following:

> "It is stated that the rocker arm lubricating oil reservoir level is monitored for high level and the level is maintained by a level control valve. No mention is made of a reservoir low-level alarm. A failure of the level control valve to maintain lubricating oil level in the rocker arm reservoir could result inadequate or no lubricating oil for the rocker arms, leading to diesel generator unavailability and/or failure. This is an unacceptable condition.

In letters dated May 17, 1984 and May 2, 1985, the applicant provided additional information. The applicant stated that

"The rocker arm lube oil reservoir level will be checked, in accordance with the manufacturer's recommendations, prior to any manual start, biweekly on engines in standby and daily on operating engines.

The staff finds the above acceptable. This issue is closed.

On the basis of its review, the staff concludes that the emergency diesel engine lubricating oil system meets the requirements of GDC 2, 4, 5, and 17, the guidance of the cited RGs and SRP Section 9.5.7, and the recommendations of NUREG/CR-0660 and industry codes and standards. Thus, it can perform its design safety function and is, therefore, acceptable.

9.5.8 Emergency Diesel Engine Combustion Air Intake and Exhaust System

In the Millstone SER the staff found the intent of the operating procedure for the access hatch which is part of the diesel engine exhaust system acceptable. However, the staff proposed ~~a~~ license condition (7) requiring the following to be included in the plant technical specifications.

1. In the event of a tornado alert or an ice storm, snow storm or freezing rain storm forecast, the access hatch in the emergency diesel generator combustion exhaust system shall be opened and shall remain open until the event has passed.

2. At least once per year, the access hatch shall be opened to verify operation of the hatch, inspected for corrosion of parts (hinges, locking mechanisms, etc.) and maintained in an operable status by replacement of corroded parts, properly lubricated, pointed, etc.

In letter dated January 7, 1985, the applicant provided additional information. The applicant stated that the design function of the access hatch is to provide an alternate exhaust path in the event that the exhaust stack is damaged by a tornado missile. He also stated that due to the meteorological climate at Millstone site, it is highly improbable that a tornado alert will occur coincidentally with a freezing rain or snow condition. Therefore, the requirement

that the access hatch be opened when an ice storm, snow storm , or freezing rain storm forecast is received is felt to be unnecessary. The staff agrees with the applicant. The applicant will require, in an abnormal operating procedure, that the hatch be periodically inspected and when such conditions exist and, if significant accumulation is observed, corrective action will be taken to ensure the hatch remains operable. The applicant will require that the access hatch be opened in the event of a tornado alert.

With regards to the second staff requirement, the applicant will require that measures taken to ensure hatch operability be addressed in the preventive maintenance procedures. The access hatch will be opened at least once per year, inspected for corrosion of parts, and maintained in an operable status.

The staff finds the above acceptable and hence removes the license condition (7) from SER Table 1.5.
This issue is closed.

A.6.11

Section 15.4.3

New Analysis for the Control Rod Drop Event

for Millstone Unit 3

15.4.3 Rod Cluster Control Assembly Malfunctions

The SER for Millstone Unit 3, in Section 15 on "Rod Cluster Control Assembly Malfunctions", indicated that a potential controller problem existed for the dropped control rod event which could lead to the imposition of operating restrictions. It also indicated that it was anticipated that a detailed analysis would show that if the transient occurs that thermal limits would not be exceeded, but that this analysis had not as yet been submitted for Millstone Unit 3. The SER also indicated that Westinghouse has developed a solution for the problem via a new methodology for analyzing the event and has documented it in a topical report (WCAP-10297P) and that this report and its methodology have been evaluated by the staff and approved. The staff evaluation was enclosed in the memorandum to F. Miraglia from L. Rubenstein, March 2, 1983, "Review of the Westinghouse Report 'Dropped Rod Methodology for Negative Flux Rate Trip Plants'". The solution requires a reactor-cycle specific analysis showing that DNB limits will not be exceeded. The Millstone Unit 3 FSAR has been revised in Amendment 12 to include a discussion of this analysis, and the results for Cycle 1 operation indicate that DNB limits will be met for this cycle for both N and N-1 loop operation. Thus operating limits will not be necessary for Cycle 1. Each future reload cycle will require similar cycle specific analysis as part of the normal reload analysis. Confirmatory item (67) is resolved.

16 TECHNICAL SPECIFICATIONS

As a result of information contained in this supplement the following corrections to Table 16.1, Technical Specification Items are made:

Item (8) and Item (13) are deleted as Technical Specification requirements.

ENCLOSURE 2

FSAR Open Items

1. Diesel Generators
2. Inservice Testing of Pumps and Valves
3. Equipment Qualification
4. Flow Measurement Capability
5. Subcompartment Analysis
6. Fire Protection
7. Inservice Exam of all Pipe Welds in Break Exclusion Area
8. Jet Impingement Effects
9. Program Evaluation (II.D.1)
10. Control Rod Drive Structural Materials
11. Preservice Inspection Program and Relief Request
12. Procedures for Actuating Hydrogen Recombiner
13. Secondary Enclosure Building
14. Sump Flow Approach Velocity
15. P.G.P, I.C.1
16. Physical Security Plan
17. Reactor Coolant Pump Trip During LOCA
18. III.D.1.1
19. QA Program Commitments
20. Containment Liner Report
21. Seismic Interaction Program