NUREG-1031
Supplement No. 2

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

September 1985

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ABSTRACT

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.

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APPENDICES

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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), 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 to that SER (SSER 2).

SSER 2 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."

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.

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*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	
(9) Loose parts detection program	Awaiting information	
(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	Under review	
(14) Fire protection	Under review	9.5.1
(15) Functional capability of ac and dc emergency lighting	Under review	
(16) Shift technical advisor training program and operating experience for startup	Under review	13.1.2
(17) Emergency Plan	Awaiting information	
(18) Limitation on overtime	Closed (SSER 2)	13.5.1.2
(19) Q list	Awaiting information	
(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	Under review	
(2) Dynamic loading	Under review	
(3) Liquefaction potential	Under review	
(4) Shoreline slope	Under review	
(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	Under review	
(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	Under review	
(14) Margins itemized in WCAP-8691	Under review	
(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	Under review	
(20) Preservice inspection program review and relief requests	Under review	
(21) Preservice and inservice inspection of steam generators	Under review	
(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	Under review	
(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	Under review	
(45) Physical separation of offsite circuits between switchyard and Class 1E system	Under review	
(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	Under review	
(51) Adequacy of station electric distribution system voltage	Under review	
(52) Routing of power cables in the cable spreading area	Deleted (SSER 1, Appendix H)	
(53) Battery charger and transformer used as isolation devices	Under review	
(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	Under review	
(57) Qualification of engine-mounted control panels	Under review	
(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	Under review	
(61) TMI Action Plan Item II.F.1.1	Under review	
(62) TMI Action Plan Item I.C.1 - procedures generation package nuclear steam supply system	Under review	
(63) Physical Security Plan	Under review	
(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	Under review	
(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	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	Under review	
(5) Moisture in air start system	Unchanged	
(6) Preheating of rocker arm lubrication oil system	Under review	
(7) Blockage of access hatch in diesel generator exhaust system	Under review	
(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.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.5 Missile Protection

3.5.2 Structures, Systems, and Components To Be Protected From Externally Generated Missiles

Section 3.5.2 of the SER identifies as an open item the lack of tornado missile protection for the two emergency exhaust pipes. Each exhaust pipe is vulnerable to tornado missile damage from a side opening and a top opening. The applicant, in a letter dated April 26, 1985, provided the results of a probabilistic risk assessment (PRA), which estimated the risk of tornado-generated missiles entering any of the four openings of concern in the emergency diesel generator enclosure. This analysis was based on a site-specific survey, and a site-specific tornado occurrence analysis and utilized the TORMIS computer code, which simulates the effects of tornado missiles. The applicant stated that the mean probability that a tornado will occur and generate missiles that will enter any one of the four emergency diesel generator enclosure openings is estimated as 1×10^{-6} per year. The probability that a tornado will occur and generate missiles that will enter one or more openings on both trains during the same tornado strike is estimated as 7×10^{-8} per year.

The staff also performed a simplified PRA, which estimated the probability of the loss of ac power in conjunction with a tornado missile striking the emergency diesel generator exhaust piping to be less than 1×10^{-8} per year. Both of the probability estimates are conservative because a missile hit does not necessarily result in sufficient damage to the exhaust pipes, upstream of the access hatch, to disable the diesel generators.

These PRAs demonstrate that the probability of significant damage to the diesel generator exhaust piping from tornado missiles causing a release of radioactivity in excess of 10 CFR 100 limits, assuming loss of offsite power, is less than 10^{-6} per year. This meets the staff's numerical acceptance criterion for a conservative PRA. Therefore, the staff concludes that the applicant has satisfactorily demonstrated compliance with the requirements of General Design Criteria (GDC) 2 and 4 (10 CFR 50, Appendix A) as related to protection of the emergency diesel generator enclosure from the effects of tornado-generated missiles. The plant design provides protection from tornado-generated missiles in conformance with the acceptance criteria of Section 3.5.2 of the Standard Review Plan (SRP, NUREG-0800).

3.10 Seismic and Dynamic Qualification of Safety-Related Electrical and Mechanical Equipment

3.10.1 Seismic and Dynamic Qualification

3.10.1.1 Introduction

As part of its review of Sections 3.9.2 and 3.10 of the applicant's Final Safety Analysis Report (FSAR), the staff evaluated the applicant's program for seismic

and dynamic qualification of safety-related electrical and mechanical equipment. The evaluation consisted of (1) a determination of the acceptability of the procedures used, standards followed, and the completeness of the program in general and (2) an audit of selected equipment to develop a basis for the judgment of the completeness and adequacy of the seismic and dynamic qualification program.

Guidance for the evaluation is provided by SRP Section 3.10, and its ancillary documents, Regulatory Guides (RGs) 1.61, 1.89, 1.92, and 1.100; NUREG-0484; and Institute of Electrical and Electronics Engineers (IEEE) Stds. 344-1975 and 323-1974. These documents define acceptable methodologies for the seismic qualification of equipment. Conformance with these criteria is required to satisfy the applicable portions of GDC 1, 2, 4, 14, and 30 (Appendix A to 10 CFR 50), as well as Appendix B to 10 CFR 50 and Appendix A to 10 CFR 100. Evaluation of the program is performed by a Seismic Qualification Review Team (SQRT), which consists of staff engineers and engineers from the Brookhaven National Laboratory (BNL), Long Island, New York.

3.10.1.2 Discussion

The SQRT reviewed the equipment seismic and dynamic qualification information contained in FSAR Sections 3.7, 3.9, and 3.10 and visited the plant site from March 4 through March 8, 1985. The purpose of the site visit was to determine the extent to which the qualification of equipment, as installed at Millstone 3, meets the criteria described above. A representative sample of safety-related electrical and mechanical equipment, as well as instrumentation, included in both nuclear steam supply system (NSSS) and balance of plant (BOP) scopes, was selected for the audit. Table 3.1 identifies the audited equipment. The plant-site visit consisted of field observation of the equipment configuration and its installation. This was followed by a review of the corresponding qualification document. The field installation of the equipment was inspected to verify and validate equipment modeling employed in the qualification program. During the audit, the applicant presented its qualification review approach and maintenance and surveillance program.

3.10.1.3 Summary

Comments based on the review of FSAR Sections 3.9.2 and 3.10 for equipment seismic and dynamic qualification are contained in the Millstone 3 SER (NUREG-1031). These comments were discussed with the applicant in several meetings. The staff is awaiting the applicant's formal response.

The comments resulting from the SQRT site audit are contained in this supplement. The equipment-specific findings identified in Table 3.1 and the generic comments are listed in the section that follows. Upon satisfactory resolution of these specific findings and generic comments and the equipment qualification open items in the SER, the seismic and dynamic qualification of safety-related equipment at Millstone 3 will meet the applicable portions of GDC 1, 2, 4, 14, and 30 (Appendix A to 10 CFR 50), Appendix B to 10 CFR 50, and Appendix A to 10 CFR 100.

3.10.1.4 Confirmatory Items

The satisfactory resolution of the specific findings identified in Table 3.1, the generic comments listed below, and the outstanding items in the SER is

required before the staff can accept the applicant's seismic qualification program for equipment:

(1) Upon completion of the seismic qualification program, the applicant must confirm that all safety-related equipment is qualified and properly installed.

(2) For pipe-mounted equipment, the applicant shall confirm that the acceleration values at equipment locations obtained from as-built piping analyses do not exceed the acceleration values used for qualification.

(3) Several pieces of equipment were observed to be installed adjacent to other safety- or non-safety-related equipment without any dynamic interaction evaluation documented in the qualification package presented during the audit. As a result, the applicant must develop a program to inspect installation of all safety-related equipment and evaluate the possible dynamic interaction between two adjacent pieces of equipment. In case a gap exists between two pieces of equipment, the physical separation should be evaluated to preclude any dynamic impact.

3.10.2 Operability Qualification of Pumps and Valves

3.10.2.1 Introduction

To ensure that an applicant has developed and implemented a program regarding the operability qualification of safety-related pumps and valves, the staff performs a two-step audit. The first step is to review FSAR Section 3.9.3.2 for the description of the applicant's pump and valve operability assurance program. The information provided in the FSAR, however, is general in nature and not sufficient by itself to provide confidence in the adequacy of the applicant's overall program for pump and valve operability qualification. To provide this confidence, the Pump and Valve Operability Review Team (PVORT), consisting of staff from BNL and the NRC, conducts an onsite audit of a small representative sample of safety-related pumps and valves and supporting documentation.

The criteria by which the audit is performed are described in SRP Section 3.10, "Seismic and Dynamic Qualification of Mechanical Electrical Equipment." Conformance with SRP Section 3.10 is required to satisfy the applicable portions of GDC 1, 2, 4, 14, and 30 (Appendix A to 10 CFR 50), Appendix B to 10 CFR 50, and Appendix A to 10 CFR 100.

3.10.2.2 Discussion

In performing the first step of the audit, the staff reviewed the methodology and procedures of the pump and valve operability assurance program contained in FSAR Section 3.9.3.2. As a result of this review, six specific areas were identified as needing further clarification and were documented in the SER. The SER concluded that the applicant had, in general, defined a pump and valve operability assurance program, but it was not possible to determine the adequacy of the overall program without a site audit.

The PVORT performed the second step of the onsite audit during the week of March 4, 1985. The purpose of this step was to (1) review the adequacy of the overall program and (2) determine the extent that the applicant meets the criteria of SRP Section 3.10. A sample of three components in the nuclear steam

supply systems and seven components in the BOP systems was selected to be audited. (Note: Two of the three nuclear steam supply system components are supplied by the architect-engineering firm and one by the reactor supplier.) The site audit also reviewed the applicant's response to the six areas of concern identified in the SER. Table 3.2 identifies the equipment audited and the audit findings. Table 3.3 lists the six areas identified as needing further clarification in the SER and their present status.

3.10.2.3 Summary and Conclusions

On the basis of the observation of the field installation, review of qualification documents, responses provided by the applicant to PVORT inquiries and to SER open items, the staff has concluded that the applicant has developed and implemented a pump and valve operability review program. Equipment-specific and SER response findings and status are identified in Tables 3.2 and 3.3, respectively. Generic findings are listed in Section 3.10.2.4, below. Upon satisfactory resolution of the specific and generic concerns, the applicant's pump and valve operability assurance program for Millstone 3 will meet the applicable portions of GDC 1, 2, 4, 14, and 30 (Appendix A to 10 CFR 50), Appendix B to 10 CFR 50, and Appendix A to 10 CFR 100.

3.10.2.4 Generic

(1) The applicant shall confirm that all pumps and valves important to safety have had their required preoperational tests completed before fuel load.

(2) The applicant shall confirm that all pumps and valves important to safety are qualified and operational for normal, accident, and postaccident operating conditions before fuel load.

(3) The applicant shall confirm that new loads resulting from a loss-of-coolant accident or analysis of as-built conditions applicable to pump and valves important to safety do not exceed those loads originally used to qualify the equipment before fuel load.

Table 3.1 Equipment audited by SQRT during plant/site visit on March 4 through March 8, 1985

SQRT ID No.	Plant ID No.	Equipment name and description	Safety function	Findings	Resolution	Status	Remarks
BOP-1	CES*MCB-MB3	Main control boards: A half-U-shaped monitoring bench-board	Used to monitor and control the safe operation and shut-down of the plant	(1) The analysis report used for demonstration of structural integrity of the control boards structure did not establish the adequacy of connecting bolts and weld. (2) The installed control boards did not have any ID, tag, or model number marked on them.	Pending	Open	
BOP-2	3ENS*SWGA	4.16-kV emergency switchgear	Supplies power to all emergency auxiliary loads, such as the 480-V load center and motor-control center (MCC) and associated load	(1) The original test reports from Wyle Labs and from General Electric were not available during the audit. (2) It was not demonstrated during the audit between the top point of the switchgear and whether there was sufficient clearance buss duct to avoid interference.	Pending	Open	
BOP-3	3FWA*P1	Motor-driven SG auxiliary feedwater pump	Provides cooling for the core during an emergency reactor shutdown and containment isolation		Qualified	Closed	
BOP-4	3CES*PNLBE10	Isolator cabinet	Performs the electrical isolation function required by RG 1.75, which states that an electrical fault condition from non-Class 1E equipment should be isolated from Class 1E circuits		Qualified	Closed	
BOP-5	3HVR*MOD72B	Motor-operated damper: Duct mounted and supported at the wall	Used for ventilation during fuel handling	The present solder joint capacitor clip in the actuator must be replaced by a seismically qualified alternative device.	Pending	Open	
BOP-6	3LMS*MOV40A	1½" globe valve assembly (with a system of structurally restraining members)	Monitors containment leak-age rate	(1) The finite element analysis in the qualification report does not address the as-built supporting conditions of the valve. (2) Qualification of the structural supporting system was not demonstrated.	Pending	Open	
BOP-7	3SCV*PNL24P	120/240-V ac single-phase 3-wire distribution panels	Provides power to safety-related equipment in the auxiliary building such as compartment heater in MCC and limit switches for motor-operated valves	Similarity between installed equipment and tested specimen was not established	Pending	Open	

Table 3.1 (Continued)

SQRT ID No.	Plant ID No.	Equipment name and description	Safety function	Findings	Resolution	Status	Remarks
BOP-8	3RHS*MV8702C	Residual heat removal system inlet isolation valve (12")	Isolates the low-power RHS from the high-pressure reactor coolant system during power operation	Operability test static deflection was not adequately demonstrated.	Pending	Open	
BOP-9	3RPS*PNLESCA	Emergency generating load sequencer cabinet	Used for loss of offsite power in general and will automatically perform the functions of load shedding and load blocking		Qualified	Closed	
BOP-10	3RHS*PNLAS	Auxiliary shutdown panel	Provides the backup for the main control board reactor shutdown controls	Field inspection revealed two loose bolts in connecting adjacent panels and one bolt missing. However, at the time of inspection, field quality control did not complete final inspection. Quality control (QC) procedures should pick up on items such as this.	Qualified	Closed	
BOP-11	3SCV*XD10	Transformer: A box-shaped wall-mounted item	Provides power supply to maintain the proper environment in MCCs and to control circuitry of heating, ventilation, and air conditioning (HVAC) dampers, one hydrogen recombiner, hydrogen recombiner valves, and a radiation monitoring microprocessor	The qualification documentation package as presented did not include the support analysis report and the equipment similarity report (Square D No. B116900).	Pending	Open	
BOP-12	3VBA*INV-1	Static inverter: A floor-mounted free-standing cabinet	Converts emergency battery power to 120-V ac and supplies power to control and instrumentation for the plant protection systems	The proximity of the installed inverter to an adjacent cabinet and the possible dynamic interaction between the two cabinets were not addressed in the qualification documents.	Pending	Open	
NSSS-1	3RCS*P	Reactor coolant pump	Supplies reactor coolant water	The as-installed stiffness matrix was not available at the time of audit. Stone & Webster (S&W) designed the support structure for the reactor coolant pump (RCP) on the basis of design drawings. This is being reevaluated according to the as-built condition.	Pending	Open	
NSSS-2	3SIH*P	Safety injection pump assembly	Provides flow during safety injection phase	Torsional frequency of assembly needs to be computed and compared to motor's operational speed.	Pending	Open	
NSSS-3	3CHS*MV	Charging pump: Discharge isolation valve (4")	Closes and isolates the chemical and volume control system in case of high energy line break	Operability test by static deflection was not adequately demonstrated.	Pending	Open	

Table 3.1 (Continued)

SQRT ID No.	Plant ID No.	Equipment name and description	Safety function	Findings	Resolution	Status	Remarks
NSSS-4	3RPS*RAKSET	7300 process protection system (PPS): 3-bay and 2-bay floor-mounted cabinets	Contains signal conditioning equipment for monitoring pressurizer water level and pressure, containment pressure, containment pressure reactor coolant flow, and temperature from various safety-related sensors	(1) The connections between the reactor protection system (RPS) cabinets and the adjacent cabinets were not addressed in the available qualification documents. (2) The implementation of Westinghouse's recommendation for field modifications of some relays (W NEUM-10583) was not confirmed. (3) Some of the qualification documents were draft copies. Westinghouse will confirm their final release. (4) Per available qualification documents, some devices are qualified for 5 years and need replacement. No document was presented during the audit to demonstrate inclusion of this requirement in the maintenance program.	Pending	Open	
NSSS-5	3RPS*RAKNIS	Nuclear instrumentation system (NIS) console: A floor-mounted free-standing four-bay cabinet	Provides alarm function, secondary control function of indicating reactor status during startup, power operation, and overpower trip protection	(1) Some devices were not qualified for Class 1E functions. Per Westinghouse, these devices are not required to perform Class 1E function. However, no document was presented to support this statement. (2) Meeting the requirement of one safe shutdown earthquake (SSE) test preceded by five operating-basis earthquake (OBE) tests was not demonstrated during the audit. (3) The installed console did not have any ID, tag, or model number marked on it.	Pending	Open	
NSSS-6	3RPS*SWG1	Reactor trip switchgear: A floor-mounted free-standing two-bay cabinet	Provides reactor trip function by tripping the control rods	(1) The implementation of Westinghouse's recommendation for field modifications of the shunt trip attachment (W NEUM-10564) was not confirmed. (2) The proximity of the installed switchgear two-bay cabinet to an adjacent non-seismic-Category I cabinet and the possible dynamic interaction between the two cabinets were not addressed in the qualification documents. (3) Per available qualification documents, some devices were qualified for 5 years and need replacement. No document was presented during the audit to demonstrate inclusion of this requirement in the maintenance program	Pending	Open	

Table 3.2 PVORT equipment audited and the audit findings

Plant ID. No.	Description	Safety function	Component	Manufacturer	Model ID	Finding/ resolution	Status
3MSS*CTV27A†	24" single-seated globe valve	Main steam isolation	Valve	Sulzer	DAS 630-B	Satisfactory	Closed
			Actuator	Sulzer	**		
3MSS*RV22A†	6"x8"x8" dual-outlet safety valve	Prevent over-pressurization of main steamline	Valve	Dresser	3707R Special	Satisfactory	Closed
3MSS*MOV18A†	8", 900-lb motor-operated gate valve	Isolation of pressure relieving valve and pressure relieving bypass valve	Valve	Walworth	N5247PSB	Satisfactory	Closed
			Actuator	Limitorque	SMB-0-25		
3SWP*P1C†	Vertical mixed-flow centrifugal pump	Provides cooling for emergency diesel generators, containment recirculation coolers, and reactor	Pump	Hayward-Tyler	24 VSN	Open items concerning mechanical performance test:	
			Driver	GE	5K6338xC119A	(1) Generated pump curves do not meet vendor or system design curves.	Open
						(2) Rundown time comparison of pumps was not within the 20% specified in procedure.	Open
						(3) Calculated horsepower (hp) was not withn 10% of pump hp determined from perormance curves.	Open
3FW*P1A†	Horizontal centrifugal pump	Provides emergency source of water for steam generators	Pump	Bingham Willamette	3x6x9 CMSD	Satisfactory	Closed
			Driver	GE	5K821051C40		
3RSS*P1A†	Deep draft centrifugal pump	Starts in response to containment depressurization actuation (CDA) signal, delivers sump water for containment spray, then switches to long-term cooling	Pump	Bingham Willamette	VCR-10x12x10B	Satisfactory	Closed
			Driver	Westinghouse	5888P36		
3SIL*MV8808A†	10" motor-operated gate valve	Remains open to mitigate design-basis accident (DBA)	Valve	Westinghouse	10GM88EEH	Satisfactory	Closed
			Actuator	Limitorque	SBD-4-200		

See footnotes at end of table.

Table 3.2 (Continued)

Plant ID. No.	Description	Safety function	Component	Manufacturer	Model ID	Finding/ resolution	Status
3SIL*MV8812B†	12" motor-operated gate valve	Closed by operator action as part of long-term recirculation mode	Valve	Pacific	G-55509	Satisfactory	Closed
			Actuator	Limitorque	SB-1		
3RSS*MOV20B†	10" motor-operated butterfly valve	Opens in response to CDA signal; during manual switchover to long-term recirculation, must be closed by remote operator action	Valve	Pratt	NMK-II	Satisfactory	Closed
			Actuator	Limitorque	HOBC/SMB		
3RCS*SV8095A††	1" solenoid-operated globe valve	Opens to provide a safety-grade letdown path for reactor coolant system for inventory control during boration	Valve assembly	Target Rock	79AB-001	Satisfactory	Closed

†BOP supplied.

**Actuated by operating medium.

††NSSS supplied.

Table 3.3 Pump and valve qualification open items

SER items	Finding/ resolution	Status
(1) The applicant did not provide the design criteria for pump and valve internal parts, such as valve discs and pump shafts. A review of qualification documents is necessary to determine whether the pump and valve internals are adequately qualified.	Satisfactory	Closed
(2) SRP Section 3.10, Paragraph II.1a(2), indicates that equipment should be tested in the operational condition; that is, normal plant loadings should be superimposed on seismic and dynamic loads, including thermal, flow-induced loads, and degraded flow conditions. The FSAR should clearly indicate how this requirement is met.	Satisfactory	Closed
(3) For those components for which qualification and/or operability assurance was provided by analysis alone, some question remains as to the confidence level ensured by this methodology. The necessity for additional component testing is being considered and cannot be established without an inspection at the plant site.	Satisfactory	Closed
(4) There should be a list of types of equipment that clearly shows the methods used for qualification. This list should also address which standards are met, in particular those cited in SRP Section 3.10.	Satisfactory	Closed
(5) Clarification of how aging was incorporated in the qualification process should be contained in the FSAR. In addition, the applicant should commit to establish a maintenance and surveillance program to maintain equipment in a qualified status throughout the life of the plant.	Satisfactory	Closed
(6) Further justification of the independent qualification of pumps, valves, prime movers, and actuators vs. their assembly qualification is also required.	Satisfactory	Closed

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance With Codes and Code Cases

5.2.1.2 Applicable Code Cases

As was noted in the SER, the NRC staff acceptance of ASME Code Cases was contingent upon the applicant supplying a confirmatory list of ASME Code Cases used in the construction of Section III, Class 1 components within the reactor coolant pressure boundary (RCPB). This information has been supplied in an amendment to the FSAR in response to inquiry 210.4.

The staff has reviewed the list of Code Cases and they are acceptable. The staff concludes that compliance with the requirements of these Code Cases will result in a component quality level that is commensurate with the importance of the safety function of the RCPB and constitutes an acceptable basis for satisfying the requirements of General Design Criterion 1 (10 CFR 50, Appendix A), and is, therefore, acceptable.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.2 Subcompartment Analyses

In the SER, the matter of subcompartment analysis was classified as open item 10. As part of this item, the staff identified a number of concerns related to the calculation of differential pressures across subcompartment walls. The specific concerns identified pertain to (1) the method of calculation of the mass and energy release rate data for the feedwater line break in the steam generator subcompartment, (2) the results of a nodalization sensitivity study for the pressurizer cubicle, and (3) a verification by the applicant of the structural capability of the refueling cavity wall. The staff has reviewed the additional information and the applicant's revised analyses to address these concerns and presents the results of its review below.

The staff is not able to complete its review on the matter of asymmetric loads on primary system components until it receives certain additional information from the applicant. The required information includes the force/moment calculations for the pressurizer and an evaluation of the margin between calculated and design loads on major component supports. In FSAR Amendment 11, the applicant provided a description of the method by which the mass and energy releases for the feedwater line break were determined. The applicant has stated that the releases were calculated using the frictionless Moody correlation for a saturated liquid, and assuming a feedwater temperature and pressure corresponding to 102% reactor power with feedwater valves wide open. The staff has reviewed the applicant's assumptions and supporting justification and finds them acceptable. On this basis, the matter of mass and energy releases for the feedwater line break is considered satisfactorily resolved.

With regard to the pressurizer cubicle, the applicant has performed a revised subcompartment analysis using a 21-node model of the containment in place of the original 8-node model. For the surge line break, which produces limiting differential pressures for the lower pressurizer cubicle, the applicant also removed the multiplying factor of 1.1 that was applied to the SATAN V mass and energy release rates in the original subcompartment analysis. The staff considers the mass and energy release rates to remain acceptable without this factor.

Using the 21-node model, the applicant calculated a maximum pressure differential across the cubicle wall of 5.8 psid for the upper cubicle (spray line break) and 20.3 psid for the lower cubicle (surge line break). These values are less than the stated design pressures for the upper and lower cubicles. The staff performed a confirmatory subcompartment analysis of the pressurizer cubicle using the COMPARE-MOD1A computer code and the applicant's revised nodalization scheme, including reported values for flow areas, K factors, and inertia. A relative humidity of 0%, temperature of 120°F, and pressure of 8.9 psia were

assumed as initial conditions in the analysis. The staff also used the frictionless Moody critical flow correlation with a 0.6 multiplier for the upper pressurizer cubicle analysis, and the isentropic critical flow model (equivalent to the homogeneous equilibrium model) for the lower pressurizer cubicle analysis. The COMPARE results indicate a maximum pressure differential of 5.6 psid for the upper cubicle and 20.6 psid for the lower cubicle, which the staff finds constitute an acceptable confirmation of the adequacy of the pressurizer cubicle design.

With regard to the reactor cavity subcompartment analysis, and the calculated differential pressure across the refueling cavity wall, the applicant provided additional information by letter dated March 1, 1985. The applicant has stated that the refueling cavity wall is designed to withstand a uniform pressure of 11.6 psid. This value is well above the differential pressure calculated to result from high energy line pipe breaks (i.e., 4.8 psid). On this basis, the staff considers the matter regarding the reactor cavity subcompartment analysis resolved.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures

The staff reported in the SER that several changes in the methodology used to calculate the mass and energy release rate data for postulated main steamline break (MSLB) accidents were made by Westinghouse during the course of the staff's review of Topical Reports WCAP-8859 and WCAP-8860, and that these model changes were not factored into the Millstone 3 MSLB analysis. In addition, the applicability of the Westinghouse methodology to the model F steam generators used in Millstone 3 was not addressed by the applicant. Consequently, the matter of the mass and energy release analysis for MSLB accidents was classified as open item 11.

The applicant does not plan to perform revised MSLB analyses using the later version of the Westinghouse methodology. The applicant's basis as stated in the Westinghouse letter from E. P. Rahe, Jr., dated February 19, 1985, is that for large, dry, and subatmosphere containments, the effect of a superheated steam blowdown on containment pressure and temperature response is not significant. Westinghouse's conclusion is based on the results of a Westinghouse analysis of a spectrum of MSLB accidents using the COCO containment response code in conjunction with the LOFTRAN blowdown code (which incorporates a detailed superheating model). A more detailed discussion of the analysis was provided by Westinghouse in a presentation to the staff at a meeting on January 25, 1985; Westinghouse is in the process of documenting this generic analysis. On the basis of the favorable findings of this study, the staff considers that open item 11 may be reclassified as confirmatory item 71. For final resolution of this item, the staff will require (1) the review and approval of the above-described Westinghouse analysis, (2) justification that the Westinghouse methodology is applicable to the Millstone 3 steam generator model, and (3) justification that the Westinghouse generic analysis is applicable to the Millstone 3 plant.

6.2.1.5 Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies

The staff reported in its SER that additional information was required from the applicant to complete the review of the applicant's minimum containment pressure

analysis. Consequently, the analysis of minimum containment pressure for emergency core cooling system performance was classified as confirmatory item 24.

In FSAR Amendment 12, the applicant provided additional information regarding the containment analysis, including mass and energy release rate data, passive heat sink data, and the calculated containment pressure response. The staff has performed a confirmatory minimum containment pressure analysis using the CONTEMPT-LT/28 computer code and the data supplied by the applicant. This analysis was carried out in accordance with the guidance set forth in Branch Technical Position (BTP) CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," Rev. 2. The CONTEMPT results are in good agreement with those provided by the applicant for the entire transient. On this basis, the staff concludes that confirmatory item 24 has been satisfactorily resolved.

6.2.4 Containment Isolation System

In the SER, the staff addressed containment isolation dependability, NUREG-0737 Item II.E.4.2, but did not address demonstration of operability of the containment purge and vent valves. The following information addresses this matter.

Demonstration of operability of the containment purge and vent valves, particularly the ability of these valves to close during a design-basis accident, is necessary to ensure containment isolation. This demonstration of operability is reflected in SRP Section 6.2.4, BTP CSB 6-4, and SRP Section 3.10 for containment purge and vent valves that are not sealed closed during operational conditions 1, 2, 3, and 4.

By letter dated April 3, 1985, the applicant committed to administratively control the closed containment isolation valves in the containment purge air subsystem during operating modes 1, 2, 3, and 4. The staff finds it acceptable that all valves greater than 3 inches in diameter that purge and vent containment be sealed closed during operating modes 1, 2, 3, and 4. Furthermore, it should be verified once every 31 days that these valves are closed. Appropriate Technical Specifications should be issued to reflect this evaluation.

6.2.6 Containment Leakage Testing Program

In the SER, the staff indicated that the applicant's proposal to leave the containment liner weld channels unvented during type A testing would be acceptable, provided the applicant demonstrates that the weld channel design is compatible with that for the steel liner. This matter was classified as confirmatory item 22, pending receipt and review of additional information from the applicant.

By letter dated October 18, 1984, the applicant indicated that for all load combinations considering emergency, test, normal, and severe operational conditions, the stress levels for the leak chase members are within the acceptance criteria as defined in FSAR Section 3.8, thereby maintaining the leaktight pressure boundary.

On the basis of a review of the information provided by the applicant, the staff concludes that the weld channels will withstand the combined pressure and temperature loads for design-basis accidents. Accordingly, the staff concludes that confirmatory item 22 has been satisfactorily resolved, and that type A testing may be performed with the weld channels unvented, as proposed by the applicant.

6.6 Inservice Inspection of Class 2 and 3 Components

6.6.3 Evaluation of Compliance With 10 CFR 50.55a(g)

10 CFR 50.55a(b)(2)(iv) requires that ASME Code, Class 2, piping welds in the residual heat removal (RHR) systems, emergency core cooling (ECC) systems, and containment heat removal (CHR) systems shall be examined. In the SER, the staff indicated that its review of the preservice inspection program plan revealed that the Class 2 portions of the high-pressure safety injection (HPSI), chemical and volume control (CVC), and containment spray (CS) systems would receive no volumetric inspection. The staff's position is that the preservice inspection program must include volumetric examination of a representative sample of welds in the RHR, ECC, and CHR systems. This was considered as an open item in the SER.

The staff's main concerns were the lack of volumetric examinations planned on piping having a wall thickness of 1/2 inch or less and the absence of any volumetric examinations at all on the 4-inch nominal pipe size (NPS) HPSI system.

The RHR, ECC, and CHR systems contain approximately 1200 circumferential welds of which about 200 have wall thicknesses 1/2 inch or greater and are subject to both volumetric and surface examinations under ASME Code Section XI. The remaining 1000 welds have wall thicknesses less than 1/2 inch and are subject to surface examination only, according to the applicable code. However, the applicant has agreed to perform volumetric examinations on a 7.5% sample of these welds distributed among the three safety systems mentioned above. The inspection does not include open-ended, dried, and vented portions of the quench spray system and recirculation spray system.

The applicant has also agreed to perform volumetric and surface examination on a 7.5% sample of those portions of the HPSI system that are 4-inch NPS and a surface examination on a 7.5% sample of those portions of this system that are 3-inch NPS, which is in excess of the requirements of the ASME Code.

In addition, because of difficulties normally experienced when performing ultrasonic examination on thin-walled material, the applicant has stated that it will develop a special non-Code examination procedure utilizing 70° longitudinal waves for these inspections. The staff finds these commitments acceptable and considers the open item resolved.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.2 Specific Findings

7.2.2.4 Design Modification for Automatic Reactor Trip Using Shunt Coil Trip Attachment

During a site visit on May 6, 1985, the staff reviewed the reactor trip elementary diagrams ESK-11A, 11B, 11C, and 11D, and also verified at the reactor trip breaker switchgear that the applicant has implemented the required modification. This resolves confirmatory item 30.

7.3 Engineered Safety Features Systems

7.3.3 Specific Findings

7.3.3.4 Steam Generator Level Control and Protection

In its SER, the staff found that the design for activation of feedwater isolation did not meet the requirements of Paragraph 4.7 of IEEE Std. 279-1971 on "Control and Protection System Interaction" in that the failure of the level channel used for control could require protective action and the remainder of the protection system channels would not satisfy the single-failure criterion. By letter dated May 4, 1984, the applicant stated that the high steam generator level trip will be changed to two-out-of-four logic. During the site audit on May 6-8, 1985, the staff reviewed the updated logic diagrams. The staff finds that this design modification meets the requirements of Paragraph 4.7 of IEEE Std. 279-1971 and is, therefore, acceptable.

7.3.3.5 IE Bulletin 80-06 Concerns

By letter dated September 11, 1984, the applicant committed to perform a confirmatory test to verify that the safety-related equipment will remain in its associated emergency mode following reset of the engineered safety features (ESF) actuation signal. In FSAR Chapter 14, preoperational/acceptance test program test numbers 66 and 67 describe the test objective and summary. The staff finds that the design is consistent with the intent of the bulletin, and the applicant's commitment to perform a confirmatory test is acceptable. This resolves confirmatory item 35.

7.3.3.8 Control Building Isolation Reset

In its SER, the staff raised a concern about the design of the reset/override features used for control building isolation signals. Because the design for this safety function is based on one-out-of-two logic for some of the initiating conditions, single failures in instrument channels associated with these functions may result in system actuation. In response to this concern, the applicant modified the design so that the use of the reset/override features, which

blocks an initiating signal for one condition, will not defeat system initiation by other initiating signals. During the site audit on May 6-8, 1985, the staff reviewed the updated logic diagrams. The staff finds that this design modification is acceptable.

7.3.3.11 Non-Class 1E Control Signals to Class 1E Control Circuits

By letter dated September 21, 1984, the applicant submitted the logic diagrams and elementary diagrams for those systems which have non-Class 1E control signals input to Class 1E control circuits. In a meeting on February 21, 1985, in Bethesda, Md., the applicant identified all the non-Class 1E signals on the drawings. These non-Class 1E signals are electrically isolated from the Class 1E control systems via a qualified isolation device (coil to contact). Furthermore, these non-Class 1E signals are bypassed when the control system receives an ESF actuation signal. During normal operation, the non-Class 1E signals provide information to the control systems so that the action taken is always in the conservative direction and the safety systems are not degraded. The staff finds this acceptable. This resolves confirmatory item 39.

7.3.3.13 Sequencer Deficiency Report

By letter dated April 30, 1985, the applicant reported a potential significant deficiency in accordance with 10 CFR 50.55(e)(1) concerning the emergency generator load sequencers (labeled by the applicant SD-45). The deficiency involved the omission of a reset function in the loss-of-coolant accident (LOCA) time delay automatic test circuitry. This deficiency will cause the mistiming and premature starting of the recirculation pumps, and possible damage to the pumps because of inadequate net positive suction head. The vendor provided a design modification package to correct this deficiency. However, during a test after the modification, it was discovered that a similar problem existed for time delay design under the loss-of-power conditions. This additional concern was reported to the resident inspector on January 21, 1985. A second design modification was provided by the vendor to correct the deficiency for the loss-of-power conditions.

The applicant completed the modifications and performed confirmatory tests to verify the design changes. The sequencers were tested successfully. The staff considers this matter closed. This resolves confirmatory item 40.

7.3.3.14 BOP Instrumentation and Control System Testing Capability

In FSAR Amendment 8, the applicant documented the testing capabilities for the reactor trip system and the ESF system which include nuclear steam supply system and balance of plant instrumentation and control systems. The staff has reviewed the schematic diagrams and finds that the design is acceptable. This resolves confirmatory item 41.

7.5 Information Systems Important to Safety

7.5.2 Specific Findings

7.5.2.4 NUREG-0737 Item II.F.1, Accident Monitoring Instrumentation, Attachments 4, 5, and 6

Attachments 4, 5, and 6 of this TMI Action Plan item require installation of extended range containment pressure monitors, containment water level monitors, and containment hydrogen concentration monitors. In the Millstone 3 SER, the staff requested that the applicant provide instrument accuracy and justify that accuracy is adequate for the intended function. By letter dated May 14, 1985, the applicant provided the following information:

(1) Containment Pressure (Extended Range)

Instrument loop accuracy during accident conditions is ±7.8% of span. During accident conditions, these instruments will provide operators with trending information to aid them in determining the effectiveness of the engineered safety features.

(2) Containment Water Level (Wide Level)

Instrument span is 16 feet starting at approximately 1 foot above the bottom of the containment recirculation sump. This range is equivalent to approximately 4000 gallons to 1,500,000 gallons. Instrument loop accuracy is ±5.0 inches. During accident conditions, the operator uses this instrument to verify that water is in the containment sump before allowing the containment spray recirculation pumps to start automatically. Before the recirculation pumps start, more than 9000 gallons (approxi-mately 8 inches of indication) will be in the sump.

(3) Containment Hydrogen Monitor

Instrument accuracy is ±2% of full scale for a range of 0 to 10% hydrogen. The operator can monitor the hydrogen concentration inside the containment to within ±0.2% hydrogen. Because the minimum flammability concentration of hydrogen in air is approximately 4%, the accuracy is adequate to enable the operator to determine how close the containment hydrogen concentration is to the flammability concentration and to monitor the function of the hydrogen recombiners.

The staff has reviewed the justifications and finds that these instruments will not cause ambiguous indication for the operator during the accident conditions. The accuracy is adequate for the intended function and is, therefore, acceptable.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.5 Overhead Heavy Load Handling System

In the SER, the staff stated that the applicant had recently provided information regarding the overhead heavy load handling system. The staff has completed its review of this information.

The overhead heavy load handling systems were reviewed in accordance with SRP Section 9.1.5. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria, except as noted below, formed the basis for the staff's evaluation of the overhead heavy load handling system with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria for the overhead heavy load handling system includes conforming to the guidelines of ANS Std. 57.1, "Design Requirements of Light Water Reactor Fuel Handling Systems," and ANS Std. 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants." The guidelines contained in the "Review Procedures" and NUREG-0162, "Control of Heavy Loads at Nuclear Plants: Resolution of Generic Technical Activity A-36," were used in lieu of ANS Stds. 57.1 and 57.2.

The overhead heavy load handling system consists of components and equipment used to move loads weighing more than one fuel assembly, and its associated handling device. The equipment includes the containment polar crane and the spent fuel shipping cask trolley for safe handling of the reactor vessel head, reactor internals, shielded plug segments, and the spent fuel cask. It also includes the new fuel receiving crane, the new fuel handling crane, and the decontamination crane in the fuel building.

The fuel handling portion of the overhead heavy load handling system is housed within the seismic Category I, flood- and tornado-protected containment and the fuel building. The containment polar crane, the spent fuel shipping cask trolley, and other critical components of fuel handling systems are designed to seismic Category I criteria so they will not fail in a manner that results in unacceptable consequences (such as fuel damage or damage to safety-related equipment). However, the cranes are not required to function following a safe shut-down earthquake (SSE). Therefore, the design satisfies the requirements of GDC 2 and the guidelines of RG 1.13, "Spent Fuel Storage Facility Design Bases," Positions C.1 and C.6, and RG 1.29,"Seismic Design Classifications," Position C.2.

Millstone 3 does not share its overhead heavy load handling system with the other Millstone units. Therefore, the requirements of GDC 5, "Sharing of Structures, Systems and Components," are not applicable.

The new fuel handling crane is used mainly to move new fuel into the fuel transfer canal but has the capacity for placing spent fuel storage racks into the spent fuel pool. This crane is equipped with electrical interlocks to prevent it from carrying any load over the spent fuel pool. The bypassing of the electrical interlocks requires written procedures and approval. The new fuel receiving crane and the spent fuel shipping cask trolley are physically incapable of carrying heavy loads over the spent fuel pool. No safety-related equipment is located along the path of travel of these cranes. The light load handling spent fuel bridge and hoist crane is used for moving spent fuel from the spent fuel pool to the cask loading area for shipping. The shipping cask is not lifted to an elevation above any structure surface high enough to cause damage that could result in unacceptable radiological release should the cask be dropped. A dropped cask cannot result in fuel damage in excess of that assumed in the design-basis fuel handling accident, nor can it damage safety-related equipment. Thus, the requirements GDC 4, "Environmental and Missile Design Bases," and GDC 61, "Fuel Storage and Handling and Radioactivity Control," and the guidelines of RG 1.13, Positions C.3 and C.5, are satisfied for handling of the spent fuel cask.

As a result of Generic Task A-36, "Control of Heavy Loads Near Spent Fuel," a set of guidelines was developed to ensure safe handling of heavy loads over structures, systems, and components important to safety. These recommendations were documented in NUREG-0612.

Following the issuance of NUREG-0612, a generic letter dated December 22, 1980, was sent to all holders of operation licenses and construction permits, requesting that responses be prepared to indicate the degree of compliance with the guidelines of NUREG-0612. The responses were made in two stages. The first response (Phase I, Section 5.1.1 of NUREG-0612) was to identify the load-handling equipment within the scope of NUREG-0612 and to describe the associated general load-handling operations such as safe load paths; procedures; operator training; special and general purpose of lifting devices; the maintenance, testing, and repair of equipment; and the handling-equipment specifications.

The applicant has responded to the December 22, 1980 generic letter. The staff and its consultant, Idaho National Engineering Laboratory (INEL), have reviewed the applicant's Phase I submittals. As a result of its review, INEL provided a Technical Evaluation Report (TER), which is included as Appendix J to this supplement. The staff has reviewed the TER and concludes that the applicant's program for resolving the Phase I concerns of NUREG-0612 for Millstone 3 is acceptable.

The applicant's response to Phase II provided analyses of load drops throughout the plant, including containment, which showed that unacceptable releases of radioactivity or damage to safety-related equipment would not occur, or that safe load paths and/or handling procedures were developed for the load.

The staff further concludes that with the completion of Phase I and on the basis of the above response and the Phase II review to date, no further action is required concerning Phase II of NUREG-0612. Therefore, the staff concludes that the requirements of GDC 4 and 61 and the guidelines of RG 1.13, Positions C.3 and C.5, have been satisfied for the overhead heavy load handling systems at Millstone 3.

In addition, the applicant has responded to the staff's concern regarding the failure of heavy loads such as concrete shield plugs during handling. The applicant stated that these structures are of reinforced concrete design with embedded lifting lugs which will not fall apart. The movement of these concrete structures follows the safe load path in accordance with procedures to preclude dropping heavy loads on safety-related equipment or spent fuel. The applicant also stated that in the event of a heavy load drop, unacceptable fuel damage or damage to safety-related equipment will not occur. Therefore, adequate protection is provided against damage to safety-related equipment from postulated load failures.

On the basis of its review, the staff concludes that the overhead heavy load handling system is in conformance with the requirements of GDC 2, 4, and 61 as they relate to its protection against natural phenomena, missile protection, safe handling of spent fuel casks, and the guidelines of RG 1.13, Positions C.1, C.3, C.5, and C.6; and RG 1.29, Positions C.1 and C.2, with respect to overhead crane interlocks and maintaining plant safety in a seismic event. The system, therefore, is acceptable. The overhead heavy load handling system meets the acceptance criteria of SRP Section 9.1.5.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

9.5.1.1 Fire Protection Program Requirements

Fire Hazards Analysis

In its SER, the staff indicated concern over whether the mechanisms by which fire protection systems may cause the simultaneous failure of redundant or diverse trains have been adequately considered in the applicant's design.

By letter dated April 2, 1985, the applicant stated that the design for components required for hot shutdown was such that a rupture or inadvertent operation of fire suppression systems will not adversely affect the operability of these components. Redundant trains of components that are susceptible to damage from water spray are physically separated so that manual fire suppression activities will not adversely affect the operability of components not involved in the postulated fire.

Automatic suppression systems have been designed and located so that operation of the systems, either intentional or inadvertent, will not cause damage to redundant trains of safety-related equipment that are needed for safe shutdown of the plant.

On the basis of its review, the staff concludes that the fire hazards analysis will meet the guidelines in Section C.1.b of Branch Technical Position (BTP) CMEB 9.5-1, and is, therefore, acceptable.

9.5.1.4 General Plant Guidelines

Building Design

In its SER, the staff stated that in the Fire Protection Evaluation Report, the applicant indicated that door openings in fire-rated barriers are provided with fire door assemblies that have ratings commensurate with the fire ratings of the walls in which they are installed; however, the applicant did not state whether the doors will be tested and labeled. The applicant has since stated that each fire door assembly will be labeled or listed by Underwriters Laboratory (UL). The applicant will document this commitment in a future amendment to the Fire Protection Evaluation Report. Until the staff receives and evaluates this information, the issue remains open.

Fire Protection for Safe Shutdown Capability

The applicant has provided a cable separation review for all rooms of the plant housing safe shutdown equipment to ensure that at least one train of this equipment and essential instrumentation is available in the event of a fire in any of these rooms.

The review identified the safety-related equipment and the redundant safe shutdown system cabling and discussed the consequences of a fire in each of these rooms. Cable and equipment were considered disabled in the area of the fire unless the fire hazard analysis assumed otherwise. The staff has reviewed the applicant's deterministic review of the plant and concludes that it provides an acceptable means of demonstrating that separation exists between redundant safe shutdown trains.

All three component cooling water pumps can be damaged by a fire in the auxiliary building. One of these pumps is required to bring the plant to cold shutdown. The applicant has committed to provide the capability to repair or replace one pump motor or cabling in either train of the component cooling water using only onsite material and power and still achieve cold shutdown within 72 hours of a reactor trip. This commitment is acceptable as it complies with BTP CMEB 9.5-1, Section C.5.b.(1)(b), "Systems necessary to achieve and maintain cold shutdown...can be repaired within 72 hours."

The applicant's fire analysis has also indicated that alternate shutdown measures were required for the control room, the instrument rack room, and the cable spreading room because these areas contain more than one division of safe shutdown cabling. To provide the alternate shutdown capability, the applicant has added a fire transfer switch panel for the orange train and an auxiliary shutdown panel with controls and instrumentation in two separate fire areas in the switchgear rooms. The controls on these panels along with certain actions at local stations provide the capability to safely shut down the plant to the cold shutdown condition without the use of the control room, instrument rack room, or the cable spreading room (see Section 9.5.1.5 of this supplement for further discussion).

The plant can go to the hot shutdown condition without repairs. Cable runs were traced through each fire area from the corresponding safe shutdown components to their power source. Additional equipment and electrical circuitry

considered as associated either because of shared common power source, common enclosure, or whose fire-induced spurious operation could affect shutdown were also identified. For the identified associated circuits, the applicant has provided circuit isolation and/or procedures to ensure that circuit failures would not prevent safe shutdown. For example, in order to prevent fire-induced spurious signals from causing loss-of-coolant accidents (LOCAs) from sources such as the residual heat removal (RHR) suction line or power-operated relief valves (PORVs), the following measures are provided. The applicant will lock out power to one of the two series RHR suction line valves during power operation. Similarly, the operator will trip the power supply breakers and remove fuses for those solenoid-operated PORVs whose controls are not provided at the auxiliary shutdown panel, thereby preventing any fire-induced spurious actuation of the PORVs.

With regard to common bus, multiple, high-impedance fault resulting from a combination of faults associated with two or more cables in a fire area, essential power supply to safe shutdown equipment may be cut off as a result of tripping of the main breaker instead of the individual circuit breakers. To this concern, the applicant stated that the operator will trip the non-required loads and reenergize the bus.

On the basis of the above, the staff concludes that the function of reactivity control, primary system inventory control, decay heat removal and pressure control are met. The staff further concludes that the post-fire, safe-shutdown systems and cable separation methodology comply with GDC 3 and the criteria of Appendix R, Section III.G, as contained in BTP CMEB 9.5-1, Section C.5.b.

In FSAR Table 1-9, the applicant requested a deviation from BTP CMEB 9.5-1, Section C.5.b, for the fire protection of safe-shutdown equipment. The deviation request was general in nature and did not specify the areas of the plant or the specific safe-shutdown system for which the applicant was requesting the deviation. In addition, the applicant did not provide adequate information to justify the deviations. The staff will require the applicant to provide protection for all safe-shutdown trains in accordance with Section C.5.b of BTP CMEB 9.5-1 or to provide adequate technical justification for any deviations.

Alternate Shutdown Capability

FSAR Section 7.4.1 and Sections 7.9 and 8.3 of the applicant's Fire Protection Evaluation Report describe the remote shutdown panel's (auxiliary shutdown and its associated transfer switch panels) design and capability. The design objective of the auxiliary shutdown panels is to provide a control point to control and monitor plant shutdown independent of the control room in the event of a fire in the control room, instrument rack room, or cable spreading room. The staff reviewed the design to the guidelines of BTP CMEB 9.5-1, Section C.5.c. The design of the fire transfer and auxiliary shutdown panels provides electrical isolation for the instrument indications and control functions for the shutdown systems from the control room. The auxiliary feedwater system, the main steam atmospheric relief valves, the reactor coolant boration system, the letdown system, and the reactor pressure can be manually controlled from the panels to achieve and maintain hot shutdown independent of the control room. Initiation of the RHR system for achieving cold shutdown is performed locally. Support system functions are initiated either at the auxiliary shutdown panel or locally.

For cold shutdown, repair of a component cooling water pump is required following a fire in an area of the auxiliary building on elevation 24 feet 6 inches (mean sea level). The applicant has committed to provide the capability to repair or replace one pump motor in either train of the component cooling water system with onsite dedicated materials and still achieve cold shutdown conditions within 72 hours of reactor trip using only onsite power.

The applicant has stated that the station emergency shutdown procedures will specify the manual actions required and will address manpower requirements. No fire brigade members will be included in the shutdown manpower requirements. The plant Technical Specifications will include requirements for periodic testing of the remote shutdown control circuit, transfer switches, and instrumentation. The applicant will rely on operator actions to place the plant in the safe-shutdown conditions. In order to complete its evaluation, the staff requires a description of the post-fire operator actions that includes the priority of the operator actions, the number of actions, the time allowed for each action, and the time in which the action must be completed.

As part of the review of the alternate shutdown capability and as a result of the issuance of IE Information Notice 85-09, "Isolation Transfer Switches and Post-Fire Shutdown Capability," the staff has indicated to the applicant that the circuitry needed for alternate shutdown should not contain a single fuse which could be blown as a result of a fire-induced short before isolation from the control room. If the fuses were blown, they would need to be replaced in order to achieve operation from the remote (auxiliary) shutdown panel. Such replacements are considered a repair, and repairs are not permitted in achieving hot shutdown. The staff has indicated to the applicant that modifications (such as installation of redundant fuses) should be provided as necessary to ensure achieving hot shutdown without repairs. The applicant has not yet responded. The staff will address compliance with the concern identified in IE Information Notice 85-09 in a future SER supplement.

Reactivity control is accomplished by a manual scram before the operator leaves the control room. Boron addition and reactor coolant makeup water are provided by the chemical and volume control system (charging systems). Reactor decay-heat removal in hot shutdown is provided through the steam generator by the auxiliary feedwater system and main steam atmospheric relief valves, and in cold shutdown by the RHR system, component cooling water system, and the safety-related service water system. Cold shutdown can be achieved within 72 hours following a fire in any plant area. The following direct reading of process variables is provided at the auxiliary shutdown panel:

(1) pressurizer pressure
(2) pressurizer level
(3) reactor coolant hot-leg temperature
(4) steam generator level
(5) steam generator pressure
(6) auxiliary feedwater flow
(7) reactor coolant pressure
(8) neutron source-range monitor
(9) neutron intermediate-range monitor

The applicant has not provided reactor coolant cold-leg temperature (T_C) indication at the auxiliary shutdown panel and proposes to use T_{sat}, the saturation temperature corresponding to the secondary-side steam generator pressure, in place of T_C for monitoring natural circulation. The applicant further stated that the procedures for cooldown from the auxiliary shutdown panel can be developed on the basis of the temperature difference between T_{sat} and reactor coolant system (RCS) hot-leg temperature (T_H), and that these procedures can take into account the time lag between T_{sat} and T_C.

The staff will require that direct reading of T_C be provided in accordance with BTP CMEB 9.5-1, Section C.5.c(2)(d): The reactor coolant temperatures, in conjunction with the RCS pressure, are essential parameters for plant shutdown and control.

The plant control elements that rely on accurate reactor coolant temperature indication are natural circulation, subcooling, and pressurized thermal shock concerns. In the natural circulation mode of operation, the hot-leg temperature, cold-leg temperature, and the difference between the hot-leg and cold-leg temperatures (T_H - T_C) provide indication by which natural circulation conditions can be determined. In order to verify that natural circulation has been established, normal plant procedures require the operator to use cold-leg temperature (T_C). The applicant suggested that the saturation temperature corresponding to the secondary-side steam generator pressure T_{sat} will approximate T_C. The staff acknowledges that such a condition can exist if natural circulation is occurring; however, if natural circulation is not occurring, the staff cannot assume T_C to be approximately T_{sat}. Cooldown is usually achieved by the operator controlling the steam generator pressure and auxiliary feedwater flow to the steam generators. Because of the inherent lag in response between the secondary and primary side, T_C cannot be inferred from T_{sat}. Natural circulation is normally determined by knowing T_H, T_C, observing that T_H and T_C are constant or decreasing, and by monitoring T_H - T_C. Since normal control room procedures require the use of T_C in confirming natural circulation, emergency procedures should not deviate from this practice. Thus, cold-leg temperature (T_C) wide-range indication is necessary for the auxiliary shutdown panel.

On the basis of its review, the staff cannot conclude that the alternative shutdown capability complies with the requirements of GDC 3 and the criteria of Appendix R, Section III.L as contained in BTP CMEB 9.5-1, Section C.5.c. The staff's review of the applicant's description of the post-fire operator actions (with time allowed for each action), compliance with the concerns identified in IE Information Notice 85-09, and provision of RCS cold-leg temperature indication at the alternative shutdown panel will be discussed in a future supplement to the SER.

9.5.1.5 Fire Detection and Suppression

Water Sprinkler and Hose Standpipe Systems

In its SER, the staff stated that all sprinkler and hose station standpipe systems have independent yard fire main connections, except for the emergency generator enclosure, service building, waste disposal building, containment building, and the auxiliary boiler room. In its SER, the staff stated it was concerned that in the above areas a single active failure or a crack in a moderate-energy line could impair both the primary and backup fire suppression systems. By letter dated April 2, 1985, the applicant provided additional information. The applicant indicated temporary hose lengths would be used to provide both primary and backup fire suppression systems. The information provided in the April 2, 1985, letter is the same that the staff had previously reviewed and reported in its SER--that the configuration of the connection for primary and backup water suppression systems to the fire main and the use of temporary hose was unacceptable.

The staff will require the applicant to provide a fire protection water supply for the emergency generator enclosure, service building, waste disposal building, containment building, and the auxiliary boiler room so that a single break or failure in the supply piping will not result in the loss of both the primary and secondary water supplies consistent with BTP CMEB 9.5-1, Section C.6.c(1).

9.5.1.6 Fire Protection of Specific Plant Areas

Cable Spreading Room

In its SER, the staff reported that the fire protection provided for the cable spreading room was not adequate.

By letters dated October 9, 1984, and April 30, 1985, the applicant demonstrated that the cable spreading room had been designed to allow manual fire fighting operations to be effective in extinguishing a fire within the cable spreading room. Access and egress routes within the cable spreading room are well defined. Cable tray arrangement and separation in the cable spreading room has been provided to allow the fire brigade ample time to effectively control and extinguish any anticipated fires. In addition, the staff visited the site and concurs with the applicant that adequate access is provided to allow the fire brigade to effectively fight anticipated fires.

In the applicant's Fire Protection Evaluation Report and by letter dated October 9, 1984, the applicant has demonstrated that the design of the CO_2 system for concentration, anoxia and toxicity, thermal shock, overpressurization, and location of detectors is in accordance with the guidelines of BTP CMEB 9.5-1, Section C.6.d.

The staff finds that the CO_2 extinguishing system with good access for manual fire fighting with hose streams will provide an adequate level of protection for the cable spreading room and is, therefore, an acceptable deviation from staff guidelines.

On the basis of its evaluation, the staff concludes that the protection provided for the cable spreading room with the approved deviation meets the guidelines of BTP CMEB 9.5-1, Section C.7.c, and is, therefore, acceptable.

9.5.1.7 Summary of Deviations From BTP CMEB 9.5-1

The following deviations from the guidelines of BTP CMEB 9.5-1 have been identified and approved:

(1) installation of a 3-hour fire-rated damper in the ductwork rather than in the wall (SER Section 9.5.1.4)

(2) fire water supply tank size (SER Section 9.5.1.5)

(3) no connection of the standpipe system to a seismic Category I water system (SER Section 9.5.1.5)

(4) no floor drains in the switchgear rooms (SER Section 9.5.1.6)

(5) use of standpipe smaller in diameter than 4 inches and 2½ inches for multiple and single hose station supplies, respectively (SER Section 9.5.1.5)

(6) automatic gas suppression system for protection of concentrated cable trays (SER Section 9.5.1.4)

(7) automatic gas suppression system for protection of the cable spreading room (SER Section 9.5.1.6)

9.5.1.8 Conclusions

The following areas are the unresolved fire protection items:

(1) safe shutdown capability (SER Section 9.5.1.4)
(2) alternate shutdown capability (SER Section 9.5.1.4)
(3) independent sprinkler and hose station connections (SER Section 9.5.1.5)
(4) qualification of fire doors (SER Section 9.5.1.4)

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

13.1.2 Plant Staff

13.1.2.1 Organization

In Section 13.1.2 of the SER, the staff stated that the applicant's plan not to have a shift technical advisor (STA) was considered to be an open item pending final Commission action on the "Commission Policy Statement on Engineering Expertise on Shift."

In a letter dated June 24, 1985, the applicant restated its position in regard to the STA and provided further information on the background and education of the current shift supervisors. In this letter, the applicant stated that six of the eight current shift supervisors hold bachelor's degrees in engineering or applied science.

At this time the staff's criteria require that an individual who will serve the dual role as shift supervisor and STA have a bachelor's degree in engineering or applied science, a senior operator license, and complete an approved STA training program. Therefore, the staff finds it acceptable for the applicant to use these six individuals in the dual role of shift supervisor/shift technical advisor provided they have senior operator licenses and have completed the STA training program. However, the acceptability of the other two shift supervisors remains an open item.

13.5 Station Administrative Procedures

13.5.1 Administrative Procedures

13.5.1.2 Limitation on Working Hours

In Section 13.5.1.2, "Limitation on Working Hours," of the SER, the staff concluded that the applicant's policy on this subject was an open item since it did not apply to all personnel performing safety-related functions as described in Generic Letter 82-12. By letter dated April 15, 1985, the applicant advised the staff that it has now implemented a nuclear engineering and operations policy statement and procedure limiting overtime for all nuclear plant personnel. The staff finds that this revised policy meets the intent of Generic Letter 82-12 and is acceptable. Therefore, the staff concludes that the applicant has established a policy governing the working hours of all nuclear plant personnel performing safety-related functions to ensure that personnel are in the proper physical condition.

13.5.1.8 Summary and Conclusions

In the SER, the staff erroneously concluded that verification of the correct performance of operating activities was an open item. Appendix H corrects this.

In addition, the staff has reviewed administrative procedures and finds that they meet the guidance in Section 5.2 of ANSI/ANS 3.2, RG 1.33, and the applicable parts of TMI Action Plan Items I.A.1.2, I.A.1.3, I.C.2, I.C.3, I.C.4, I.C.5, and I.C.6.

15 ACCIDENT ANALYSES

15.9 TMI Action Plan Requirements

15.9.13 II K.3.30 Revised Small-Break LOCA Methods To Show Compliance With 10 CFR 50, Appendix K

In its SER, the staff stated that the revised Westinghouse model for small-break loss-of-coolant-accident (SBLOCA) analysis was under staff review. On May 21, 1985, the staff approved the new Westinghouse SBLOCA model, NOTRUMP, for use in satisfying TMI Action Plan Item II.K.3.30. This section presents the staff's basis for approval of NOTRUMP.

NUREG-0737 is a report transmitted by a letter from D. G. Eisenhut, Director of NRC's Division of Licensing of the Office of Nuclear Reactor Regulation, to licensees of operating power reactors and applicants for operating reactor licenses forwarding TMI Action Plan requirements which have been approved by the Commission for implementation. Section II.K.3.30 of Enclosure 3 to NUREG-0737 outlines the Commission's requirements for the industry to demonstrate its SBLOCA methods continue to comply with the requirements of Appendix K to 10 CFR 50.

The technical issues to be addressed were outlined in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants." In addition to the concerns listed in NUREG-0611, the staff requested that licensees with U-tube steam generators assess their computer codes with the Semiscale S-UT-08 experimental results. This request was made to validate the code's ability to calculate the core coolant-level depression as influenced by the steam generators before loop seal clearing.

In response to TMI Action Plan Item II.K.3.30, the Westinghouse Owners Group (WOG) has elected to reference the Westinghouse NOTRUMP code as its new licensing SBLOCA model. Referencing the new computer code did not imply deficiencies in WFLASH to meet the Appendix K (10 CFR 50) requirements. The decision was based on desires of the industry to perform licensing evaluations with a computer program specifically designed to calculate SBLOCAs with greater phenomenological accuracy than would be capable by WFLASH.

The material that follows documents the staff's evaluation of the WOG response to TMI Action Plan Item II.K.3.30 confirmatory items.

15.9.13.1 Summary of Requirements

NUREG-0611 required licensees and applicants with Westinghouse nuclear steam supply system (NSSS) designs to address the following concerns:

(1) Provide confirmatory validation of the SBLOCA model to adequately calculate the core heat transfer and two-phase coolant level during core uncovery conditions.

(2) Validate the adequacy of modeling the primary side of the steam generators as a homogeneous mixture.

(3) Validate the condensation heat transfer model and effects of non-condensable gases.

(4) Demonstrate, through noding studies, the adequacy of the SBLOCA model to calculate flashing during system depressurization.

(5) Validate the polytropic expansion coefficient applied in the accumulator model.

(6) Validate the SBLOCA model with LOFT tests L3-1 and L3-7. In addition, validate the model with the Semiscale Test S-UT-08 experimental data.

Detailed responses to these six items are documented in WCAP-10054, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code."

15.9.13.2 Evaluation

The staff evaluated the TMI Action Plan Item II.K.3.30 requirements outlined above.

(1) Core Heat Transfer Models

The WOG referenced the NOTRUMP computer code as its new computer program for SBLOCA evaluation. NOTRUMP was benchmarked against core uncovery experiments conducted at the Oak Ridge National Laboratory (ORNL). These tests were performed under NRC sponsorship. The good agreement between the calculations and the data confirmed the adequacy of the drift flux model used for core hydraulics as well as the core heat transfer models of cladding temperature predictions.

The staff finds the core thermal-hydraulic models in NOTRUMP acceptable.

(2) Steam Generator Mixture Level Model

NUREG-0611 requested licensees and applicants with Westinghouse-designed NSSSs to justify the adequacy of modeling the primary system of the steam generators as a homogeneous mixture. This question was directed to the WFLASH code. NOTRUMP, the new SBLOCA licensing code, models phase separation and incorporates flow regime maps within the steam generator tubes. The adequacy of this model was demonstrated through benchmark analyses with integral experiments, in particular with Semiscale Test S-UT-08.

The staff finds the steam generator model in NOTRUMP acceptable.

(3) Noncondensable Effects On Condensation Heat Transfer

NUREG-0611 requested validation of the condensation heat transfer correlations in the Westinghouse SBLOCA model and an assessment of the consequences of noncondensable gases in the primary coolant. The condensation heat transfer model used in NOTRUMP is based on steam experiments performed by Westinghouse on a 16-tube pressurized-water reactor (PWR) steam generator model. For two-phase conditions, an empirical correlation developed by Shah (see Westinghouse report WCAP-10079) is applied.

The staff finds the condensation heat transfer correlation in NOTRUMP acceptable.

The influence of noncondensable gases on the condensation heat transfer was demonstrated by degrading the heat transfer coefficient in the steam generators. The heat transfer degradation was calculated using a boundary layer approach. For this calculation, the noncondensable gases generated within the primary coolant system were collected and deposited on the surface of the steam generator tubes. The sources of noncondensables considered were:

- air dissolved in the refueling water storage tank
- hydrogen dissolved in the primary system
- hydrogen in the pressurizer vapor space
- radiolytic decomposition of water

With a degradation factor on the heat transfer coefficient, the limiting SBLOCA was reanalyzed for a typical PWR. The WOG thereby concluded that formation of noncondensable gases in quantities that may reasonably be expected for a 4-inch cold-leg-break LOCA presents no serious detriment to the PWR system response in terms of core uncovery or system pressure. What perturbation was observed was minor in nature.

The staff finds acceptable the Westinghouse submittal on the influences of noncondensable gases on design-bases SBLOCA events. The staff's conclusion is based on the limited amount of noncondensable gases available during a design-basis SBLOCA event, as well as results obtained from Semiscale experiments which reached similar conclusions while injecting noncondensable gases in excess amount expected during an SBLOCA design-basis event.

(4) Nodalization Studies for Flashing During Depressurization

As a consequence of the staff's experience with modeling SBLOCA events with NRC-developed computer codes (in particular the TMI-2 accident), the staff questioned the adequacy of the nodalization in the licensing model to calculate the depressurization of the primary system. The staff therefore requested validation of the Westinghouse evaluation model to properly calculate the depressurization expected during an SBLOCA event.

Through nodalization studies and validation of the NOTRUMP licensing model with integral experiments (e.g., LOFT and Semiscale), Westinghouse demonstrated the acceptability of the nodalization and nonequilibrium models.

The staff finds the Westinghouse model acceptable for calculating depressurization during SBLOCA events.

(5) Accumulator Model

WFLASH, the previous Westinghouse SBLOCA analysis code, applied a polytropic gas expansion coefficient of 1.4 to the nitrogen in the accumulators. The WOG was requested to validate this accumulator model in light of data obtained through the LOFT experimental programs for SBLOCAs. Westinghouse reviewed the applicable LOFT data and determined the need to perform full-scale accumulator tests. On the basis of these tests, Westinghouse modified the polytropic expansion coefficient to a more realistic value. Of interest is Westinghouse's conclusion

that the selection of either a high or low expansion coefficient had negligible effect on the calculated peak cladding temperature (PCT). This insensitivity is only appropriate to NOTRUMP, with its nonequilibrium assumptions.

The staff finds acceptable the polytropic expansion coefficient in the NOTRUMP code.

(6) Code Validation

Following the Three Mile Island event of 1979, staff analyses with NRC-developed computer codes led to concerns that detailed nodalization was required to simulate realistic systems responses to postulated SBLOCAs. As a consequence, licensees and applicants with Westinghouse plants were asked to validate their licensing tools with integral experiments. Specifically, the NRC requested that the computer codes be validated with the LOFT L3-1 and L3-7 experimental data. In addition, the staff also requested that the code be benchmarked with the Semiscale S-UT-08 experimental data.

Westinghouse performed the above benchmark analyses. For the LOFT tests, Westinghouse showed good agreement between the NOTRUMP calculations and the experimental data. For the S-UT-08 test, Westinghouse demonstrated that NOTRUMP did a reasonable job calculating the experimental data. However, this required a more detailed nodalization of the steam generators than used in the licensing model. With the less detailed licensing nodalization, the pre-loop seal-clearing core-level depression phenomenon, as observed in the S-UT-08 data, was not conservatively calculated for very small breaks. However, the calculated peak cladding temperature was demonstrated to be higher (more conservative) with the coarse nodalization. The staff, therefore, finds acceptable the NOTRUMP computer code and the associated nodalization for SBLOCA design-basis evaluation.

15.9.13.3 Conclusion

The Westinghouse Owners Group, by referencing WCAP-10079 and WCAP-10054, has identified NOTRUMP as its new thermal-hydraulic computer program for calculating SBLOCAs. The staff finds acceptable the use of NOTRUMP as the new Westinghouse licensing tool for calculating SBLOCAs for Westinghouse NSSS designs.

The responses to NUREG-0611 concerns, as evaluated within this SER supplement, have also been found acceptable.

This SER supplement completes the requirements of TMI Action Plan Item II.K.3.30 for licensees and applicants with Westinghouse NSSS designs who were members of the WOG and who referenced WCAP-10079 and WCAP-10054 as their response to this item.

Within 1 year of receiving this supplement, the licensees and applicants with Westinghouse NSSS designs are required to submit plant-specific analyses utilizing NOTRUMP, as required by TMI Action Plan Item II.K.3.31. Per Generic Letter 83-35, compliance with TMI Action Plan Item II.K.3.31 may be submitted generically. The staff requires that the generic submittal include validation that the limiting break location has not shifted away from the cold legs to the hot or pump suction legs.

16 TECHNICAL SPECIFICATIONS

In the SER the staff identified certain issues that must be included in the Technical Specifications as a condition of staff acceptance. These issues were identified in Table 16.1 and were discussed further in sections of the report as indicated in parentheses after each item.

As a result of additional information in this supplement Table 16.1 is modified to include an additional item, Technical Specification Item (16).

Table 16.1 Technical Specification items (revised from SER Table 16.1)

Item	SER Section
(16) Verification that containment isolation valves are closed	6.2.4

APPENDIX A

CONTINUATION OF CHRONOLOGY OF THE NRC STAFF RADIOLOGICAL REVIEW OF THE MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

February 12, 1985	Letter to applicant concerning use of ASME Code Case N-407.
February 14, 1985	Letter from applicant concerning "in furtherance" of pollution certification.
February 19, 1985	Letter from applicant concerning oesign verification activities.
February 20, 1985	Representatives from NRC and Northeast Utilities meet in Bethesda, Md., to discuss responses to unresolved items and plans for Seismic Qualification Review Team audit to be held during the week of March 4-8, 1985.
February 28, 1985	Letter from applicant transmitting a response to SER open item 3.
March 1, 1985	Letter from applicant concerning partial response to SER open item 10.
March 1, 1985	Letter from applicant transmitting a response to SER confirmatory item 70.
March 1, 1985	Letter from applicant requesting an amendment to CPPR-113 for Millstone Unit 3 to reflect an exemption requested to General Design Criterion (GDC) 4 (10 CFR 50, Appendix A), as it relates to the dynamic effects associated with the reactor coolant system main loop pipe breaks.
March 4, 1985	Letter from applicant transmitting Amendment 12 to the Final Safety Analysis Report (FSAR).
March 4-8, 1985	Representatives from NRC and Northeast Utilities meet on March 4 at the construction offices of the Millstone 3 site in Waterford, Conn., and on March 5-8 at the emergency operations facility, Millstone 3 site, for the staff to perform Seismic Qualification Review Team and pump and valve operability audits to confirm the applicant's implementation of the qualification programs.

March 7, 1985	Letter to applicant requesting additional information for N-1 loop operation and the ultrasonic testing demonstration at Millstone 3 on November 9, 1984.
March 12, 1985	Representatives from NRC and Northeast Utilities meet in Bethesda, Md., to discuss emergency lighting illumination levels in the auxiliary shutdown panel area/purple switchgear room, control room, diesel generator control panel areas, and safety-related areas, and access and egress paths to those areas.
March 12, 1985	Letter from applicant transmitting a response to SER confirmatory item 18.
March 12, 1985	Letter from applicant transmitting Revision 1 to the Physical Security Plan for receipt of nuclear fuel.
March 14, 1985	Letter from applicant transmitting responses to SER confirmatory items.
March 18, 1985	Letter to applicant requesting additional information for the safety parameter display system audit.
March 22, 1985	Letter from applicant transmitting responses to SER confirmatory items.
March 26, 1985	Representatives from NRC and Northeast Utilities meet in Bethesda, Md., to discuss power-operated relief valves, block valves, and associated power trains and how overpressure protection will be achieved while operating at low temperature.
March 27, 1985	Letter to applicant certifying pollution control facilities.
April 1, 1985	Letter to applicant concerning physical security plan.
April 1, 1985	Letter from applicant concerning SER confirmatory items 44, 45, 46, and 50.
April 2, 1985	Letter from applicant transmitting responses to SER open items.
April 3, 1985	Letter from applicant transmitting response to question 260.58.
April 3, 1985	Letter from applicant transmitting response to question 271.1.

April 9, 1985	Representatives from NRC and Northeast Utilities meet in Bethesda, Md., to discuss status of Technical Specification review and schedule for review during the remaining 7 months before the projected license issuance in November 1985.
April 9, 1985	Letter from applicant transmitting the 1984 Annual Financial Report of the Connecticut Light and Power Company and Western Massachusetts Electric Company.
April 10-12, 1985	Representatives from NRC and Northeast Utilities meet in Waterford, Conn., to evaluate implementation of the onsite power systems designed as installed at the Millstone 3 site.
April 11, 1985	Letter from applicant concerning SER open item 13.
April 11, 1985	Letter from applicant concerning diesel generator air start system high energy pipe break, SER open item 2, and question 430.76.
April 11, 1985	Letter from applicant concerning final long forms for seismic Qualification Review Team/Pump and Valve Operability Review Team items.
April 15, 1985	Letter from applicant concerning SER open item 18.
April 16, 1985	Representatives from NRC and Northeast Utilities meet in Bethesda, Md., to discuss applicant's responses to staff's questions concerning conformance to RG 1.97 relevant to emergency response capability.
April 16, 1985	Letter from applicant transmitting a response to SER confirmatory item 27.
April 17, 1985	Letter from applicant concerning management changes
April 19, 1985	Letter from applicant transmitting a response to SER open item 10.
April 22, 1985	Letter from applicant concerning NUREG-0612.
April 26, 1985	Letter from applicant requesting Code case approval.
April 26, 1985	Letter from applicant transmitting a response to SER open item 2.
April 29, 1985	Letter from applicant transmitting Amendment 13 to the FSAR.
April 30, 1985	Letter from applicant transmitting responses to questions 492.8 through 492.12, dated March 7, 1985.

May 2, 1985	Letter from applicant transmitting a response to SER confirmatory item 14.
May 2, 1985	Letter from applicant transmitting a revised response to SER outstanding item 2.
May 2-8, 1985	Representatives from NRC and Northeast Utilities meet in Waterford, Conn., at the Millstone 3 site to evaluate actual implementation of instrumentation and control systems design as installed at Millstone 3 site.
May 6, 1985	Letter from applicant transmitting Amendment 9 to the Environmental Report.
May 7, 1985	Letter from applicant transmitting a response to question 250.12.
May 7, 1985	Letter from applicant requesting a schedular exemption from GDC 4.
May 10, 1985	Letter from applicant transmitting a response to NRC comments on Physical Security Plan.
May 13, 1985	Letter from applicant transmitting the annual reports of major electric utilities, licensees, and others for 1984 (no cover letter).
May 14, 1985	Letter from applicant transmitting responses to SER confirmatory items.
May 14, 1985	Representatives from NRC and Northeast Utilities meet in Bethesda, Md., to discuss fragility analysis of retaining wall, stability of beach sand slope, emergency generator enclosure building, and buried service water piping system.
May 20, 1985	Letter to applicant requesting additional information for Emergency Plan, Draft 2, to Revision 0, and Generic Letter 83-28.
May 20, 1985	Letter from applicant transmitting a response to SER confirmatory item 10.
May 31, 1985	Letter from applicant transmitting Revision 0 of the Millstone Nuclear Power Station Physical Security Plan.
June 3, 1985	Letter from applicant transmitting response to SER confirmatory items.

June 4, 1985	Representatives from NRC and Northeast Utilities meet in Bethesda, Md., to discuss deviations from Branch Technical Position CMEB 9.5-1 of the Standard Review Plan, Section 9.5.1.
June 5, 1985	Letter to applicant concerning environmental assessment.
June 5, 1985	Letter to applicant transmitting an exemption from GDC 4 regarding the need to analyze large primary loop pipe ruptures as the structural design basis for Millstone 3.
June 6, 1985	Letter from applicant transmitting a response to SER confirmatory item 61.
June 6, 1985	Letter to applicant concerning small-break loss-of-coolant accident analysis for Millstone 3, TMI Action Item II.K.3.30.
June 6, 1985	Letter to applicant transmitting requests for additional information from staff.
June 7, 1985	Letter from applicant responding to SER open item 2.2.
June 7, 1985	Letter from applicant concerning emergency operating procedures generation package response to NRC comments.
June 10, 1985	Letter to applicant transmitting the first draft of Technical Specifications.
June 10, 1985	Letter to Westinghouse withholding from public disclosure WCAP-10587, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for Millstone Unit 3."
June 11, 1985	Letter from applicant transmitting a response to SER confirmatory item 31.
June 11, 1985	Letter from applicant responding to question 410.32.
June 17, 1985	Letter from applicant responding to SER confirmatory item 23.
June 17, 1985	Representatives from NRC, Northeast Utilities, and Lawrence Livermore National Laboratory meet in Waterford, Conn., to discuss Technical Specifications for which additional information may be required to ensure that the final plant Technical Specifications are documented in the FSAR as accepted in the SER and reflected in the as-built plant design.

June 21, 1985	Representatives from NRC and Northeast Utilities meet in Bethesda, Md., to discuss Seismic II/I design considerations for Millstone 3.
June 25, 1985	Letter from applicant responding to Generic Letter 85-02 on steam generator tube integrity.
June 27, 1985	Letter to applicant transmitting a typed version of the first draft of the Millstone 3 Technical Specifications.
June 28, 1985	Letter from applicant transmitting a change to Revision 0 of the Physical Security Plan.
July 1, 1985	Letter to applicant requesting additional information on Emergency Plan, Draft 2 to Revision 0.
July 1, 1985	Letter from applicant concerning ultransonic inspection demonstration.
July 1, 1985	Letter from applicant responding to NRC question 410.32, isolation transfer switches and post-fire shutdown capability.
July 2, 1985	Representatives from NRC and Northeast Utilities meet in Bethesda, Md., to discuss Millstone 3 Environmental Qualification Report.
July 5, 1985	Letter from applicant transmitting the Draft Standard Technical Specifications.
July 11, 1985	Representatives from NRC and Northeast Utilities meet in Bethesda, Md. to discuss deviations from Branch Technical Position CMEB 9.5-1 of Standard Review Plan, Section 9.5.1.
July 11, 1985	Letter from applicant responding to materials engineering questions.
July 12, 1985	Letter from applicant transmitting a response to requests for additional information on structural and geotechnical engineering.
July 12, 1985	Letter from applicant transmitting a response to SER confirmatory item 58.
July 15, 1985	Letter from applicant transmitting a revised response to question 492.7.
July 18, 1985	Letter from applicant transmitting responses to questions 440.72 through 440.77.
July 18, 1985	Letter from applicant concerning functional capability of emergency lighting (SER open item 15).

July 19, 1985	Letter to applicant transmitting second draft of Technical Specifications for Millstone Unit 3.
July 24, 1985	Letter to applicant transmitting Amendment No. 12 to CPPR-113 to incorporate the partial exemption from GDC 4 of 10 CFR 50 Appendix A granted per applicant request.
July 29, 1985	Letter from applicant commenting on draft NUREG-0844, "Steam Generator Tube Integrity."
July 29 and 30, 1985	Representatives from NRC and Northeast Utilities meet in Berlin, Conn., to perform Design Verification and Design Validation Audit of Millstone 3 SPDS.
August 2, 1985	Letter from applicant concerning small-break LOCA analysis, TMI Action Item II.K.3.30.
August 6, 1985	Letter to applicant concerning Revision 0 to the Physical Security Plan.
August 6, 1985	Letter from applicant concerning Seismic Qualification Review Team (SQRT) and Pump and Valve Operability Review Team (PVORT) audits.
August 6, 1985	Representatives from NRC and Northeast Utilities meet in Bethesda, Md., to discuss applicant's response to questions resulting from the staff's review of the Millstone Emergency Plan, Draft 2 to Revision 0, dated January 1985.
August 6, 1985	Letter from applicant transmitting a revised response to question 260.58.

APPENDIX B

BILBIOGRAPHY

Rahe, E. P., Jr. (W), letter to H. R. Denton (NRC), "Steam Superheating During SLB for Dry Containments," February 19, 1985.

U.S. Nuclear Regulatory Commission, Generic letter from D. G. Eisenhut to all licensees of operating plants, applicants for operating licenses, and holders of construction permits, "Control of Heavy Loads," December 22, 1980.

---, Generic Letter 82-12 from R. Purple (for D. Eisenhut) to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits, "Nuclear Power Plant Staff Working Hours," June 15, 1982.

---, Generic Letter 82-33, see NUREG-0737, Supplement 1.

---, Generic Letter 83-35, from D. G. Eisenhut to all licensees of operating reactors, applicants for operating licenses and holders of construction permits, "Clarification of TMI Action Plan Item II.K.3.31," November 2, 1983.

---, NUREG-0484, Rev. 1, "Methodology for Combining Dynamic Responses," May 1980.

---, NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," January 1980.

---, NUREG-0612, "Control of Heavy Loads at Nuclear Plants: Resolution of Generic Technical Activity A-36," July 1980.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980; Supplement 1, December 17, 1982.

---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981 (contains Branch Technical Positions).

---, Office of Inspection and Enforcement, Bulletin 80-06, "Engineered Safety Features (ESF) Reset Controls," March 13, 1980.

---, Office of Inspection and Enforcement, Information Notice 85-09, "Isolation Transfer Switches and Post-Fire Shutdown Capability," January 31, 1985.

Westinghouse Corporation, WCAP-8859-(Nonproprietary), "TRANFLO Steam Generator Code Description," September 1976.

---, WCAP-8860 (Nonproprietary), "Mass and Energy Releases Following a Steam Line Rutpure," September 1976.

---, WCAP-10054, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," December 19, 1982. Submitted by letter, NS-EPR-2681, November 12, 1982 from E. P. Rahe (W) to C. O. Thomas (NRC).

---, WCAP-10079, "NOTRUMP, A Nodal Transfer Small Break and General Network Code," November 14, 1982.

APPENDIX D

ABBREVIATIONS

ASME	American Society of Mechanical Engineers
BNL	Brookhaven National Laboratory
BOP	balance of plant
BTP	Branch Technical Position
CDA	containment depressurization actuation
CFR	Code of Federal Regulations
CHR	containment heat removal
CS	containment spray
CVC	chemical and volume control
DBA	design-basis accident
ECC	emergency core cooling
ESF	engineered safety features
FSAR	Final Safety Analysis Report
GDC	General Design Criterion(a)
GE	General Electric
HPSI	high-pressure safety injection
HVAC	heating, ventilation, and air conditioning
ID	identification
IEEE	Institute of Electrical and Electronics Engineers
INEL	Idaho National Engineering Laboratory
LOCA	loss-of-coolant accident
MCC	motor control center
MSLB	main steamline break
NIS	nuclear instrumentation system
NPS	nominal pipe size
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OBE	operating-basis earthquake
ORNL	Oak Ridge National Laboratory
PCT	peak cladding temperature
PORV	power-operated relief valve

PPS	process protection system
PRA	probabilistic risk assessment
PVORT	Pump and Valve Operability Review Team
PWR	pressurized-water reactor
QC	quality control
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RPS	reactor protection system
SBLOCA	small-break loss-of-coolant accident
SER	Safety Evaluation Report
SQRT	Seismic Qualification Review Team
SRP	Standard Review Plan
SSE	safe shutdown earthquake
SSER	Supplement to Safety Evaluation Report
STA	shift technical advisor
S&W	Stone & Webster
TER	Technical Evaluation Report
TMI-2	Three Mile Island, Unit 2
UL	Underwriters Laboratory
WOG	Westinghouse Owners Group

APPENDIX F

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This supplement is a product of the NRC staff and its consultants. The NRC staff members listed below were principal contributors to this report.

NRC STAFF MEMBERS

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APPENDIX H

ERRATA TO MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3,
SAFETY EVALUATION REPORT

Page 9-8, Section 9.1.4, paragraph 5, line 3

DELETE: "including the spent fuel racks and spent fuel cask"

Page 9-8, Section 9.1.4, paragraph 7, line 2

CHANGE: "transfer"

TO: "handling"

Page 13-21, Section 13.5.1.8, line 12

DELETE: "verification of the correct performance of operating activities."

Page 13-21, Section 13.5.1.8, line 14

CHANGE: "two open items"

TO: "open item"

APPENDIX J

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS
MILLSTONE UNIT 3

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS
MILLSTONE UNIT 3
Docket No. [50-0423]
(Phase I)

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ABSTRACT

The Nuclear Regulatory Commission (NRC) has requested that all nuclear plants, either operating or under construction, submit a response of compliancy with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." EG&G Idaho, Inc., has contracted with the NRC to evaluate the responses of those plants presently under construction. This report contains EG&G's evaluation and recommendations for Millstone Unit 3.

EXECUTIVE SUMMARY

Millstone 3 is consistent with the intent of Article 5.1.1 of NUREG 0612.

CONTENTS

TABLES

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS

MILLSTONE UNIT 3

(Phase I)

1. INTRODUCTION

1.1 Purpose of Review

This technical evaluation report documents the EG&G Idaho, Inc., review of general load-handling policy and procedures at Millstone Unit 3. This evaluation was performed with the objective of assessing conformance to the general load-handling guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" [1], Section 5.1.1.

1.2 Generic Background

Generic Technical Activity Task A-36 was established by the U.S. Nuclear Regulatory Commission (NRC) staff to systematically examine staff applicant criteria and the adequacy of measures in effect at operating nuclear power plants to assure the safe handling of heavy loads and to recommend necessary changes to these measures. This activity was initiated by a letter issued by the NRC staff on May 17, 1978 [2], to all power reactor applicants, requesting information concerning the control of heavy loads near spent fuel.

The results of Task A-36 were reported in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The staff's conclusion from this evaluation was that existing measures to control the handling of heavy loads at operating plants, although providing protection from certain potential problems, do not adequately cover the major causes of load-handling accidents and should be upgraded.

In order to upgrade measures for the control of heavy loads, the staff developed a series of guidelines designed to achieve a two-phase objective using an accepted approach or protection philosophy. The first portion of the objective, achieved through a set of general guidelines identified in NUREG-0612, Article 5.1.1, is to ensure that

all load-handling systems at nuclear power plants are designed and operated such that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed. The second portion of the staff's objective, achieved through guidelines identified in NUREG-0612, Articles 5.1.2 through 5.1.5, is to ensure that, for load-handling systems in areas where their failure might result in significant consequences, either (a) features are provided, in addition to those required for all load-handling systems, to ensure that the potential for a load drop is extremely small (e.g., a single-failure-proof crane) or (b) conservative evaluations of load-handling accidents indicate that the potential consequences of any load drop are acceptably small. Acceptability of accident consequences is quantified in NUREG-0612 into four accident analysis evaluation criteria.

The approach used to develop the staff guidelines for minimizing the potential for a load drop was based on defense in depth and is summarized as follows:

- Provide sufficient operator training, handling system design, load-handling instructions, and equipment inspection to assure reliable operation of the handling system

- Define safe load travel paths through procedures and operator training so that, to the extent practical, heavy loads are not carried over or near irradiated fuel or safe shutdown equipment

- Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

Staff guidelines resulting from the foregoing are tabulated in Section 5 of NUREG-0612.

1.3 Plant-Specific Background

On December 22, 1980, the NRC issued a letter [3] to Northeast Nuclear Energy Company (NNECO), the applicant for Millstone Unit 3 requesting that the applicant review provisions for handling and control of heavy loads at Millstone Unit 3, evaluate these provisions with respect to the guidelines of NUREG-0612, and provide certain additional information to be used for an independent determination of conformance to these guidelines. On May 15, 1984, NNECO provided the initial response [4] to this request. EG&G received comments from NRC in September 1984 [9]. Additional information was received in March 1985 [10], and April 1985 [11].

2. EVALUATION AND RECOMMENDATIONS

2.1 Overview

The following sections summarize NNECO's review of heavy load handling at Millstone Unit 3 accompanied by EG&G's evaluation, conclusions, and recommendations to the applicant for bringing the facilities more completely into compliance with the intent of NUREG-0612. The applicant has indicated the weight of a heavy load for this facility (as defined in NUREG-0612, Article 1.2) as 1800 lb.

2.2 Heavy Load Overhead Handling Systems

This section reviews the applicant's list of overhead handling systems which are subject to the criteria of NUREG-0612 and a review of the justification for excluding overhead handling systems from the above mentioned list.

2.2.1 Scope

"Report the results of your review of plant arrangements to identify all overhead handling systems from which a load drop may result in damage to any system required for plant shutdown or decay heat removal (taking no credit for any interlocks, technical specifications, operating procedures, or detailed structural analysis) and justify the exclusion of any overhead handling system from your list by verifying that there is sufficient physical separation from any load-impact point and any safety-related component to permit a determination by inspection that no heavy load drop can result in damage to any system or component required for plant shutdown or decay heat removal."

A. Summary of Applicant's Statements

The applicant's review of overhead handling systems identified the cranes and hoists shown in Table 2.1 as those which handle heavy loads in the vicinity of irradiated fuel or safe shutdown equipment.

TABLE 2.1 APPLICABLE OVERHEAD LOAD HANDLING SYSTEMS

Equipment No.	Identification	Location
3MHR-CRN1	Polar Crane	Containment
3MHF-CRN1	Spent Fuel Shipping Cask Trolley	Fuel Building
3MHF-CRN2	New Fuel Handling Crane	Fuel Building
3MHF-CRN3	New Fuel Receiving Crane	Fuel Building
3MHF-CRN4	Fuel Building Decontamination Crane	Fuel Building
3MHP-CRN1	Auxiliary Building Filter Handling Crane/Monorail	Auxiliary Building
3MHP-CRN2A,B,C	Auxiliary Building Charging Pump Trolley	Auxiliary Building
(-)	Reactor Plant Component Cooling Water Heat Exchanger Monorail	Auxiliary Building
3MHS-CRN,B,1	Spent Fuel Bridge and Hoist	Fuel Building

The applicant has also identified numerous other cranes that have been excluded from satisfying the criteria of the general guidelines of NUREG-0612.

B. EG&G Evaluation

From a study of the March 14, 1985 submittal [10], it appears that the applicable overhead handling systems have been included in Table 2-1. Crane identified as "Excluded Overhead Load Handling Systems" in the submittal [10] have had their exclusion adequately justified.

C. EG&G Conclusions and Recommendations

Based on the information provided, EG&G concludes that the applicant has included all applicable hoists and cranes in their list of handling systems which must comply with the requirements of the general guidelines of NUREG-0612.

2.3 General Guidelines

This section addresses the extent to which the applicable handling systems comply with the general guidelines of NUREG-0612, Article 5.1.1. EG&G's conclusions and recommendations are provided in summaries for each guideline.

The NRC has established seven general guidelines which must be met in order to provide the defense-in-depth approach for the handling of heavy loads. These guidelines consist of the following criteria from Section 5.1.1 of NUREG-0612:

- Guideline 1--Safe Load Paths

- Guideline 2--Load-Handling Procedures

- Guideline 3--Crane Operator Training

- Guideline 4--Special Lifting Devices

- Guideline 5--Lifting Devices (not specially designed)

- Guideline 6--Cranes (Inspection, Testing, and Maintenance)

- Guideline 7--Crane Design.

These seven guidelines should be satisfied for all overhead handling systems and programs in order to handle heavy loads in the vicinity of the reactor vessel, near spent fuel in the spent-fuel pool, or in other areas where a load drop may damage safe shutdown systems. The succeeding paragraphs address the guidelines individually.

2.3.1 Safe Load Paths [Guideline 1, NUREG-0612, Article 5.1.1(1)]

> "Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent-fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee."

A. Summary of Applicant's Statements

NNECO provided figures which "identify, as much as practical, the location of safe load paths, spent fuel, and safe shutdown equipment in the areas of concern.

The safe load paths shown on these figures will not be permanently marked on the plant flooring. This is due to the possibility that when loads are being moved, the flooring may be covered with disposable polyvinyl sheeting. In lieu of the permanent markings a supervising load director will be available to verify the load path and help direct the crane operator."

NNECO also stated that deviation from procedures will require an approved procedural change.

B. EG&G Evaluation

NNECO's response to this guideline is brief but seems to meet the intent of the guideline. Load paths are defined and a load director will verify and direct the load handling operation to ensure that load paths are followed.

C. EG&G Conclusions and Recommendations

Millstone Unit 3 is consistent with the intent of Guideline 1.

2.3.2 Load-Handling Procedures [Guideline 2, NUREG-0612, Article 5.1.1(2)]

"Procedures should be developed to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of NUREG-0612. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe path; and other special precautions."

A. Summary of Applicant's Statements

Administrative procedures will include the general guidelines and evaluation requirements of NUREG-0612. The safe load paths shown in this report will be used as the load-handling paths. Any deviation from these operational procedures will require an approved procedural change.[4]

Procedures for the lifting of all heavy loads will incorporate the guidance of NUREG-0612 [10].

B. EG&G Evaluation

NNECO states, "Procedures for the lifting of all heavy loads will incorporate the guidance of NUREG-0612" [10]. This commitment covers the requirements of NUREG-0612.

C. EG&G Conclusions and Recommendations

Millstone Unit 3 is consistent with the intent of Guideline 2.

2.3.3 Crane Operator Training [Guideline 3, NUREG-0612, Article 5.1.1(3)]

"Crane operators should be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, 'Overhead and Gantry Cranes' [5]."

A. Summary of Applicant's Statements

An operator training program is currently being developed and, along with operator qualification and conduct, will be consistent with the intent of ANSI B30.2-1976.

B. EG&G Evaluation

NNECO has committed to compliance with Guideline 3.

C. EG&G Conclusions and Recommendations

Millstone Unit 3 is consistent with the intent of Gudieline 3.

2.3.4 Special Lifting Devices [Guideline 4, NUREG-0612, Article 5.1.1(4)]

"Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, 'Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials' [6]. This standard should apply to all special lifting devices which carry heavy loads in areas as defined above. For operating plants, certain inspections and load tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used. This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6 which bases the stress design factor on only the weight (static load) or the load and of the intervening components of the special handling device."

A. Summary of Applicant's Statements

The two special lifting devices, the reactor vessel head lifting device and the upper internals lifting rig assembly, were both designed prior to the publishing of ANSI N14.6-1978.

The ANSI N14.6 document has been reviewed in detail in WCAP-10669 and compared to the requirements used to design and manufacture the reactor vessel head lift rig, the reactor vessel internals lift rig, load cell, and the load cell linkage.

The following conclusions are apparent as a result of this comparison:

1. The ANSI N14.6 requirements for design, fabrication and quality assurance are generally in agreement with those used for these special lift devices.

2. The ANSI N14.6 criteria for stress limits associated with certain stress design factors for most tensile and shear stresses are adequately satisfied.

3. These devices are not in strict compliance only with the ANSI N14.6 requirements for acceptance testing, maintenance and verification of continuing compliance. Recommendations are included to identify actions that should enable these devices to be considered in compliance with the intent of ANSI N14.6.

4. The application of the ANSI N14.6 criteria for stress design factor of 3 and 5 are only for shear and tensile loading conditions. Other loading conditions are to be analyzed to other appropriate criteria.

   A response to the requirement of ANSI N14.6, Section 5.2.1, requiring an initial acceptance load test prior to use equal to 150 percent of the maximum load is that the 125 percent of maximum load test that was performed be accepted in lieu of the 150 percent load test.

A response to ANSI N14.6 Section 5.3 which requires, annually, either a 150 percent maximum load test or dimensional, visual and non-destructive testing of major load carrying welds and critical areas follows. (Since the 150 percent load test is very impractical, the approach identified in the following recommendation is to perform a minimum of non-destructive testing.)

a. Reactor Vessel Head Lift Rig:

Prior to use and after reassembly of the spreader assembly, lifting lug, and upper lifting legs to the upper portion of the lift rig, visually check all welds. Raise the vessel head slightly above its support (maximum of 6 inches) and hold for 10 minutes. Visually inspect the sling block lugs to the lifting block welds, and spreader lug to spreader arm weld. If no problems are apparent, continue to lift, monitoring the load cell readout at all times.

b. Reactor Vessel Internals Lift Rig

Prior to use, visually inspect the rig components and welds while on the storage stand for signs of cracks or deformation. Check all bolted joints to ensure that they are tight and secure. After connection to the upper or lower internals, raise the assembly slightly off its support (a maximum of 6 inches) and hold for 10 minutes. Visually inspect the sling block lugs to the lifting block welds. If no problems are apparent, continue to lift, monitoring the load cell readout at all times.

Dimensional checking is not included since these structures are large (about 16 feet diameter by 50 ft high) and the results of dimensional checking would always be questionable. Other checks on critical load path parts such as pins, are also not included since an examination of these items would require disassembly of the special lift devices.

A periodic non-destructive surface examination of critical welds and/or parts will be performed once every ten years as part of an inservice inspection outage.

B. EG&G Evaluation

EG&G concurs that the special lifting devices at Millstone Unit 3 meet the requirements of ANSI N14.6, except as noted by NNECO.

EG&G accepts the 125 percent acceptance load test and the proposed alternatives to maintenance and verification of continuing compliance. The latter is in accordance with the "Synopsis of Issues, Associated with NUREG-0612."

C. EG&G Conclusions and Recommendations

Millstone Unit 3 is consistent with the intent of Guideline 4.

2.3.5 Lifting Devices (Not Specially Designed) [Guideline 5, NUREG-0612, Article 5.1.1(5)]

"Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, 'Slings' [7]. However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load. The rating identified on the sling should be in terms of the 'static load' which produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used."

A. Summary of Applicant's Statements

Millstone 3 has evaluated the potential routine dynamic loading for lifting devices not specifically designed and found them to be a relatively small fraction of the static load. The evaluation has been made on the basis of the crane speeds which are all below 30 ft/minute except for one crane which has a speed of 32 ft/min. Therefore, Millstone 3 is taking exception to the requirement to select slings in accordance with the maximum working load tables of ANSI B30.9 considering the sum of static and dynamic loads.[10]

Millstone 3 will select and use slings in accordance with ANSI B30.9.[11]

B. EG&G Evaluation

Millstone 3's exception to using dynamics loads is acceptable per "Synopsis of Issues Associated with NUREG 0612.

C. EG&G Conclusions and Recommendations

Millstone 3 is consistent with the intent of Guideline 5.

2.3.6 Cranes (Inspection, Testing, and Maintenance) [Guideline 6, NUREG-0612, Article 5.1.1(6)]

"The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' with the exception that tests and inspections should be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency (e.g., the polar crane inside a PWR containment may only be used every 12 to 18 months during refueling operations, and is generally not accessible during power operation. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, test, and maintenance should be performed prior to their use)."

A. Summary of Applicant's Statements

Crane inspection, testing, and maintenance procedures will comply with the intent of the guidelines of ANSI B30.2-1976, Chapter 2-2. Should any deviations from this standard be required, they will be equivalent to the requirements of ANSI B30.2-1976.

B. EG&G Evaluation

NNECO's response is brief but adequate.

C. EG&G Conclusions and Recommendations

Millstone Unit 3 is consistent with the intent of Guideline 6.

2.3.7 Crane Design [Guideline 7, NUREG-0612, Article 5.1.1(7)]

"The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' and of CMAA-70, 'Specifications for Electric Overhead Traveling Cranes' [8]. An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied."

A. Summary of Applicant's Statements

The containment polar crane (3MHR-CRN1), the spent fuel shipping cask trolley (3MHF-CRN1), the new fuel receiving crane (3MHF-CRN3), and the decontamination area crane (3MHF-CRN4) have been designed to meet the criteria and guidelines of CMAA-70, Specification for Electrical Overhead Traveling Cranes, and ANSI B30.2-1967. Although these cranes have been designed to the 1967 ANSI standard, they have been reviewed for compliance with the 1976 standard and there are no significant differences between the two ANSI standards which would affect the operation of the cranes. The new fuel handling crane (3MHF-CRN2) has been designed to comply with the guidelines of CMAA-70 and ANSI B30.2-1976.

The balance of the load-handling devices are not cranes, so CMAA-70 and ANSI B30.2-1976 were not used in their design. Instead, ANSI B30.11, Standard Monorail System and Underhung Cranes, and ANSI B30.16, Standard Overhead Hoists, were used.

B. EG&G Evaluation

The above statements are brief but indicate that the applicable cranes and hoists were designed in accordance with standards equivalent to those specified in the guideline.

C. <u>EG&G Conclusions and Recommendations</u>

Millstone Unit 3 is consistent with the intent of Guideline 7.

3. CONCLUDING SUMMARY

3.1 Applicable Load-Handling Systems

The list of cranes and hoists supplied by the applicant as being subject to the provisions of NUREG-0612 is complete (see Section 2.2.1).

3.2 Guideline Recommendations

Compliance with the seven NRC guidelines for heavy load handling (Section 2.3) is satisfied at Millstone Unit 3. This conclusion is represented in tabular form as Table 3.1.

Guideline	Recommendation
1. Section 2.3.1 Guideline 1	Consistent with the intent of Guideline 1.
2. Section 2.3.2 Guideline 2	Consistent with the intent of Guideline 2.
3. Section 2.3.3	Consistent with the intent of Guideline 3.
4. Section 2.3.4 Guideline 4	Consistent with the intent of Guideline 4.
5. Section 2.3.5 Guideline 5	NNECO is consistent with the intent of Guideline 5.

6.	Section 2.3.6 Guideline 6	Consistent with the intent of Guideline 6.
7.	Section 2.3.7 Guideline 7	Consistent with the intent of Guideline 7.

TABLE 3.1. MILLSTONE UNIT 3, COMPLIANCE MATRIX

Identification	Load Rating (tons)	Safe Loads Paths	Load Handling Procedures	Crane Operator Training	Special Lifting Devices	Lifting Devices not Special Design	Crane Inspection Test Maintenance	Crane Design
3 MHF-CRN 1 Polar Crane	Bridge-434 Trolley 1-217 Trolley 2-217 Aux Hook-30	C	C	C	C		C	C
3 MHR-CRN 1 Spent Fuel Shipping Cask Trolley	125	C	C	C	--		C	C
3 MHF-CRN 2 New Fuel Handling Crane	10	C	C	C	--		C	C
3 MHF-CRN 3 New Fuel Receiving Crane	10	C	C	C	--		C	C
3 MHF-CRN 4 Fuel Building Decontamination Crane	5	C	C	C	--		C	C
3 MHP-CRN 1 Auxiliary Building Filter Handling Crane/Monorail	10	C	C	C	--		C	C
3 MHP-CRN 2A, B, C Auxiliary Building Charging Pump Trolley	5	C	C	C	--		C	C
(-) Reactor Plant Component Cooling Water Heat Exchanger Monorail	Information not supplied	C	C	C	--		C	C
3 MHS-CRN B1 Spent Fuel Bridge and Hoist	3	C	C	C	--		C	C

C = Applicant action complies with NUREG 0612 Guideline.

NC = Applicant action does not comply with NUREG 0612 Guideline.

R = Applicant has proposed revision/modifications designed to comply with NUREG 0612 Guidelines.

I = Insufficient information provided by the applicant

4. REFERENCES

1. NUREG-0612, Control cf Heavy Loads at Nuclear Power Plants, NRC.

2. V. Stello, Jr. (NRC), Letter to all applicants. Subject: Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, May 17, 1978.

3. USNRC, Letter to NNECO. Subject: NRC Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, December 22, 1980.

4. W. G. Council (NNECO), Letter to B. J. Yougblood (NRC). Subject: Response to Auxiliary Systems Branch Draft SER Open Items, May 15, 1984.

5. ANSI B30.2-1976, "Overhead and Gantry Cranes".

6. ANSI N14.6-1978, "Standard for Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or more for Nuclear Materials".

7. ANSI B30.9-1971, "Slings".

8. CMAA-70, "Specifications for Electric Overhead Traveling Cranes".

9. Amarjit, Singh (NRC), letter to T. H. Stickley (EG&G), Comments on Millstone 3, September 24, 1984.

10. W. G. Counsil (NNECO), letter to B. J. Youngblood (NRC) Subject: Millstone Nuclear Power Station, Unit No. 3, March 14, 1985.

11. W. G. Counsil (NNECO), letter to B. J. Youngblood (NRC), Subject: Millstone Nuclear Power Station, Unit No. 3, April 22, 1985.

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14. ABSTRACT (200 words or less)

The Safety Evaluation Report issued in August 1984 provided the results of the NRC staff review of Northeast Nuclear Energy Company's application for a license to operate the Millstone Nuclear Power Station, Unit No. 3. Supplement No. 1 to that report, issued in March 1985 updated the information contained in the Safety Evaluation Report and addressed the ACRS Report issued on September 10, 1984.

This Report, Supplement No. 2 updates the information contained in the Safety Evaluation Report and Supplement No. 1 and addresses prior unresolved items.

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