NUREG-1031
Supplement No. 5

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

January 1986

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ABSTRACT

This report supplements the Safety Evaluation Report (NUREG-1031) issued in July 1984, Supplement 1 issued in March 1985, Supplement 2 issued in September 1985, Supplement 3 issued in November 1985, and Supplement 4 issued in November 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 (licensee 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.

The supplement provides more recent information regarding resolution of license conditions identified in the SER. Because of the favorable resolution of the items discussed in this report, the staff concludes that Millstone Nuclear Power Station, Unit No. 3, can be operated by the licensee at power levels greater than 5% without endangering the health and safety of the public.

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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 (NNECO, the licensee), 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), in September 1985 by Supplement 2 (SSER 2), in November 1985 by Supplement 3 (SSER 3) and in November 1985 by Supplement 4 (SSER 4); these documented the resolution of several outstanding and confirmatory items and license conditions in further support of the licensing activities. The present report, Supplement 5 to the SER (SSER 5), 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 supplements, and supports the license for operation at power levels greater than 5%.

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 licensee and the NRC staff is listed chronologically in this appendix. Appendices B and D list references and abbreviations, respectively. Appendix F lists principal staff members who contributed to this supplement. Appendix N adds to the SER the staff's evaluation of the licensee's request for relief from certain ASME Code requirements.

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 PWR Licensing-A, 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 licensee. The status of these items is listed Table 1.1 (an updated version of SER Table 1.3). As Table 1.1 indicates, none of these items are considered open items.

1.6 Confirmatory Items

The staff identified confirmatory items in the 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). As Table 1.2 indicates, none of these items are considered confirmatory items.

1.7 License Condition Items

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

The license conditions are listed in Table 1.3 (an updated version of SER Table 1.5). If the staff has removed the license condition, the notation "closed" so indicates. If the license condition has been revised, that is noted, too, in Table 1.3. The section of this supplement in which the change is reported is listed in Table 1.3.

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	Closed (SSER 4)	
(3) Protection against postulated pipe breaks outside containment	Changed to License Condition 12 (SSER 4)	
(4) Loading combinations	Closed (SSER 1)	
(5) Design and construction of component supports	Closed (SSER 1)	
(6) Inservice testing of pumps and valves	Closed (SSER 4)	
(7) Equipment qualification	Changed to License Condition 13 (SSER 4)	
(8) Flow measurement capability	Closed (SSER 4)	
(9) Loose parts detection program	Closed (SSER 3)	
(10) Subcompartment analysis	Closed (SSER 4)	
(11) Mass and energy release analysis	Changed to confirmatory item (71) (SSER 2)	
(12) Volumetric inspection of Class 2 components	Closed (SSER 2)	
(13) Power-operated relief valve and block valve, compliance with NUREG-0737 (TMI Action Plan)	Closed (SSER 3)	
(14) Fire protection	Changed to License Condition 14 (SSER 4)	
(15) Functional capability of ac and dc emergency lighting	Closed (SSER 3)	
(16) Shift technical advisor training program and operating experience for startup	Changed to License Condition 15 (SSER 4)	
(17) Emergency Plan	Closed (SSER 4)	

Table 1.1 (Continued)

Item	Status	Section*
(18) Limitation on overtime	Closed (SSER 2)	
(19) Q list	Closed (SSER 3)	
(20) Detailed Control Room Design Review	Closed (SSER 4)	

*Section of this supplement where item is discussed.

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

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

Table 1.2 (Continued)

Item	Status	Section*
(20) Preservice inspection program review and relief requests	Closed (SSER 4)	
(21) Preservice and inservice inspection of steam generators	Closed (SSER 3)	
(22) Containment liner weld channel venting	Closed (SSER 2)	
(23) Maximum external differential pressure on containment	Closed (SSER 4)	
(24) Minimum containment pressure for emergency core cooling system	Closed (SSER 2)	
(25) Procedures for actuating hydrogen recombiner	Closed (SSER 3)	
(26) Secondary enclosure building	Closed (SSER 4)	
(27) Sump flow approach velocity	Closed (SSER 4)	
(28) Compliance with GDC 51	Closed (SSER 3)	
(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)	
(31) Reactor coolant pump underspeed trip	Closed (SSER 4)	
(32) Conformance with BTP ICSB-26	Closed (SSER 1)	
(33) Test of engineering safeguard P-4 interlock	Closed (SSER 1)	
(34) Steam generator level control and protection	Closed (SSER 2)	
(35) Confirmatory test related to IE Bulletin 80-06	Closed (SSER 2	
(36) Control building isolation reset	Closed (SSER 2)	
(37) Power lockout feature for motor-operated valves	Closed (SSER 1)	

Table 1.2 (Continued)

Item	Status	Section*
(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)	
(40) Sequencer deficiency report	Closed (SSER 2)	
(41) Balance-of-plant instrumentation and control system testing capability	Closed (SSER 2)	
(42) Instrument accuracy related to Positions [Attachments] 4, 5, and 6, TMI Action Plan Item II.F.1	Closed (SSER 2)	
(43) Description and analysis demonstrating compliance with GDC 5	Closed (SSER 1)	
(44) Physical separation of offsite circuits within a common right of way	Closed (SSER 3)	
(45) Physical separation of offsite circuits between switchyard and Class 1E system	Closed (SSER 3)	
(46) Generation rejection scheme	Closed (SSER 3)	
(47) Description and analysis demonstrating compliance with GDC 17	Closed (SSER 1)	
(48) Description and analysis demonstrating compliance with GDC 18	Closed (SSER 1)	
(49) Positive statement of compliance with BTP PSB-1	Closed (SSER 1)	
(50) Compliance with Position 1 of BTP PSB-1	Closed (SSER 3)	
(51) Adequacy of station electric distribution system voltage	Closed (SSER 4)	
(52) Routing of power cables in the cable spreading area	Deleted (SSER 1, Appendix H)	

Table 1.2 (Continued)

Item	Status	Section*
(53) Battery charger and transformer used as isolation devices	Closed (SSER 3)	
(54) Design criteria of associated circuits from isolation device to load	Deleted (SSER 1, Appendix H)	
(55) Core damage procedure (TMI Action Plan Item II.B.3, Criterion 2)	Closed (SSER 1)	
(56) Control of concrete dust	Closed (SSER 3)	
(57) Qualification of engine-mounted control panels	Closed (SSER 3)	
(58) 7-day fuel oil storage of each diesel generator	Closed (SSER 4)	
(59) Airborne radioactivity monitoring	Closed (SSER 3)	
(60) Process control program for solidification of wet wastes	Closed (SSER 3)	
(61) TMI Action Plan Item II.F.1.1	Closed (SSER 3)	
(62) TMI Action Plan Item I.C.1 procedures generation package nuclear steam supply system	Changed to License Condition 10 (SSER 4)	
(63) Physical Security Plan	Closed (SSER 3)	
(64) Initial test program	Closed (SSER 3)	
(65) Reactor coolant pump trip during loss-of-coolant accident	Closed (SSER 4)	
(66) TMI Action Plan Item III.D.1.1	Closed (SSER 4)	
(67) Analysis of dropped control rod	Closed (SSER 3)	
(68) Steam generator tube rupture	Deleted (SSER 4)	
(69) No failure in emergency core cooling system (ECCS) is not most limiting case in evaluating EECS	Deleted (SSER 2)	
(70) QA program commitments	Closed (SSER 3)	

Table 1.2 (Continued)

Item	Status	Section*
(71) Mass and energy release analysis	Changed to License Condition 9 (SSER 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	Revised (SSER 4)	
(2) Compliance with NUREG-0612 ("Heavy Load Handling")	Closed (SSER 2)	
(3) Installation of postaccident sampling system	Closed**	
(4) Sediment control during fuel oil storage tank refill	Closed (SSER 3)	
(5) Moisture in air start system	Revised (SSER 3)	
(6) Preheating of rocker arm lubrication oil system	Closed (SSER 3)	
(7) Blockage of access hatch in diesel generator exhaust system	Closed (SSER 3)	
(8) Plant-specific analyses utilizing NOTRUMP (TMI Item II.K.3.31)	Added (SSER 2)	
(9) Mass and energy release analysis	Closed (SSER 5)	6.2.1.4
(10) TMI Action Plan Item I.C.1 - procedures generation package nuclear steam supply system	Closed (SSER 5)	13.5.2
(11) N-1 loop operation	Unchanged (SSER 4)	
(12) Protection against postulated pipe breaks outside containment	Closed (SSER 5)	3.11
(13) Equipment qualification	Closed (SSER 5)	3.11
(14) Fire protection	Closed (SSER 5)	9.5.1.5
(15) Shift technical advisor training program and operating experience for startup	Revised (SSER 5)	13.1.2
(16) Seismic interaction program	Closed (SSER 5)	3.9.2
(17) Hazards program	Closed (SSER 5)	17

*Section of this supplement where item is discussed.

**Letter from J. F. Opeka (NNECO) to V. Noonan (NRC), dated January 24, 1986.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

In the "Background" to Branch Technical Position (BTP) MEB 3-1 as presented in Standard Review Plan (SRP) Section 3.6.2 (NUREG-0800), the staff position on pipe break postulation acknowledged that pipe rupture is a rare event that may only occur under unanticipated conditions such as those that might be caused by possible design, construction, or operation errors, unanticipated loads, or unanticipated corrosive environments. The BTP MEB 3-1 pipe break criteria were intended to utilize a technically practical approach to ensure that an adequate level of protection had been provided to satisfy the requirements of General Design Criterion (GDC) 4 (10 CFR 50, Appendix A). Specific guidelines were developed in BTP MEB 3-1 to define explicitly how the requirements of GDC 4 were to be implemented. The SRP guidelines in BTP MEB 3-1 were not intended to be absolute requirements, but represent viable approaches considered to be acceptable by the staff.

The SRP provides a well-defined basis for performing safety reviews of light-water reactors. The uniform implementation of design guidelines in BTP MEB 3-1 ensures that a consistent level of safety will be maintained during the licensing process. Alternative criteria and deviations from the SRP are acceptable provided an equivalent level of safety can be demonstrated. Acceptable reasons for deviations from SRP guidelines include changes in emphasis of specific guidelines as a result of new developments from operating experience or plant-unique design features not considered when the SRP guidelines were developed.

The SRP presents the most definitive basis available for specifying NRC's design criteria and design guidelines for an acceptable level of safety for light-water-reactor facility reviews. The SRP guidelines resulted from many years of experience gained by the staff in establishing and using regulatory requirements in the safety evaluation of nuclear facilities. The SRP is part of a continuing regulatory standards development activity that not only documents current methods of review, but also provides a basis for an orderly modification of the review process when the need arises to clarify the content, correct any errors, or modify the guidelines as a result of technical advancements or an accumulation of operating experience. Proposals to modify the guidelines in the SRP are considered for their impact on matters of major safety significance.

The staff has recently received a request from the licensee to consider an alternate approach to the guidelines in SRP Section 3.6.2, BTP MEB 3-1, regarding the postulation of intermediate pipe breaks (letter dated January 14, 1986). For all high-energy piping systems identified in the January 14 letter, the licensee proposes to eliminate from design considerations those breaks generally referred to as "arbitrary intermediate breaks" (AIBs) which are defined as those break locations which, based on piping stress analysis results, are below the stress and fatigue limits specified in BTP MEB 3-1, but are selected to provide a minimum of two postulated breaks between the terminal ends of a piping system. The licensee has stated that occupational radiation exposure during inspection, maintenance, and repair will be reduced over the life of the plant. The licensee is requesting approval of alternative pipe break criteria to eliminate the requirement to evaluate the jet impingement effects from six AIB locations and to eliminate the need to design and protect from the effects of AIBs during future plant modifications. However, the licensee has stated that the elimination of AIBs will not impact the environmental qualification of safety-related equipment. The break postulation for environmental effects is performed independently of break postulation for pipe whip and jet impingement.

In the early 1970s when the pipe break criteria in BTP MEB 3-1 were first drafted, the advantages of maintaining low stress and usage factor limits were clearly recognized, but it was also believed that equipment in close proximity to the piping throughout its run might not be adequately designed for the environmental consequences of a postulated pipe break if the break postulation proceeded on a purely mechanistic basis using only high stress and terminal end breaks. As the pipe break criteria were implemented by the industry, the impact of the pipe break criteria became apparent on plant reliability and costs as well as on plant safety. Although the overall criteria in BTP MEB 3-1 have resulted in a viable method which assures that adequate protection has been provided to satisfy the requirements of GDC 4, it has become apparent that the particular criterion requiring the postulation of arbitrary intermediate pipe breaks can be overly restrictive and may result in an excessive number of pipe rupture protection devices which do not provide a compensating level of safety.

At the time the BTP MEB 3-1 criteria were first drafted, high-energy leakage cracks were not being postulated. In Revision 1 to the SRP (NUREG-0800), the concept of using high-energy-leakage cracks to mechanistically achieve the environment desired for equipment qualification was introduced to cover areas which are below the high-stress/fatigue-limit break criteria and which would otherwise not be enveloped by a postulated break in a high-energy line. In the proposed elimination of AIBs, the staff believes that the essential design requirement of equipment qualification is not only being retained but is being improved, since all safety-related equipment is to be qualified environmentally and, furthermore, certain elements of construction which may lead to reduced reliability are being eliminated.

In addition, some requirements which have developed over the years as part of the licensing process have resulted in additional safety margins that overlap the safety margin provided in the pipe break criteria. For example, the criteria in BTP MEB 3-1 include margins to account for the possibility of flaws that might remain undetected in construction and to account for unanticipated piping steady-state vibratory loadings not readily determined in the design process. However, inservice inspection requirements for the life of the plant to detect flaws before they become critical, and staff positions on the vibration monitoring of safety-related and high-energy piping systems during preoperational testing, further reduce the potential for pipe failures occurring from these causes.

Because of the recent interest expressed by the industry to eliminate the AIB criteria and, particularly, in response to the submittals provided by several utilities including Northeast Nuclear Energy Company (NNECO), the staff has reviewed the BTP MEB 3-1 pipe break criteria to determine where such changes may be made.

3.6.2.1 Bases for the Elimination of Arbitrary Intermediate Pipe Breaks

In a letter dated January 14, 1986, the licensee presented its request for the elimination of AIBs and the technical bases for its proposal. There is a general consensus in the nuclear industry that current knowledge and experience support the conclusion that designing for the arbitrary intermediate pipe breaks is not justified. The reasons for this conclusion are discussed in the paragraphs that follow.

(1) Operating Experience Does Not Support Need for Criteria

The combined operating history of commercial nuclear plants (extensive operating experience in more than 80 operating U.S. plants and a number of similar plants overseas) has not shown the need to provide protection from the dynamic effects of AIBs.

(2) Piping Stresses Well Below ASME Code Allowables

Currently, AIBs are postulated to provide a minimum of two pipe breaks at the two highest stress locations between piping terminal ends. Consequently, AIBs are postulated at locations in the piping system where pipe stresses and/or cumulative usage factors are well below ASME Code allowables. Such postulation necessitates the installation and maintenance of complicated mitigating devices to afford protection from dynamic effects such as pipe whip and/or jet impingement. When these selected break locations have stress levels only slightly greater than the rest of the system, installation of mitigating devices lends little to enhance overall plant safety.

(3) Unanticipated Thermal Expansion Stress

Unanticipated stresses from restraint of thermal expansion can be introduced into the piping system if pipe rupture protection devices come into contact with the pipes. The potential for this happening is greater than that for mechanistic failure at an arbitrary break point. To prevent a consequent decrease in the overall reliability of the pipe system, an additional as-built verification step is involved in the design process for each installed pipe whip restraint. Elimination of AIBs would significantly reduce the effort involved in designing and installing pipe rupture protection devices.

(4) Improved Inservice Inspection

Pipe whip restraints are normally located adjacent to or surrounding the welds at changes in pipe direction. Access during plant operation of inservice inspection activities can be improved by eliminating congestion created by these pipe rupture protection devices and the supporting structural framing associated with arbitrary pipe breaks.

(5) Substantial Lost Savings and Reduction in Radiation Exposure

The estimated cost savings in design, material, and construction over the 40-year plant life is estimated to be well in excess of $1 million and the savings in personnel radiation exposure in excess of 100 person-rem for Millstone 3. These figures are probably conservative in light of the uncertainty involved in predicting the amount of future plant modifications.

(6) <u>Decrease in Heat Loss</u>

The elimination of pipe whip restraints associated with arbitrary breaks will preclude the requirement for cutback insulation or special insulating assemblies near the closefitting restraints. This will reduce the heat loss to the surrounding environment, especially inside containment, and will result in improved operational efficiency.

3.6.2.2 Staff Evaluation of the Bases for the Elimination of Arbitrary Breaks

The technical bases for the elimination of the AIB criteria as discussed in the preceding section of this report provided many arguments supporting the licensee's conclusion that the current SRP guidelines on this subject should be changed. However, it is not apparent that a unilateral position by the utility (concluding an unconditional deletion of the AIB criteria) can be justified without a clear understanding of the safety implications that may result for the various classes of high-energy piping systems involved. In this section, the staff discusses the bases behind the current AIB criteria from an ASME Code design standpoint and put into perspective the uncertainty factors on which the need to postulate AIBs should be evaluated.

The ASME Code design requirements for Class 1 piping systems differ from those for Class 2 and 3 piping systems; there are, however, other design considerations that are common to Class 1, 2, and 3 systems. These other design considerations [viz., (1) intergranular stress corrosion cracking, (2) water/steam hammer, and (3) thermal fatigue] can affect the safety of the systems in which AIBs are eliminated. Therefore, while evaluating the acceptability of the licensee's proposed deviation from SRP Section 3.6.2, the staff examined the significance of the above three additional design considerations for the specific Millstone 3 piping systems proposed by the licensee for elimination of AIBs.

(1) <u>ASME Code Class 1 Piping Systems</u>

In accordance with BTP MEB 3-1 [paragraph B.1.c.(1)] breaks in ASME Code Class 1 piping should be postulated at the following locations in each piping and branch run:

(a) at terminal ends;

(b) at intermediate locations where the maximum stress range as calculated by Eq. (10) and either Eq. (12) or (13) of ASME Code NB-3650 exceeds 2.4 S_m;

(c) at intermediate locations where the cumulative usage factor exceeds 0.1;

(d) if two intermediate locations cannot be determined by (b) and (c) above, two highest stress locations based on Eq. (10) should be selected.

The AIB criteria are stated in (d) above. It should be noted that the request for alternative criteria does not propose to deviate from the criteria in (a), (b), and (c) above. Pipe breaks will continue to be postulated at terminal ends irrespective of the piping stresses.

Pipe breaks are to be postulated at intermediate locations where the maximum stress range as calculated by Eq. (10) and either (12) or (13) exceeds 2.4 S_m. The stress evaluation in Eq. (10) represents a check of the primary plus secondary stress intensity range from ranges of pressure, moments, thermal gradients, and combinations thereof. Equation (12) is intended to prevent formation of plastic hinges in the piping system caused only by moments resulting from thermal expansion and thermal anchor movements. Equation (13) represents a limitation for primary plus secondary membrane plus bending stress intensity excluding thermal bending and thermal expansion stresses; this limitation is intended to ensure that the K_e-factor (strain concentration factor) is conservative. The K_e-factor was developed to compensate for absence elastic shakedown when primary plus secondary stresses exceed S_m.

With respect to piping stresses, the pipe break criteria were not intended to imply that breaks will occur when the piping stress exceeded 2.4 S_m (80% of the primary plus secondary stress limit). The staff believes, however, that if a pipe break were to occur (on one of those rare occasions), it is more likely to occur at a piping location where there is the least margin to the ultimate tensile strength.

Similarly, from a fatigue strength standpoint, the staff believes that a pipe break is more likely to occur where the piping is expected to experience large cyclic loadings. Although the staff concurs with the industry belief that a cumulative usage factor of 0.1 is a relatively low limit, the uncertainties involved in the design considerations with respect to the actual cyclic loadings experienced by the piping tend to be greater than the uncertainties involved in the design considerations used for the evaluation of primary and secondary stresses in piping systems. The staff finds that the conservative fatigue considerations in the current SRP guidelines provide an appropriate margin of safety against uncertainties for those locations where fatigue failures are likely to occur (e.g., at local welded attachments).

(2) <u>ASME Code Class 2 and 3 Piping Systems</u>

In accordance with BTP MEB 3-1 [paragraph B.1.c(2)], breaks in ASME Code Class 2 and 3 piping should be postulated at the following locations:

(a) at terminal ends

(b) at intermediate locations selected by one of the following criteria:

(i) at each pipe fitting, welded attachment, and valve

(ii) at each location where the stresses exceed 0.8 (1.2 S_h + S_A), but at not less than two separated locations chosen on the basis of highest stress.

In its proposal the licensee has not proposed changing criterion (a) above. Postulation of pipe breaks at terminal ends will not be eliminated in the proposed SRP deviation for Class 2 and 3 piping systems.

The AIB criterion is stated in criterion (b)(ii) above where breaks are to be postulated at intermediate locations where the stresses exceed 0.8 (1.2 S_h + S_A)

but "at not less than two separated locations chosen on the basis of highest stress." The stress limit provided in the above pipe break criterion represents the stress associated with 80% of the combined primary and secondary stress limit. Thus, a break is required to be postulated where the maximum stress range as calculated by the sum of Eq. (9) and (10) of NC/ND-3652 of the ASME Code, Section III, exceeds 80% of the combined primary and secondary stress limit, when one considers those loads and conditions for which level A and level B stress levels have been specified in the system's design specification (i.e., sustained loads, occasional loads, and thermal expansion), including an operating basis earthquake (OBE) event. However, the Class 2 and 3 pipe break criteria do not have a provision for the postulation of pipe breaks based on a fatigue limit since an explicit fatigue evaluation is not required in the ASME Code for these classes of construction because of favorable service experience and lower levels of operating cyclic stresses.

For those Class 2 and 3 piping systems which experience a large number of stress cycles (e.g., main steam and feedwater systems), the ASME Code has provisions which are intended to address these types of loads. The rules governing considerations for welded attachments in ASME Class 2 and 3 piping which do preclude fatigue failure are partially given in paragraph NC/ND-3645 of the ASME Code. The Code states:

> External and internal attachments to piping shall be designed so as not to cause flattening of the pipe, excessive localized bending stresses, or harmful thermal gradients in the pipe wall. It is important that such attachments be designed to minimize stress concentrations in applications where the number of stress cycles, due either to pressure or thermal effect, is relatively large for the expected life of the equipment.

Code rules governing the fatigue effects associated with general bending stresses caused by thermal expansion are addressed in NC/ND-3611.2(e) and are generally incorporated into the piping stress analyses in the form of an allowable stress reduction factor.

Thus it can be concluded that when the piping designers have appropriately considered the fatigue effects for Class 2 and 3 piping systems in accordance with NC/ND-3645, the likelihood of a fatigue failure in Class 2 and 3 piping caused by unanticipated cyclic loadings can be significantly reduced.

(3) Additional Design Considerations

In its presentation to the Advisory Committee on Reactor Safeguards (ACRS) on June 9, 1983, and in an October 5, 1983, meeting between a group of pressurized-water-reactor (PWR) near-term operating license utilities and the NRC staff, the staff indicated that the elimination of AIBs was not to apply to piping systems in which stress corrosion cracking, large unanticipated dynamic loads such as steam or water hammer, or thermal fatigue in fluid mixing situations could be expected to occur. In addition, the elimination of AIBs was to have no effect on the requirement to environmentally qualify safety-related equipment and in fact this requirement was to be clarified to assure positive qualification requirements.

(a) Intergranular Stress Corrosion Cracking (IGSCC)

As discussed in the licensee's letter of January 14, 1986, the IGSCC potential is likely to be reduced if the following factors are controlled: high residual tensile stresses, susceptible piping material and a corrosive environment. Although any stainless or carbon steel piping will exhibit some degree of residual stress and material susceptibility, the licensee has demonstrated (January 14, 1986, letter) that Millstone 3 minimizes the potential for IGSCC by utilizing piping material with low susceptibility to stress corrosion and by preventing the existence of a corrosive environment. The likelihood of stress corrosion cracking in stainless steel increases with carbon content. Consequently, only the lower carbon content stainless steel (304, 304L, 316, 316L) have been used for the primary systems at Millstone 3. For the secondary piping systems, carbon steel has been used for the piping, fittings, and valve bodies forming the pressure boundaries. The piping is flushed with demineralized water subject to limits on total dissolved solids, conductivity, chlorides, fluorides, and pH. Water chemistry for preoperational testing is controlled by written specifications.

During plant operation, primary- and secondary-side water chemistry are monitored. Containment concentrations are maintained below the thresholds known to be conducive to stress corrosion cracking. The water chemistry control standards are included in operating procedures for the systems where arbitrary breaks are being eliminated.

On the basis of the above information, the staff has concluded that adequate protection against IGSCC has been provided by the licensee.

(b) Water/Steam Hammer

Because of the susceptibility of main feedwater (MFW) systems to water hammer, the licensee has incorporated several water hammer prevention/minimization features into the design of the MFW piping at Millstone 3. As discussed in the January 14, 1986, letter and in SER Section 10.4.7, the potential for water hammer caused by rapid condensation of a steam bubble in the steam generator (SG) feedring has been reduced by installing J-tubes in the feedring to prevent drainage of water during low steam generator water level. Also, the feedwater (FW) connections to the steam generators utilize a downward-turned 90-degree elbow that does not present a horizontal pipe run immediately upstream of the FW nozzles. This configuration is intended to prevent formation of steam pockets during steam generator low water level conditions and to minimize the volume of water external to the SG which could pocket a steam bubble. This piping arrangement follows Westinghouse design guidelines.

On the basis of the above information and the inclusions in SER Section 10.4.7, the staff concludes that the design features and operating procedures described above will minimize the potential for water hammer occurrence in the main FW piping system.

(c) Thermal Fatigue

The staff has concluded that the Millstone 3 systems for which AIBs are to be eliminated are not susceptible to thermal fatigue and mixing for the following reasons:

(i) The fatigue analysis performed by the licensee for Class 1 piping systems shows that all of the Class 1 AIB locations involve cumulative usage factors below the AIB postulation limit of 0.1 (licensee's letter of January 14, 1986). For Class 2 and 3 piping components, fatigue failure protection is assured by the ASME Code-allowable stress range checks and a stress range reduction factor for thermal expansion stress. The mandatory breaks are postulated at 80% of the Code-allowable stresses, even after eliminating the AIBs identified in the January 14, 1986, letter).

(ii) The plant systems are designed to minimize thermal cycling and thermal shock. The FW system is arranged so that during normal operation, mixing (backflow from the steam generator) of hot water in the steam generator with the lower temperature FW is localized in the area of the FW inlet nozzle. The FW piping geometry has a downward elbow directly off the steam generator nozzle which forms a loop seal geometry to minimize mixing. The inverted J-tubes located on the FW sparger further reduce the mixing effect. Feedwater flow surges are precluded by control valve design. The auxiliary FW ties into the vertical main FW piping just upstream of the loop seal geometry. The location of the auxiliary FW connection minimizes the length of piping subjected to the lower temperature auxiliary FW. Thermal sleeves are provided in the branch connection to the main FW piping and in the steam generator nozzle to minimize the thermal transient effects of auxiliary FW initiation.

(4) Class 1 Piping Systems Evaluation

For Class 1 piping, a considerable amount of quality assurance in design, analyses, fabrication, installation, examination, testing, and documentation is provided which ensures that the safety concerns associated with the uncertainties discussed above are significantly reduced. On the basis of the staff evaluation of the design considerations given to Class 1 piping, the stress and fatigue limits provided in the BTP MEB 3-1 break criteria, and the relatively small degree of uncertainty in unanticipated loadings, the staff finds that the need to postulate arbitrary intermediate pipe breaks in ASME Code Class 1 piping in which large unanticipated dynamic loads, stress corrosion cracking, and thermal fatigue such as in mixing situations are not present and in which all equipment has been environmentally qualified is not compensated for by an increased level of safety. In addition, systems may actually perform more reliably for the life of the plant if the SRP criterion to postulate arbitrary intermediate breaks for ASME Code Class 1 piping is eliminated. The staff has concluded that the above-described requirements are present for those ASME Code Class 1 piping systems identified in the licensee's submittal of January 14, 1986.

(5) Class 2 and 3 Piping Systems Evaluation

On the basis of the staff evaluation of the design considerations given to Class 2 and 3 piping, the stress limits provided in the SRP break criterion, and the relatively small degree of uncertainty in unanticipated loadings, the staff finds that dispensing with arbitrary intermediate pipe breaks is justified for Class 2 and 3 piping in which stress corrosion cracking, large unanticipated dynamic loads, or thermal fatigue in fluid mixing situations are not expected to occur provided (a) the piping designers have appropriately considered the effects of local welded attachments per NC/ND-3645 and (b) all safety-related equipment in the vicinity of Class 2 and 3 piping systems has been environmentally qualified for the nondynamic effects of a nonmechanistic pipe break with the greatest consequences on the equipment. The staff has concluded that the above-described requirements are present for those ASME Code Class 2 and 3 piping systems identified in the licensee's letter dated January 14, 1986.

(6) Piping Systems Not Included in Proposal

For those piping systems, or portions thereof, which are not included in the licensee's submittal of January 14, 1986, the staff requires that the existing guidelines in BTP MEB 3-1 of the SRP (NUREG-0800), Revision 1, be met. However, should other piping lines which are not specifically identified in the licensee's submittal subsequently qualify for the conditions described above, the implementation of the proposed elimination of the AIB criteria may be used provided those additional piping lines are appropriately identified to the staff.

3.6.2.3 Conclusion

The licensee has proposed a deviation from the current guidelines of the SRP by requesting relief from postulating arbitrary intermediate pipe breaks in high-energy piping systems which are not susceptible to intergranular stress corrosion cracking, steam or water hammer effects, and thermal fatigue in fluid mixing. The SRP guideline which requires that two intermediate breaks be postulated even when the piping stress is low resulted from the need to ensure that equipment qualified for the environmental consequences of a postulated pipe break was provided over a greater portion of the high-energy piping run.

This proposal is based, in part, on the condition that all equipment in the spaces traversed by the fluid system lines, for which AIBs are being eliminated, is qualified for the environmental (nondynamic) conditions that would result from a nonmechanistic break with the greatest consequences on surrounding equipment. In addition, the licensee has committed to perform preoperational testing of all the systems identified in the letter of January 14, 1986, and also monitor those systems for vibration during preoperational and startup testing.

The staff has evaluated the technical bases for the proposed deviation with respect to satisfying the requirements of GDC 4. Furthermore, the staff has considered the potential problems identified in NUREG/CR-2136 which could impact overall plant reliability when excessive pipe whip restraints are installed. On the basis of its review, the staff finds that when those piping system conditions as stated above are met, there is a sufficient basis for concluding that an adequate level of safety exists to accept the proposed deviation.

Thus, on the basis that the piping systems have satisfied the above conditions, the staff concludes that the pipe rupture postulation and the associated effects are adequately considered in the design of Millstone 3 and, therefore, the deviation from the Standard Review Plan is acceptable.

3.9 Mechanical Systems and Components

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

In Section 3.9.2 of SSER 4, the staff identified a confirmatory item regarding the implementation of the seismic interaction program. The licensee had previously submitted for staff review its seismic interaction program for addressing potential interactions between seismic Category I and non-seismic Category I piping and equipment. The staff evaluation provided in SSER 4 found the program to be acceptable. However, the implementation of the program had not been completed at that time. Thus, the staff considered the issue to be acceptable contingent upon a satisfactory implementation of the program, the acceptability of the results, and the completion of any corrective actions which may be required as a results of its implementation.

In a letter from J. F. Opeka to H. Denton dated January 14, 1986, the licensee provided the results of its seismic interaction program. The results of the program found that the seismic interactions which could occur between seismic Category I and non-seismic Cateogry I piping and equipment will not adversely affect the ability of seismic Category I piping and equipment to perform their safety functions. The calculated stresses induced in the seismic Category I piping and equipment resulting from the seismic interactions were all found to be within the acceptance criteria previously accepted by the staff in SSER 4, except as noted below.

As a confirmatory measure, the staff performed an audit of the seismic interaction program results at the Sargent & Lundy offices on January 9, 1986. The audit focused on the basis for concluding that the piping analyses performed are bounding for the evaluation of non-seismic Category I piping systems. Furthermore, the staff reviewed the program results to evaluate the appropriateness of overall conclusions drawn from those piping subsystems where high stresses and large lateral deflections were found, in order to determine the need for generic recommendations or additional corrective actions.

During the audit, the staff reviewed the resolution of high piping stresses and large lateral deflections as noted in several piping subsystems. The staff found that the use of damping values per ASME Code Case N-411 in conjunction with refined analytical techniques resulted in acceptable piping stresses. Except for two cases, the piping stresses were found to meet the ASME Code Service Level D allowable values as stated in the 1983 Edition (including Winter 1984 Addenda).

The two cases in which Service Level D allowables were exceeded occurred in a chilled water subsystem (AX-107X) and in a boron recovery subsystem (SL-7A). Further evaluation by the licensee of the AX-107X overstress condition determined that the formation of a plastic hinge would occur in a branch line pipe elbow. However, the maximum angular rotation at the hinge was calculated to be 2.2 degrees and would not lead to total collapse of the piping subsystem. Furthermore, the high stress was primarily caused by a main header displacement and not by the branch pipe inertia acceleration. If this secondary displacement component of the pipe stress were removed from the stress evaluation, the calculated primary stress would meet ASME Code-allowable values.

The second condition of overstress occurred in subsystem SL-7A which consisted of mainly 1-inch- and 2-inch-diameter piping. Two overstress conditions were found at socket welded tees. However, when the actual size of the fillet welds was used in the calculations, the recalculated stresses were found to be within Level D-allowable values. The staff finds that the additional evaluations performed by the licensee adequately address the overstress conditions and are acceptable.

Excessive piping lateral deflections were identified as a generic concern by the licensee because of the potential for the piping to slip off certain types of supports. Thus, the licensee initiated a walkdown of all affected piping systems and implemented modifications to those supports to preclude the potential of pipe slipoff. All modifications to these supports will be completed before 5% power is exceeded. The staff finds the actions taken by the licensee to address large lateral pipe deflections to be acceptable.

The staff also reviewed the methodology used to evaluate the effect of upward loads on supports designed for downward loads only and found the approach adequate. The staff also noted that the seismic accelerations in the vertical direction, in general, tend to be less than the dead weight of the piping.

For the evaluation of pipe supports, the staff reviewed the qualification of certain standard component supports in which the vendor-recommended allowable values were exceeded. For U-bolts and U-straps, the licensee performed analyses to qualify the U-bolts and U-straps to higher loads. The analysis results were also compared with recent test results available from the vendor for further substantiation and were found to be in agreement. For concrete expansion anchor bolts, the licensee had previously conducted extensive testing to qualify the anchor bolts for the specific bolt sizes and concrete strengths used at Millstone 3. For the non-seismic Category I piping systems, the licensee used faulted (seismic Level D) allowables for the anchor bolts, which resulted in a minimum factor of safety of 2.95. Although the safety factor is less than the vendor-recommended design factor of safety of 4.0, the staff found the lesser factor of safety which could result in a slight slippage of the anchor bolt (but not complete pullout failure) to be acceptable for the faulted condition of non-seismic Category I piping systems where structural integrity, not piping functionality, is the primary consideration.

On the basis of the staff's review of the results of the seismic interaction program, the staff concludes that the bounding analyses of the non-seismic Category I piping systems and the supplemental walkdown measures, including corrective actions taken by the licensee to resolve potentially unacceptable seismic/non-seismic interactions, provide a reasonable basis to conclude that the non-seismic Category I piping systems will maintain their structural integrity and will not adversely affect the ability of seismic Category I piping and equipment to perform their safety functions. Thus, the staff finds the implementation of the seismic interaction program to be adequate and the results to be acceptable. All modifications to preclude potentially unacceptable swing/sway interactions will be completed before 5% power is exceeded. The piping stress reports have yet to be completed to include reconciliation of these support modifications. However, the staff believes the impact of these modifications will not likely change the acceptability of the program results nor the conclusions reached by the staff. The licensee should inform the staff when the final reconciliation is completed. If further modifications are required, the staff will pursue this matter at that time. Thus, License Condition 2.C(4) of the Millstone 3 Operating License, NPF-44, has been adequately satisfied.

3.11 Environmental Qualification of Electrical Equipment Important to Safety and Safety-Related Mechanical Equipment

License Condition 2.C(3) of the Millstone 3 Low Power License stated "By December 27, 1985, the licensee shall submit a revised compartment analysis using the mass and energy release data provided by the Westinghouse Owners Group Program." By letters dated December 20, 1985, and January 7 and 14, 1986, the licensee provided the requested information.

To calculate pressure and temperature transients following a main steamline break (MSLB), the licensee used mass and energy data from Westinghouse Topical Report WCAP-10961-P to account for the effect of superheated steam release due to steam generator tube uncovery. By a letter from J. F. Opeka (Northeast Utilities) to V. S. Noonan (NRC), dated January 17, 1986, the licensee provided for staff review one copy of WCAP-10961-P entitled "Steam Line Break Mass/Energy Release for Equipment Environmental Qualification Outside Containment." In the interim, Westinghouse is preparing a formal submittal of the topical report in accordance with the requirements of 10 CFR 2.790.

The staff finds that Millstone 3 Model F steam generators and its main steamline break protection system are included in the studies of the Westinghouse Topical Report. The report includes six categories of Westinghouse Owners Group plants in which Millstone 3 is categorized as Category 1. In each category, a spectrum of breaks in different sizes, power levels, break locations, and auxiliary feedwater models was studied in detail. The report is still under review; however, the review has progressed sufficiently for the staff to conclude there is reasonable assurance that the concerns regarding Millstone 3 superheated steam blowdown have been resolved pending the generic approval of the topical report.

In addition, the licensee stated that all Millstone 3 equipment which is required to function to mitigate the consequences of a main steamline break accident is qualified to function at the maximum compartment temperature of 325°F at steamline isolation. The licensee also stated that the equipment will remain in its safe position regardless of the fact that it will be exposed to temperatures above the qualification temperature. The staff reviewed all the information provided by the licensee and found it acceptable. Accordingly, the staff concludes that the aforementioned license condition has been satisfied.

4 REACTOR

4.4 Thermal-Hydraulic Design

4.4.4 Operating Abnormalities

4.4.4.2 Crud Deposition and Flow Measurement Uncertainty

In Section 4.4.4.2 of SSER 3 for Millstone 3, the staff reviewed the licensee's July 15, 1985, analysis for flow measurement uncertainty. It was concluded that enough detail was not presented in the analysis to enable the staff to find the flow uncertainty values acceptable. Also, contrary to past experience, the flow measurement uncertainty for both four- and three-loop operation was presented as being the same.

A revised flow measurement analysis was presented (letter from J. F. Opeka (NNECO) to B. J. Youngblood (NRC), dated September 26, 1985). The revised flow measurement uncertainty analysis included both four-loop and three-loop operation as well as elbow tap error. The total flow uncertainities with two indicators per loop were given as: 2.4% (four loops in operation) and 2.76% (three loops in operation). These values did not include the 0.1% additional penalty to account for venturi fouling. The licensee committed to add inspection ports upstream and downstream of the venturis during the first refueling outage.

Before the start of each cycle (before performing the calorimetric measurement for flow), the venturi meters will be verified to be clear [by performing a visual inspection (borascope, photography, etc.)] and will be cleaned when necessary. However, if the venturi meters are not inspected, the licensee committed to add an additional 0.1% to the total RCS flow measurement uncertainty.

The staff reviewed the Millstone 3 plant-specific flow measurement uncertainty analysis and compared the results with a generic Westinghouse flow measurement analysis. The Millstone 3 values were found to be conservative, but a number of details were still missing that were required for the analysis; however, because of the conservative results presented in this analysis, the staff accepted the flow measurement uncertainities of 2.4% (four-loop operation) and 2.76% (three-loop operation), subject to receiving a more detailed analysis for confirmation. These values were specified in the Technical Specifications and were also used for the minimum allowed reactor coolant flow Technical Specification.

The Technical Specifications also state the conditions for which the 0.1% venturi fouling penalty would be eliminated if the venturi meters were determined to be cleaned. The licensee subsequently presented the detailed flow measurement uncertainty analysis for confirmation in a letter from J. F. Opeka (NNECO) to V. S. Noonan (NRC), dated January 7, 1986. This analysis provided sufficient detail. However, the flow measurement uncertainities were changed from the previous analysis in a less conservative direction. The new values are given as: 2.0% (four-loop operation) and 2.3% (three-loop operation).

The staff has not completed its review of the January 7, 1986, submittal. Therefore, until this review is complete, the licensee will be required to use the flow uncertainty values and resulting Technical Specifications based on the September 26, 1985, submittal. These values are: 2.4% (for four-loop operation) and 2.76% (for three-loop operation). These numbers do not include a possible venturi fouling uncertainty as discussed previously in this section.

4.4.8 Instrumentation for Detection of Inadequate Core Cooling

4.4.8.1 Clarification of Requirements

A clarification of requirements for inadequate core cooling instrumentation (ICCI) was provided in Item II.F.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements." In November 1982, the Commission determined that an instrumentation system for detecting inadequate core cooling (ICC) consisting of upgraded subcooling margin monitors (SMMs), core exit thermocouples (CETs), and a reactor coolant inventory tracking system (ITS) is required for the operating pressurized water reactor facilities.

4.4.8.2 Inadequate Core Cooling Instrumentation System Design

The Millstone 3 licensee, Northeast Nuclear Energy Co. (NNECO), has provided information in the revised FSAR Section 4.4.6.5 (Amendment 9) and in a letter dated June 14, 1984, from W. G. Counsil (NNECO) to B. J. Youngblood (NRC) in response to staff concerns regarding ICCI.

The Millstone 3 ICC monitoring system consists of three instrumentation subsystems: (1) the subcooling margin monitor (SMM), (2) the core exit thermocouples (CETs), and (3) the reactor coolant inventory monitoring system. The Millstone 3 ICC system has been designed as Class 1E with redundant trains, each containing standalone processing electronics and displays to monitor, alarm, and trend ICC. The monitoring system was tested and calibrated before fuel load. Millstone 3 is provided with two ICC information display systems. Redundant Class 1E cabinets are provided outside the control room and display SMM, CET, and coolant level information. In the control room, the primary ICC information is provided through the safety parameter display system (SPDS). All the ICC information transmitted to the SPDS has been provided with optical isolation. The SPDS displays are designed to incorporate accepted human factors principles so that the displayed ICC information can be readily understood by plant operators during normal and abnormal plant conditions.

4.4.8.3 Subcooling Margin Monitor

The SMM system uses reactor coolant system (RCS) temperatures and pressures to calculate subcooling (to 300°F) and superheat (to 45°F) either in terms of temperature or pressure. The calculation is based on the most conservative values of the temperature and pressure input. The calculation of the subcooling/superheat is performed by the SPDS using input from the T_{hot} and T_{cold} resistance temperature devices (RTDs), CET, the unheated junction temperatures of the heated junction thermocouple (HJTC), and the RCS pressure. Signal validation techniques are utilized to ensure the quality of the input variables. Saturation/superheat trouble alarms are provided on the main control board from the ICC cabinet.

4.4.8.4 Core Exit Thermocouples

The core exit thermocouple (CET) monitoring system consists of two redundant independent trains that monitor the 50 chromel-alumel CETs. All CETs are provided with the required cold junction temperature compensation which consists of an RTD providing a signal to the ICC processor in the Class 1E cabinet and display. The CET temperature range is from 200°F to 2300°F. All CETs are displayed on a digital panel meter selectable from a switch panel. All CETs are uniformly dispersed in the core, therefore, satisfying the requirement of a minimum of four CETs per quadrant. A CET high alarm is provided in the main control boards from the ICC cabinet.

4.4.8.5 Reactor Vessel Level Instrumentation System

The heated junction thermocouple system monitors coolant inventory in the region above the core. Redundant strings of heated junction thermocouples are arranged in the reactor vessel head area to provide an indication on conditions at eight distinct levels. The system includes two channels, each consisting of a string of eight equidistant sensors. The system indicates percent of level in the plenum and the head areas. One of the ways of displaying the ICC information is provided through the SPDS. However, the SPDS display is not Class 1E. During the staff's audit of the SPDS on July 29, 1985, the licensee provided a "Design Availability Calculation" for the SPDS estimating the availability at 99.54%. The plant-specific ICC procedures will be based on Combustion Engineering reports CEN-185, CEN-152, and the CEOG letter to D. M. Crutchfield dated June 1, 1982, and will be incorporated into the Westinghouse emergency response guidelines.

The generic Combustion Engineering topical report on inadequate core cooling instrumentation using heated junction thermocouples for reactor vessel level measurement has been reviewed by the staff and was found acceptable. The evaluation was published in NUREG/CR-2627, March 1982.

4.4.8.6 Evaluation and Conclusions

The staff has reviewed the licensee's submittal dated June 14, 1984 [(W. G. Counsil (NNECO) to B. J. Youngblood (NRC)], the revised FSAR Section 4.4.6.5 (Amendment 9), and the licensee's submittal dated January 20, 1986 [J. F. Opeka (NNECO) to V. S. Noonan (NRC)].

The staff concludes that the ICC system design is acceptable for an operating license and to satisfy the requirements of NUREG-0737, Item II.F.2. The ICC primary and backup display instrumentation is acceptable with the SPDS being non-Class 1E but with an estimated availability of over 99%.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

This evaluation supplements conclusions in this section of the SER (NUREG-1031) which addressed the definition of examination requirements and evaluation of compliance with 10 CFR 50.55a(g). In previous supplements to NUREG-1031, Section 5.2.4, the staff reported that the Preservice Inspection Program for the systems and components within the reactor coolant pressure boundary is consistent with the applicable regulation and Code requirements and the review is a confirmatory issue contingent upon the licensee completing the required examinations and identifying all impractical preservice inspection requirements with a supporting technical justification.

In a letter dated December 23, 1985, the licensee submitted relief requests from ASME Code Section XI requirements which the licensee has determined to be impractical to perform at Millstone 3 and provided supporting information pursuant to 10 CFR 50.55a(a)(3). The staff evaluated the ASME Code-required examinations that the licensee determined to be impractical and the staff found that the licensee has demonstrated that compliance with the requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. On the basis of review of the licensee's submittals and granting of relief from the preservice examination requirements, the staff concludes that the Preservice Inspection Program for systems and components within the reactor coolant pressure boundary at Millstone 3 is acceptable and in compliance with 10 CFR 50.55a(g)(3). The detailed evaluation supporting this conclusion is provided in Appendix N to this report.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

In the Safety Evaluation Report for Millstone 3, the staff evaluated the adequacy of the containment structure functional design (SER Section 6.2.1.1). In SSER 2, the staff evaluated the adequacy of the pressurizer cubicle design (SSER 2, Section 6.2.1.2). After SSER 2 was issued, the licensee submitted several amendments to the FSAR that alter the analyses on which the staff's above reviews are based. A brief description of the key changes made in each amendment, and the impact of the changes on the licensee's licensing basis calculations, is provided below.

6.2.1.1 Containment Structure

In FSAR Amendment 15, the licensee (1) revised the mass and energy release rate data for the limiting pipe breaks for containment depressurization (generally lower release rates for the double-ended pump suction guillotine (DEPSG), and the 0.6 DEPSG), (2) modified certain containment recirculation cooler parameters (service water flow to each cooler changed from 65,000 gpm to 6,230 gpm, overall heat transfer (UA) for each cooler changed from 3.86 X 10^6 to 3.79 X 10^{10} Btu/hr/°F), and (3) changed the initial containment pressure for depressurization analysis from 9.76 psia to 9.81 psia. In FSAR Amendment 17, the licensee (1) shifted the recirculation spray pump starting time from 220 seconds to 670 seconds, (2) took credit for approximately 260,000 square feet of additional steel heat sink, and (3) made additional modifications to the mass and energy release rate data for the limiting pipe breaks for containment depressurization. The licensee reported that as a result of these changes, the peak containment pressure decreased from 39.4 psig to 36.09 psig (because of additional heat sinks), the maximum calculated time to reestablish a subatmospheric condition decreased from 3350 seconds to 2560 seconds (largely from revisions in the mass and energy release rate data), and the maximum subatmospheric peak pressure increased from -0.13 psig to -0.07 psig.

The staff has performed revised confirmatory calculations to reflect the above changes. The staff's calculations result in a peak containment pressure (for the hot-leg double-ended rupture) of 36.8 psig using the CONTEMPT-LT/28 computer code, and a maximum depressurization time (for the 0.6 DEPSG) of 2720 seconds using the CONTEMPT-4, MOD6 computer code. The staff's analyses approximately confirm the applicant's calculations. The staff has also performed a revised minimum containment pressure analysis using the CONTEMPT-LT/28 computer code to determine the impact of the additional heat sink area reported by the licensee. The CONTEMPT results remain in good agreement with those originally provided by the licensee for the entire transient. On the basis of these calculations, the staff concludes that the licensee has satisfactorily demonstrated the adequacy of the containment functional design, and the minimum containment pressure analysis.

6.2.1.2 Subcompartment Analyses

In FSAR Amendment 16, the licensee reported results of analyses for two additional pipe break locations in the lower pressurizer cubicle subcompartment, and revised the subcompartment design pressure from 20.5 psid to 27.3 psid. As a result of the change in pipe break location, the limiting differential pressure for the lower pressurizer cubicle (surge line double-ended rupture) increased from 20.31 psid to 24.15 psid. This peak calculated differential pressure is below the revised design value for the subcompartment.

The staff has performed a confirmatory analysis of the postulated pipe break using the COMPARE-MOD1A computer code. The staff's calculation also produces a peak differential pressure of 24.1 psid. On this basis, the staff finds the licensee's subcompartment analyses acceptable.

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

License Condition 2.C(9) of the Millstone 3 Low-Power License states that "NNECO shall justify the applicability of the Westinghouse LOFTRAN methodology to the Model F steam generator as contained in Topical Report WCAP-8822-P-S2." By letter from J. F. Opeka (NNECO) to V. S. Noonan (NRC) dated January 13, 1986, the licensee provided the requested justification. The licensee states that the primary factors which influence containment response to superheat conditions are independent of steam generator type, and consequently, the Westinghouse LOFTRAN methodology is applicable to Millstone 3. The staff has reviewed the licensee's justification and concurs with the conclusion that the LOFTRAN methodology is applicable to Millstone 3 with the Model F steam generator. Therefore, the staff concludes that its concerns expressed in License Condition 2.C(9) are resolved.

6.6 Inservice Inspection of Class 2 and 3 Components

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

This evaluation supplements conclusions in Section 6.6.3 of the SER (NUREG-1031) which addressed the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g). In previous SER supplements, the staff reported in Section 6.6 that the Preservice Inspection Program for Class 2 and 3 systems and components is consistent with the applicable regulation and Code requirements, and the review is a confirmatory issue contingent upon the licensee completing the required examinations and identifying all impractical preservice inspection requirements with a supporting technical justification.

In a letter dated December 23, 1985, the licensee submitted relief requests from ASME Code Section XI requirements which the licensee determined to be impractical to perform at Millstone 3 and provided supporting information pursuant to 10 CFR 50.55a(a)(3). The staff evaluated the ASME Code-required examinations that the licensee determined to be impractical and the staff found that the licensee has demonstrated that compliance with the requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. On the basis of a review of the licensee's submittals and the granting of relief from the preservice examination requirements, the staff concludes that the Preservice Inspection Program for systems and components within the reactor coolant pressure boundary at Millstone 3 is acceptable and in compliance with 10 CFR 50.55a(g)(3). The detailed evaluation supporting this conclusion is provided in Appendix N to this report.

9 AUXILIARY SYSTEMS

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

9.5.1.5 Fire Detection and Suppression

Carbon Dioxide Suppression System

In SSER 4 the staff stated that the license would be conditioned to require the licensee to demonstrate the operability of the gaseous (CO_2) fire suppression system before initial criticality. By letter from J. F. Opeka (NNECO) to V. S. Noonan (NRC) dated December 3, 1985, the licensee submitted the results of the acceptance tests on the carbon dioxide systems. These results confirmed that the design concentration of carbon dioxide gas was achieved and maintained at all test probes. On this basis, the staff concludes that the carbon dioxide fire suppression systems conform with Section C.6.d of BTP CMEB 9.5-1 and are, therefore, acceptable.

9.5.3 Lighting System

In SSER 3, the staff incorrectly stated that the licensee committed, in its letter of July 18, 1985, to increase the dc lighting systems illumination level to a minimum of 10 foot-candles in certain areas of the purple switchgear room, control room, orange switchgear room, and diesel generator room. The licensee instead committed to provide a maintained average of 10 foot-candles in those areas. The staff finds this acceptable.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

13.1.2 Plant Staff

As stated in Low-Power License Condition 2.C(14), training of the shift advisers and the shift crew by the licensee shall be completed and approved by the NRC before 5% power is exceeded.

This safety evaluation presents the staff's assessment of the licensee's provisions for hot participation experience on shift at Millstone 3 during low-power testing and power operations.

Shift Advisor Qualifications

Three shift advisor candidates were selected. All are employees or previous employees of Northeast Utilities (NU), having served at other NU nuclear plants. Each has had extensive previous nuclear experience, fully meeting the minimum experience levels for shift advisors set forth in Generic Letter 84-16. Each shift advisor meets the medical requirements for licensed operators. The staff concluded that the shift advisor candidates were acceptable.

Shift Advisor Duties and Responsibilities

The licensee provided a copy of Station Operating Procedure OP-3262, Operations Shift Advisor, which sets the duties, responsibilities, and qualifications of the shift advisor. The staff reviewed this procedure and concluded that it acceptably describes the duties and responsibilities of the shift advisors. However, during a telephone conversation with representatives of the NU training department and licensing staff on November 6, 1985, the staff pointed out that Section 6.1 of the procedure should be expanded to include a requirement that shift advisors participate in the licensed operator requalification training program so that they can be cognizant of changes in procedures design and license conditions. In addition, the shift advisors should participate in shift training, including training on the Millstone 3 simulator. On November 8, 1985, the staff was informed by the licensee that Procedure OP-3262 would be expanded as noted above. This commitment was confirmed by a letter from the licensee dated November 12, 1985. The staff concluded that the licensee's description of shift advisor duties and responsibilities was acceptable.

Shift Advisor Training Program

The licensee described the initial training program to be presented to the shift advisor candidates to qualify them to serve as shift advisors at Millstone 3. The staff's initial review of this program indicated that it was generally acceptable. The staff did identify, however, a number of areas in which some revision to the program was in order. The staff discussed these matters with a representative of the licensee's training department during a telephone conversation on November 6, 1985. On November 8, 1985, the staff was informed by the licensee that these revisions to the training program would be made; this commitment subsequently was confirmed by letter dated November 12, 1985. On this basis, the staff concluded that the shift advisor training program was acceptable.

Systems Training

The staff reviewed the training objectives described in the training program for the equipment and systems described, and considered these objectives suitable for shift advisor training. However, staff review of the systems portion of the training program concluded that the systems training did not include training on a number of systems that are vital to normal plant operation and others of which the shift advisors should be knowledgeable to provide advice to the operating shifts. These include:

- reactor coolant pumps and motors, including lubrication, cooling, and monitoring equipment
- residual heat removal
- component cooling
- reactor protection
- non-nuclear instrumentation, including incore temperature monitoring
- containment, including normal and emergency cooling
- reheat steam and feedwater heating
- main turbine, generator, and condenser, including auxiliary systems
- ac and dc vital power supply
- service water systems for normal and emergency operations
- area and process radiation monitors
- fire protection systems

In the letter of November 12, 1985, the licensee committed to include these systems in the training program. The staff concluded that the proposed systems training was acceptable.

Technical Specification Training

NU proposed to train shift advisors on the technical specifications that apply to Millstone 3 during operations in modes 1 and 2. However, once assigned, the services of shift advisors are required whenever the unit is in operating modes 1-4. Therefore, the staff required that the training be broadened to include training in technical specifications applicable to operating modes 1-4. Furthermore, the staff felt that the objectives of the technical specification training should include the bases for the technical specifications. The licensee committed orally to broaden this training to include technical specifications applicable to operation in modes 1-4. This commitment was confirmed in the November 12, 1985, letter, and the staff thus concluded that the technical specification training was acceptable.

Procedure Training

The staff questioned whether too much emphasis would be given to abnormal and emergency operation procedures in lieu of concentrating on procedures for normal plant operation. In discussions with the NU staff, NU clarified that the intent of this segment is to concentrate on procedures which are germane to the shift advisor duties for normal operations, while providing a suitable background for abnormal and emergency conditions that may occur. This was acceptable. The NRC staff also questioned the apparent lack of training in administrative procedures which are germane to shift operations. The NU representatives clarified this issue by reference to a September 20, 1985, letter which describes the similarity of administrative procedures at NU nuclear plants. By virtue of the extensive previous experience of each of the shift advisor candidates at one or more of the other NU nuclear plants, NU considered that special training in administrative procedures was not required. The NRC staff agreed.

Simulator Training

The staff review did not disclose any provisions in the training program for the shift advisor candidates to practice problem solving involving use of their knowledge of the procedures, systems, and technical specifications. The staff recommended that problem solving be introduced into both the classroom and simulator portions of the program and be included in the final certification examination.

The simulator training for shift advisors consisted of observation of reactor and plant startups, power escalation of at least 25% above 20% power, and normal plant and reactor shutdown. In the letter of November 12, 1985, the licensee confirmed plans to incorporate problem solving specific to Millstone 3 through the use of instrument-failure-induced transients in the simulator training program. The staff concluded that the planned simulator training was appropriate for shift advisors.

General

The staff believed that knowledge of the previous checkout and test experience at Millstone 3 and an awareness of the overall startup and test program planned for Millstone 3 would enable the shift advisors to perform their duties better. The staff recommended that a brief overview of the past experience and future plans be incorporated into the training program.

In the November 12, 1985, letter, the licensee stated that the shift advisors would be briefed on the past experience at the plant and that they would be briefed on the upcoming startup and test programs at the briefings conducted for their respective shifts. The staff concluded that this was acceptable.

On December 10 and 11, 1985, the staff reviewed the written examination for the shift advisors and witnessed simulator exercises conducted with the shift advisors at the Millstone 3 plant-referenced simulator.

The staff reviewed the written examination which was administered to the shift advisor candidates on December 4, 1985; the answer key was also reviewed. In addition, the staff reviewed grading of the examination by the Millstone 3 training staff. The examination consisted of 21 questions requiring 39 responses, and covered the areas of Millstone 3 systems, procedures, and technical specifications. The questions were developed from the objectives contained in lesson plans for the shift advisor training program and are considered to be at the appropriate level for the candidates.

Review of the grading of the written examinations was conducted using criteria contained in NUREG-1021, "Operator Licensing Examination Standards." The grading was acceptable.

In a letter of January 10, 1986, the licensee provided copies of the written examination, answer key, and grades of the advisors. All candidates passed the written examination.

On December 11, 1985, the staff observed the shift advisors during exercises at the Millstone 3 simulator. The exercises were conducted over a 5-hour period and consisted of a normal plant shutdown from 60% power with a number of malfunctions. The plant shutdown was performed by an operating crew consisting of personnel who were preparing for operator licensing examinations. Each shift advisor was paired with a licensed or certified instructor/evaluator who also acted as an intermediary with the crew. The evaluations were conducted by use of simulator checklists and were submitted in the letter of January 10, 1986.

During the exercise period, the shift advisors were evaluated in the areas of plant systems, use of procedures, and technical specifications. The shift advisors also observed the planning and progress of the plant shutdown by the operating crew. During the shutdown and after malfunctions, the shift advisors often questioned the crew's method of resolving the malfunctions and its influence in continuing the planned shutdown. The staff found the setting of the evaluations was appropriate and that it presented ample opportunities for the shift advisors to act and be evaluated in the role of advisor. The staff also found that the evaluation process was conducted in a manner that met the guidance for simulator examinations, as applicable for shift advisors, contained in NUREG-1021.

In the letter of January 10, 1986, the licensee provided a simulator training performance summary for each of the advisors. The summary consisted of an evaluation of each advisor's

- ability to follow normal plant operations using plant procedures
- knowledge of Millstone 3 systems, control instrumentation, and reference material
- ability to evaluate abnormal and emergency conditions, including problem solving and establishing priorities

Although conclusions contained in the summaries are positive, they also include recommendations for additional training or familiarization. The staff concludes that the evaluations were conducted in an acceptable manner and represent a thorough assessment of the shift advisors.

Operating Shift Crew Training

In a November 1, 1985, letter, the licensee provided an outline of the planned training of operating shift crews to ensure they understand the role of shift advisors. The staff reviewed the program and determined the training was appropriate. During a telephone conversation with the licensee's staff on January 17, 1986, the staff requested confirmation that crew training had been completed. The licensee plans to inform the staff by letter of the completion of operating crew training.

Conclusion

In a Janury 10, 1986, letter confirming the certification of shift advisors, the licensee also provided an update of the activities of the shift advisors which includes: additional simulator training with their assigned crews; participation in the Millstone 3 Licensed Operator Requalification Program; and continuing involvement in the daily activities of their operating shifts. Subject to receipt of a letter from the licensee confirming completion of operating crew training, the staff concludes that the shift advisor training program has been completed in an acceptable manner and that the shift advisors are properly trained and qualified to perform advisory duties.

13.5 Station Administrative Procedures

13.5.2 Operating and Maintenance Procedures

13.5.2.3 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

Section 13.5.2.3 of SSER 4 discussed three open issues related to the Millstone 3 Emergency Response Guidelines (ERGs). These items are

(1) guidance for use of the reactor coolant system (RCS) loop isolation (stop) valves during recovery from a steam generator tube rupture (SGTR) event

(2) correcting degraded core cooling guidelines (EOP-35, FR-C.2).

(3) including in the Millstone 3 Emergency Operating Procedures (EOPs) reactor vessel level monitoring system (RVLMS) setpoints corresponding to 50% steam-water mixture as provided in the Westinghouse Generic ERGs using the reactor vessel liquid inventory system (RVLIS)

In a letter from J. F. Opeka (NNECO) to V. S. Noonan (NRC) dated January 14, 1986, the licensee addressed these items. The staff's evaluation of each follows:

(1) Use of Loop Stop Valves

The licensee provided for the use of loop isolation valves after a steam generator tube rupture (SGTR). Among the conditions required for stop valve use are achievement of stabilization of transient effects of the SGTR event, availability of offsite power, and adequate subcooling and level. The staff finds the guidance acceptable for immediate implementation based on the limitations for its use assuring an acceptably low likelihood for negative effects and the potential benefits of loop isolation for some beyond-design-basis scenarios.

However, the staff recommends that (a) the guidance be retitled to clarify the limitations of its use (by whom, how, and when) and (b) copies of the guidance be appended to, but not be made a part of the EOPs to facilitate communications between operators and technical staff when considering loop isolation.

In the longer term, consistent with resolution of N-1 operation considerations, the staff requires reassessment of this guidance, with commensurate modifications including explicit identification of loop isolation considerations as discussed in SSER 4 and identification of situations where loop isolation might be beneficial (relative to other options).

(2) Degraded Core Cooling Guideline (EOP-35, FR-C.2)

In its submittal, the licensee has stated that the level check (for exiting FR-C.2) would be restored to steps 5 and 7a to effect consistency between the approved generic ERGs and the Millstone 3 ERGs. By doing so, the typographical errors noted in the licensee's submittal would be corrected. This action is consistent with the staff's requirement and is acceptable.

(3) RVLMS Setpoints With Pumps

Running--In its submittal, the licensee stated that readings using RVLMS are not influenced by the operating status of RCS pumps as are those using the RVLIS in the approved referenced ERGS. The licensee stated that in the RVLIS the deleted steps compensated for pump status and that the same functional requirement of monitoring core covery is accomplished by the 19% level indication using RVLMS. The licensee indicated that the RVLMS level criterion would be included in the Millstone 3 EOPs. The staff finds this acceptable, since the licensee states that the function of the level criterion will be satisfied using the 19% RVLMS indication.

Conclusions

The staff finds that the licensee has resolved the concerns identified in SSER 4 regarding the Millstone 3 ERGs, and therefore Millstone 3 Low-Power License Conditions 2.C(15)(a), (b), and (c) have been satisfied. The staff's longer term requirement for improvements to loop isolation guidance for N-1 loop operation must be resolved before the staff approves N-1 loop operation as presented in Low-Power License Condition 2.C(6).

15 ACCIDENT ANALYSES

15.3 Decrease in Reactor Coolant Flow Rate

15.3.3/15.3.4 Reactor Coolant Pump Rotor Seizure and Shaft Break

(1) Locked Rotor Accident

The licensee has provided an analysis of the postulated locked rotor accident assuming that the atmospheric dump valve on one of the four steam generators fails open. The analysis assumes that the steam generator with the failed-open valve can be isolated within 30 minutes and, as a result of the departure from nucleate boiling, 3% of the fuel rods experience cladding failure. In addition, the licensee assumed (a) a steam dump of 159,000 lb from the affected steam generator before isolation, (b) a steam release from the remaining three steam generators, of 384,000 lb during the first 2 hours of the accident and 1,363,000 lb for the remaining 6 hours of the accident, (c) a 1-gpm leak from the primary side to the secondary side, and (d) an iodine partition coefficient of 100 for the unaffected steam generators and a partition coefficient of 1 for the affected steam generator.

The staff reviewed the licensee's analysis and then performed an independent dose assessment based on the parameters listed in Table 15.1. The staff analysis considered two cases: (a) all the leakage from the primary to the secondary side event goes into the affected steam generator and (b) all of the primary to secondary leakage goes into the unaffected steam generators. These two cases give an upper and lower bound for the locked rotor accident. The results of the staff analysis are presented in Table 15.2.

The staff concludes that the distances to the exclusion area and to the low-population zone (LPZ) boundaries for Millstone 3 are sufficient to provide reasonable assurance that the calculated radiological consequences of a potential locked rotor accident, consistent with the failure of an atmospheric dump valve and loss of offsite power, would not exceed 10 CFR 100.11 dose guidelines.

15.6 Decrease in Reactor Coolant Inventory

15.6.3 Steam Generator Tube Rupture

By letter dated October 25, 1985, the licensee proposed to use the results of the Westinghouse Owners Group (WOG) generic program for steam generator tube rupture (SGTR) to resolve the Millstone 3 SGTR licensing issue for both four-loop and three-loop operation.

The subgroup of WOG submitted WCAP-10698 in December 1984, and Supplement 1 to WCAP-10698 in May 1985, to support the resolution of the licensing issues associated with an SGTR accident. The subgroup also plans to submit an evaluation of the consequences of steam generator overfill resulting from an SGTR.

The licensee has indicated that the results of the generic program can be used to demonstrate the margin available for steam generator overfill for both four-loop and three-loop operations. On the basis of the comparison of the preliminary estimates of the time to overfill for the two plant types presented in WCAP-10698 and a comparison of the other factors that may affect the margin to overfill, the licensee stated that the evaluation would demonstrate increased steam generator overfill margin for four-loop operation. The overfill margin for three-loop operation may be slightly less than for four-loop operation as a result of the differences in initial plant conditions and recovery times associated with the reduced power level for three-loop operation. The staff concludes that the proposed methodology to evaluate an N-1-loop SGTR event is acceptable pending the results of the review of WCAP-10698.

Supplement 1 to WCAP-10698 has been reviewed by the staff (Mueller, October 22, 1985). The staff concluded that the dose analysis methodology used in the evaluation is acceptable, with the exception of the iodine transport models which were not provided by the licensee.

WCAP-10698 is currently being reviewed by the staff, and the staff will report its findings when the review is completed.

Table 15.1 Locked rotor accident assumptions

Parameter	Value
Power	3636 MWt
Primary to secondary side leak rate	1 gpm
Unaffected steam generator	
0-2-hour steam dump	384,000 lb
2-8-hour steam dump	1,363,000 lb
Iodine partition coefficient	100
Affected steam generator	
0-30-min steam dump	159,000 lb
Iodine partition coefficient	1
0-2-hour EAB meteorology	5.3×10^{-4} sec/m^3
0-8-hour LPZ meteorology	2.7×10^{-5} sec/m^3

Table 15.2 Locked rotor accident dose calculations

	EAB		LPZ	
Case	Thyroid	(whole body)	Thyroid	(whole body)
1: 0-2 hr	48	0.2	2.5	0
0-8 hr	-	-	-	-
Total	48	0.2	2.5	0
2: 0-2 hr	10	0.2	0.5	0
2-8 hr	-	-	6.4	0
Total	10	0.2	6.9	0

17 QUALITY ASSURANCE

17.6 Independent Design Verification

17.6.4 Staff Assessment

17.6.4.3 Hazard Analysis Program

In SSER 4, the staff stated that Stone & Webster Engineering Corporation (SWEC) had reviewed the applicant's "Hazards Review Program Summary," NERM-69, Revision 0 (September 17, 1985) and determined that when the program is successfully implemented, it will correct conditions identified in the action items and audit observations discussed in SSER 4. The summary report identified the criteria and procedures to be followed to document the status of the hazards program, including high-energy line breaks (HELBs) and moderate-energy line breaks (MELBs) (pipe whip, jet impingement, spray wetting, and flooding) and internal missile postulation.

Since the hazards program had not been completed at the time the Low-Power License was issued, License Condition 2.C(18) of the Low-Power License required the licensee to complete its hazards program and provide a schedule for making any required modifications before exceeding 5% power.

In a letter from J. F. Opeka (NNECO) to V. S. Noonan (NRC) dated January 23, 1986, the licensee stated that the hazards program was complete and the results showed that modifications were necessary.

On the basis of this information, the staff concludes that Low-Power License Condition 2.C(18) has been satisfied.

18 HUMAN FACTORS ENGINEERING

18.2 Safety Parameter Display System

In Supplement 1 to NUREG-0737, the staff identified five critical safety functions that would provide an overview of the safety status of the plant. These are

(1) reactivity control
(2) reactor core cooling and heat removal from the primary system
(3) reactor coolant system integrity
(4) radioactivity control
(5) containment conditions

The specific parameter or variables selected to be displayed to represent the critical safety functions are reviewed by the staff for adequacy and basis.

The licensee has chosen different critical safety functions to be consistent with the Westinghouse Owners Group Emergency Response Guidelines and provide displays of the critical safety function status trees used by the guidelines. These are

(1) subcriticality
(2) core cooling
(3) heat sink
(4) integrity
(5) containment
(6) reactor coolant system inventory
(7) radiation release

Upon review of these critical safety functions and the specific parameters selected to be displayed to represent the critical safety functions, the staff concluded that four additional parameters were required to adequately present an overview of the safety status of the plant. The staff stated in SSER 4 that:

> "The status of the following variables may not be available on the SPDS proposed for Millstone 3:
>
> RHR flow
> Containment isolation
> Containment hydrogen concentration
> Hot leg temperature
>
> "During the RHR and emergency core cooling system (ECCS) modes of cooling when steam generators are not available, the viability of the heat removal system should be monitored. The applicant has cited variables used to monitor the 'core cooling safety function' to address 'heat removal safety function' monitoring. In drawing a distinction between 'core cooling' and

'heat removal' safety functions, a more direct variable(s) should be considered to monitor the heat removal safety function. RHR flow may be sufficient to address both of the above cooling modes (RHR and ECCS); however, the applicant should further evaluate the 'heat removal' critical safety function and discuss the SPDS variable(s) that will assess its status during the RHR and ECCS modes of cooling.

"Containment isolation is an important parameter for use in making a rapid assessment of 'Containment Conditions.' In particular, a determination that known process pathways through containment have been secured provides a significant additional assurance of containment integrity. Containment hydrogen concentration is a key parameter used in the emergency guidelines to monitor combustible gas control and to indicate a compromise of the 'Containment Conditions' safety function.

"Hot leg temperature is a key indicator used in the ERGs (ES-0.1, Attachment A, Generic Instrumentation, Revision 1, Page 3) to determine the viability of natural circulation as a mode of heat removal. NUREG-0737, Supplement 1 gives 'RCS Temperature' as a proposed variable, but does not specify hot leg temperature.

"Opeka (May 1985) [letter from J. F. Opeka (NNECO) to B. J. Youngblood (NRC), dated May 24, 1985] indicates that, although RHR flow, containment hydrogen concentration, and hot leg temperature can be displayed at the SPDS console through various plant computer monitoring systems, the above four variables are not considered necessary to the SPDS by the applicant, and as such are not included in the Millstone SPDS.

"The above variables do, for given scenarios, provide unique inputs to determinations of status for their respective CSFs [critical safety functions], which have not been discussed by the applicant as being satisfied by other variables in the proposed Millstone SPDS list. The applicant should address these variables and their functions by: (1) adding these variables to the Millstone SPDS, (2) providing alternative added variables along with justifications that these alternates accomplish the same safety functions for all scenarios, (3) providing justification that variables currently on the Millstone SPDS do in fact accomplish the same safety functions for all scenarios, or (4) identifying that these variables are in fact available from the SPDS variables." [(5) The licensee can also provide justification acceptable to the staff for not including these variables.]

The licensee has provided the following responses to the SER. The staff's position regarding these responses is also included.

- RHR Flow

For Millstone 3, the RHR system is not used for containment recirculation; therefore, it is not an appropriate variable.

The staff agrees with the licensee's response. However, the licensee should propose an alternate variable for monitoring heat removal during this ECCS mode.

· Containment Isolation

The licensee's position is that containment isolation status is not a symptom-oriented variable but is rather a system status, and that the adequacy of containment isolation can be monitored by measuring radiation inside and outside containment. In addition, Millstone 3 operators can easily observe cortainment isolation status by looking at the engineered safety feature (ESF) status panel directly across from the primary SPDS station.

The staff's response is that Supplement 1 to NUREG-0737 calls for the SPDS to display sufficient variables to monitor several critical safety functions that reflect plant safety status including containment conditions. Supplement 1 does not specify that these variables should be symptom-oriented variables. Nor does Supplement 1 forbid the use of system status variables. In the staff's judgment, an adequate assessment of containment conditions must include information regarding the isolation status of containment, in addition to radiation measures inside and outside of containment, because of the function of containment; that is, because the function of containment is to provide a barrier, the status of known process pathways through the containment must be known to assess containment conditions and their possible effect on plant safety status.

The staff would find the current containment isolation (CI) display (on the ESF panel) acceptable on these conditions: (1) it should be defined as part of the SPDS system, (2) a commitment should be made to always retain the relative physical location of the primary SPDS and the CI display or to otherwise ensure that the CI display always remains easily viewable from the primary SPDS station, and (3) a commitment should be made to confirm that the CI display is pattern recognizable and that pattern-recognition aspects of the display will be retained in the future if addition or deletion of display tiles is necessary.

· Containment Hydrogen Concentration

This variable is not monitored until recombiners are turned on in containment and, therefore, does not lend itself to input to one of the critical safety functions on the primary SPDS display.

The staff's position is that containment hydrogen concentration should be added to the containment critical safety function using available sensors when they are in operation (turned on) or any new sensors that may be required in resolution of other regulatory issues. It would also be acceptable to add a separate indicator on the primary SPDS display alerting the operator to various conditions of hydrogen concentration in containment.

· Hot-Leg Temperature

This parameter has been added to the inadequate core cooling (ICC) display. However, it does not feed into any of the critical safety functions (as defined at Millstone 3) that make up the primary SPDS displays.

The staff's position is that this is not satisfactory and that there should be an indicator on the primary SPDS display alerting the operator to abnormal hot-leg temperature readings.

Summary

Unresolved items identified in SSER 4, have not yet been resolved and therefore will remain as a condition in the full-power license.

APPENDIX A

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

November 12, 1985	Letter from licensee concerning training program for shift advisors.
November 25, 1985	Letter to licensee issuing NPF-44 for 5% power and testing for Millstone Unit 3. Transmittal included Facility Operating License N.NPF-44, Appendix A and B (Technical Specifications and Environmental Protection Plan, respectively), Amendment No. 17 to Indemnity Agreement No. B-39, and Assessment of the Effects of License Duration on Matters Discussed in the FES.
December 3, 1985	Letter to Westinghouse withholding from public disclosure "Justification for the Use of the W-3 Correlation for the Millstone Unit 3 Steamline Break Analysis," CAW-85-71.
December 3, 1985	Letter from licensee concerning Fire Protection Audit--CO_2 Systems Test Results.
December 6, 1985	Letter to licensee transmitting two copies of Supplement No. 4 to the SER (SSER 4).
December 17, 1985	Letter to licensee transmitting 20 bound copies of SSER 4.
December 17, 1985	Letter to licensee transmitting FSAR Amendment 17.
December 18, 1985	Letter to licensee concerning 10 CFR 50.54(f) on station blackout.
December 20, 1985	Letter from licensee concerning evaluation of environmental effects of main steamline break outside containment.
December 23, 1985	Letter from licensee concerning reactor coolant pump rotor seizure event.
December 23, 1985	Letter from licensee submitting request for relief from preservice inspection.
December 23, 1985	Letter to licensee transmitting corrected page for Millstone Unit 3 Technical Specifications originally issued November 25, 1985.
December 24, 1985	Letter to licensee concerning Environmental Effects of Main Steamline Break and Seismic Interaction Program.

December 26, 1985	Letter from licensee responding to information request regarding station blackout.
January 7, 1986	Letter from licensee concerning evaluation of environmental effects of main steamline break outside containment.
January 10, 1986	Letter from licensee concerning training evaluation and certification of shift advisors.
January 13, 1986	Letter from licensee concerning applicability of Westinghouse LOFTRAN methodology to Model F steam generator.
January 13, 1986	Letter to licensee requesting information concerning station blackout for Millstone 3.
January 14, 1986	Letter from licensee concerning Seismic Interaction Program at Millstone 3.
January 14, 1986	Letter from licensee concerning evaluation of environmental effects of main steamline break outside containment.
January 14, 1986	Letter from licensee concerning Procedures Generation Package.
January 17, 1986	Letter from licensee concerning evaluation of environmental effects of main steamline break outside containment.
January 20, 1986	Letter from licensee concerning installation, testing, and calibration of inadequate core cooling system.
January 23, 1986	Letter from licensee concerning Hazards Review Program.

APPENDIX B

REFERENCES

Combustion Engineering Owners Group (CEOG), letter to D. M. Crutchfield (NRC), June 1, 1982.

Muller, D. R. (NRC), Memorandum to D. M. Crutchfield, October 22, 1985.

Northeast Nuclear Energy Company (NNECO), NERM-69 (Rev. 0), "Hazards Review Program Summary," September 17, 1985.

U.S. Nuclear Regulatory Commission (NRC), Generic Letter 84-16, "Adequacy of On-Shift Operating Experience for Near-Term Operating License Applicants," June 27, 1984.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980; Supplement 1, January 1983.

---, NUREG-0800 (Rev. 1), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981.

---, NUREG-1021 (Rev. 1), "Operator Licensing Examination Standards," October 1, 1984.

---, NUREG/CR-2136, "Effect of Postulated Event Devices on Normal Operation of Piping Systems in Nuclear Power Plants," May 1981.

---, NUREG/CR-2627, "Inadequate Core Cooling Instrumentation Using Heated Junction Thermocouples for Reactor Vessel Level Measurement," March 1982.

Westinghouse Corp., WCAP-8822-P-S2, "Mass and Energy Releases Following a Steamline Rupture," September 1976.

---, WCAP-10698, "Steam Generator Tube Rupture Analysis Methodology To Determine the Margin to Steam Generator Overfill," December 1984.

---, WCAP-10698 (Supplement 1), "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," May 24, 1985.

---, WCAP-10961-P, "Steamline Break Mass/Energy Releases for Equipment Environmental Qualification Outside Containment," October 1985.

APPENDIX D

ABBREVIATIONS

ACRS Advisory Committee on Reactor Safeguards
AIB arbitrary intermediate break
ASME American Society of Mechanical Engineers

BTP Branch Technical Position

CET core exit thermocouple
CFR Code of Federal Regulations
CI containment isolation
CRV Code-required volume
CSF critical safety function

DEPSG double-ended pump suction guillotine

EAB exclusion area boundary
ECCS emergency core cooling system
EOP emergency operating procedure
ERG emergency response guideline
ESF engineered safety feature

FSAR Final Safety Analysis Report
FW feedwater

GDC General Design Criterion (a)

HELB high-energy line break
HJTC heated junction thermocouple

ICC inadequate core cooling
ICCI inadequate core cooling instrumentation
IGSCC intergranular stress corrosion cracking
IR inner radius
ISI inservice inspection
ITS inventory tracking system

LPZ low-population zone

MELB moderate-energy line break
MFW main feedwater
MSLB main steamline break

NNECO Northeast Nuclear Energy Company
NRC U.S. Nuclear Regulatory Commission
NU Northeast Utilities

OBE operating basis earthquake

PSI	preservice inspection
PWR	pressurized-water reactor
QA	quality assurance
RCS	reactor coolant system
RHR	residual heat removal
RTD	resistance temperature device
RVLIS	reactor vessel liquid inventory system
RVLMS	reactor vessel level monitoring system
SER	Safety Evaluation Report
SG	steam generator
SGTR	steam generator tube rupture
SMM	subcooling margin monitor
SPDS	safety parameter display system
SRP	Standard Review Plan
SSER	supplement to Safety Evaluation Report
SWEC	Stone & Webster Engineering Corporation
WOG	Westinghouse Owners Group
WRV	weld required volume

APPENDIX F

NRC STAFF CONTRIBUTORS

This supplemental safety evaluation report is a product of the NRC staff. The NRC staff members listed below were principal contributors to this report.

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H. Brammer	Senior Mechanical Engineer	Engineering (PWR-A)
J. Buzy	Senior Reactor Engineer	Maintenance and Training
R. Giardina	Mechanical Engineer	Plant Systems (PWR-A)
R. Goel	Mechanical Engineer	Plant Systems (PWR-A)
C. Li	Mechanical Engineer	Plant Systems (PWR-A)
R. Lobel	Section Leader	Reactor Systems (PWR-A)
F. Orr	Reactor Systems Engineer	Facilities Operation (PWR-A)
R. Palla	Mechanical Engineer	Regulatory Improvements
D. Terao	Mechanical Engineer	Engineering (PWR-B)
H. Walker	Mechanical Engineer	Electrical Instrumentation, and Control Systems (PWR-A)
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* Reflects reorganization since Supplement 4 was issued.

APPENDIX N

PRESERVICE INSPECTION RELIEF REQUEST EVALUATION

INTRODUCTION

For nuclear power facilities whose construction permit was issued on or after July 1, 1974, 10 CFR 50.55a(g)(3) specifies that components shall meet the preservice examination requirements set forth in editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code applied to the construction of the particular component. The provisions of 10 CFR 50.55a(g)(3) also state that components (including supports) may meet the requirements set forth in subsequent editions and addenda of this Code which are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein.

In a submittal dated December 23, 1985, the licensee requested relief from ASME Code Section XI requirements which the licensee has determined to be impractical and provided supporting information pursuant to 10 CFR 50.55a(a)(3). Therefore, the staff evaluation consisted of reviewing the licensee's submittal to the requirements of the applicable Code edition and addenda and determining if relief from the Code requirements was justified.

TECHNICAL REVIEW CONSIDERATIONS

The construction permit for Millstone 3 was issued on August 9, 1974. In accordance with 10 CFR 50.55a(g)(3), components (including supports) which are classified as ASME Code Class 1 and 2 have been designed and provided with access to enable the performance of required preservice examinations.

Verification of as-built structural integrity of the primary pressure boundary is not dependent on the Section XI preservice examination. The applicable construction codes to which the primary pressure boundary was fabricated contain examination and testing requirements which by themselves provide the necessary assurance that the pressure boundary components are capable of performing safely under all operating conditions reviewed in the FSAR and described in the plant design specification. As a part of these examinations, all of the pressure boundary full penetration welds were volumetrically examined (radiographed) and the system was subjected to hydrostatic pressure tests.

The intent of a preservice examination is to establish a baseline reference before the initial operation of the facility. The results of subsequent inservice examination can then be compared with the original condition to determine whether changes have occurred. If the inservice inspection results show no change from the original condition, no action is required. Should no baseline data be available, all flaws must be treated as new flaws and evaluated accordingly. Section XI of the ASME Code contains acceptance standards that may be used as the basis for evaluating the acceptability of such flaws.

Other benefits of the preservice examination include providing redundant or alternative volumetric examination of the primary pressure boundary using a test method different from that employed during the component fabrication. Successful performance of the preservice examination also demonstrates that the welds so examined are capable of being effectively inspected during the subsequent inservice examinations using a similar test method.

In the case of Millstone 3, a large portion of the preservice examination required by the ASME Code was performed. Failure to perform a 100% preservice examination of the welds identified below will not significantly affect the assurance of the initial structural integrity.

In some instances where the required preservice examinations were not performed to the full extent specified by the applicable ASME Code, the staff may require that these examinations or supplemental examinations be conducted as part of the inservice inspection program. The performance of supplemental examinations, such as surface examinations, in areas where volumetric examination is difficult will be more meaningful after a period of operation. Acceptable preoperational integrity has already been established by similar ASME Code Section III fabrication examinations.

Several of the preservice inspection relief requests involve limitations to the examination of the required volume of a specific weld. The inservice inspection (ISI) program is based on the examination of a representative sample of welds to detect generic service-induced degradation. In the event that the welds identified in the preservice inspection (PSI) relief requests are required to be examined again, the possibility of augmented inservice inspection will be evaluated during review of the licensee's initial 10-year ISI program. An augmented program may include increasing the extent and/or frequency of examination of accessible welds.

EVALUATION OF RELIEF REQUESTS

The licensee requested relief from specific preservice inspection requirements in a submittal dated December 23, 1985. On the basis of the information submitted by the licensee and the staff's review of the design, geometry, and materials of construction of the components, certain preservice inspection requirements of the ASME Boiler and Pressure Vessel Code, Section XI have been determined to be impractical to perform. The licensee has demonstrated that either (1) the proposed alternative would provide an acceptable level of quality and safety or (2) compliance with the specified requirements of this section would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3), conclusions that these preservice requirements are impractical are justified as follows. Citations of the Code refer to the ASME Code, Section XI, 1980 Edition including Addenda through Winter 1980.

(A) Relief Request PR-1, Examination Categories B-B and B-D, Pressure-Retaining Welds in the Reactor Vessel

For Millstone 3 a volumetric (ultrasonic) examination of essentially 100% of the weld length shall be conducted for the following items in accordance with the ASME Code, Article IWB-2500:

Item	Description
B.1.12	Longitudinal shell welds
B.1.21	Head circumferential welds
B.1.22	Lower head meridional welds
B.1.30	Shell-to-flange weld
B.3.90	Nozzle-to-shell welds
B.3.100	Nozzle inner radius areas

Pursuant to 10 CFR 50.55(a)(g)5(iii), relief is requested from performing the preservice volumetric examination of the inaccessible portions of the subject vessel welds.

Because geometric configuration and permanent obstructions affected the matrix of welds (see December 23, 1985, submittal), a 100% volumetric examination was not possible. The PSI limitations and the specific relief as they apply to each weld, are noted in the submittal.

The licensee offered the following support for relief/alternative examinations proposed:

(1) The subject welds received both volumetric examination by radiography and surface examinations during fabrication, in accordance with ASME Code Section III requirements which provide adequate assurance of the structural integrity of the welds.

(2) A preservice hydrostatic test was conducted successfully on the class/pressure boundary of which these welds are a part thereof (IWB-2500-1).

(3) Inservice system leakage tests will be performed per Category B-P, IWB-2500-1.

Staff Evaluation

This relief request is acceptable based on the following considerations:

(1) All of the reactor pressure vessel welds passed volumetric examinations during fabrication in accordance with the rules of ASME Code Section III for Class 1 components.

(2) All of the identified welds will be subject to a system pressure test in accordance with Section XI Class 1 requirements.

(3) Accessible portions of the above-listed welds received a preservice volumetric examination in accordance with the ASME Code Section XI.

(4) The limited Section XI ultrasonic examination, the radiography performed during fabrication, and the hydrostatic test provide an acceptable level of preservice structural integrity.

(B) Relief Request PR-2, Examination Category B-A, Pressure-Retaining Weld in Reactor Vessel Closure Head

For Millstone 3, a volumetric examination of essentially 100% of the weld length shall be conducted for the following items in accordance with the ASME Code, Article IWB-2500:

Item	Description
B1.21	Circumferential head weld
B1.22	Meridional head weld
B1.40	Head-to-flange weld

Pursuant to 10 CFR 50.55(a)(g)5(iii), relief is requested from performing the preservice volumetric examination of the inaccessible portions of the subject vessel welds.

Geometric configuration and permanent obstructions limited the volumetric examination of the following three welds. Examination data sheets and limitation sketches provided in the December 23, 1985, submittal depict the affected areas. Relief is therefore requested from complying with the 100% WRV (weld required volume) coverage of these welds.

(1) Weld No. 101-101, Head-to-Flange Weld

Access to this weld is limited to essentially one side only because of the forged flange configuration. Additional limitations from the top side of the weld are due to permanently attached head lifting lugs. Required volume not examinable, ~38%.

(2) Weld No. 103-101, Circumferential Head Weld

Permanently attached head lifting lugs prevented volumetric examination of ~7% of the WRV.

(3) Weld No. 101-104D, Meridional Head Weld

A 2.7-in.-diameter repair area (surface concavity) on the weld centerline prohibited sufficient coverage of the WRV in that area. Required volume not examinable, ~2%.

The licensee offered the following support for relief/alternative examinations:

(1) The subject welds received both volumetric examination by radiography and surface examinations during fabrication in accordance with ASME Code Section III requirements which provide adequate assurance of the structural integrity of the welds.

(2) A preservice hydrostatic test was conducted successfully on the Class 1 pressure boundary of which these welds are a part (IWB-2500-1).

(3) In-service system leakage tests will be performed per Category B-P, IWB-2500-1.

Staff Evaluation

This relief request is acceptable based on the following considerations:

(1) All of the reactor pressure vessel welds passed volumetric examinations during fabrication in accordance with the rules of ASME Code Section III for Class 1 components.

(2) All of the identified welds will be subject to a system pressure test in accordance with Section XI Class 1 requirements.

(3) Accessible portions of the above-listed welds received a preservice volumetric examination in accordance with the ASME Code Section XI.

(4) The limited Section XI ultrasonic examination, the radiography performed during fabrication, and the hydrostatic test provide an acceptable level of preservice structural integrity.

(C) Relief Request PR-3, Examination Category B-B, Pressure-Retaining Welds in Steam Generators

For Millstone 3, a volumetric examination of essentially 100% of the weld length shall be conducted for the following items in accordance with the ASME Code, Article IWB-2500:

Item	Description
B2.11	Circumferential shell-to-head welds (pressurizer)
B2.12	Longitudinal shell welds (pressurizer)
B2.40	Tubesheet-to-head welds (steam generators)

Pursuant to 10 CFR 50.55(a)(g)5(iii), relief is requested from performing the preservice volumetric examination of the inaccessible portions of the subject vessel welds.

Geometric configuration and permanent obstructions limited the volumetric examination of the following listed welds. Examination data sheets and limitation sketches depict the affected areas. Relief is therefore requested on complying with the 100% WRV coverage of the welds.

Pressurizer

(1) 03-007-SW-J, Shell-to-Upper-Head Weld

Permanently installed insulation support ring obstructed part of the required scanning area.

Required volume not examinable, ~9%.

(2) 03-007-SW-F, Shell-to-Lower-Head Weld

Permanent obstructions (alignment target pads and instrumentation lines) and the geometric configuration--weld transition between plate thickness variations prohibited complete coverage.

Required volume not examinable, ~30%.

(3) 03-007-SW-A, Longitudinal Seam Weld

Permanently installed insulation support ring obstructed part of the required scanning area.

Required volume not examinable, ~5%.

Steam Generators

03-003-SW-Z	S/G A
03-004-SW-Z	S/G B
03-005-SW-Z	S/G C
03-006-SW-Z	S/G D

Permanent obstructions--permanent I-beam support columns for each generator restricted scans as shown in the December 23, 1985, submittal.

Required volume not examinable, ~30%.

The licensee offered the following support for relief/alternative examinations:

(1) The subject welds received both volumetric examination by radiography and surface examinations during fabrication in accordance with ASME Code Section III requirements which provide adequate assurance of the structural integrity of the welds.

(2) A preservice hydrostatic test was conducted successfully on the Class I pressure boundary of which these welds are part thereof (IWB-2500-1).

(3) Inservice system leakage tests will be performed per Category B-P, IWB-2500-1.

Staff Evaluation

This relief is acceptable for PSI based on the following considerations:

(1) The subject welds received both volumetric (radiography) and surface examinations during fabrication in accordance with ASME Code Section III requirements.

(2) The required volumes not examinable of welds 03-007-SW-J and 03-007-SW-A represent a relatively small percentage of the total.

(3) The examinations performed demonstrate an acceptable level of preservice structural integrity.

(D) Relief Request PR-4, Examination Categories B-L-2 and B-M-2, Internal Surfaces of Pump Casings and Valve Bodies

For Millstone 3, a visual examination (VT-3) of the internal surfaces shall be conducted for the following items in accordance with the ASME Code, Article IWB-2500:

Item	Description
B12.40	Valve bodies exceeding 4 in. (nominal pipe size)
B12.20	Pump casings

Note: Examinations are limited to:

- One valve within each group of valves that are of the same design, manufacturing method, and are performing similar functions in the system.

- One pump per group performing similar functions.

Pursuant to 10 CFR 50.55(a)(g)5(iii), relief is requested from performing a preservice visual examination (VT-3).

The licensee offered the following support for relief/alternative examinations:

For the reactor coolant pump casings and 35 valve bodies (see December 23, 1985, submittal) in the reactor coolant, pressurizer, safety injection, and residual heat removal systems, relief is requested from disassembly of an operable valve or pump for performing a preservice visual examination (VT-3).

The requirement to disassemble an operable valve or pump for the sole purpose of performing a visual examination (VT-3) of the internal pressure-retaining boundary is impractical and not commensurate with the increased safety achieved by this inspection. Class 1 valves and pumps are installed in their respective systems, and many have completed functional testing. To disassemble these items would provide a very small potential for increasing plant safety margins with a very disproportionate impact on expenditures of plant manpower and resources.

The manufacturer's test data will be used in lieu of a preservice visual examination (VT-3). This includes documentation of examinations performed during fabrication and installation of the subject valves. The examinations performed may include volumetric, surface, and visual examinations, as required by ASME Code Section II, "Material Specifications for Ferrous and Nonferrous Material," and ASME Code Section III, "Construction and Installation Requirements."

Class 1 valves and pumps are subjected to numerous types of nondestructive testing and a rigorous quality assurance program during all stages of fabrication, storage, and installation. These valves and pumps have been found acceptable by the manufacturer, the ASME Authorized Nuclear Inspector, and Northeast Utilities' Quality Assurance group. During maintenance of Class 1 pumps, a visual examination (VT-3) will be performed.

Staff Evaluation

The purpose of the VT-3 inspection of the internal surfaces of pumps and valves is to determine if severe degradation of the materials is occurring or has occurred over a period of time. It is unlikely that the valves and pumps in Millstone 3 have experienced material degradation, considering the materials from which the components were fabricated and the inservice performance of similar components in other facilities. To disassemble the valves and pumps at this time solely to perform the required Section XI preservice visual examination of the internal surfaces is impractical. The staff has determined that the nondestructive examinations and functional tests performed to date significantly exceed the requirements of the Section XI visual examination and, therefore, these examinations and tests are an acceptable alternative to the Code requirement.

(E) Relief Request PR-5, Examination Categories B-D and C-B, Nozzle Inner Radius Sections of Steam Generators and Pressurizer

For Millstone 3, a volumetric (ultrasonic) examination of nozzle inner radius (IR) sections for the steam generator and pressurizer shall be conducted in accordance with the ASME Code, Articles IWB-2500 and IWC-2500, for Code Classes 1 and 2, respectively.

Pursuant to 10 CFR 50.55(a)(g)5(iii), relief is requested from performing the preservice volumetric examination of the nozzle inner radius sections of the steam generator and pressurizer as listed below:

Class	Component	Nozzle (IR)
1	Pressurizer	Surge 03-007-SW-S(IR) Spray 03-007-SW-E(IR)
1	Pressurizer	Relief 03-007-SW-D(IR)
1	Pressurizer	Safety 03-007-SW-A(IR)
1	Pressurizer	Safety 03-007-SW-B(IR)
1	Pressurizer	Safety 03-007-SW-C(IR)
1	Steam generator	Primary inlet 03-003-IR Primary inlet 03-004-IR Primary inlet 03-005-IR Primary inlet 03-006-IR
1	Steam generator	Primary outlet 03-003-IR Primary outlet 03-004-IR Primary outlet 03-005-IR Primary outlet 03-006-IR
2	Steam generator	Feedwater 03-053-SW-R-IR Feedwater 03-054-SW-R-IR Feedwater 03-055-SW-R-IR Feedwater 03-056-SW-R-IR
2	Steam generator	Steam outlet 03-053-SW-T-IR Steam outlet 03-054-SW-T-IR Steam outlet 03-055-SW-T-IR Steam outlet 03-056-SW-T-IR

Reasons for the licensee's request are:

(1) Currently available ultrasonic examination techniques do not provide results which yield a meaningful baseline for comparison with subsequent examinations. This is a result of complex geometrical configuration, long metal paths, limited accessible scan areas, and cladding on some of the nozzles. (Refer to the drawings of the applicable nozzles depicting the complex geometries in the December 23, 1985, submittal.)

(2) The areas involved have received extensive surface examination during fabrication and reveal no indications of cracking. In addition, the general area has been interrogated with ultrasonics during PSI nozzle-to-vessel weld examinations.

(3) Cracking in the inner radius areas is generally attributed to thermal cycling when it has occurred. Hence, for PSI, there is little or no safety impact as a result of not performing these examinations.

The licensee proposed the following alternative examinations:

(1) Before operation, the licensee will assemble a file of as-built information for each of the steam generator and pressurizer nozzles to support application of new examination techniques during ISI. This information will include factors which could impact a sound beam directed at this area from the OD.

(2) Follow industry progress with technique development and work toward the performance of a meaningful ultrasonic examination during ISI.

Staff Evaluation

This relief request is acceptable for PSI based on the following considerations:

(1) All pressure-retaining components were hydrostatically tested to the requirements of ASME Code Section III before plant startup.

(2) The staff review of the design configuration of the nozzle inner radius has concluded that the Code-required volumetric examination is impractical. The staff has determined that performing the ASME Section III hydrostatic test along with the surface examination is an acceptable alternative.

(3) The staff will continue to evaluate the development of new or improved procedures and will require that these procedures be made part of the ISI examination requirements.

(F) Relief Request PR-7, Examination Category C-B, Nozzle-to-Shell Welds and Nozzle Inside Radius Sections of the Residual Heat Removal Heat Exchangers

For Millstone 3, volumetric and surface examination of the nozzle-to-shell weld (C2.21) and volumetric examination of the nozzle inside radius section (C2.22) for the residual heat removal heat exchangers shall be conducted in accordance with the ASME Code, Article IWC-2500.

Pursuant to 10 CFR 50.55(a)(g)5(iii), relief is requested from performing the preservice volumetric examination of the nozzle inner radius sections and surface and volumetric examinations on the nozzle-to-shell welds of the residual heat removal heat exchangers.

The licensee gave the following reasons for requesting relief:

(1) The residual heat removal heat exchanger nozzle-to-shell welds and nozzle inside radius areas are totally inaccessible to RT, UT, and surface techniques because permanently welded reinforcement plates have been placed over the welds. (See Figure IWC-2500-4(c) from the 1983 Edition of Section XI which accurately depicts the construction.)

(2) The 1983 Edition of Section XI has recognized and taken action to resolve the need for relief as this is a generic problem with many heat exchanger designs. Since the NRC has accepted the 1983 Edition for use, the licensee is utilizing this to support its request for relief.

The licensee proposed the following alternative examinations:

(1) The subject welds received both volumetric examination by radiography and surface examinations during fabrication in accordance with ASME Code Section III requirements which provide adequate assurance of the structural integrity of the welds.

(2) A preservice hydrostatic test was conducted successfully on the Class 2 pressure boundary of which these welds are a part thereof (IWC-2500-1).

(3) Inservice system leakage tests will be performed per Category C-H, IWC-2500-1.

Staff Evaluation

This relief request is acceptable for PSI based on the following considerations:

(1) Although the 1983 Edition of Section XI has not been accepted for use by the NRC at this time as stated by the licensee, the staff is in agreement with the examination requirements of the 1983 Edition for welds covered by reinforcement plates.

(2) The staff's review of the design configuration of the nozzle inner radius has concluded that the Code-required volumetric examination is impractical. The staff has determined that the ASME Code Section III examinations demonstrate an acceptable level of preservice structural integrity.

(3) The subject weld area received radiographic examination and a hydrostatic test during fabrication in accordance with ASME Code Section III requirements. An ultrasonic examination has been performed on the nozzle-to-vessel welds per ASME Code Section XI requirements.

(G) Relief Request PR-8, Examination Category B-D, Nozzle-to-Vessel Welds in Steam Generators and Pressurizer

For Millstone 3, volumetric (ultrasonic) examination of 100% of the full penetration nozzle-to-vessel welds for the steam generator and pressurizer shall be conducted in accordance with the ASME Code, Article IWB-2500 for Code Class 1.

Pursuant to 10 CFR 50.55(a)(g)5(iii), relief is requested from performing the preservice volumetric examination on the inaccessible portions of the nozzle-to-shell welds of the steam generator and pressurizer.

The licensee requested relief because:

(1) Geometric configuration of the nozzle-to-shell welds listed below and their close proximity to one another limits the volume that can be examined.

(2) Scanning is limited to one side only with a ½-V technique. Restriction on axial scan is due to the close proximity of the welds to each other.

Pressurizer

03-007-SW-A	03-007-SW-C	03-007-SW-E (spray nozzle)
03-007-SW-B	03-007-SW-D	03-007-SW-S (surge nozzle)

Required volume not examinable, ~80%.

Steam Generators

Coverage is from both sides of weld with ½-V technique. Restriction on axial scan is due to the steam generator supports integral extensions.

03-003-SW-V inlet	03-004-SW-II outlet	03-006-SW-V inlet
03-003-SW-II outlet	03-005-SW-V inlet	03-006-SW-II outlet
03-004-SW-V inlet	03-005-SW-II outlet	

Required volume not examinable, ~10%.

The licensee supported its request for relief/alternative examinations as follows:

(1) The subject welds received both volumetric examination by radiography and surface examinations during fabrication in accordance with ASME Code Section III requirements which provide adequate assurance of the structural integrity of the welds.

(2) A preservice hydrostatic test was conducted successfully on the Class 1 pressure boundary of which these welds are a part thereof (IWB-2500-1).

(3) Inservice system leakage tests will be performed per Category B-P, IWB-2500-1.

Staff Evaluation

This relief request is acceptable for PSI based on the following considerations:

(1) All pressure-retaining components were hydrostatically tested to the requirements of ASME Code Section III before plant startup.

(2) The staff's review of the design configuration of the nozzle inner radius has concluded that the Code-required volumetric examination is impractical. The staff has determined that performing the ASME Code Section III hydrostatic test along with the surface examination is an acceptable alternative.

(3) The staff will continue to evaluate the development of new or improved procedures and will require that these procedures be made part of the ISI examination requirements.

(H) Relief Request PR-10, Examination Category B-J, Pressure-Retaining Welds in Main Coolant Piping System

For Millstone 3, volumetric (ultrasonic) and surface examination of essentially 100% of the length of each weld in the main coolant piping system shall be conducted in accordance with the ASME Code, Article IWB-2500.

Pursuant to 10 CFR 50.55(a)(g)5(iii), relief is requested from performing the preservice volumetric examination of the inaccessible portions of the welds in Table 1 of the December 23, 1985, submittal for the following reasons:

(1) Geometric configuration, permanent obstructions, and structural interferences prohibit 100% volumetric exam coverage of the Code-required examination volume. Relief is therefore requested from performing preservice examination on the inaccessible portions of the volume required as noted in Table 1.

(2) It should be noted that the Westinghouse-developed UT technology for CCSS piping was utilized in performing all examinations on the main coolant piping. Refer to J. F. Opeka's letter to B. J. Youngblood, B11576, dated May 7, 1985, for NUSCO's response to the staff question (SER Question 250.12) which addresses NRC concerns.

(3) A 0° longitudinal beam examination was conducted on all CCSS welds to map ID geometry contours. This was done in addition to Section XI requirements to aid in the performance and evaluation of angle beam examination results.

The licensee proposed the following alternative examinations:

(1) The subject welds received both volumetric examination by radiography and surface examinations during fabrication in accordance with ASME Code Section III requirements. Having met these requirements, adequate assurance of the structural integrity of the subject welds is provided.

(2) A preservice hydrostatic test was conducted successfully on the Class 1 pressure boundary, of which these welds are a part thereof, per IWB-2500-1 requirements.

(3) Inservice system leakage tests will be performed per Category B-P, IWB-2500-1, as well as surface and volumetric exams as required by Section XI, "Selection Criteria." Any advances in UT technology will be evaluated to determine its application for achieving maximum volume coverage and results.

Staff Evaluation

The staff has reviewed the December 23, 1985, submittal including the tables identifying the welds for which relief is being requested. These tables list the weld number, configuration (pipe-to-elbow, pipe-to-nozzle, pipe-to-pump casing, etc.), material type, and the licensee's estimate of the percentage of the Code-required examination that was completed.

Complete examinations meeting the requirements of ASME Code Section XI were performed on welds of similar configurations which utilized the same weld techniques, procedures, and materials. The inspected welds will be subject to the same operating and environmental conditions as the partially inspected welds or the uninspected weld. Therefore, the acceptable preservice examination results of the inspectable welds provide reasonable assurance by sampling of the structural integrity of the subject welds.

On this basis, the staff has concluded that the limited Section XI volumetric examination and the Section III fabrication examinations provide an acceptance level of preservice structural integrity and that compliance with the specific requirements of Section XI would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

(I) Relief Request PR-11, Examination Category B-J, Pressure-Retaining Welds in Piping

For Millstone 3, volumetric (ultrasonic) and surface examination of essentially 100% of the length of each Code Class 1 piping weld $\geq$ 4-in. nominal pipe size shall be conducted in accordance with the ASME Code, Section XI, Article IWB-2500.

Pursuant to 10 CFR 50.55(a)(g)5(iii), relief is requested from performing the preservice volumetric examination of the inaccessible portions of the welds listed in Table 1 of the December 23, 1985, submittal.

Permanent structural interferences prohibit 100% volumetric examination coverage of the Code-required volume (CRV). Relief is therefore requested from performing preservice examinations on the inaccessible portions of the volume required as noted in Table 1 of the December 23, 1985, submittal.

The licensee proposed the following alternative examinations:

(1) The subject welds received both volumetric examination by radiography and surface examinations during fabrication in accordance with ASME Code Section III requirements. Having met these requirements, adequate assurance of the structural integrity of the subject welds is provided.

(2) A preservice hydrostatic test was conducted successfully on the Class 1 preservice boundary, of which welds are a part thereof, per IWB-2500-1 requirements.

(3) Inservice system leakage tests will be performed per Category B-P, IWB-2500-1, as well as surface and volumetric exams as required by Section XI selection criteria. Any advances in UT technology will be evaluated to determine its application for achieving maximum volume coverage and results.

<u>Staff Evaluation</u>

The staff has reviewed the December 23, 1985, submittal including the welds, configurations, limitations, and percent coverage of the required examinations. The examinations performed during fabrication to Section III requirements, the preservice hydrostatic test, and the percentage of each weld examined during preservice examination provide adequate bases for acceptance of the welds' structural integrity.

(J) <u>Relief Request PR-12, Examination Category C-F, Pressure-Retaining Welds in Piping</u>

For Millstone 3, volumetric (ultrasonic) and surface examination of essentially 100% of the length of each weld requiring examination (> 4-in. nominal pipe size, > ½-in. thickness) shall be conducted in accordance with the ASME Code, Article IWC-2000.

Pursuant to 10 CFR 50.55(a)(g)5(iii), relief is requested from performing the preservice volumetric and/or surface examinations on the inaccessible portions of the welds listed in Table 1 of the December 23, 1985, submittal.

Geometric configuration, permanent obstructions, and/or structural interferences prohibit 100% examination coverage of the Code-required volume or area. Relief is therefore requested from performing preservice examinations on the inaccessible portions.

The licensee proposed the following alternative examinations:

(1) The subject welds received volumetric examination by radiography during fabrication in accordance with ASME Code Section III requirements. Having met these requirements, adequate assurance of the structural integrity of the subject welds is provided.

(2) A preservice hydrostatic test was conducted successfully on the Class 2 preservice boundary, of which welds are a part thereof, per IWC-2500-1 requirements.

(3) Inservice system leakage tests will be performed per Category C-H, IWC-2500-1 as well as surface and volumetric exams as required by Section XI selection criteria. Any advances in UT technology will be evaluated to determine its application for achieving maximum volume coverage and results.

Staff Evaluation

This relief request is acceptable for PSI based on the following considerations:

(1) The subject welds received volumetric examination by radiography during fabrication in accordance with ASME Code Section III requirements.

(2) The welds were subjected to a preservice hydrostatic test.

(3) The percentage of each of the required volumes of the welds to be examined by UT is relatively high, ranging from 74% to 98%.

CONCLUSIONS

On the basis of the foregoing, pursuant to 10 CFR 50.55a(a)(3), the staff has determined that certain Section XI-required preservice examinations are impractical. The licensee has demonstrated that either (1) the proposed alternatives would provide an acceptable level of quality and safety or (2) compliance with the requirements would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety.

The staff's technical evaluation has not identified any practical method by which the existing Millstone Nuclear Power Station, Unit 3 can meet all the specific preservice inspection requirements of Section XI of the ASME Code. Requiring exact compliance with all Section XI-required examinations would delay startup of the plant in order to redesign a significant number of plant systems, obtain sufficient replacement components, install the new components, and repeat the preservice examination of these components. Even after the redesign efforts, complete compliance with the preservice examination requirements probably could not be achieved. However, the as-built structural integrity of the existing primary pressure boundary has already been established by the construction code fabrication examinations.

On the basis of the staff's review and evaluation, it is concluded that the public interest is not served by imposing certain provisions of Section XI of the ASME Code that have been determined to be impractical. Pursuant to 10 CFR 50.55a(a)(3), relief is allowed from these requirements which are impractical to implement.

NRC FORM 335 (6 83) U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1 REPORT NUMBER (Assigned by TIDC, add Vol No., if any)
NUREG-1031 Supplement No. 5

2 Leave blank

3 TITLE AND SUBTITLE
SAFETY EVALUATION REPORT RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

4 RECIPIENT'S ACCESSION NUMBER

5 DATE REPORT COMPLETED
MONTH: JANUARY YEAR: 1986

6 AUTHOR(S)

7 DATE REPORT ISSUED
MONTH: JANUARY YEAR: 1986

8 PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Pressurized Water Reactor Licensing-A
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

9 PROJECT/TASK/WORK UNIT NUMBER

10 FIN NUMBER

11 SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Same As 8. above

12a TYPE OF REPORT
Technical

12b PERIOD COVERED (Inclusive dates)
November 1985 - January 1986

13 SUPPLEMENTARY NOTES
Docket No. 50-423

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. Supplement Nos. 2 and 3 dated September 1985 and November 1985, respectively, updated the information contained in the Safety Evaluation Report and Supplement No. 1 and addressed prior unresolved items. Supplement No. 4, issued in November 1985 addressed the items concerning the issuance of a low power license (5%). The low power license, NPF-44 was issued on November 25, 1985.

This Supplement No. 5 addresses the items relating to the issuance of a full power license (100%).

The facility is located in Waterford Township, New London County, Connecticut.

15a KEY WORDS AND DOCUMENT ANALYSIS

15b DESCRIPTORS

16 AVAILABILITY STATEMENT
UNLIMITED

17 SECURITY CLASSIFICATION (This report)
UNCLASSIFIED

18 NUMBER OF PAGES

19 SECURITY CLASSIFICATION (This page)
UNCLASSIFIED

20 PRICE
$

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

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