NORTHEAST UTILITIES

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

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 665-5000

November 18, 1985

Docket No. 50-423
B11880

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

Dear Mr. Youngblood:

Millstone Nuclear Power Station, Unit No. 3
Engineering Assurance Audit Open Items

Enclosed is the information requested by the NRC Staff in a November 12, 1985 telephone conversation concerning the Millstone Unit No. 3 Engineering Assurance (EA) audit items. Included in this package are:

- o An updated Corrective Action Status report.
- o FSAR changes resulting from:
  - A0 166, Item 1 (S-051)
  - A0 171, Item I.5 (H-019, H-023)
  - A0 171, Item I.7 (H-011)
- o A0 164, Item 7 (P-026), Leakage Detection
  - LCN #760
  - PCR #S-1017
  - PCR #S-1018

Another item, a status on the Hazards Program which identifies all interactions that have not been resolved, has been provided under separate cover.

B001
3/40

We trust that this information addresses your remaining concerns regarding the EA audits. If you have any questions regarding this information, please contact our licensing representative.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY
et. al.

BY NORTHEAST NUCLEAR ENERGY COMPANY
Their Agent

J. F. Opeka
J. F. Opeka
Senior Vice President

cc: Mr. E. V. Imbro
NRC Quality Assurance Branch

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

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

Lorraine J. D'Amico
Notary Public

My Commission Expires March 31, 1988

Millstone Nuclear Power Station, Unit No. 3
Engineering Assurance Audits
CORRECTIVE ACTION STATUS REPORT

References: (1) Engineering Assurance Technical Audit Report, Millstone Unit No. 3 Project, Audit No. 59, dated August 16, 1985, (including Errata dated August 22, 1985).

(2) Engineering Assurance Corrective Action Status Report, Millstone Unit No. 3 Project, dated November 1, 1985.

This provides an updated status of corrective actions resulting from the Stone & Webster Engineering Corporation (SWEC), Engineering Assurance (EA) Technical Audit of Millstone Unit No. 3 as reported in Reference (1).

Forty-nine (49) audit items were open at the time of issue of the previous EA Corrective Action Status Report (Reference (2)). The actions required to resolve these items have been completed and verified by the EA audit team. All items resulting from the EA Technical Audit are considered closed.

(S-51)

ATTACHMENT A

STONE & WEBSTER ENGINEERING CORPORATION
LICENSING DOCUMENT CHANGE REQUEST FORM

666

NAM 15-1, Rev 1

PROJECT Millstone 3	REQUEST NO.
CLIENT NUSCo JO. NO. 12179	REVISION

ORIGINATOR: see NUSCo cover sheet (TITLE) (STAFF GROUP/DISCIPLINE) DATE: 8/19/85

COGNIZANT LDE W J Briggs (NAME) see NUSCo cover sheet (SIGN & DATE)

LEAD LICENSING ENGINEER: WFEmerson / RG Joshi (NAME)

LICENSING DOCUMENT(S) AFFECTED:

FSAR Table 3.8-3

REASON(S) FOR PROPOSED CHANGE:

See NUSCo cover sheet

DESCRIPTION OF PROPOSED CHANGE (ABSTRACT):

See attached sheets

ATTACH CHANGE IN FINAL LICENSING DOCUMENT FORM, 6 PAGES ATTACHED

EVALUATION OF PROPOSED CHANGE - INITIAL UNDER CHANGE CATEGORY EVALUATION

(LIST)	MINOR	MAJOR	SUBSTANTIAL
LDE (S) M. Sinha	x MKSinha		
W. Emerson W.Emerson	WF		
LLE R Joshi			
PE			

ADDITIONAL COMMENTS (ATTACH ADDITIONAL SHEETS IF NECESSARY):

1. This change supercedes #619 dated 7/5 (NAM 15-1)
2. Comments resolved w/ SONVC & D. Robinson 10/29/85 WF

NORTHEAST UTILITIES

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

General Offices • Selden Street, Berlin, Connecticut

P.O. BOX 270
HARTFORD, CONNECTICUT 06141-0270
(203) 666-6911

Licensing Document C.R. No. NAM 15-1 rev 1.

Millstone Nuclear Power Station, Unit No. 3
FSAR and EROLS CHANGE REQUEST

The following page(s) of Table 3.8-3, 1 thru 4 of the FSAR/~~EROLS~~ (circle one) are attached showing the proposed change(s) written in black pen; initialed; and dated.

*Reason for Proposed Change(s):

See attached sheets which are to provide SRP loading conditions in addition to or instead of original loading conditions.

Originator: [signature]

Date: 8/14/85

Originator's Supervisor: Walter J. Bragg

Date: 8/15/85

Licensing Lead: Carol J. Shaffer

Date: 8/15/85

Additional Review: (if necessary)

Date:

*Note: Proposed changes to the FSAR/EROLS text must be written in a manner such that they constitute a revision to the text and read as you would want them to appear. Comments and/or questions are not acceptable. Proposed changes should be thoroughly researched prior to submitting them to Licensing to ensure their validity. If you have questions, please contact Licensing.

TABLE 3.8-3

LOADS AND LOADING COMBINATIONS

1. Concrete Structures (Containment Internal Structures and Category I Structures, other than the containment mat, shell, and dome).

Loads and loading combinations are based on ACI 318, and AEC Enclosure 3 - Structural Design Criteria for Evaluating Effects of High Energy Pipe Breaks on Category I Structures Outside the Containment, Structural Engineering Branch, Directorate of Licensing.

1. U = 1.4D + 1.7L

2. U = 1.4D + 1.7L + 1.7H
2a. U = 0.9D + 1.7H

3. U = 1.4D + 1.7L + 1.4F
3a. U = .9D + 1.4F

4. U = 0.75 (1.4D + 1.7L + 1.7W)
4a. U = 0.90D + 1.3W

5. U = 0.75 (1.4D + 1.7L + 1.7 x 1.1 (1/2 SSE)
5a. U = 0.90D + 1.3 x 1.1 (1/2 SSE)

6. U = 1.1 D + 1.1 L + 1.1 SSE
6a. U = 0.9D + 1.1 SSE

7. U = 1.1 D + 1.1 L + 1.0 Wt
7a. U = 0.9D + 1.0 Wt 11

8. U = D + L + Ta + Ra + 1.5Pa

9. U = D + L + Ta + Ra + 1.25Pa + 1.25 OBE + 1.0 (Yr + Yj + Ym)

10. U = D + L + Ta + Ra + Pa + SSE + 1.0 (Yr + Yj + Ym)

Notes - Concrete Structures

(1) U is the required section strength based on strength design methods described in ACI 318-71.

(2) In combinations 8, 9, and 10, the maximum values of Pa, Ta, Ra, Yj, Yr, and Ym, including an appropriate dynamic load factor, shall be used unless a time-history analysis is performed to justify otherwise.

(3) For load combinations 9 and 10, local section strengths and stresses may be exceeded under the concentrated loads Yr, Yj,

TABLE 3.8-3 (Cont)

and Ym, provided there will be no loss of function of any safety related system.

(4) For load combinations 7 and 7a, local section strengths and stresses may be exceeded under the tornado missile load provided there will be no loss of function of any safety related system.

INSERT (A)

2. Steel Structures

A. Elastic Working Stress Design Service Load Conditions

1 1. 1.0 S = D + L

2. 1.0 S = D + L + OBE

3. 1.33 S = D + L + W

Factored Load Conditions

4. 1.6 S = D + L + SSE

5. 1.6 S = D + L + Wt

6. 1.6 S = D + L + Ta + Ra + Pa

7. 1.6 S = D + L + Ta + Ra + Pa + OBE + 1.0 (Yr+Yj+Ym)

8. 1.6 S = D + L + Ta + Ra + Pa + SSE + 1.0 (Yr+Yj+Ym)

B. Plastic Design

Factored Load Conditions

9. 0.9Y = D + L + Ta + Ra + 1.5 Pa

9a. 1.0Y = D + L + Ta + Ra + 1.5 Pa

10. 0.9Y = D + L + Ta + Ra + 1.25 Pa + 1.25 OBE + 1.0 (Yr+Yj+Ym)

10a 1.0 = D + L + Ta + Ra + 1.25 Pa + 1.25 OBE + 1.0 (Yr + Yj + Ym)

11. 0.9Y = D + L + Ta + Ra + Pa + SSE + 1.0 (Yr+Yj+Ym)

11a = D + L + Ta + Ra + Pa + SSE + 1.0 (Yr + Yj + Ym)

Notes - Steel Structures

(1) S is the required section strength based on the elastic design methods and allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings."

(2) Y is the section strength required to resist design loads based on plastic design methods described in Part 2 of the AISC

INSERT (A)

2. STEEL STRUCTURES

A. ELASTIC WORKING STRESS DESIGN SERVICE LOAD CONDITIONS

1. $1.0S = D + L$

1a. $1.5S = D + L + T_o + R_o$ (If T_o & $R_o = 0$ use Equation 1.)

2. $1.0S = D + L + OBE$

2a. $1.5S = D + L + T_o + R_o + OBE$ (If T_o & $R_o = 0$ Use Equation 2

3. $1.33S = D + L + W$ [redacted]

3a. $1.5S = D + L + W + T_o + R_o$ (If T_o & $R_o = 0$ Use Equation

FACTORED LOAD CONDITIONS

4. $1.6S = D + L + T_o + R_o + SSE$

5. $1.6S = D + L + T_o + R_o + W_t$

6. $1.6S = D + L + T_a + R_a + P_a$

7. $1.6S = D + L + T_a + R_a + P_a + 1.0(Y_r + Y_j + Y_m) + OBE$

8. $1.7S = D + L + T_a + R_a + P_a + 1.0(Y_r + Y_j + Y_m) + SSE$

TABLE 3.8-3 (Cont)

"Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings."

(3) Both cases of L having its full value or being completely absent are checked for load combinations 1, 1a, 2, 2a, 3, 3a, 4, and 5.

(4) In combinations 4 to 8 and 9 to 11, thermal loads are neglected when it can be shown that they are secondary and self-limiting in nature, or where the material is ductile.

(5) In combinations 6, 7, ~~and~~ 8, ~~and~~ 9, 10, and 11, the maximum values of Pa, Ta, Ra, Yr, Yj, and Ym, including an appropriate dynamic factor, are used unless a time-history analysis is performed to justify otherwise.

(6) Combination 5 shall be satisfied without the tornado missile load. Combinations 7, 8, 10 and 11 shall be first satisfied without Yr, Yj, and Ym. When considering these loads, however, local section strengths may be exceeded under the effect of these concentrated loads, provided there will be no loss of function of any safety related system. Furthermore, in computing the required section strength, S, the plastic section modulus of steel shapes may be used for combinations 7 and 8.

(7) Insert (B)

Loads, Definition of Terms, and Nomenclature

1. Normal Loads - Those loads encountered during normal plant operations and shutdown. They include the following:

D - Dead loads or their related internal moments and forces including any permanent equipment loads + OK

L - Live loads or their related internal moments and forces, including any movable equipment loads and other loads which vary with intensity and occurrence

F - Lateral and vertical pressure of liquids, or their related internal moments and forces. ~~[illegible]~~ F is included in D for steel structures.

H - Lateral earth pressure, or its related internal moments and forces. ~~[illegible]~~ H is included in L for steel structures.

To - Thermal loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition. ~~[illegible]~~ To is included in D for concrete structures equations 1, 2, 2a, 3, 3a, 4, 4a, 5, 5a, 6, 6a, 7 and 7a.

Ro - Pipe loads during operating or shutdown conditions, based on the most critical transient or steady-state condition. ~~[illegible]~~ Ro is included in D for concrete structures equations 1, 2, 2a, 3, 3a, 4, 4a, 5, 5a, 6, 6a, 7, and 7a.

Table 3.8-3

INSERT B.

(7) Combinations 1a, 2a, 3a, 9a, 10a and 11a were added for use in design evaluations since September 1985. These combinations are consistent with NUREG-0800, sect. 3.84, (Rev. 1, July 1981)

TABLE 3.8-3 (Cont)

2. Severe Environmental Loads - Those loads that could infrequently be encountered during the plant life. They include:

   OBE - Loads generated by the operating basis earthquake

   W - Loads generated by the design wind specified for the plant site (Section 3.3.1)

3. Extreme Environmental Loads - Those loads which are credible but highly improbable. They include:

   SSE - Loads generated by the safe shutdown earthquake

   Wt - Loads generated by the design tornado specified for the plant site (Section 3.3.2)

4. Abnormal Loads. Those loads generated by a postulated high-energy pipe break accident within a building and/or compartment thereof. Included in this category are the following:

   Pa = Maximum differential pressure load generated by a postulated break

   Ta = Thermal loads under accident conditions generated by a postulated break

   Ra = Pipe and equipment reactions under accident conditions generated by a postulated break

   Yr = Loads on the structure generated by the reaction on the broken high-energy pipe during a postulated break

   Yj = Jet impingement load on a structure generated by a postulated break

   Ym = Missile impact load on a structure generated by or during a postulated break, such as pipe whipping.

(H-019, H-023)

LCR # 745

STONE & WEBSTER ENGINEERING CORPORATION
LICENSING DOCUMENT CHANGE REQUEST FORM

PROJECT: MILLSTONE III
CLIENT: NUSCO
J.O. NO. 12179
REQUEST NO. 745
REVISION 1

ORIGINATOR: MICHAEL HEANEY (TITLE) WNL (STAFF GROUP/DISCIPLINE)
DATE: OCTOBER 30, 1985

COGNIZANT LDE: M ROBERT BAIN (NAME) [signature] 10/30/85 (SIGN & DATE)

LEAD LICENSING ENGINEER: Mr WALTER EMERSON (SWEC) / R JOSHI (NUSCo) (NAME)

LICENSING DOCUMENT(S) AFFECTED:

FSAR VOLUME 5 SECTION 3.5.1.1

REASON(S) FOR PROPOSED CHANGE:

INVOKE SIMILAR CRITERIA OUTSIDE CONTAINMENT PRECLUDING VALVE STEM EJECTION FOR HIGH ENERGY SYSTEM VALVES THAT ARE BACK SEATED.

DESCRIPTION OF PROPOSED CHANGE (ABSTRACT):

INCLUDE WORDING PROVIDED IN SECTION 3.5.1.2.1 IN SECTION 3.5.1.1 CONCERNING PRECLUDING VALVE STEM MISSILE EJECTION FOR VALVES WHICH HAVE BACK SEATS.

ATTACH CHANGE IN FINAL LICENSING DOCUMENT FORM, 2 PAGES ATTACHED

EVALUATION OF PROPOSED CHANGE - INITIAL UNDER CHANGE CATEGORY EVALUATION

	(LIST)	MINOR	MAJOR	SUBSTANTIAL
LDE (S)	R. BAIN	RAB		
	W. EMERSON	W		
LLE	R. JOSHI			
PE				

ADDITIONAL COMMENTS (ATTACH ADDITIONAL SHEETS IF NECESSARY):

See Attached comments for proposed submittal. C Meisel 11/6/85

tied to EA audit # 171 Item I.5

Rev 1, identified on Insert A. W

Chap recommended by EA AUDIT TEAM MEMBER and (H019 / H023)

Discussed with Project Hazards Review Team

PShelh 11/11/85 M. Heaney W Emerson 11/11/85

3.5 MISSILE PROTECTION

3.5.1 Missile Selection and Description

Systems and components located both inside and outside the containment have been examined to identify and classify potential missiles. Two broad categories of systems and components are reviewed to determine the potential for generating missiles; pressurized components and high speed rotating machinery. Only designs where a single failure could lead to missile ejection are considered. The basic approach to ensure missile protection of systems and components both inside and outside of containment involves the following considerations:

1. Examination of systems in order to identify and classify potential missile sources

2. Evaluation of the design adequacy of equipment to preclude generation of missiles

3. Evaluation of the effects of the generation of missiles where the potential exists and provisions for protection against them

The objective is to ensure design adequacy against generation of missiles and means of protecting safety related equipment should a missile be generated.

3.5.1.1 Internally Generated Missiles (Outside Containment)

The design bases consider missiles generated outside the containment but internal to the plant site. These shall not cause damage that may affect the safe shutdown or cause radiation release during operating conditions, and postulated accident conditions associated with the effects of missile formation. Table 3.5-1 identifies the safety related structures, systems, and components outside the containment required for safe shutdown of the reactor under all conditions of plant operation.

Valves in high energy fluid systems are evaluated as potential missile sources. Valves are typically designed with parts which are removable for maintenance. It is these removable parts which present the most significant potential for missile producing failures. Our review of valves located outside containment has indicated that all the valves in the high energy systems can be categorized as follows: PLEASE INSERT HERE

1. Valves that are isolated or enclosed from safety related equipment. These valves will not impose any danger.

2. Valves that are not enclosed or isolated. Low probability study is conducted to establish whether the missile will inflict damage on safety related equipment. If the damage would affect the safe shutdown of the plant, reorientation of the source is performed or barrier design is established STET

Rev 1
CHG #745
[illegible] 11/11/[illegible]

INSERT A

valves provided with back seated stems
~~VALVE STEMS IN MOTOR-OPERATED (MOV) OR AIR-OPERATED (AOV) VALVES~~ ARE NOT CONSIDERED CREDIBLE SOURCES OF MISSILES. ~~THESE TYPES OF VALVES HAVE THE STEMS BACK SEATED.~~ THIS DESIGN FEATURE EFFECTIVELY ELIMINATES THE POSSIBILITY OF EJECTING VALVE STEMS EVEN IF THE STEM THREADS FAIL. VALVE BONNETS ARE CONSIDERED CREDIBLE SOURCES FOR MISSILES IN CASES WHERE THE BONNET IS BOLTED TO THE BODY. THE BONNET, AND THE CONNECTION BOLTS ARE POSTULATED TO BE EJECTED. VALVE BONNET MISSILES ARE NOT CONSIDERED CREDIBLE WHERE THE BONNET IS WELDED TO TH BODY, ~~OR~~ the bonnet IS INTEGRAL WITH THE BODY, the valve has a large inertia valve operator, or the bonnet bolts are torqued in a controlled manner.

EA AUDIT TEAM
C. Morrell 11/6/95

PROJECT HAZARDS GRO
P Sheld[illegible] 11/6/95
Michael [illegible] 11/6/95

(H-011) # 759

STONE & WEBSTER ENGINEERING CORPORATION
LICENSING DOCUMENT CHANGE REQUEST FORM

PROJECT Millstone III
CLIENT NUSCO JO. NO. 12179

REQUEST NO. Am 18-
REVISION

ORIGINATOR: P Sheldon (TITLE) PWR (STAFF GROUP/DISCIPLINE) DATE: 11/12/85

COGNIZANT LDE M SCANLON (NAME) M Scanlon 11/12/85 (SIGN & DATE)

LEAD LICENSING ENGINEER: W. Emerson (SWEC) / R Joshi (NUSCo) (NAME)

LICENSING DOCUMENT(S) AFFECTED: FSAR Section 1.9
TABLES 1.9-1, 1.9-2

REASON(S) FOR PROPOSED CHANGE: To Provide Justification For deviations from SRP 3.6.1 BTP ASB 3-1 Regarding Seismic qualification of Pipe Rupture mitigating equipment.

DESCRIPTION OF PROPOSED CHANGE (ABSTRACT):

See Attached Sheets
Revision To Section 1.9 (SRP 3.6.1)

ATTACH CHANGE IN FINAL LICENSING DOCUMENT FORM, 5 [struck through] PAGES ATTACHED

EVALUATION OF PROPOSED CHANGE - INITIAL UNDER CHANGE CATEGORY EVALUATION

(LIST)	MINOR	MAJOR	SUBSTANTIAL
LDE (S) M R Scanlon	MS		
R Bain	MB		
W. Emerson (NOTED NOV 13 1985 W. Emerson)	S		
LLE R Joshi			
PE			

ADDITIONAL COMMENTS (ATTACH ADDITIONAL SHEETS IF NECESSARY):

EA AUDIT ITEM H-011; ACTION ITEM 171, 1.7 response to close item by adding clarification to FSAR, SECTION 1.9

TABLE 1.9-1 (Cont)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
3.4.2 (Rev. 2)	II. 1 - Use of DBF or highest groundwater level in design.	DBF or normal groundwater level used.	3.4.2
3.5.1.3 (Rev. 2)	II - Missile protection for cold shutdown maintenance components.	Cold shutdown maintenance components not specified as targets.	3.5.1.3
3.5.1.5 (Rev. 1)	III.3 - Definition of P_p.	Missiles that produce secondary missiles which could damage vital equipment not considered.	3.5.1.5, 3.2.3.2.1
3.5.1.6 (Rev. 2)	III.2 - Inflight crash rate of 4×10^{-10}.	NUREG-75/087 crash rate of 3×10^{-9} used.	3.5.1.6
3.6.1 (Rev. 1)	BTP ASB 3-1, B.1.a(1) - Requires an arbitrary split be postulated on the main steam and the feedwater systems at a location proximate to essential systems.	FSAR does not commit to postulate this arbitrary split.	3.6.1.3.3
	BTP ASB 3-1, B.1.a(2) - Suggests main steam and feedwater pipes not be routed in the vicinity of the control room.	Main steam and feedwater pipes are routed in the vicinity of the control room.	3.6.1.3.3
insert x Chg#759 →			
3.6.2 (Rev. 1)	III.2.a - Requires use of pressure and temperature values corresponding to the greater contained energy at hot standby or at 102 percent power.	FSAR uses internal pressure, and temperature conditions in the piping system during reactor operation at 100 percent power.	3.6.2.2.1
	III.2.a - Requires that the allowable capacity for crushable material shall be limited to 80 percent of its rated energy absorbing capacity as determined by dynamic testing.	FSAR uses an allowable of 80 percent of energy absorbing capacity based on static testing.	3.6.2.2.1
	BTP MEB 3-1, B.1.e - States particular criteria for postulating through wall leakage cracks in high energy Class 1, 2,3, and non-nuclear piping.	FSAR does not postulate cracks in high energy piping.	3.6.2.1.2
			← insert A Chg#513
3.7.2 (Rev. 1)	II.4 - Finite element and half space representation of subgrade soil stiffness.	FSAR does not address half space representation.	3.7.2

TABLE 1.9-1 (Cont)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
3.4.2 (Rev. 2)	II. 1 - Use of DBF or highest groundwater level in design.	DBF or normal groundwater level used.	3.4.2
3.5.1.3 (Rev. 2)	II - Missile protection for cold shutdown maintenance components.	Cold shutdown maintenance components not specified as targets.	3.5.1.3
3.5.1.5 (Rev. 1)	III.3 - Definition of P_p.	Missiles that produce secondary missiles which could damage vital equipment not considered.	3.5.1.5, 3.2.3.2.1
3.5.1.S (Rev. 2)	III.2 - Inflight crash rate of 4×10^{-10}.	NUREG-75/087 crash rate of 3x10[illegible] used.	3.5.1.6
3.6.1 (Rev. 1)	BTP ASB 3-1, B.1.a(1) - Requires an arbitrary split be postulated on the main steam and the feedwater systems at a location proximate to essential systems.	FSAR does not commit to postulate this arbitrary split.	3.6.1.3.3
	BTP ASB 3-1, B.1.a(2) - Suggests main steam and feedwater pipes not be routed in the vicinity of the control room.	Main steam and feedwater pipes are routed in the vicinity of the control room.	3.6.1.3.3
3.6.2 (Rev. 1)	III.2.a - Requires use of pressure and temperature values corresponding to the greater contained energy at hot standby or at 102 percent power.	FSAR uses internal pressure, and temperature conditions in the piping system during reactor operation at 100 percent power.	3.6.2.2.1
	III.2.a - Requires that the allowable capacity for crushable material shall be limited to 80 percent of its rated energy absorbing capacity as determined by dynamic testing.	FSAR uses an allowable of 80 percent of energy absorbing capacity based on static testing.	3.6.2.2.1
	BTP MEB 3-1, B.1.e - States particular criteria for postulating through wall leakage cracks in high energy Class 1, 2,3, and non-nuclear piping.	FSAR does not postulate cracks in high energy piping.	3.6.2.1.2
3.7.2 (Rev. 1)	II.4 - Finite element and half space representation of subgrade soil stiffness.	FSAR does not address half space representation.	3.7.2

insert X Chg#759

insert A Chg#513

Insert X

Specific SRP Acceptance Criteria	Summary Description of Differences	Corresponding FSAR Section
BTP ASB 3-1, B.2.a - States that essential systems and components should be designed to meet the seismic design Criteria of R.G. 1.29.	In the Aux Steam and Hot Water Heating systems, only the electrical detection and actuation devices for the isolation valves are qualified to class 1E requirements, and located in a Seismic Cat I building.	3.6.1.3.3

ADD TO:

P. 2 of Table 1.9-1

TABLE 1.9-2 (Cont)

SRP 3.5.1.6

SRP TITLE: AIRCRAFT HAZARDS

A. Actual difference between FSAR and SRP

SRP 3.5.1.6, Paragraph III.2, indicates an inflight crash rate of 4×10^{-10} per year. FSAR analysis uses the old SRP crash rate of 3×10^{-9}.

B. Justification for differences from SRP

A response to this difference has been provided in the Applicant's response to NRC Acceptance Review Request Number 311.4.

SRP 3.6.1

SRP TITLE: PLANT DESIGN FOR PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT

A. Actual differences between FSAR and SRP

1. BTP ASB 3-1, B.1.a(1), requires an arbitrary split be postulated on the main steam and the feedwater systems at a location proximate to essential systems. The split must be postulated regardless of whether the break exclusion requirements of BTP MEB 3-1, Item B.1.6 are met. The FSAR does not commit to postulate this split.

2. BTP ASB 3-1, B.1.a(2), states that main steam or feedwater piping should not be routed in the vicinity of the control room. The FSAR states that the main steam and feedwater pipes are routed in the vicinity of the control room.

B. Justification for differences from SRP

1. Essential systems, components, or structures are not located within the main steam or feedwater containment penetration area. Environmental effects are of no consequence. The design basis for environmental effects in these areas is given in FSAR Section 3.11, Appendix B.

2. Pipe rupture restraints are provided to prevent main steam pipe whip into the control building wall. The feedwater pipe does not impact the control building in accordance with the discussion in FSAR Section 3.6.1.3.3.

insert Y Chg #759

Insert Y

Add the following to SRP - 3.6.1
TABLE 1.9-2

A.3 BTP-ASB 3.1, B.2.a States that essential systems and components should be designed to meet the seismic design criteria of Regulatory Guide 1.29. Appendix A of BTP-ASB 3-1 defines essential systems and components as those Required to shut down the reactor and mitigate the consequences of a postulated piping failure, without off site power. FSAR section 3.6.1.3.1 identifies two high energy line break isolation systems for the auxiliary steam and hotwater heating systems where the isolation valves are in a non seismic piping system in a non seismic AREA.

B.3 The electrical detection and actuation devices for the two isolation systems fully meet the seismic design criteria for mitigating components and are located in the Auxiliary Building. The isolation valbes have been located in an area to provide optimal isolation capability for the area being isolated. This Design is adequate under the following criteria:

BS 11/13/85

1. The Auxiliary Steam and Hotwater heating piping, located in the Auxiliary Building,

lose pressure boundary integrity
PS —

is designed to meet the seismic design criteria of Regulatory Guide 1.29 regulatory position C.2. This ensures that the subject piping will not fail during a seismic event, therefore isolation valve operation would not be required during the seismic event.

2. The auxiliary steam and hot water heating piping is postulated to break during normal plant operating conditions, therefore the redundant isolation valves will function properly to terminate the event.

3. To provide additional assurance of continued isolation following the HELB event, manual valves will be closed to isolate the affected piping. This will ensure that any seismic interactions occuring some time after the HELB event will not compromise the isolation afforded by the automatic valves.

(P-026)

STONE & WEBSTER ENGINEERING CORPORATION
LICENSING DOCUMENT CHANGE REQUEST FORM

760

PROJECT Millstone 3 CLIENT NUSCO JO. NO. 12179	REQUEST NO. Am18- REVISION
ORIGINATOR: DL Borska (TITLE) / Nuclear Group (STAFF GROUP/DISCIPLINE) DATE: 11/13/85	
COGNIZANT LDE M. R. Scanlon (NAME) [signature] 11/13/85 (SIGN & DATE)	
LEAD LICENSING ENGINEER: W. Emerson (SWEC) / R Joshi (NUSCO) (NAME)	
LICENSING DOCUMENT(S) AFFECTED: FSAR Section 6.3.2.5	

REASON(S) FOR PROPOSED CHANGE:

To clarify those areas of the ECCS flow paths which have detection and isolation provisions.

DESCRIPTION OF PROPOSED CHANGE (ABSTRACT):

See attached paragraph which is to be added to pgs. 6.3-14 and 6.3-15. This paragraph should replace the current statement "Means are are also provided to detect and isolate such leaks in the emergency core cooling flow path within 30 minutes".

ATTACH CHANGE IN FINAL LICENSING DOCUMENT FORM, 3 WR PAGES ATTACHED

EVALUATION OF PROPOSED CHANGE - INITIAL UNDER CHANGE CATEGORY EVALUATION

(LIST)	MINOR	MAJOR	SUBSTANTIAL
LDE (S) [signature]	[initials]		
NOTED NOV 13 1985 W.Emerson	[initials]		
LLE R. Joshi			
PE			

ADDITIONAL COMMENTS (ATTACH ADDITIONAL SHEETS IF NECESSARY):

EA AUDIT ITEM P. 026 A.O. # 164, ITEM 7

2. Either of the two subsystems can be isolated and removed from service in the event of a leak outside the containment.

3. Adequate redundancy of check valves is provided to tolerate failure of a check valve during the long term as a passive component.

4. Should one of the two subsystems be isolated in this long term period, the other subsystem remains operable.

5. Provisions are also made in the design to detect leakage from components outside the containment, collect this leakage, and to provide for maintenance of the affected equipment.

Thus, for the long-term emergency core cooling function, adequate core cooling capacity exits with one flow path removed from service.

Subsequent Leakage from Components in ECCS System

With respect to piping and mechanical equipment outside the containment, considering the provisions for visual inspection and leak detection, leaks will be detected before they propagate to major proportions. A review of the equipment in the system indicates that the largest sudden leak potential would be the sudden failure of a pump shaft seal. Evaluation of leak rate assuming only the presence of a seal retention ring around the pump shaft showed flows less than 50 gpm would result. Piping leaks, valve packing leaks, or flange gasket leaks have been of a nature to build up slowly with time and are considered less severe than the pump seal failure.

Larger leaks in the ECCS are prevented by the following:

1. The piping is classified ANS Safety Class 2 and, therefore, must comply with the corresponding quality assurance program associated with this safety class.

2. The piping, equipment, and supports are designed to ANS Safety Class 2 seismic classification permitting no loss of function resulting from the design basis earthquake.

3. The system piping is located within a controlled area on the plant site.

4. The piping system receives periodic pressure tests and is accessible for periodic visual inspection.

5. The piping is austenitic stainless steel which is not susceptible to brittle fracture during operating conditions.

Based on this review, the auxiliary and engineered safety features buildings and related equipment are designed to be capable of handling leaks up to a maximum of 50 gpm. Means are also provided to

(Replace with insert Z.

see insert Z - Chg#760 5/13/05

detect and isolate such leaks in the emergency core cooling flow path within 30 minutes.

Potential Boron Precipitation

Boron precipitation in the reactor vessel can be prevented by a back-flush of cooling water through the core to reduce boil-off and resulting concentration of boric acid in the water remaining in the reactor vessel.

Three flow paths are available for hot leg recirculation of sump water. Each safety injection pump can discharge to two hot legs with suction taken from the containment recirculation pump discharge. In addition, the recirculation pumps can discharge through the common cross-connect line and inject water through two hot legs.

Loss of one pump or one flow path will not prevent hot leg recirculation, since two redundant flow paths are available for use.

Safety Grade Cold Shutdown Function

During a safety grade cold shutdown the ECCS is relied upon to provide one of the two redundant flow paths for boration and makeup. The ECCS high head injection header provides this function. The redundant flow path is the normal charging header which is part of the chemical and volume control system. Two independent subsystems each consisting of a charging pump and the associated valves and piping are provided and are powered by redundant emergency buses in a manner that ensures that at least one subsystem is always operable. A solenoid valve provided in each subsystem and located in the chemical and volume control system ensures that the remote throttling capability necessary for RCS inventory control and shutdown is available. Provisions are also included in the ECCS design to ensure that the accumulators can be either isolated or vented so that RCS depressurization can be accomplished. Details of the cold shutdown design are discussed in Section 5.4.7.

6.3.2.6 Protection Provisions

The provisions taken to protect the system from damage that might result from dynamic effects associated with postulated rupture of piping, are discussed in Section 3.6. The provisions taken to protect the system from missiles are discussed in Section 3.5. The provisions to protect the system from seismic damage are discussed in Sections 3.7, 3.9, and 3.10. Thermal stresses on the RCS are discussed in Section 5.2

6.3.2.7 Provisions for Performance Testing

Test lines are provided for performance testing of the ECCS system as well as individual components. These test lines and instrumentation are shown on Figures 6.2-37 and 6.3-2. All pumps have miniflow lines for use in testing operability. Additional information on testing can be found in Section 6.3.4.2.

Change p. 6.3-14 & 15 from

Means are also provided to detect and isolate such leaks in the emergency core cooling flow path within 30 min.

TO: [INSERT Z]

⇒ Means are also provided to detect and isolate such leaks in the emergency core cooling equipment cubicles within 30 minutes. In the pipe tunnel areas of the ESF and Fuel Buildings, detection and isolation for ECCS fluid leakage is not required because conservative piping design and valve surveillance precludes leakage postulation. All ECCS piping within these areas meets the low stress limits required of MEB 3.1 for break exclusion areas for moderate energy piping. While the crossover ~~areas~~ valves located in the ESF pipe tunnel are subject to quarterly surveillance during pump inservice testing to identify any incipient leakage.

MILLSTONE UNIT 3

FIGURE 1
NEAM 117, Rev 3

PROJECT CHANGE REQUEST*

PROPOSED CHANGE: See attahced sheets for details

PCR No. S-1017

Revise the EOP's to secure ECCS sump pumps.

INITIATOR M. Scanlon DATE November 7, 1985

AFFECTED DOCS

DWGS:

SPECS:

TURNOVER:

NO:_____ DATE:______ SYS GR:______

ASSOCIATED DOC.

WPCN________
NMCN________
BKFT________
NWDI________
VAR NO.________

REASON: In response to EA tech audit #4 concern P-26

PRELIMINARY REVIEWER

1.0 John E Kreckty 11/13/85
SWEC/NUSCO APE DATE

PCR REQUIRED ☒
PCR NOT REQUIRED ☐

REVIEWER

PRELIMINARY EVALUATION

2.0
John E Kreckty 11/13/85
SWEC DATE
________ ______
NUSCO APE DATE
________ ______
NNECO STARTUP COORDINATOR DATE

FUTURE MOD PROJECT LIST ☐
CONTINUE DETAILED EVALUATION ☒
APPROVED ☐
DISAPPROVED ☐

RECOMMENDED ACTION

3.0
________ ______
SWEC DATE
________ ______
NUSCO APE DATE
________ ______
NNECO STARTUP COORDINATOR DATE

APPROVE PCR INCORPORATION ☐
DISAPPROVED ☐
FUTURE MOD PROJECT LIST ☐

DRB

4.0
________ ______
SWEC PE DATE
________ ______
NUSCO PE DATE

	SWEC	NUSCO
APPROVED	☐	☐
DISAPPROVED	☐	☐

PM

5.0
________ ______
SWEC PM DATE
________ ______
NUSCO PM DATE

	SWEC	NUSCO
APPROVED	☐	☐
DISAPPROVED	☐	☐

FUTURE MOD PROJECT LIST

6.0
________ ______
NNECO SUPT DATE

APPROVED ☐
DISAPPROVED ☐

*DOES THIS PCR APPROVE A PLANT DESIGN CHANGE? YES____ NO____

IMPLEMENTING DOCUMENT ISSUE DATE

__________ ________

A concern was identified during the recent Technical Audit #4 that involves an insensitivity of the leakage detection system which would result in an unmonitored transfer of contaminated ECCS System fluid leakage to the waste disposal building storage tanks.

The current design of the aerated drains system (DAS) includes automatic actuation of non-safety related sump pumps on increasing cubicle sump level. In areas where potential ECCS system leakage could occur due to a long term passive failure, the existing leakage detection system could be insensitive to small leakage rates enveloped by the DAS sump pump capacities.

The leakage detection system had previously been evaluated for effectiveness based on leakage rates associated with passive piping failures in the form of moderate energy pipe cracks as defined in BTP-MEB 3-1. The resultant flow rates, in all cases, were found to be far greater than the capacities of the non-safety related local area sump pumps. As a result, the safety-related level switches installed in all ECCS pump cubicles would alert the operator to the problem for corrective actions. This evaluation is conservative for assessment of flooding effects but is potentially non-conservative from the point of view of ECCS leakage detection and containment within the SLCRS boundary.

In response to the technical audit observation, we have evaluated the consequences of the transfer of sump fluid when the leak rate is within the capacity of the sump pumps (25 gpm).

We have identified three main areas of concern related to this undetected, unmonitored transfer of ECCS leakage:

1. Radiological consequences of transferring sump water outside the secondary containment boundary increasing the potential for an unmonitored, unfiltered release.

2. Potential for degradation of core cooling capabilities in the long term due to the depletion of containment sump fluid via the undetected leakage.

3. Potential for contamination of plant areas leading to limitations on accessibility and increased exposure hazards to plant operating staff.

The radiological consequences of a prolonged transfer of sump fluid at 25 gpm outside the SLCRS boundary will increase site boundary dose rates.

Long term core cooling capabilities will remain unaffected by prolonged ECCS leakage due to the large volume of the containment sump when compared to the small transfer rate (25 gpm). Eventual detection and termination of ECCS leakage/transfer will be sufficient from this point of view.

Undetected and unmonitored transfer of ECCS leakage poses not only a long term plant clean up problem but will limit accessibility to the waste disposal building, post-accident.

Based on offsite dose rate and accessibility concerns discussed above, we propose the following corrective actions:

> For both the ESF and Aux Building sump pumps the Emergency Operating procedures should be ammended to alert the Emergency Assessment Team to evaluate the ramifications of the operation of these pumps and to secure them if deemed necessary. This would be accomplished by deenergizing the pumps at the local MCC or the associated bus, whichever is accessible.

MILLSTONE UNIT 3 FIGURE 1

NEAM 117, Rev 3

PROJECT CHANGE REQUEST*

PCR No. S-1018

PROPOSED CHANGE: Adding surveillance requirements to 3RSS*MV8838A/B, 3RSS*MV8837A/B, 3RSS*MV8804A/B during quaterly Ist testing of the RSS pumps. ← See Attached SIR Confirmation Ltr #66

INITIATOR M. Scanlon DATE November 11, 1985

AFFECTED DOCS — TURNOVER: Tied to Refueling cycle. — ASSOCIATED DOC.

DWGS:

SPECS: NO:____ DATE:____ SYS GR:____

WPCN____ NMCN____ BKFT____ NWDI____ VAR NO.

REASON: These valves should be monitored for any incipient stem leakage in order to meet the commitments made in LCN 760 generated in response to Tech. Audit #4 A.O. 164 Item 7. PCR Approved on basis of NRC approved test program as attached letter. Referenced RSS valves should be specifically addressed in test procedure to ascertain no visible stem packing leakage.

PRELIMINARY REVIEWER

1.0 John E Kruelty 11/13/85
SWEC/NUSCO APE DATE

PCR REQUIRED [X]
PCR NOT REQUIRED [ ]

REVIEWER

2.0 PRELIMINARY EVALUATION

SWEC John E Kruelty	DATE 11/13/85	FUTURE MOD PROJECT LIST [ ]
NUSCO APE NR	DATE	CONTINUE DETAILED EVALUATION [X]
NNECO STARTUP COORDINATOR NR	DATE	APPROVED [ ]
		DISAPPROVED [ ]

3.0 RECOMMENDED ACTION

SWEC	DATE	APPROVE PCR INCORPORATION [ ]
NUSCO APE NR	DATE	DISAPPROVED [ ]
NNECO STARTUP COORDINATOR	DATE	FUTURE MOD PROJECT LIST [ ]

4.0 DRB

			SWEC	NUSCO
SWEC PE [signature]	DATE 11-14-85	APPROVED	[X]	[X]
NUSCO PE [signature]	DATE 11/14/85	DISAPPROVED	[ ]	[ ]

5.0 PM

			SWEC	NUSCO
SWEC PM NR	DATE	APPROVED	[ ]	[ ]
NUSCO PM NR	DATE	DISAPPROVED	[ ]	[ ]

6.0 FUTURE MOD PROJECT LIST

NNECO SUPT [signature] DATE 11/14/85

APPROVED [ ]
DISAPPROVED [X] For Future Mod List. — Cycle to R (Refueling) For Program Development

*DOES THIS PCR APPROVE A PLANT ~~DESIGN~~ CHANGE? YES X NO____

IMPLEMENTING DOCUMENT Procedure ISSUE DATE

Surveillance Procedure