6/21/76

Docket No. 50-336

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Northeast Nuclear Energy Company
ATTN: Mr. D. C. Switzer, President
P. O. Box 270
Hartford, Connecticut 06101

Gentlemen:

In response to your request dated March 2, 1976, the Commission has issued the enclosed Amendment No. 11 to Facility Operating License No. DPR-65 for the Millstone Nuclear Power Station, Unit 2.

The amendment consists of changes in the Technical Specifications to require operability and surveillance of hydraulic snubbers necessary to protect the primary coolant system and all other safety related systems and components. During our review of the proposed change, we found that certain modifications to the application were necessary. These modifications were found mutually acceptable to you and the NRC staff and have been incorporated into the Technical Specifications.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 11
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

OFFICE	ORB#3	ORB#3	ORB#4	OELD	ORB#3	DOR
SURNAME	CParrish	DJaffe	DSnaider	SHFess	GLear	JMcGough
DATE	4/13/76	4/13/76	4/14/76	4/16/76	4/29/76	4/15/76

cc:

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

THE CONNECTICUT LIGHT AND POWER COMPANY,
THE HARTFORD ELECTRIC LIGHT COMPANY,
WESTERN MASSACHUSETTS ELECTRIC COMPANY, AND
NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
License No. DPR-65

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by The Connecticut Light and Power Company, The Hartford Electric Light Company, Western Massachusetts Electric Company, and Northeast Nuclear Energy Company (the licensees), dated March 2, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

   E. An environmental statement or negative declaration need not be prepared in connection with the issuance of this amendment.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment.

3. The license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: June 21, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 11

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Revise the following pages of the Appendix "A" Technical Specifications with the enclosed pages. Revised pages are identified by Amendment Number and contain vertical lines indicating the area of change. Corresponding overleaf pages are also provided to maintain document completeness.

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3/4 4-27
3/4 4-28
3/4 4-29
3/4 7-21 (added)
3/4 7-22 (added)
3/4 7-23 (added)
3/4 7-24 (added)
3/4 7-25 (added)
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Pages

3/4 7-27 (added)
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3/4 7-32 (added)
B 3/4 4-12
B 3/4 7-5
B 3/4 7-6 (added)

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

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SURVEILLANCE REQUIREMENTS (Continued)

An initial report of any abnormal degradation of the structural integrity of the Safety Class 1 components detected during the above required inspections shall be made within 10 days after detection and the detailed report shall be submitted pursuant to Specification 6.9.1 within 90 days after completion of the surveillance requirements of this specification.

The Inservice Inspection Program shall be reviewed every 5 years to assure that the equipment, techniques and procedures being utilized are current and applicable. The results of these reviews shall be reported in Special Reports to the Commission pursuant to Specification 6.9.2 within 90 days of completion.

b. Inspections Following Repairs or Replacements The structural integrity of the reactor coolant system shall be demonstrated after completion of all repairs and/or replacements to the system by verifying the repairs and/or replacements meet the requirements of Articles IS-400 and IS-500 (Summer 1971 Addenda) of Section XI of the ASME Boiler and Pressure Vessel Code. When repairs and/or replacements are made which involve new strength welds on components greater than 4 inch diameter, the new welds shall receive a surface and 100 percent volumetric examination and meet applicable code requirements. When repairs and/or replacements are made which involve new strength welds on components 4 inch diameter or smaller, the new welds shall receive a surface examination and meet applicable code requirements.

c. Inspections Following System Opening The structural integrity of the reactor coolant system shall be demonstrated after each closing by performing a leak test, with the system pressurized to at least 2250 psia, in accordance with Article IS-500 (Summer 1971 Addenda) of Section XI of the ASME Boiler and Pressure Vessel Code and the Pressure/Temperature limits of Specification 3.4.9.1.

TABLE 4.4-4

INSERVICE INSPECTION PROGRAM - SAFETY CLASS 1 COMPONENTS

Item No.*	Examination Category *	Components and Parts to be Examined	Examination Method *	Extent of Examination 10 Year Interval
		Section 1. Reactor Vessel and Closure Head		
1.1	A	Longitudinal and circumferential shell welds in core region	Volumetric	Note 11
1.2	B	Longitudinal and circumferential welds in shell (other than those of Category A and C) and meridional and circumferential seam welds in bottom head and closure head (other than those of Category C)	Volumetric	Note 11
1.3	C	Vessel-to-flange and head-to-flange circumferential welds	Volumetric	Note 1
1.4	D	Primary nozzle-to-vessel welds and nozzle-to-vessel inside radiused section	Volumetric	Note 12
1.5	E	Partial Penetration Welds Including Control Rod Drive Penetrations	Visual	Note 1
1.6	O	Control Rod Drive Housing Pressure Retaining Welds	Volumetric	Note 1
1.7	F	Primary nozzles to safe-end-welds	Visual and Surface and Volumetric	Note 10
1.8	G-1	Closure studs and nuts	Volumetric and Visual or Surface	Note 1

TABLE 4.4-4 (Cont)

Item No.*	Examination Category *	Components and Parts to be Examined	Examination Method *	Extent of Examination 10 Year Interval
		Steam Generators (Cont)		
3.6	H	Integrally-welded vessel supports	Visual and Volumetric	Note 1
3.7	I-2	Vessel cladding	Visual	Note 1
		Section 4. Piping Pressure Boundary		
4.1	F	Vessel, pump and valve safe-ends to primary pipe welds and safe-ends in branch piping welds	Visual and Surface and Volumetric	Note 1
4.2	J-1	Circumferential and longitudinal pipe welds and branch pipe connections welds larger than 4 inches in diameter	Visual and Volumetric	Note 1
4.3	G-1	Pressure-retaining bolting	Visual and Volumetric	Note 1
4.4	G-2	Pressure-retaining bolting	Visual	Note 1
4.5	K-1	Integrally-welded supports	Visual and Volumetric	Note 1
4.6	K-2	Piping support and hanger (except hydraulic snubbers)	Visual	Note 1
4.7	J-2	Circumferential and longitudinal pipe welds and branch pipe connections welds	Visual	Note 1

TABLE 4.4-4 (Cont)

Item No.*	Examination Category *	Components and Parts to be Examined	Examination Method *	Extent of Examination 10 Year Interval
		Piping Pressure Boundary (Cont)		
4.8	J-1	Socket welds and pipe branch connections welds 4 inches diameter and smaller	Visual and Surface	Note 1
		Section 5. Pump Pressure Boundary and Pump Flywheels		
5.	L-1	Pump casing welds	Visual and Volumetric	Note 1
5.2	L-2	Pump casings	Visual	Note 7
5.3	F	Nozzle-to-safe end welds	Visual and Volumetric	Note 9
5.4	G-1	Pressure-retaining bolting	Visual and Volumetric	Note 1
5.5	G-2	Pressure-retaining bolting	Visual	Note 1
5.6	K-1	Integrally-welded supports	Visual and Volumetric	Note 7
5.7	K-2	Supports and hangers (except hydraulic snubbers)	Visual	Note 1
5.8	-	Flywheels	Volumetric	Note 14
		Section 6. Valve Pressure Boundary		
6.1	M-1	Valve-body welds	Visual and Volumetric	Note 8

TABLE 4.4-4 (Cont)

Item No.*	Examination Category *	Components and Parts to be Examined	Examination Method *	Extent of Examination 10 Year Interval
		Valve Pressure Boundary (Cont)		
6.2	M-2	Valve bodies	Visual	Note 9
6.3	F	Valve-to-safe end welds	Visual and Volumetric	Note 10
6.4	G-1	Pressure-retaining bolting	Visual and Volumetric	Note 5
6.5	G-2	Pressure-retaining bolting	Visual	Note 1
6.6	K-1	Integrally-welded supports	Visual and Volumetric	Note 6
6.7	K-2	Supports and Hangers (except hydraulic snubbers)	Visual	Note 1
		Section 7. Secondary Side of Steam Generators		
7.1	C-A	Steam generator secondary side circumferential shell welds 1) lower shell to tube sheet weld 2) shell to lower cone section weld 3) shell to upper cone section weld	Volumetric	Note 13

TABLE 4.4-4 (Cont'd)

NOTES

* The item number (except 5.8), examination category and examination method are listed in Table IS-261 of Section XI of the ASME Boiler and Pressure Vessel Code.

1. The extent and frequency of examinations listed in Table IS-251 of the 1971 edition (including the Summer 1973 Addendum) of Section XI of the ASME Boiler and Pressure Vessel Code shall be performed for the applicable examination category.

2. Since no bolting less than two inches in diameter is provided, no examination is required.

3. Since nozzle-type supports are provided, the area to be examined shall be the weld connection between the nozzle and the vessel shell as described in Examination Category D.

4. Since all penetrations fulfill Part C of the IS-121 exclusion criteria, all penetrations shall be examined in accordance with Examination Category E.

5. Since no bolting equal to or greater than two inches in diameter are provided, no examination is required.

6. Since no integrally welded supports are provided, no examination is required.

7. Since the pump casings are fabricated as cast stainless steel sections joined with austenitic welds, present volumetric techniques and procedures are no amendable to both pre-service and in-service applications. As a result, in-service examinations of the pumps, visual as well as volumetric, may be delayed until the tenth year as permitted by Section XI of the ASME Boiler and Pressure Vessel Code. Should a volumetric technique be available at that time, examinations shall be performed in accordance with the applicable item number.

8. Since the valve bodies are fabricated as stainless steel castings, with no welds, no examination is required.

9. Those valves and pumps which may be disassembled without causing an interruption in shutdown cooling shall be examined in accordance with the applicable item number.

10. Since no such safe-end welds are present, no examination is required.

3/4.7.8 HYDRAULIC SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8.1 All hydraulic snubbers listed in Table 3.7-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more hydraulic snubbers inoperable, restore the inoperable snubber(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Hydraulic snubbers shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program:

a. Each hydraulic snubber with seal material fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment and approved as such by the NRC, shall be determined OPERABLE at least once after not less than 4 months but within 6 months of initial criticality and in accordance with the inspection schedule of Table 4.7-3 thereafter, by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors. Initiation of the Table 4.7-3 inspection schedule shall be made assuming the unit was previously at the 6 month inspection interval.

b. Each hydraulic snubber with seal material not fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment shall be determined OPERABLE at least once per 31 days by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown, a representative sample of at least 10 hydraulic snubbers or at least 10% of all snubbers listed in Table 3.7-1, whichever is less, shall be selected and functionally tested to verify correct piston movement, lock up and bleed. Snubbers greater than 50,000 lb. capacity may be excluded from functional testing requirements. Snubbers selected for functional testing shall be selected on a rotating basis. Snubbers identified as either "Especially Difficult to Remove" or in "High Radiation Zones" may be exempted from functional testing provided these snubbers were demonstrated OPERABLE during previous functional tests. Snubbers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each snubber found inoperable during these functional tests, an additional minimum of 10% of all snubbers or 10 snubbers, whichever is less, shall also be functionally tested until no more failures are found or all snubbers have been functionally tested.

TABLE 3.7-1

SAFETY RELATED HYDRAULIC SNUBBERS*

HANGER NO. (1)	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
310022	HPSI-46S/21E/-7	I	No	No
312015 (2)	MS-4N/41W/+66	I	No	Yes
312016 (2)	MS-4N/41E/+66	I	No	Yes
312017	MS-4N/41E/+64	I	No	Yes
312018	MS-4N/41W/+64	I	No	Yes
312019	MS-4N/41E/+63	I	No	Yes
401008	HPSI-F.2/18.9/-37	A	Yes	No
401024	CS-F.8/17.7/-30	A	No	No
401025 (2)	HPSI-H.2/17.2/-30	A	No	No
401106	HPSI-H.4/17.7/-11	A	No	Yes
401107	HPSI-H.4/17.7/-13	A	No	Yes
402009	LPSI-F.2/18.9/-32	A	Yes	No
402013	HPSI-F.3/18.9/-40	A	Yes	No

TABLE 3.7-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

HANGER NO. (1)	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
402022	LPSI-F.2/18.1/-31	A	Yes	No
402056 (2)	LPSI-H.4/17.6/-32	A	Yes	No
402083	CS-H4/18.4/-20	A	No	Yes
402[illegible]	SIT-42S/41W/+25	I	No	No
402113 (2)	SDC-18S/50W/-2	I	Yes	No
402115 (2)	SDC-18S/50W/+3	I	Yes	No
4030[illegible]8	SFP-E.5/18.1/-36	A	Yes	No
403070	SFP-L.5/18.9/+10	A	Yes	No
403090	SFP-M.4/18.9/+7	A	Yes	No
405388	MS-E.5/19.6/+53	A	No	No
405618 (2)	RBCCW-J.7/16.6/-13	A	No	No
410004 (2)	HPSI-57S/10W/-13	I	No	No

TABLE 3.7-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

HANGER NO. (1)	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
410012	SIT-27S/42W/+13	I	Yes	No
410014	SIT-30S/47W/+15	I	No	No
410015	SIT-30S/47W/+15	I	No	No
410017	SIT-38S/48W/+1	I	No	No
410019	SIT-32S/47W/+8	I	No	No
410021	SIT-6N/47W/+9	I	No	No
410022	SIT-9N/62W/+5	I	No	No
410027	SIT-6N/47W/+10	I	No	No
410028	SIT-6N/47W/+15	I	No	Yes
410029	SIT-6N/47W/+15	I	No	No
410031 (2)	SIT-1N/37W/+15	I	Yes	No
410061	SDC-20S/28W/-5	I	Yes	Yes
410065 (2)	SIT-53S/33E/-4	I	No	Yes

TABLE 3.7-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

HANGER NO. (1)	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
410067	SIT-30S/47E/+6	I	No	No
410083	SIT-30S/54E/-1	I	No	No
410086 (3)	SIT-30S/47E/+15	I	No	1 Yes 2 No
410103	SIT-11N/61W/+5	I	No	No
411010	FEED-E/22/+41	A	No	No
411011 (2)	FEED-E/22/+47	A	No	No
411023	FEED-E/22/+34	A	No	No
411026	FEED-E.5/22/+31	A	No	No
411028	FEED-E.5/22/+34	A	No	No
412002	MS-4N/41W/+63	I	No	Yes
412003	MS-M.4/18.9/+55	A	No	No
412004	MS-E.5/20/+50	A	No	No
412013	FEED-7S/50E/+50	I	No	No

TABLE 3.7-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

HANGER NO. (1)	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
412015	FEED-10S/50W/+49	I	No	No
412016 (2)	MS-M.4/20/+55	A	No	No
412018	FEED-3S/51W/+45	I	No	Yes
~~413009~~	MS-F.8/18.9/+55	A	No	No
413011 (2)	MS-E.5/18/+40	A	No	No
413018 (2)	MS-H.4/18.9/+55	A	No	No
413019 (2)	MS-E.5/17/+40	A	No	No
413021	MS-D/16/+43	A	No	No
413022 (2)	MS-E/16/+41	A	No	No
413024 (2)	MS-E.5/17/+40	A	No	No
413025 (2)	MS-E.5/17/+40	A	No	No
413028	MS-K.7/18.9/+55	A	No	No
413029 (2)	MS-E.5/18.5/+48	A	No	No
413030 (2)	MS-E.5/18.5/+46	A	No	No

TABLE 3.7-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

HANGER NO. (1)	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
413031	MS-E.5/18.5/+47	A	No	No
413032 (2)	MS-E.5/19.6/+56	A	No	No
413036	MS-E/18/+48	A	No	No
413039	MS-E/21/+49	A	No	No
413041	MS-C/16/+44	A	No	No
413046 (2)	MS-E/16/+41	A	No	No
413081	MS-E.5/19/+46	A	No	No
413082 (2)	MS-E.5/19/+46	A	No	No
413162	FEED-E/23/+64	A	No	No
413163	FEED-E.5/23/+64	A	No	No
413172	FEED-E.5/19/+49	A	No	No
413179	FEED-J.7/18.9/+50	A	No	No
413181	FEED-K.7/18.9/+50	A	No	No
413192 (2)	FEED-F.2/18.9/+50	A	No	No

TABLE 3.7-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

HANGER NO. (1)	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
413199 (2)	FEED-L.5/19.8/+50	A	No	No
416014 (2)	CS-8S/61E/+7	I	No	No
416020	CS-23S/56E/+10	I	No	No
416023 (2)	CS-30S/60W/+7	I	No	No
416025	CS-18S/60W/+8	I	No	No
416027	CS-5S/60W/+5	I	No	No
427075	SW-L.5/17.2/-13	A	No	No
427097 (2)	SW-L.5/17.2/-11	A	No	No
427106	SW-L.5/17.2/-11	A	No	No
427115 (2)	SW-L.5/15.9/-14	A	No	No
450071	RBCCW-J.7/17.2/-13	A	No	No
490001 (2)	MS-D/19/+28	A	No	No
490002 (2)	MS-D/19/+25	A	No	No
490003 (2)	MS-C/19/+27	A	No	No
490004 (2)	MS-D/19/+25	A	No	No

TABLE 3.7-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

HANGER NO. (1)	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
490005 (2)	MS-C/19/+33	A	No	No
490006	MS-D/19/+33	A	No	No
490007	MS-D/19/+32	A	No	No
490008	MS-C/19/+32	A	No	No
490018 (2)	MS-D/17/+33	A	No	No
490019	MS-D/17/+33	A	No	No
490031	MS-B/17.1/+46	A	No	No
501022 (2)	HPSI-F.8/18.9/-29	A	Yes	Yes
502032	CS-E.5/19.6/+2	A	Yes	No
504002	HPSI-F.2/18.5/-42	A	Yes	No
504003	HPSI-F.2/17.2/-42	A	Yes	No
505166 (2)	RBCCW-J.7/17.2/-16	A	No	No
507004	HPSI-F.2/18.5/-42	A	Yes	No
510017	SIT-6N/47E/+15	I	No	No

TABLE 3.7-1 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

HANGER NO. (1)	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
510018	SIT-6N/47E/+15	I	No	No
511001	FEED-D/22/+37	A	No	No
513023 (2)	SG-K.6/19.6/-2	A	Yes	No
513032	MS-E/19/+48	A	No	No
SS1-SS8 (8)	SG #1	I	Yes	Yes
SS1-SS8 (8)	SG #2	I	Yes	Yes
51119-R27 (2)	MS-D/17/+45	A	No	No
51119-R28 (2)	MS-B/17/+45	A	No	No
51119-C29	MS-D/17/+45	A	No	No

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-1 provided that safety evaluations, documentation and reporting are provided in accordance with 10 CFR 50.59 and that a proposed revision to Table 3.7-1 is included with the next License Amendment request.

** Modifications to this table due to changes in high radiation areas shall be submitted to the NRC as part of the next License Amendment request.

(1) The hanger number is listed. Where more than one snubber is associated with a given hanger, it is so indicated in parentheses.

TABLE 4.7-3

HYDRAULIC SNUBBER INSPECTION SCHEDULE

NUMBER OF SNUBBERS FOUND INOPERABLE DURING INSPECTION OR DURING INSPECTION INTERVAL*	NEXT REQUIRED INSPECTION INTERVAL**
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3 or 4	124 days ± 25%
5, 6, or 7	62 days ± 25%
$\geq$ 8	31 days ± 25%

* Snubbers may be categorized into two groups, "accessible" and "inaccessible." This categorization shall be based upon the snubber's accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

** The required inspection interval shall not be lengthened more than one step at a time.

for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The required inspection programs for the Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent practicable, the inspection program for the Reactor Coolant System components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code "Inservice Inspection of Nuclear Reactor Coolant Systems" dated July 1, 1971.

All areas scheduled for volumetric examination have been pre-service examined using equipment, techniques and procedures anticipated for use during post-operation examinations. To assure that consideration is given to the use of new or improved inspection equipment, techniques and procedures, the Inservice Inspection Program will be periodically reviewed on a 5 year basis.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of most reactor coolant loop components except the reactor vessel. The reactor vessel requires special consideration because of the radiation levels.

The techniques anticipated for inservice inspection include visual inspections, ultrasonic, radiographic, magnetic particle and dye penetrant testing of selected parts.

The nondestructive testing for repairs on components greater than 4 inches diameter gives a high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. Repairs on components 4 inches in diameter or smaller receive a surface examination which assures a similar standard of integrity. In each case, the leak test will ensure leak tightness during normal operation.

For normal opening and reclosing, the structural integrity of the Reactor Coolant System is unchanged. Therefore, satisfactory performance of a system leak test at 2250 psia following each opening and subsequent reclosing is acceptable demonstration of the system's structural integrity. These leak tests will be conducted within the pressure-temperature limitations for Inservice Leak and Hydrostatic Testing of Specification 3.4.9.1 and Figure 3.4-2.

Inspection of the pipe hangers and supports provides assurance that these devices are operated within permissible travel and/or loading limits.

3/4.4.11 CORE BARREL MOVEMENT

This specification is provided to ensure early detection of excessive core barrel movement if it should occur. Core barrel movement will be detected by using four excore neutron detectors to obtain Amplitude Probability Distribution (APD) and Spectral Analysis (SA). Baseline core barrel movement Alert Levels and Action Levels at nominal THERMAL POWER levels of 20%, 50%, 80% and 100% of RATED THERMAL POWER will be determined during the reactor startup test program.

A modification to the required monitoring program may be justified by an analysis of the data obtained and by an examination of the affected parts during the plant shutdown at the end of the first fuel cycle.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

3/4.7.8 HYDRAULIC SNUBBERS

The hydraulic snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. The only snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety related system.

The inspection frequency applicable to snubbers containing seals fabricated from materials which have been demonstrated compatible with their operating environment is based upon maintaining a constant level of snubber protection. Therefore, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during an inspection of these snubbers determines the time interval for the next required inspection of these snubbers. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

BASES

To provide further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Observed failures of these sample snubbers will require functional testing of additional units. To minimize personnel exposures, snubbers installed in high radiation zones or in especially difficult to remove locations may be exempted from these functional testing requirements provided the OPERABILITY of these snubbers was demonstrated during functional testing at either the completion of their fabrication or at a subsequent date.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 11 TO LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE UNIT 2

DOCKET NO. 50-336

Introduction

During the summer of 1973, inspections at two reactor facilities revealed a high incidence of inoperable hydraulic shock suppressors (snubbers) manufactured by Bergen Paterson Pipesupport Corporation. As a result of those findings, the Office of Inspection and Enforcement required each operating reactor licensee to immediately inspect all Bergen Paterson snubbers utilized on safety systems and to reinspect them 45 to 90 days after the initial inspection. Snubbers supplied by other manufacturers were to be inspected on a lower priority basis.

Since a long term solution to eliminate recurring failures was not immediately available, the Division of Reactor Licensing sent a letter dated October 2, 1973, to operating facilities utilizing Bergen Paterson snubbers specifying continuing surveillance requirements and requesting a submittal within one year of proposed Technical Specifications for a snubber surveillance program. On March 2, 1976, Northeast Nuclear Energy Company (NNECO) proposed Technical Specifications for hydraulic snubbers at Millstone Unit 2 as requested by our letter of December 24, 1975. During our review of the proposed change, we found that certain modifications were necessary. These modifications were discussed with the licensee and have been incorporated into the proposed Technical Specifications.

DISCUSSION

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal movement during startup and shutdown.

The consequence of an inoperable snubber is an increase in the probability of structural damage to piping resulting from a seismic or other postulated event which initiates dynamic loads. It is, therefore, necessary that snubbers installed to protect safety system piping be operable during reactor operation and be inspected at appropriate intervals to assure their operability.

Examination of defective snubbers at reactor facilities has shown that the high incidence of failures observed in the summer of 1973 was caused by severe degradation of seal materials and subsequent leakage of the hydraulic fluid. The basic seal materials used in Bergen Paterson snubbers were two types of polyurethane; a millable gum polyester type containing plasticizers and an unadulterated molded type. Material tests performed at several laboratories(1) established that the millable gum polyurethane deteriorated rapidly under the temperature and moisture conditions present in many snubber locations. Although the molded polyurethane exhibited greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. The investigation indicated that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

An extensive seal replacement program has been carried out at many reactor facilities. Experience with ethylene propylene seals has been very good with no serious degradation reported thus far. Although the seal replacement program has significantly reduced the incidence of snubber failures, some failures continue to occur. These failures have generally been attributed to faulty snubber assembly and installation, loose fittings and connections and excessive pipe vibrations. The failures have been observed in both PWRs and BWRs and have not been limited to units manufactured by Bergen Paterson. Because of the continued incidence of snubber failures, we have concluded that snubber operability and surveillance requirements should be incorporated into the Technical Specifications. We have further concluded that these requirements should be applied to all safety related hydraulic snubbers, regardless of manufacturer, in all light water cooled reactor facilities.

We have developed Technical Specifications and Bases to provide additional assurance of satisfactory snubber performance and reliability. The specifications require that snubbers be operable during reactor power operation, startup, hot standby and hot shutdown conditions. Because snubber protection is required only during relatively low probability events, a period of 72 hours is allowed for repair or replacement of one or more defective units or the reactor must be in hot standby within 6 hours and in cold shutdown within 30 hours.

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(1) Report, H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974, Subject Hydraulic Shock Sway Arrestors

An inspection program is specified to provide additional assurance that the snubbers remain operable. The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The longest inspection interval allowed in the Technical Specifications after a record of no snubber failures has been established as 18 months. Experience at operating facilities has shown that the required surveillance program should provide an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment. Snubbers containing seal material which has not been demonstrated to be compatible with the operating environment are required to be inspected every 31 days.

To further increase the level of snubber reliability, the Technical Specifications require functional tests of at least 10 snubbers (or 10% of those listed in the Technical Specifications, whichever is less) at least once per 18 months. The tests will verify proper movement, lock up and bleed.

Evaluation

The proposed Technical Specifications submitted by NNECO on March 2, 1976, closely conformed to the sample Technical Specifications which we sent to NNECO in our letter dated December 24, 1975. Accordingly, we find the proposed Technical Specifications to be acceptable, as modified.

In addition to adding Technical Specifications for operability and inspection of snubbers as Technical Specifications 3/4.7.8, the previous inspection program contained in section 4.4.10 is being deleted since it duplicated information contained in Technical Specification 3/4.7.8.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4), that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 21, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-336

NORTHEAST NUCLEAR ENERGY COMPANY,
THE CONNECTICUT LIGHT AND POWER COMPANY,
THE HARTFORD ELECTRIC LIGHT COMPANY, AND
WESTERN MASSACHUSETTS ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY
OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 11 to Facility Operating License No. DPR-65 issued to Northeast Nuclear Energy Company, The Connecticut Light and Power Company, The Hartford Electric Light Company, and Western Massachusetts Electric Company, which revised Technical Specifications for operation of the Millstone Nuclear Power Station, Unit 2, located in the Town of Waterford, Connecticut. The amendment is effective as of the date of issuance.

The amendment modifies the Technical Specifications to require the operability and surveillance of hydraulic snubbers necessary to protect the primary coolant system and all other safety related systems and components.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chpater I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 2, 1976, (2) Amendment No. 11 to License No. DPR-65, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Waterford Public Library, Rope Ferry Road, Waterford, Connecticut 06385.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 21 day of June 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors