Northeast Nuclear Energy

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The Northeast Utilities System

MAR 25 1996

Docket No. 50-336
B15615

Re: 10 CFR 50.73

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

This letter forwards Licensee Event Report (LER) 96-008-00 documenting an event that occurred at Millstone Nuclear Power Station, Unit No. 2 on February 22, 1996. This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(ii).

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

For: P. M. Richardson
Director - Millstone Unit No. 2

By: J. M. Bergin
Assistant Operations Manager
Millstone Unit No. 2

Attachment: LER 96-008-00

cc: T. T. Martin, Region I Administrator
P. D. Swetland, Senior Resident Inspector, Millstone Unit No. 2
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2

IE22 1/1

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)
Containment Sump Screen Enclosure Has Openings Larger Than Design

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
02	22	96	96	008	00	03	25	96	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): 5
POWER LEVEL (10): 0%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	X 50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
G. P. van Noordennen, Nuclear Licensing Supervisor	(860)440-[illegible]

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On February 22, 1996, at 1422 hours, with the plant in Mode 5 at 0% power, it was concluded that the condition of the wire mesh screen enclosure over the two containment recirculation suction pipes (containment sump screen) created a condition that was outside the design basis of the plant. The wire mesh is fabricated from 304 stainless steel 0.080 diameter wire, designed to provide 0.187 inch openings. A total of ten locations that were inconsistent with the specified opening size of 0.187" were identified. The cause of this event was a construction/installation error resulting from inadequate administrative controls. The screen mesh on the ends of the enclosure appear to have been installed during original plant construction. Based on a judgment that debris may pass through the screen openings and impede ECCS or containment spray flow, it has been concluded that the as-found configuration of the containment sump screens placed the plant in a condition outside of its design basis. The screen enclosure is being replaced with one consistent with the original design. Administrative controls have been strengthened through issuance and revision of the Design Control Manual. Millstone Unit No. 2 is currently performing a review of the plant's design basis, design processes and programs, which will ensure that lessons learned from this sump screen issue are factored into the plan to ensure proper field configuration. There were no automatic or manually initiated safety systems activated as a result of this event.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

I. Description of Event

On February 22, 1996, at 1422 hours, with the plant in Mode 5 at 0% power, it was concluded that the condition of the screen enclosure over the two containment recirculation suction pipes (containment sump screen) created a condition that was outside the design basis of the plant. Some screen mesh panels opening sizes were larger than design and some holes were found in the enclosure. The plant had been previously shut-down due to a concern that the designed openings in the containment sump screen were larger than the openings in the High Pressure Safety Injection (HPSI) injection valves. It was postulated that if debris the size of the designed sump screen opening was introduced into the HPSI injection valves, it could partially clog the valve throttling disc and prevent the valve(s) from passing the required flow during a LOCA.

As part of the follow-up activities for the HPSI valve plugging concern, a visual inspection of the containment sump screen enclosure was performed when the plant was shutdown. The containment sump screen enclosure consists of a steel angle iron structure which is divided into two compartments and is anchored to the containment floor. The structure is approximately twelve feet long, seven feet wide, and three and three quarters feet high. This structure is covered with 1.25 inch grating and wire mesh cloth; the partition in the center that creates the two compartments also consists of grating and wire mesh cloth. The wire mesh is fabricated from 304 stainless steel 0.080 inch diameter wire, designed to provide 0.187 inch openings. Each compartment covers the area where the suction piping for the Emergency Core Cooling System (ECCS) penetrates the floor. Two 24 inch lines penetrate the floor in the southwest quadrant of containment and extend up approximately eleven inches. The assembly is sized to maintain low fluid velocities of approximately 0.305 ft./sec. through the side walls (assuming that the top side of the sump screen is completely blocked by debris). This inspection identified that portions of the enclosure were constructed with wire mesh screen that had larger openings than required by the design drawing, and various holes were discovered that were larger than the design screen opening. Specifically, the wire mesh on the two end panels had 0.25 inch openings and the wire mesh of the center partition had 0.375 inch openings; the holes consisted of gaps between the mesh and frame panels ranging in size from approximately 0.25 inch by two feet long to 0.75 inch by 0.5 inch. A total of ten locations that are inconsistent with the specified opening size of 0.187 inch were identified.

There were no immediate operator actions required in response to this event. Additionally, there were no automatic or manually initiated safety systems activated as a result of this event.

II. Cause of Event

The cause of this event was a construction/installation error resulting from inadequate administrative controls. The screen mesh on the ends of the enclosure appear to have been installed during original plant construction as no record could be located to indicate that a modification or change had been performed. The center partition screen mesh was determined to be missing and was installed in January of 1988. Review of the documentation for this installation indicates that the proper screen was indicated but a 0.375 inch mesh was provided and installed. No specific verification of screen mesh size was documented during the installation.

A contributing cause of this event was inadequate design drawing details which did not provide all the construction details necessary for the installation of the screen enclosure. The original design drawing provided general requirements for the enclosure. The details for the screen/grating attachment and for the center partition were not provided on the drawings.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

III. Analysis of Event

This event is being reported pursuant to the requirements of 10 CFR 50.73 (a)(2)(ii)(B), "any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant."

The containment sump screen is designed to preclude debris from impeding ECCS and containment spray system flow during the recirculation phase of a LOCA. Following discovery of the sump screen holes being larger than specified in the Final Safety Analysis Report (FSAR), the structural adequacy of the screen was evaluated and determined to be acceptable. The potential for plugging the HPSI injection valves and/or containment spray nozzles was re-evaluated. The results of the evaluation concluded that the as-found configuration of the containment sump screen placed the plant in a condition outside of its design basis. This conclusion is based on a judgment that debris may pass through the screen openings and impede ECCS or containment spray flow.

The type of debris that would pass through the screen openings would tend to be of low density and low structural strength. Material of this type would be reduced in size as it passes through the HPSI or containment spray pumps. The differential pressure across the HPSI injection valves and containment spray nozzles would tend to force any material that is marginally capable of obstructing flow, through the valves or nozzles. Based on these factors, and the low probability of a LOCA requiring recirculation, the safety significance of this event is low.

IV. Corrective Action

The screen enclosure is being replaced with one that is consistent with the original design.

Administrative controls have been strengthened via issuance and revision of the Design Control Manual to ensure that any changes to the plant are properly documented and tested.

Millstone Unit No. 2 is currently performing a verification of the plant's design basis. This includes a review of the plant's design basis, design processes and programs, and will ensure that lessons learned from this sump screen issue are factored into the plan to ensure proper field configuration.

V. Additional Information

During the HPSI injection valve evaluation, it was determined that the best estimate time to initiate the containment Sump Recirculation Actuation Signal (SRAS) was shorter than was indicated in the FSAR: 39 minutes vs. 44 minutes. This condition was initially reported as a prompt report per 10 CFR 50.72. Upon further review, it was determined to be not reportable, since the worst case design basis time to initiate SRAS had not been compromised. The system was still capable of removing the core residual heat with a 39 minute SRAS initiation time.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

Similar Events

None

Manufacturer Data

EIIS Codes:

BE-SCN-B130
BQ-SCN-B130
BP-SCN-B130