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Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362-9637

Operated by Nuclear Management Company LLC

February 19, 2001

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

LER 2001-002
Failure to Comply with Technical Specification and ASME Code Section XI Inservice Inspection Requirements

A Licensee Event Report for this occurrence is attached. This report contains no new NRC commitments.

Contact Patrick Burke, Project Manager, at (763) 295-1661 if you require further information.

Byron Day

Byron Day
Plant Manager
Monticello Nuclear Generating Plant

c: Regional Administrator - III NRC
NRR Project Manager, NRC
Sr Resident Inspector, NRC
Minnesota Department of Commerce

Attachment

IE22

NRC FORM 366 (1-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4)
Failure to Comply with Technical Specification and ASME Code Section XI Inservice Inspection Requirements

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
01	19	2001	2001	- 002 -	00	02	19	2001	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)							
			20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
POWER LEVEL (10)	100		20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
			20.2203(a)(1)		50.36(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
			20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
			20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
			20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
			20.2203(a)(2)(v)	X	50.73(a)(2)(i)(B)	X	50.73(a)(2)(vii)		
			20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
			20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Patrick Burke	(763) 295-1661

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

CAUSE	SYSTEM	COMPONET	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE).		NO		03	29	01

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

On January 19, 2001, following a request by the NRC resident inspector for work documentation related to a snubber replacement on the High Pressure Coolant Injection (HPCI) system, the Monticello plant staff became aware that the requisite NIS-2 form had not been generated as required by paragraph IWA-7520 of the 1986 Edition of the ASME Code Section XI. Further investigation revealed repair and replacement plans and NIS-2 forms had not been generated for replacement activities involving other ASME Code Section XI snubbers and safety-relief valve topworks.

Monticello Technical Specification 3.15.A requires that components in quality groups A, B, and C (Class 1, 2, and 3) be declared inoperable if they do not comply with requirements of ASME Section XI.

On January 24, 2001, the Limiting Condition for Operation (LCO) described in Technical Specification Section 3.15.A was entered for the snubbers not meeting code requirements. The snubbers were considered inoperable. Technical Specification 3.6.H was entered which allows 72 hours to perform an engineering evaluation to demonstrate snubber acceptability. On January 25, 2001, it was determined the LCO should have been entered on January 19, 2001, when the problem was first identified.

On January 29, 2001, safety relief valve (SRV) topwork replacements were found to have been performed without complying with code requirements. Therefore, the SRVs were declared inoperable and Technical Specification LCO 3.6.E.2 was entered. A Notice of Enforcement Discretion was requested and verbal approval granted on January 30, 2001.

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

Description

On January 19, 2001, while operating at 100% power, following a request by the NRC resident inspector for work documentation related to a snubber[1] replacement on the High Pressure Coolant Injection (HPCI)[2] system, the Monticello plant staff became aware that the requisite NIS-2 form had not been generated as required by paragraph IWA-7520 of the 1986 Edition of the ASME Code Section XI. Further investigation revealed repair and replacement plans and NIS-2 forms had not been generated for replacement activities involving other ASME Code Section XI snubbers and safety-relief valve (SRV)[3,4] topworks. It was also determined that the Authorized Nuclear Inservice Inspector (ANII) was not given the opportunity for review and approval of the repair and replacement plans or NIS-2 forms in accordance with the ASME Code Section XI 1986 edition.

On January 24, 2001, the Limiting Condition for Operation (LCO) described in Technical Specification Section 3.15.A was entered for the snubbers not meeting code requirements. Monticello's Technical Specifications include an LCO that indicates that failure to comply with the requirements of ASME Code Section XI for quality groups A, B, and C (Class 1, 2, and 3) components renders the components inoperable. Since the snubbers were inoperable, Technical Specification 3.6.H was entered which allows 72 hours to complete an engineering evaluation to demonstrate snubber acceptability. On January 25, 2001, it was determined that the LCO should have been entered on January 19, 2001 when the problem was first identified. As a result, supported systems were promptly declared inoperable as required by Technical Specification 3.6.H and a plant shutdown was initiated to conform to the limiting Technical Specification requirement (i.e., be in a hot shutdown condition within 12 hours). Since the engineering evaluation which demonstrated snubber acceptability was completed prior to reaching hot shutdown, the shutdown was halted.

On January 29, 2001, SRV topwork replacements were found to have been performed without complying with code requirements. Therefore the SRVs were declared inoperable and Technical Specification LCO 3.6.E.2 was entered. A plant shutdown was initiated. A Notice of Enforcement Discretion was requested and verbal approval granted on January 30, 2001. The shutdown was halted after the plant was granted the request for enforcement discretion.

Other components have been found that do not meet code requirements for similar reasons. These components have been entered into the corrective action process and evaluated for operability in accordance with Generic Letter (GL) 91-18 and the Notice of Enforcement Discretion. This LER will be supplemented to discuss the reportability aspect of these other non-conformances in accordance with the reportability and Technical Specification requirements in effect at the time of discovery.

1 EIIS System Code = SPT
2 EIIS System Code = BJ
3 EIIS Code = TA
4 EIIS Code = RV

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

Event Analysis

Analysis of Reportability

This event is being reported as required by 10CFR50.73(a)(2)(i)(B) as a condition prohibited by Technical Specification 3.15.A and 3.6.E.2 for SRV topworks and 3.6.H for snubbers. These components were replaced, in violation of the ASME Code Section XI, without preparation of repair and replacement plans, NIS-2 forms and ANII involvement.

This event is being reported as required by 10 CFR50.73(a)(2)(vii) since the snubbers were declared inoperable, as required by Technical Specifications 3.15.A, and affected redundant systems were also considered inoperable. Subsequent analysis demonstrated the snubbers to be acceptable and the affected systems to be operable.

Additionally, this event is being reported as required by 10CFR50.73(a)(2)(i)(B) as a condition prohibited by Technical Specifications since the LCO for Section 3.6.H (snubbers) was entered on January 24, 2001 instead of on January 19, 2001 when the problem was first identified. Therefore, that LCO was not satisfied.

Other components have been identified after January 26, 2001 that have similar non-conformances with the ASME Code Section XI. This LER will be supplemented to discuss the reportability aspect of these other non-conformances in accordance with the reportability and Technical Specification requirements in effect at the time of discovery.

Safety Significance

The ANII involvement provides and documents third party review of technical and quality requirements of the code. The Monticello quality assurance, quality control and work control processes compensate for (but do not substitute for) the lack of ANII involvement in the repairs or replacements. Therefore, we believe that there is very low safety significance to this event.

The recently identified ASME Code Section XI non-conformances have been evaluated in accordance with NRC GL 91-18 to ensure that component operability is not adversely affected. Evaluation of operability per NRC GL 91-18 is consistent with practices for other instances of degraded or non-conforming conditions and is incorporated as an integral part of the Monticello Corrective Action Program.

A bounding quantitative probabilistic risk assessment (PRA) has been performed as a sensitivity study to show that the potential increase in risk associated with failure to involve the ANII as required by the ASME Code is small. The PRA analysts believe that the additional likelihood for failure of the SRVs to perform their functions is less than 1% over their currently assumed failure rate. A result of 1.46 E-05/yr core damage frequency (CDF) is obtained by assuming a 10% increase in the failure rates. This can be compared to a baseline CDF of 1.44 E-05/yr. This amounts to less than a 1.5% increase in CDF due to the exaggerated degradation in reliability of SRVs to perform their function. In conclusion, there is less than minimal increase in risk due to lack of the ANII involvement since the SRVs are able to perform their intended function. The sensitivity study shows that any potential increase in risk is very small.

Similar assessments for other non-conformances (where operability is demonstrated) would be expected to show similar results.

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

Cause

Interpreting ASME Code Section XI requirements often requires expert knowledge. Although the plant staff responsible for work planning has a good general knowledge of ASME Code Section XI requirements for repair and replacement activities for reactor pressure boundary components (e.g. pumps and valves), they lacked expertise to assure compliance for replacement of component supports and other components.

The work control documents did not provide sufficient direction for completing repair and replacement plans, NIS-2 forms, and obtaining ANII involvement.

The Technical Specifications were not literally followed when, on January 19, 2001, it was first realized, that ASME Code Section XI requirements for ISI were not fully met. Technical Specification LCO 3.15.A and 3.6.H should have been entered at that time.

An investigation team is currently in the process of determining the root cause of this event.

Corrective Actions

Plans have been developed to determine the full extent of condition for ASME Code Section XI non-compliance and to restore compliance. Independent ASME Code experts have completed a Self-Assessment. A root cause investigation team has been formed to identify the root cause. Findings and actions are being entered into the Corrective Action Program for disposition.

Management expectations have been reinforced regarding literal compliance with Technical Specifications.

Evaluations have been conducted which demonstrated that the affected components are operable.

Process changes, training and procedural improvements are being formulated.

A License Amendment Request, "Relocation of ASME Inservice Inspection Requirements to a Licensee Program" has been submitted on an exigent basis.

Failed Component Identification

none

Similar Events

LER 97-010, "Failure to Include Some Supports on the Reactor Head Vent Line in the ISI Program in the 2nd 10 Year Interval Due to Inaccurate Drawings and Failure to Report This Event in a Timely Manner Due to Personnel Error"

LER 97-004, "Failure to Submit Relief Requests for Limited Inservice Inspection Examinations"

The corrective actions for these LERs did not prevent this event because they did not focus on the lack of organizational knowledge regarding the AMSE Code Section XI non-conformances and the extent of condition. These two similar events were reported as a condition prohibited by Technical Specification 3.15.A. However, the affected systems were not declared inoperable as required by Technical Specification LCO 3.15.A. Since these events were discovered after January 23, 2001 and occurred over 3 years ago, additional reporting is not required.