NSP

Northern States Power Company

Monticello Nuclear Generating Plant
2807 West Hwy 75
Monticello, Minnesota 55362-9637

June 7, 1995

Report Required by
10 CFR Part 50, Section 50.73

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

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

LER 94-011, Revision 1
Inoperable Safety/Relief Valves Resulting in Violation of Plant Technical Specifications

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

Please contact Tom Parker at (612) 295-1014 if you require further information.

William J Hill

William J. Hill
Plant Manager
Monticello Nuclear Generating Plant

c: Regional Administrator - III NRC
Sr Resident Inspector, NRC
NRR Project Manager, NRC
State of Minnesota,
Attn: Kris Sanda

Attachment

IE22
/1

NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
MONTICELLO NUCLEAR GENERATING PLANT	05000 - 263	1 OF 6

TITLE (4)
Inoperable Safety/Relief Valves Resulting in Violation of Plant Technical Specifications

EVENT DATE (5)			LER NUMBER (6)			REPORT NUMBER (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
09	29	94	94	011	01	06	07	95	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	0%		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
			20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		below and in Text, NRC
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Tom Parker	612-295-1014

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
X	RV	BLL	T020	YES						

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	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)
NCR FORM 366 (5-91)

On September 29, 1994, during a refueling outage, it was discovered that potentially two Safety/Relief Valves were inoperable at the same time during the previous cycle. Technical Specification 3.6.E.1.a and 2.4.B require 7 of the 8 Safety/Relief Valves to be operable whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F.

On January 19, 1994, during normal plant operation, a bellows leaking alarm was received on the "H" Safety/Relief Valve (SRV) resulting in its self-actuation function being declared inoperable. On September 29, 1994 during the next refueling outage, ASME Section XI testing of the "D" Safety/Relief Valve determined its self-actuation pressure setpoint to be out of acceptance range. Thus, the self-actuation function of "D" Safety/Relief Valve was also inoperable for an undetermined portion of the previous operating cycle. Failure mechanism for "H" Safety/Relief Valve was determined to be a leaking bellows O-ring. The high lift pressures of "D" Safety/Relief Valve were attributed to an increase in the abutment gap between pilot stem and pilot disc. Both Safety/Relief Valves were replaced with spare units.

Description:

On September 29, 1994, during a refueling outage, it was discovered that potentially two Safety/Relief Valves were inoperable at the same time during the previous cycle. Technical Specification 3.6.E.1.a and 2.4.B require 7 of the 8 Safety/Relief Valves to be operable whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F.

On January 19, 1994, during normal plant operations, a bellows (EIIS Component Code: BLL) leaking alarm was received on the "H" Safety/Relief Valve (EIIS Component Code: RV). The self-actuation mode of this Safety/Relief Valve was declared inoperable on January 19, 1994, and the plant continued normal operation as allowed by Technical Specifications 3.6.E.1.a and 2.4.B.

On September 29, 1994, during the performance of valve bench checks required by Section XI of the ASME Code, the self-actuation, set pressure of the "D" Safety/Relief Valve was determined to be 1145 psig which exceeded the Technical Specification limit of:

1120 psig + 1% tolerance (See Technical Specification Bases 3.6.E), or
1120 psig + 11.2 psig = 1131.2 psig

This condition of the set pressure exceeding the allowed value by 14 psi indicates the self-actuation function of "D" Safety/Relief Valve was inoperable for an undetermined portion of the previous operating cycle, Cycle 16. This resulted is two Safety/Relief Valves potentailly being inoperable at the same time.

Cause:

Based on the results and response of the "D" Safety/Relief Valve (S/RV) during as-found testing, the high lifts could only be attributed to a change, defect, or malfunction of the valves pilot stage. Therefore, a root cause for the high lifts is not obvious. The following were considered in combination or in singular as possible causes for the high lifts:, pilot seat leakage, friction, sticking, wear, setspring adjusting nut movement, foreign matter, pilot disc-to-pilot stem mating surface irregularities, and pilot disc-to-pilot stem connection looseness.

The method used for determining S/RV lift points assumes that lift occurs at the point where a step increase in bellows movement is seen during test pressure ramp up since pilot lift can not be measured directly. Some interpretation of test data is required to find this point, especially if excessive pilot leakage is present since it can make the step increase point indistinguishable. However, the test results and procedure were reviewed and indicated that the step increases were distinct and pilot leakage appeared to be minimal. Thus, pilot seat leakage did not contribute to this event.

Increased friction or sticking between components in the pilot stage are not suspected as causes because there were no wear marks present and as-found lifts were consistently high (usually, the initial lift is only high if friction or sticking is present). Wear was ruled out as a cause because no evidence of wear was found and the pilot stage had not been operated since the previous as-left test.

Movement of the setspring adjusting nut during locking pin installation or from vibration were considered. However, the as-found lifts were 30 - 50 PSI higher than the previous as-left results. This would have required a tightening of the adjusting nut by about one full turn. Installation of the pin is witnessed by Quality Control personnel and no rotation of the adjusting nut is allowed during the installation. Regardless, there would have been no need to move the adjusting nut by this amount to install the pin because of the numerous pin holes in the adjusting nut. Tightening of the nut due to vibration could not have occurred because the locking pin prevents significant rotation of the nut.

The lifts were approximately 30 - 50 psi higher than the previous as-left results. The valve manufacturer has stated that the magnitude of this increase would correspond to any change in the pilot stage components that would result in an increase in bellows movement or abutment gap of about 1 mil. It is surmised that the high as-found lifts of the "D" Safety/Relief Valve were caused by an increase in the abutment gap between pilot stem and pilot disc. Pilot stage lift (S/RV setpoint) occurs when the abutment gap between the pilot disc and pilot stem are reduced from expansion of the pilot bellows due to increase in pressure and the pilot stem lifts the pilot disc off its seat. An increase in abutment gap could have been caused by pilot disc rotation in which uneven mating surface contact points changed during pilot disc rotation, by foreign matter on mating surfaces which was disengaged during subsequent inservice operation, or by a loose pilot disc-to-pilot stem connection.

Any unevenness that may have been present in the mating surfaces could nct be accurately measured due the small size of the imperfections and the design of the pilot stem and disc. A total indicated runout of 8 thousandths was measured on the pilot stem which could have contributed to uneven mating surface contact. A maximum runout of 10 thousandths is recommended by the manufacturer, but this value is only to assure adequate clearances between components. Foreign matter could have been present. Care is always taken to thoroughly clean the valves and prevent foreign matter from entering the valve prior to testing. However, minute particles may be present in the steam or come loose from the internal surfaces of the valves. The pin connection was inspected and did not appear to be loose but it would only have to be loose enough to cause a 1 mil change in axial movement of the stem.

We feel, based on the thorough inspection and evaluation performed, that one or more of the factors discussed above contributed to the increase in the pilot stem to pilot disc abutment gap, and the increased lift pressure of the S/RV. Methods and controls are in place to provide a high level of assurance that these factors will not have an adverse impact on S/RV lift setpoint. We feel that this is an isolated occurrence and not symptomatic of any recurring phenomenon.

Analysis:

This report is being submitted in accordance with 10 CFR Part 50, Section 50.73(a)(2)(i)(B).

It is unknown when the "D" Safety/Relief Valve set point shifted. The "D" Safety/Relief Valve was last proven to meet the set pressure requirement on July 22, 1991 during bench testing, and was installed in the Pressure Relief System in February 1993. If it is conservatively assumed that the "D" Safety/Relief Valve exceeded the self-actuation, set pressure requirement since the valve topworks was installed, then we have unknowingly been in non-compliance with the Technical Specifications since January 19, 1994, when the self actuation function of "H" Safety/Relief Valve was declared inoperable due to the bellows leak alarm.

The bellows leak was not by itself large enough to cause the Safety/Relief Valve to become inoperable. However, because the external side of the bellows is contained within the bellows leakage detection system boundary, back pressure from leakage reduces the differential pressure across the bellows thereby raising the system pressure that will open the Safety/Relief Valves. The Bellows Leakage alarm is set to alarm at 5 psig. This 5 psig backpressure will cause the Safety/Relief Valves to self-actuate at 5 psi higher than the set point. It is not known how high the back pressure became. Based on the time response of the alarm after venting off the pressure, the as found condition of the bellows leak detection line, and the maximum temperature attainable in the bellows leak detection line, engineering judgment suggests that the maximum pressure rise in the bellows leak detection line was well below 20 psig. During this outage's bench testing of Safety/Relief Valve "H", the "as found" setpoint was found to be 1111 psig. Therefore, the "H" Safety/Relief Valves probably would have been operable, i.e., it would have lifted below the Technical Specification limit of 1120 + 11.2 = 1131.2 psig. However, since we do not know the exact back pressure that the leak applied to the bellows assembly, we have conservatively considered the valve to be inoperable.

In addition to the self-actuation feature, the "H" valve is opened at 1052 +/- 3 psig by the Low-Low Set instrumentation. If this setpoint is exceeded following a reactor scram, the Low-Low Set logic opens the valve. This feature was unaffected by the

bellows leak and would have opened the valve below the self-actuation setpoint of 1120 psig. Similarly, the "D" Safety/Relief Valve was available for actuation by the Automatic Depressurization System logic, which is used to mitigate the small break Loss of Coolant Accident.

A bounding analysis previously performed by the Nuclear Analysis Department, which assumed that 3 Safety/Relief Valves were totally inoperable for self-actuation, demonstrated compliance with all vessel over pressure and Minimum Critical Power Ratio acceptance criteria. Other analyses were reviewed and all were found to be acceptable.

Based on the analysis presented above, it is concluded that there were no consequences that affected public health or safety from the concurrent failures of Safety/Relief Valves "D" and "H".

Corrective Action:

1. The topworks assemblies were replaced for Safety/Relief Valves "D" and "H" during the 1994 refueling outage.

2. Preventative maintenance Procedure, PM 4280-3 has been changed to require a detailed inspection of the metal O-ring seating surface for scratches, etc.

3. The pilot stage stem, disc, and preload spacer for the topworks removed from Safety/Relief Valve "D" were replaced. The topworks was then successfully as-left tested and placed into stock as a spare.

Additional Information:

Failed Component Identification:
Three stage Safety/Relief Valves
Manufactured by Target Rock Corporation
Part Number 67F, 6" inlet, 10" outlet

Previous Similar Event: None.