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

Operated by Nuclear Management Company LLC

May 25, 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 2001-009
Construction Error Results in Failure to Perform Periodic Testing of One Instrument Line Excess Flow Check Valve

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

Contact David Musolf, Consulting Production Engineer, at (763) 295-1201 if you require further information.

John Grubb For Jeff Forbes

Jeff S. Forbes
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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Monticello Nuclear Generating Plant	05000263	1 OF 5

TITLE (4)
Construction Error Results in Failure to Perform Periodic Testing of One Instrument Line Excess Flow Check Valve

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
03	28	2001	2001	– 009	– 0	05	25	2001	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	N
POWER LEVEL (10)	0

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)
	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)		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)
David Musolf	(763) 295-1201

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

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

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)

On March 28, 2001, a discrepancy was discovered between the Piping & Instrumentation Diagram (P&ID) and the as-built piping associated with the B reactor vessel fuel zone water level channel. The discrepancy was discovered while working on a design change to connect the two reactor vessel fuel zone instrument channels to upgraded reference columns. It was found that two instrument lines between instrument racks on elevations 935 feet and 962 feet in the Reactor Building had been crossed in a vertical pipe run during original plant construction. As a result, some instrumentation connected to containment penetration X-29A via excess flow check valve XFV-26 was actually connected to excess flow check valve XFV-57 and vice versa. This error presented no operability concerns for the affected instrumentation. However, a review of the procedure used to periodically test these valves revealed that the piping error prevented a valid test of XFV-57 from being performed. The B leg piping was rerouted to conform to the P&ID. XFV-57 was tested and was found to be operable. Technical Specification 4.7.D.1.b requires excess flow check valves to be tested every operating cycle. The fact that XFV-57 had not been properly tested since original plant startup constitutes a violation of this Technical Specification.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Monticello Nuclear Generating Plant	05000263	2001 –	009 –	00	2 OF 5

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

Description

On March 28, 2001, during a mid-cycle plant outage, installation of Modification 01Q075 was in progress to upgrade the fuel zone reactor vessel water level channels[1] to utilize the safeguards vessel level reference columns previously modified to be more resistant to reference leg "flashing."

During work on piping[2] associated with the B fuel zone channel, it was found that the as-built piping connected to vessel penetration X-29A was different than shown on the Piping & Instrumentation Diagram (P&ID) used in the modification package. This condition was confirmed by running demineralized water through the various isolated piping runs connected to penetration X-29A.

The P&ID shows instrument piping downstream of containment penetration X-29A splitting into two branches. The first branch connects to feedwater level instrumentation on panel C56 and fuel zone level and safety/relief valve (SRV) low-low set pressure switches[3] on panel C122 via excess flow check valve[4] XFV-26. The second branch connects to residual heat removal (RHR) permissive interlock pressure switches[5] on panel C122 via excess flow check valve XFV-57. Panel C56 is on elevation 962' and panel C122 is on elevation 935' in the Reactor Building. Refer to Figure 1.

Investigation found that the piping between C56 and C122 were rolled in the vertical pipe run between floors. This resulted in the RHR pressure switches actually being connected to XFV-26 and the fuel zone and SRV low low set pressure switches connected to XFV-57. The operability of instrumentation connected to XFV-26 and XFV-57 was not affected by the piping error since it made no difference which excess flow check valve supplied each instrument.

Review of the excess flow check valve surveillance test procedure showed that the low side instrument tap of feedwater level transmitter LT-6-52B was being used to bleed off pressure when testing XFV-26. The piping routing error did not prevent a valid test of XFV-26. The instrument tap for RHR permissive Interlock pressure switch PS-2-3-49B was used in the procedure to bleed off pressure when testing XFV-57. Because the RHR permissive interlock pressure switch was actually connected to XFV-26 due to the piping error, a valid test of XFV-57 was not performed and XFV-26 was actually being tested twice.

[1] EIIS System Code: IP
[2] EIIS Component Code: TBG
[3] EIIS Component Code: PS
[4] EIIS Component Code: FCV
[5] EIIS Component Code: PS

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

Modification 01Q075 was revised to include the restoration of the B instrumentation piping to conform to the original design specified on the P&ID.

Event Analysis

Analysis of Reportability

Technical Specification 4.7.D.1.b states: "At least once per operating cycle the primary system instrument line flow check valves shall be tested for proper operation." Also, Technical Specification 4.15.B.1 states: "Inservice testing of quality group A, B, and C pumps and valves shall be performed in accordance with the requirements for ASME Code Class 1, 2, and 3 pumps and valves, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g) except where relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55(a)(g)(6)(i), or where alternate testing is justified in accordance with Generic Letter 89-04."

10 CFR 50.73(a)(2)(i)(B) requires reporting of "any operation or condition which was prohibited by the plant's Technical Specifications except when ... (2) the event consisted solely of a case of a late surveillance test where the oversight was corrected...." It was conservatively determined that this event did not meet the criteria for a missed surveillance test because of the long period during which XFV-57 was not tested. The event is therefore being reported in accordance with Section 50.73(a)(2)(i)(B).

Safety Significance

We believe that this event has a low safety significance based on the following factors:

1. The instrument piping routing error did not result in the inoperability of any equipment.
2. Following discovery of this event, XFV-57 was removed and bench tested successfully.
3. An extent of condition evaluation revealed no other similar excess flow check valve P&ID discrepancies.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Monticello Nuclear Generating Plant	05000263	2001	– 009	– 00	4 OF 5

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

Cause

The piping associated with the A reactor vessel fuel zone level instrumentation was tested and found to match the reactor vessel instrumentation P&ID. A contact made with General Electric confirmed that the intended piping design was shown on the P&ID.

Field investigation confirmed that the true cause of the occurrence was a construction error resulting from "rolling" of the B leg piping in passing through the concrete structure of the Reactor Building between elevations 935' and 962'.

A review of all revisions to the P&IDs for piping in this area confirmed that no changes had been made since the original construction of the plant. It is believed that the error was made at that time.

Corrective Actions

The scope of Modification 01Q075 was expanded to include routing of the B reactor vessel instrumentation piping to eliminate the error caused by "rolling" of the piping between elevations 935' and 962'. Modification 01Q075 was satisfactorily completed on the B leg piping.

Demin water was injected at various instrument taps with the instruments and associated trips bypassed to confirm that the revisions to the B leg piping were correctly installed.

The configuration of the piping associated with the A reactor vessel fuel zone level instrumentation was tested and verified to match the P&ID. Modification 01Q075 was satisfactorily completed on the A leg piping.

An extent of condition evaluation was completed and no other similar excess flow check valve P&ID discrepancies were found. The results of this evaluation are document in a Condition Report.

Failed Component Identification

None

Similar Events

None

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Monticello Nuclear Generating Plant	05000263	2001 –	009 –	00	5 OF 5

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

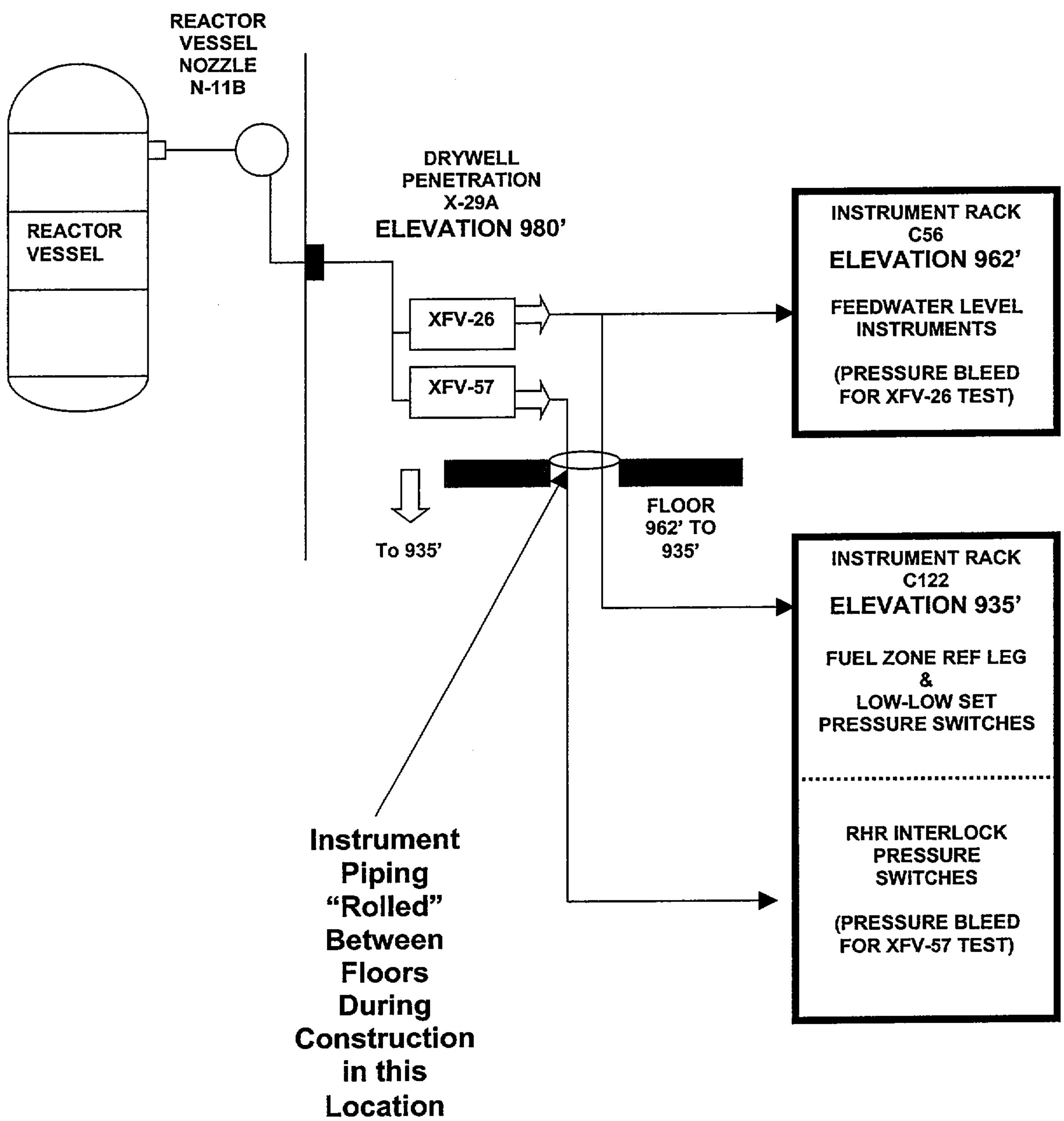

Figure 1
Simplified Drawing of As-Designed B Reactor Vessel Instrument Piping

Monticello Nuclear Generating Plant
Condition Report 20011860

Report Type: GEN GENERAL CONDITION
Task : APPROVE REPORT
CR Level: 1
Id Method : S Site-Self Identified

Title: Construction Error Results in Failure to Perform Periodic Testing of One Instrument Line Excess Flow Check Valve.

Topics:

Equipment:
XFV-57 RPV LVL/PRESS PEN X-29A EXCESS FLOW ISOL VLV

Systems:
RPV RX Pressure Vessel

	Source	Type		Number	Rev	Supplement
Doc Number :	MONTI	LER	-	200109	-	-

Disposition: RWORK REWORK (RETURNED TO ORIGINAL SPECS)
Project ID : 01Q075 FUEL ZONE LEVEL REFERENCE COLUMN MOD
Related WO : 0106899 Bench Test XFV-57 to meet Section XI and Tech Spec
Related RI :

Operability Eval: N **NRC Reportable :** Y **Unplanned LCO :** N
ASME XI : Y **Supplier Related:** N **Nuclear Network:** N
MRFF : N

Process Codes:
ECD EQUIP CONT - OPERATIONS OF EQUIP
STB SURVEILLANCE PROGRAM

Cause Codes:
B4B TECHNICAL INACCURACIES

Resp Group : CNSTM CONSTRUCTION MECHANICAL
SB/RB/KB : Rule Based

Actions: 1

Paper Attachments:

Doc Handler Attachments:
gen20011860.doc

Monticello Nuclear Generating Plant
Condition Report 20011860

Special Reviews:

OCR	SORENSEN, TERESA	OPERATIONS COMMITTEE REVIEW

Cross-Referenced Controlled Documents:

Cross-Referenced Issues:

Occurrence Date: 03/28/01
Due Date : 05/27/01

Initiator :	NUELK, RANDY L	**Date Initiated:**	03/28/01
Screener :	NUELK, RANDY L	**Date Screened :**	03/28/01
Supervisor:	ENGELKE, STEPHEN A	**Date Assigned :**	03/31/01
Assessor :	NUELK, RANDY L	**Date Assessed :**	04/30/01
Reviewer :	ENGELKE, STEPHEN A	**Date Reviewed :**	05/22/01
Approver :	GRUBB, JOHN C	**Date Approved :**	

Monticello Nuclear Generating Plant
Condition Report 20011860
Action 20012394

Action Type: APR PREVENT RECURRENCE
Task : PERFORM ACTION
Priority : 2

NRC Commitment : N
Request for Training: N
Outage Related : N

Topics:

Related Condition Report: 20011860 CR Type: GEN
Report Title: Construction Error Results in Failure to Perform Periodic Testing of One Instrument Line Excess Flow Check Valve.

Action to Perform:

Revise surveillance 0255-20-ID-1, Excess Flow Check Valve Test Procedure to assure XFV-57 is properly tested.

Completed Action:

Paper Attachments:

Doc Handler Attachments:

Special Reviews:

Equipment:

Cross-Referenced Controlled Documents:

Cross-Referenced Issues:

Action Date : 04/26/01
Action Due Date : 09/01/01
NRC Commitment Date:
Action Group : ELECT/I&C SYSTEM ENGINEERING

Proposed By : NUELK, RANDY L
Action Supervisor: ENGELKE, STEPHEN A
Action Performer : NUELK, RANDY L
Action Approver :

Date Proposed : 04/26/01
Date Assigned : 04/26/01
Date Performed:
Date Approved :

I. Initiation

Description

During Reactor Water Level modification 01Q075, the piping associated with Reference Leg 2-3-3B via penetration X-29A was found different than the P&ID on which the modification package was based. Running demin water through the various isolated piping runs proved that the plant configuration is different than the P&ID.

According to the P&ID, penetration X-29A splits to feed the feedwater level instruments (PI-2-3-60B, LT-6-52B, and PT-6-53A) on C56 and the fuel zone level instruments (LT-2-3-112A and LIS-2-3-73B) and the Low Low Set pressure instruments (PT-4067 A/B/C/D) on C122 via XFV-26 and the RHR Permissive Interlock pressure switches (PS-2-3-49B, PS-2-3-50B, and PS-2-3-53B) on C122 via XFV-57. C56 is on level 962' while C122 is on 935'.

It was found that the two pipe runs between C56 and C122 were rolled in the vertical run. The result is that the RHR pressure switches were actually connected to XFV-26 and the fuel zone and Low Low Set pressure switches were actually connected to XFV-57.

Review of surveillance 0255-20-ID-1, Excess Flow Check Valve Test Procedure, shows that the low side instrument tap of feedwater level transmitter LT-6-52B was used to test XFV-26. It shows that the instrument tap of RHR Permissive Interlock pressure switch PS-2-3-49B was used to test XFV-57. However, since the RHR Permissive Interlock pressure switches were actually connected to XFV-26, no checks of XFV-57 were done.

Technical Specification 4.7.D.1.b states: "At least once per operating cycle the primary system instrument line flow check valves shall be tested for proper operation." Also, 4.15.B.1 states: "Inservice testing of quality group A, B, and C pumps and valves shall be performed in accordance with the requirements for ASME Code Class 1, 2, and 3 pumps and valves, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10CFR50, Section 50.55a(g) except where relief has been granted by the Commission pursuant to 10CFR50, Section 50.55(a)(g)(6)(I), or where alternate testing is justified in accordance with Generic Letter 89-04."

Method of Discovery

During modification 01Q075 as described above.

Reactor Power 0% s/d

Shift Manager Notification

SM: Bruce MacKissock **Date** 3/28/01

Immediate Actions Taken

Review of surveillance 0255-20-IA-1 and P&IDs M-115, M-116, and M-116-1 and walkdown of piping. No other immediate actions since the safeguards, feedwater, and fuel zone instruments are isolated per the modification.

II. Screening

A. Prompt Operability Determination

		Yes	No
1.	Does the condition affect safety-related SSCs or SSCs covered by the Technical Specifications?	☒ Yes Go to II.A.2	☐ No Go to II.B
2.	Have the involved SSCs been declared inoperable?	☒ Yes Go to II.B	☐ No Go to II.A.3
3.	Are the involved SSCs CLEARLY operable?	☐ Yes Go to II.B	☐ No Go to II.A.4
4.	Is there REASONABLE EXPECTATION the SSCs are operable?	☐ Yes Perform an operability evaluation	☐ No Notify Operations and go to II.B
5.	Check this box to insert an operability evaluation.	☐	

B. Compliance with Technical Specifications

Since the plant is in cold shutdown, the check valves are not required to be operable. Therefore, we are not currently in violation of technical specifications. However, we were not in compliance with technical specifications 4.7.D.1.b and 4.15.B.1, as noted in the description, during past operation.

C. NRC 10CFR50.72 or State notification required

Not required.

D. Personnel Qualifications Affected

None.

E. Initial Extent of Condition Review

1. Identify any immediate actions necessary for other safety-related equipment as a result of this condition.

   It is possible that the same condition exists on the A reference leg. That leg is protected during work on the B reference leg. Determination must await completion of the B leg work.

   Testing on 3/30/01 showed that the configuration on A reference leg configuration matches the P&ID and therefore does not have the problem.

F. 10CFR Part 21 Screening

1. Is the equipment associated with this Condition Report safety related? ☒ Yes ☐ No
2. Does the equipment contain a defect, does it not meet purchase specifications or has there been a failure to comply with NRC regulations? ☐ Yes ☒ No

If both of the above = Yes, and this has not already been reported to the NRC, you should insert a Part 21 evaluation.

3. Has the NRC been formally notified of this issue? ☐ Yes ☒ No
4. Check this box to insert a Part 21 evaluation. ☐

III. Assign Assessor (Provide management expectations, person hours, etc.)

John Grubb is assigned as the management sponsor for this CR. John has determined that the condition requires an LER and that an investigation report is not necessary IAW 4AWI-10-01-05; the LER is considered the investigation report.

IV. Assessment

A. Detailed Description (if necessary to complement the Initiator's Description)

   Description is accurate.

B. Operability (for Safety-Related SSCs and SSCs covered by Tech Specs)

1. Changes to the Prompt Operability Determination (if any).

   None required; plant was in cold shutdown at the time of determination.

2. Identify any Inoperable SSCs and the length of time of inoperability (Include any safety significance in the section below).

   Removal and testing of the single excess flow check valve following the determination that it had not been previously tested showed that the valve operated properly and within specification. The conclusion is that no SSCs were ever inoperable.

C. Reportability

10CFR50.73(a)(2)(I)(B) states: "Any operation or condition which was prohibited by the plant's technical specifications except when (1) the TS is administrative in nature; (2) the event consisted solely of a case of a late surveillance test where the oversight was corrected, the test was performed, and the equipment was found to be capable of performing its specified safety functions; or (3) the TS was revised prior to discovery of the event such that the operation or condition was no longer prohibited at the time of discovery of the event." In NUREG-1022 Rev. 2 on page 33, it states "Reporting is not required if an event consists solely of a case of a late surveillance test where the oversight is corrected, the test is performed, and the equipment is found to be capable of performing its safety functions." A survey of NMC members resulted in essentially two recommendations. Most felt that the issue did not warrant an LER under the 50.73 rule while some felt that even if the event didn't meet the technical requirements for reporting, it would be a good idea to voluntarily submit a report. The plant has decided to submit an LER for the event.

D. Final Extent of Condition Review

1. Determine if this condition has generic implications on other equipment, systems or programs.

   Division A does not have the same problem. This was confirmed prior to the work on the A instrument leg.

   Review of walkdown drawings performed for seismic qualification of small bore piping for Project 88Z007 did not show any other significant piping or documentation problems (CR 20011840).

2. Scope of Review

   Demin water was injected at various instrument taps with the instruments and associated trips bypassed to determine that actual piping paths prior to the work on the A leg.

   Review of CR 20011840, Apparent Discrepancies on P&ID Lead to Connecting to Wrong Instrument Line.

E. Cause Determination

1. Evaluation

   The cause of this occurrence was an original construction error that resulted in a difference between the instrument lines in the plant and the P&ID. The first presumption was that the P&ID was incorrect. However, after further analysis of other drawings and the results of testing on the A leg and after conversations with GE on plant design, it is concluded that the P&ID shows the piping the way it was originally designed. The actual pipe connections on the B leg were "rolled" between elevations 935' and 962'. A review of all revisions of the P&IDs also shows that piping in this area had not been changed. Thus, it is concluded that this condition has existed since the plant was constructed.

2. Scope of Review

Review of all revisions of the applicable P&ID.

Walkdowns of the actual pipe runs which illustrate the difficulties encountered when the piping trays traverse concrete structures.

Analysis of surveillance 0255-20-ID-1, Excess Flow Check Valve Test Procedure to determine which valves may have gone unchecked due to the discrepancies.

Review of CR 20011840, Apparent Discrepancies on P&ID Lead to Connecting to Wrong Instrument Line.

F. Safety Significance

Since the affected excess flow check valve tested successfully, the safety significance of missing previous tests is negligible. Since the extend of condition evaluation did not identify any other similar drawing or P&ID discrepancies, the safety significance is none.

G. Actions Taken

Excess Flow Check Valve XFV-57 was detached and bench tested successfully (WO 0106899).

APR 20012394 written to revise surveillance 0255-20-ID-1, Excess Flow Check Valve Test Procedure to assure XFV-57 is properly tested.

H. Maintenance Rule Functional Failure ☐ Yes ☒ No

V. Due Date Extensions

Old Due Date	New Due Date	Date that the Due Date was Changed	Reason for the Change	Who Authorized the Change
4/27/01	5/29/01	4/30/01	To match the due date of the associated LER	S A Engelke

VI. Review Comments (if any)

VII. Approver Comments (if any)

VIII. Other Comments

FINAL REVIEW AT OC MTG 2269 ON 5/23/01.