



**UNITED STATES  
NUCLEAR REGULATORY COMMISSION**

REGION III  
2443 WARRENVILLE ROAD, SUITE 210  
LISLE, ILLINOIS 60532-4352

October 16, 2019

Mr. Christopher Church  
Site Vice President  
Monticello Nuclear Generating Plant  
Northern States Power Company, Minnesota  
2807 West County Road 75  
Monticello, MN 55362-9637

**SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT—TRIENNIAL INSPECTION  
OF EVALUATION OF CHANGES, TESTS AND EXPERIMENTS BASELINE  
INSPECTION REPORT 05000263/2019011**

Dear Mr. Church:

On September 4, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Monticello Nuclear Generating Plant and discussed the results of this inspection with you and other members of your staff. The results of this inspection are documented in the enclosed report.

One Severity Level IV violation without an associated finding is documented in this report. We are treating this violation as a Non-Cited Violation (NCV) consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violation or significance or severity of the violation documented in this inspection report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement; and the NRC Resident Inspector at Monticello Nuclear Generating Plant.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at <http://www.nrc.gov/reading-rm/adams.html> and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

***/RA/***

Robert C. Daley, Chief  
Engineering Branch 3  
Division of Reactor Safety

Docket No. 05000263  
License No. DPR-22

Enclosure:  
As stated

cc: Distribution via LISTSERV®

Letter to Christopher Church from Robert C. Daley dated October 16, 2019

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT—TRIENNIAL INSPECTION  
 OF EVALUATION OF CHANGES, TESTS AND EXPERIMENTS BASELINE  
 INSPECTION REPORT 05000263/2019011 DATED OCTOBER 16, 2019

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**U.S. NUCLEAR REGULATORY COMMISSION  
Inspection Report**

Docket Number: 05000263

License Number: DPR-22

Report Number: 05000263/2019011

Enterprise Identifier: I-2019-011-0040

Licensee: Northern States Power Company - Minnesota

Facility: Monticello Nuclear Generating Plant

Location: Monticello, MN

Inspection Dates: June 3, 2019, to June 7, 2019

Inspectors: M. Holmberg, Senior Reactor Inspector  
J. Robbins, Reactor Inspector  
D. Szwarc, Senior Reactor Inspector

Approved By: Robert C. Daley, Chief  
Engineering Branch 3  
Division of Reactor Safety

## SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) continued monitoring the licensee's performance by conducting a triennial inspection of evaluation of changes, tests and experiments baseline inspection at Monticello Nuclear Generating Plant in accordance with the Reactor Oversight Process. The Reactor Oversight Process is the NRC's program for overseeing the safe operation of commercial nuclear power reactors. Refer to <https://www.nrc.gov/reactors/operating/oversight.html> for more information.

### List of Findings and Violations

Failure to Report Changes in Peak Clad Temperature for Postulated Loss-of-Coolant Accidents			
Cornerstone	Significance	Cross-Cutting Aspect	Report Section
Not Applicable	NCV 05000263/2019011-01 Open/Closed	Not Applicable	71111.17T
The inspectors identified a Severity Level IV Violation of Title 10 of the <i>Code of Federal Regulations</i> (CFR) 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Lightwater Nuclear Power Reactors" for the licensee's failure to report to the NRC changes in the limiting peak clad temperature (PCT) for postulated loss-of-coolant- accident (LOCA) scenarios identified in revision 2 of calculation 13-055 "Core Spray and LPCI Flow Delivered to Reactor Vessel for Safety Analysis."			

### Additional Tracking Items

None.

## INSPECTION SCOPES

Inspections were conducted using the appropriate portions of the inspection procedures (IPs) in effect at the beginning of the inspection unless otherwise noted. Currently approved IPs with their attached revision histories are located on the public website at <http://www.nrc.gov/reading-rm/doc-collections/insp-manual/inspection-procedure/index.html>. Samples were declared complete when the IP requirements most appropriate to the inspection activity were met consistent with Inspection Manual Chapter (IMC) 2515, "Light-Water Reactor Inspection Program - Operations Phase." The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel to assess licensee performance and compliance with Commission rules and regulations, license conditions, site procedures, and standards.

## REACTOR SAFETY

### 71111.17T - Evaluations of Changes, Tests, and Experiments

#### Sample Selection (IP Section 02.01) (24 Samples)

The inspectors reviewed the following evaluations, screenings, and/or applicability determinations for Title 10 of the *Code of Federal Regulations* (CFR) 50.59:

- (1) SCR-10-0143, Implement the Crossflow Ultrasonic Flow Measurement System, Revision 2
- (2) SCR-14-0147, Issue 120V Distribution Panel Voltage Drop Analysis, Revision 0
- (3) SCR-15-0131, Operation with HPCI Steam Line Drain Trap Bypass (CV-2043) in the Open Position and Annunciator 3-B-10 Non-Functional, Revision 2
- (4) SCR-15-0378 (Evaluation), Replace Standby Gas Treatment Controllers to Improve Reliability, Revision 1
- (5) SCR-15-0438, Freeze Seal for SW-37 Replacement, Revision 0
- (6) SCR-16-0043 (Evaluation), ECCS-LOCA Analysis Input Changes, Revision 0
- (7) SCR-16-0076, Incorporate Updated MOV DC Curves into MOV Calculations, Revision 1
- (8) SCR-16-0096, Add Steps to Reduce Impact on LS-23-90 when Unisolating HPCI Steam Lines, Revision 0
- (9) SCR-16-0119, Owners Acceptance of Zachery Nuclear Core Spray and Residual Heat Removal (RHR) Piping, Revision 0
- (10) SCR-16-0162, Revise USAR 12.02 for Clarification and to Better Align with NEI 08-05 FSAR Update Guidance, Revision 1
- (11) SCR-16-0169, Evaluate Refuel Floor Loading when DSC Fuel Cask Staged on Floor, Revision 0
- (12) SCR-16-0199, Operate System with Increased Levels in Tank, Revision 2
- (13) SCR-16-0213, Turbine Bypass Valve Technical Specification Bases Clarification, Revision 1
- (14) SCR-16-0307, Internal Flooding Analysis, Revision 0
- (15) SCR-16-0354, 13 RHR Pump Motor Cooler Flow Rate During Surveillance Testing, Revision 0
- (16) SCR-16-0377, Replace HPCI Level Switch LS-23-98 Topworks and Disable Auto Pumpdown Feature, Revision 1
- (17) SCR-17-0054, Evaluation of Localized Thinning for 2017 Essential Service Water (ESW) Piping Inspections, Revision 0

- (18) SCR-17-0147, Revision to USAR Section 8.8.7 Cables, Revision 0
- (19) SCR-17-0154, PS17-3" HELB Crack/Break Elimination, Revision 0
- (20) SCR-17-034 (Evaluation), Increase GE14 Nuclear Fuel from 35 to 37 GWD/MTU for Core Average End of Cycle Exposure for Radiological Consequences Evaluations, Revision 0
- (21) SCR-18-0090, HPCI Room Transient Temperature, Revision 0
- (22) SCR-18-0132, Outboard C MSIV Fast Closure, Revision 0
- (23) SCR-18-0146, Monticello Calculation 04-133, Outboard Main Steam Isolation Valve Calculation Update, Revision 0
- (24) SCR-19-0021, Revise Division 2 RHRSW Pump and Valve Tests, Revision 0

**INSPECTION RESULTS**

Failure to Report Changes in Peak Clad Temperature for Postulated Loss-of-Coolant Accidents			
Cornerstone	Severity	Cross-Cutting Aspect	Report Section
Not Applicable	Severity Level IV NCV 05000263/2019011-01 Open/Closed	Not Applicable	71111.17T
<p>The inspectors identified a Severity Level IV Violation of Title 10 of the <i>Code of Federal Regulations</i> (CFR) 50.46 "Acceptance Criteria for Emergency Core Cooling Systems for Lightwater Nuclear Power Reactors" for the licensee's failure to report to the NRC changes in the limiting peak clad temperature (PCT) for postulated loss-of-coolant- accident (LOCA) scenarios identified in revision 2 of calculation 13-055 "Core Spray and LPCI Flow Delivered to Reactor Vessel for Safety Analysis."</p> <p><u>Description:</u> On February 12, 2016, the licensee issued revision 0A of calculation 1180, "ECCS [Emergency Core Cooling System]-LOCA [Loss-of-Coolant-Accident] SAFER/GESTR," to accept vendor General Electric Hitachi (GEH) records as site quality assurance records including GEH analysis 0000-0163-2998-R0 "Monticello ECCS LOCA Evaluation for Modified Low Pressure ECCS Injection Performance Curves (LPCS [Low Pressure Core Spray] and LPCI [Low Pressure Coolant Injection])."</p> <p>On April 13, 2016, the licensee issued Revision 2 to calculation 13-055, "Core Spray and LPCI Flow Delivered to Reactor Vessel for Safety Analysis," which provided LPCS and LPCI reactor vessel injection flow versus reactor vessel-to-torus differential pressure curves suitable for use as replacements for those used in current safety analyses and as new inputs in future safety analyses (applicable to GEH 14C fuel in service at the Monticello Nuclear Generating Plant). In calculation 13-055, the licensee concluded that the GEH supplemental analysis 0000-0163-2998-R0 demonstrated improved run out flow for LPCS and resulted in acceptable margins for PCT in design basis LOCA and small break LOCA scenarios. Additionally, in calculation 13-055 and GEH analysis 0000-0163-2998-R0, the licensee identified various scenarios that changed PCT by more than 50 °F for the limiting LOCA transients from that recorded for the last acceptable model, but none of the PCT changes resulted in exceeding the PCT limit of 2200°F identified in 10 CFR 50.46. Subsequently, the licensee confirmed that in 2016, the revised LPCS pump flow parameters were selected and incorporated into Updated Final Safety Analysis Report Table 14.7-8, "Core Spray System Parameters" that were consistent with Case 2 from the GEH supplemental analysis 0000-0163-2998-R0 and which resulted in a -14 °F change in PCT.</p>			

NRC regulation 10 CFR 50.46 identifies the acceptance criteria for emergency core cooling systems in lightwater nuclear power reactors and requires that cooling system performance be calculated in accordance with an acceptable evaluation model for a number of postulated LOCAs of different sizes, locations, and other properties to ensure the most severe postulated LOCA is evaluated. This regulation also requires that for each change or error discovered in the evaluation model or in the application of such a model, the effect on the limiting ECCS analysis, be reported to the NRC at least annually. In this case, the inspectors identified that the licensee had made changes associated with application of the ECCS model and had not reported this result in an annual report to the NRC. Specifically, in Revision 2 of calculation 13-055, the licensee adopted revised LPCS pump performance curves for use in ECCS scenarios that impacted the limiting PCT analysis and did not report this information to the NRC.

Corrective Actions: The licensee entered the issue in their corrective action system on June 4, 2019.

Corrective Action References: 501000028099, "50.59 missed 10 CFR 50.46 report"

Performance Assessment: The inspectors determined this violation was associated with a minor performance deficiency. Consistent with the guidance in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors answered "no" to the more-than-minor screening questions. The inspectors also reviewed the examples of minor issues in IMC 0612, Appendix E, "Examples of Minor Issues" and found no examples related to this issue.

Enforcement: The Reactor Oversight Process (ROP)'s significance determination process does not specifically consider the regulatory process impact in its assessment of licensee performance. Therefore, it is necessary to address this violation which impedes the NRC's ability to regulate using traditional enforcement to adequately deter non-compliance.

Severity: In accordance with examples included in Section 6.9.d of the NRC Enforcement Policy, this issue was screened as a Severity Level IV Violation.

Violation: Title 10 CFR 50.46 "Acceptance Criteria for Emergency Core Cooling Systems for Lightwater Nuclear Power Reactors"

Title 10 CFR 50.46a(3) (ii) requires in part, for each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the licensee, shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in 10 CFR 50.4 or 10 CFR 52.3 of this chapter, as applicable.

Contrary to the above, between April 13, 2016, and September 4, 2019, the licensee failed to report the nature of a change in the application of a model that affects the temperature calculation, and its estimated effect on the limiting ECCS analysis to the Commission at least annually. Specifically, on April 13, 2016, in Revision 2 of calculation 13-055, the licensee evaluated and accepted modified ECCS injection pump performance curves that resulted in a -14F change to the PCT and failed to report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission.

Enforcement Action: This violation is being treated as a non-cited violation, consistent with Section 2.3.2 of the Enforcement Policy.



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## **EXIT MEETINGS AND DEBRIEFS**

The inspectors confirmed that proprietary information was controlled to protect from public disclosure.

- On September 4, 2019, the inspectors presented the triennial inspection of evaluation of changes, tests and experiments baseline inspection results to Mr. Christopher Church, and other members of the licensee staff.
- On June 7, 2019, the inspectors presented the Interim Exit inspection results to Mr. Don Barker and other members of the licensee staff.

## DOCUMENTS REVIEWED

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
71111.17T	Calculations	01-174	Minimum Required RHRSW Pressure at RHR Heat Exchanger	3
		07-045	RHR Pump Motor Model 5K511DT5410 Cooling Coil Minimum Flow Evaluation	2A
		13-055	Core Spray and LPCI Flow Delivered to Reactor Vessel For Safety Analyses	2
		14-014	Distribution Panel Y20 Voltage Drop Analysis	0
		16-072	480V Coordination Study	0
		Calculation 11-180	ECCS-LOCA SAFER/GESTR	0A
	Corrective Action Documents	500000087543	50.59 - Formal Evaluation for HPCI Drain Line	03/16/2016
		500001516105	50.59 - HPCI SR Test Inconsistent with TS Bases	03/17/2016
		500001560841	Cables not protected from fault on the circuit	06/12/2017
		501000004050	EC28685 missing required 50.59 screening	10/19/2017
		501000010972	50.59 Screening Graded a 1 at QRT	04/19/2018
		501000011895	50.59 AD Need Revision	05/11/2018
		501000014807	Outboard MSIV Calculation 04-133 Issues	07/25/2018
	Corrective Action Documents Resulting from Inspection	601000000116	PS17-3" HELB Cracks/Breaks Elimination	08/22/2017
		501000028099	19-50.59-Missed 10 CFR 50.46 Report	06/04/2019
		501000028218	19-50.59-Piping Corrosion Allowance Doc	06/06/2019
		501000028233	19-50.59-Typo in Superseded Calculation 14-014	06/06/2019
		501000028239	19-50.59 Replaced ESW pipe below Calculated Tmin	06/06/2019
		501000028260	19-50.59 Pipe Replaced after Calculation Remaining Service Life	06/06/2019
	Drawings	501000028261	19-50.59 SW-MIC Service Life Calculation	06/06/2019
		NH-36249	P&ID High Pressure Coolant Injection System (Steam Side)	84
		NX-13142-43	RCIC Primary Steam,	6
	Engineering Changes	25889	OPERATING WITH HPCI CV-2043 (STEAM TRAP BYPASS) OPEN	000
		26866	EVALUATION OF IMPACT OF HPCI DRAIN LINE BYPASS FLOW TO CONDENSER	000

Inspection Procedure	Type	Designation	Description or Title	Revision or Date
		27905	REPLACE HPCI LEVEL SWITCH LS-23-98 TOPWORKS AND DISABLE AUTO PUMP DOWN FEATURE	0
	Engineering Evaluations	0000-0163-2998-R0	Monticello ECCS LOCA Evaluation for Modified Low Pressure ECCS Injection Performance Curves (LPCS and LPCI)	0
		608000000236	MSIV Steam Leakage Evaluation	0
		SCR-15-0408	Replace RHR Pressure Switches PS-10-101A/B/C/D	1
		SCR-17-189	Replace DPI-2994C and DPI-2994D	0
	Miscellaneous	3830-17-041	Fire Protection Change Review for PCR 01550124	4
		Regulatory Issue Summary 2007-24	NRC Staff Position on use of the Westinghouse Crossflow Ultrasonic Flow Meter for Power Uprate OR Power Recovery	09/27/2007
	Procedures	0255-05-IA-1-2	"B" RHRSW QUARTERLY PUMP AND VALVE TESTS	87
		0255-05-III-2A	COMPREHENSIVE 12 RHRSW PUMP AND VALVE TESTS	33
		C.6-003-B-02	HPCI TURBINE EXH HI DRAIN POT LEVEL	5
		C.6-003-B-10	HPCI TURBINE INLET HI DRAIN POT LEVEL	9