U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Reports No. 50-266/92016(DRP);
50-301/92016(DRP)

Docket Nos. 50-266; 50-301

Licenses No. DPR-24; DPR-27

Licensee: Wisconsin Electric Company
231 West Michigan
Milwaukee, WI 53201

Meeting Conducted: July 7, 1992

Meeting At: Region III Office, Glen Ellyn, Illinois

Type of Meeting: Enforcement Conference

Inspection Conducted: Onsite May 27 through June 14, 1992
In-office review of additional material on
July 7-8, 1992

Inspectors: K. R. Jury
J. Gadzala

Approved By: I N Jackiw
I. N. Jackiw, Chief
Reactor Projects Section 3A

7-10-92
Date

Meeting Summary

Enforcement Conference on July 7, 1992, (Reports No. 50-266/92016(DRP); No. 50-301/92016(DRP))

Areas Discussed: A review of the apparent violations and areas of concern identified during the inspection, and corrective actions taken or planned by the licensee. The enforcement options pertaining to the apparent violations were also discussed with the licensee.

DETAILS

1. Persons Present at the Conference

Wisconsin Electric Power Company (WEPCO)

R. E. Link, Vice-President Nuclear Power
G. J. Maxfield, Plant Manager - Point Beach
J. C. Reisenbuechler, Manager - Operations & Technical Support
N. L. Hoefert, Manager - Operations
D. F. Johnson, Manager - Regulatory Affairs
J. F. Becka, Manager - Regulatory Services
G. Morin, Nuclear Engineer, Regulatory Affairs
M. Baumann, Project Engineer, Nuclear Regulation

U. S. Nuclear Regulatory Commission, Region III

A. B. Davis, Regional Administrator
W. L. Forney, Deputy Director, Division of Reactor Projects
J. N. Hannon, Director, Project Directorate III-3, NRR
R. W. DeFayette, Director, Enforcement and Investigation Coordination Staff
L. R. Greger, Chief, Reactor Projects Branch 3
R. B. Samworth, Licensing Project Manager, NRR
I. N. Jackiw, Chief, Reactor Projects Section 3A
K. R. Jury, Senior Resident Inspector, Point Beach
C. H. Weil, Enforcement Specialist
B. A. Berson, Regional Counsel

2. Enforcement Conference

An enforcement conference was held in the NRC Region III office on July 7, 1992. This conference was conducted as a result of the preliminary findings of the inspection conducted on May 27 through June 14, 1992, in which apparent violations of NRC regulations and license conditions were identified. The findings relate to the circumstances surrounding the Unit 1 overcooling event that occurred on May 27, 1992. Inspection findings are documented in Inspection Reports No. 50-266/92014; 50-301/92014, transmitted to the licensee by letter dated June 30, 1992.

The purpose of this conference was to (1) discuss the apparent violations, causes, and the licensee's corrective actions; (2) discuss several areas of concern; (3) determine if there were any escalating or mitigating circumstances; and (4) obtain any information which would help determine the appropriate enforcement action.

The licensee's representatives did not contest any of the apparent violations and were in agreement with the NRC's understanding of the areas of concern.

The licensee's representatives described the events which led to the violations, including root causes and corrective actions taken. In summary the immediate corrective actions were (1) discontinue crevice flushing activities during the rest of the refueling outage; (2) perform an engineering evaluation to determine the effects of the thermal transient on the structural integrity of the limiting region of the reactor vessel; (3) commit to not pressurize above the Low Temperature Overpressure Protection limit until completion and review of the engineering evaluation; (4) review the previous Unit 2 evolution to determine if a similar transient occurred; (5) commit to revise the procedure, clarify heatup and cooldown administrative limits, evaluate joint technical review of multi-disciplinary procedures, add heatup and cooldown rate alarms, improve cooldown rate indication, and conduct additional operator training.

At the conclusion of the meeting, the licensee was informed that they would be notified in the near future of the final enforcement action.

Attachment: Licensee Presentation Slides

WISCONSIN ELECTRIC POWER COMPANY
MANAGEMENT MEETING WITH NRC
July 7, 1992

ENFORCEMENT CONFERENCE
AGENDA

INTRODUCTION	Greg Maxfield
Description of SG Crevice Cleaning Evolution	Norm Hoefert
Unit 2 SG Crevice Cleaning Securing Decay Heat Removal	Norm Hoefert
Unit 1 SG Crevice Cleaning Excessive Cooldown	Norm Hoefert
Engineering Analysis Excessive Cooldown	Doug Johnson
Root Causes & Contributing Factors	Jim Reisenbuechler
Corrective Actions	Jim Reisenbuechler
SUMMARY	Greg Maxfield

EA 92-120

DESCRIPTION OF STEAM GENERATOR CREVICE CLEANING

Hot Soak

Fill steam generators to 80"

Heat up to 325-335°F using RCPs

Open atmospheric dump valves and boil for 30 minutes, maintaining level >60"

Shut atmospherics. Secure 1 RCP. Cool down to 175-190°F

Drain steam generators

Cleaning Cycle

Fill steam generators to 24-30"

Heat up to 290-300°F using RCPs

Secure RCPs

Minimize cooldown by operating 1 RHR pump and bypassing RHR heat exchanger using flow control valves

Fully open both atmospheric dump valves and boil for 60 minutes. Add water to keep tubesheet from drying out

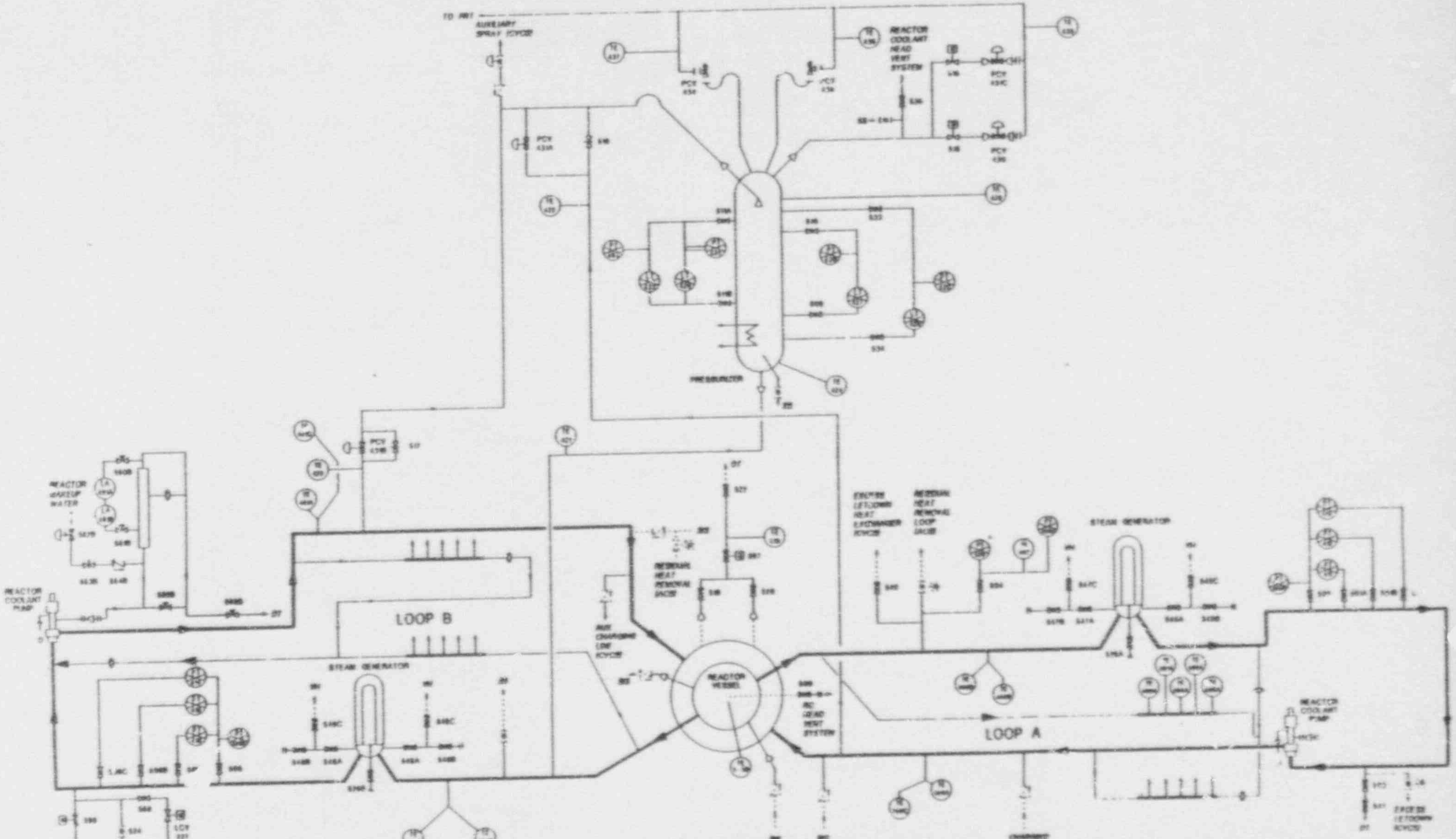
Start 1 RCP

Cool down to 175-190°F

Drain steam generators

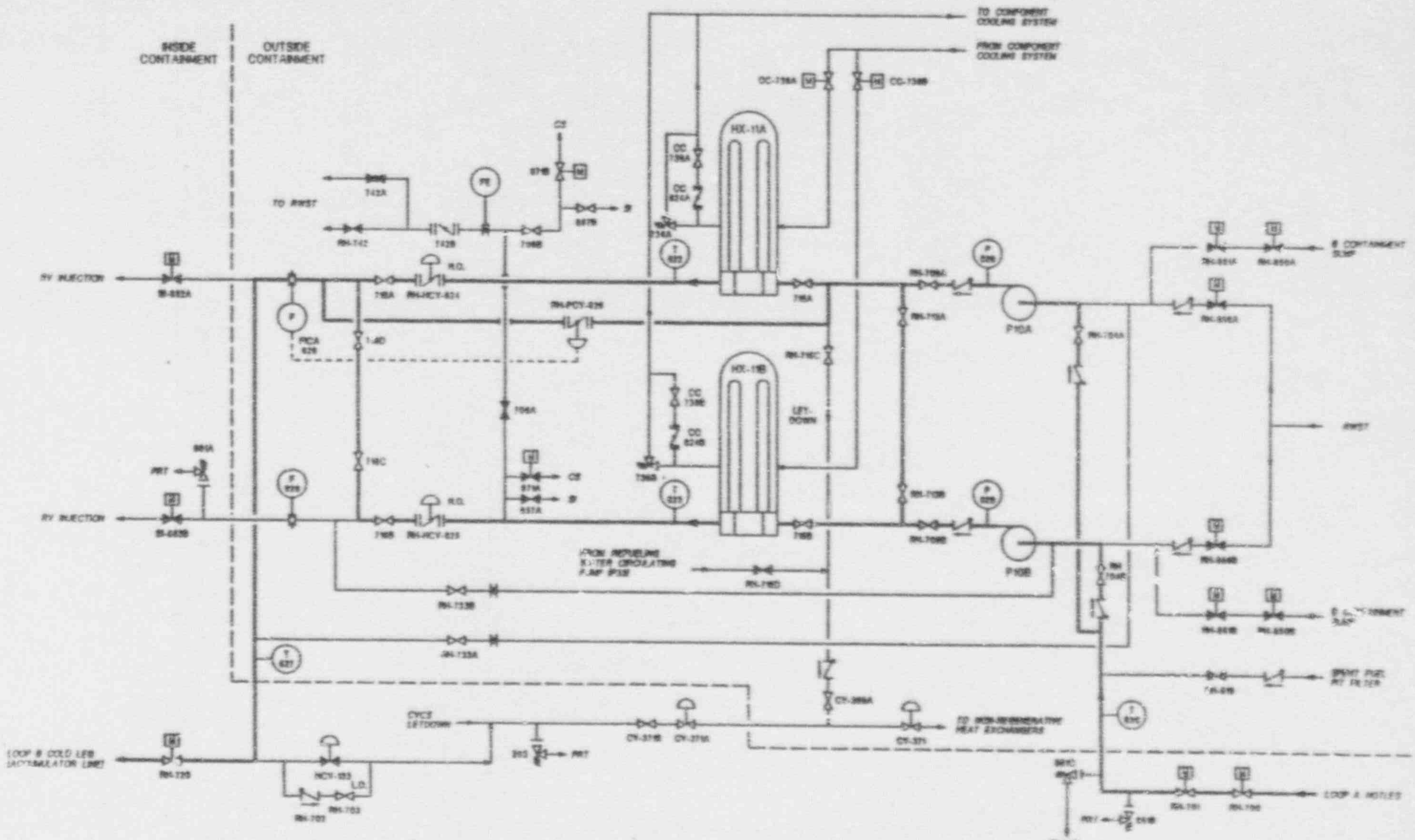
UNITS 1 AND 2
CREVICE CLEANING METHOD HISTORY

Year	Unit	Method/Procedure	Temp ~°F
1992	1	RP-6B (Hot)	300
1991	1	RP-6A (Vacuum)	195
	2	WMTP 11.19 (Hot)	300
1990	1	RP-6A (Vacuum)	195
	2	RP-6A (Vacuum)	195
1989	1	RP-6A (Vacuum)	195
	2	RP-6A (Vacuum)	195
1988	1	WMTP 11.19 (Hot)	250
	2	RP-6A (Vacuum)	195
1987	1	WMTP 11.19 (Hot)	250
	2	WMTP 11.19.2 (Vacuum)	195
1986	2	WMTP 11.19 (Hot)	250



REACTOR COOLANT SYSTEM FLOW DIAGRAM

FOR INFORMATION ONLY



FOR INFORMATION ONLY

RESIDUAL HEAT REMOVAL SYSTEM

UNIT 2 CREVICE CLEANING SECURING DECAY HEAT REMOVAL

- Performed on November 10, 1991, per Wisconsin Michigan Test Procedure WMTP 11.19, Revision 7, "Steam Generator Crevice Cleaning" (predecessor to RP-6B, "Steam Generator Crevice Cleaning").

- WMTP 11.19, Revision 7 had increased temperature range from 200-250°F to 175-300°F for more effective cleaning per EPRI study. Did not include "hot soak" subsequently incorporated into RP-6B.

- Operators controlled cooldown by different methods, including throttling component cooling to RHR heat exchangers; throttling or shutting RHR heat exchanger manual valves and shutting off RHR pump(s).

- TS states, "At least one of the above decay heat removal methods shall be in operation except when required to be secured for testing."

- Using this TS, allowance to temporarily secure both RHR pumps, the cooling action of the crevice cleaning, and the ability to quickly restore RHR flow by turning a control switch, operators concluded that shutting off both RHR pumps was an <u>allowed</u> and <u>prudent</u> action.

- Six cleaning cycles performed on Unit 2. Approximate cooldown rate was 55°F/hr; most cycles <50°F/hr.

- Requirement to keep 1 RHR pump operating at all times was included in RP-6B because it was conservative for TS compliance.

15.3 LIMITING CONDITIONS FOR OPERATION

15.3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps*

a. When the reactor is critical, except for tests, at least one reactor coolant pump shall be in operation.

(1) Reactor power shall not be maintained above 3.5% of rated power unless both reactor coolant pumps are in operation.

(2) If either reactor coolant pump ceases operating, immediate power reduction shall be initiated under administrative control as necessary to reduce power to less than 3.5% of rated power.

(3) If both reactor coolant pumps cease operating and power is greater than 3.5% of rated power, but less than 10% of rated power, reactor shutdown will commence immediately and verify the reactor trip breakers are opened within one hour.

b. When the reactor is subcritical and the average reactor coolant temperature is greater than 350°F, except for tests, at least one reactor coolant pump shall be in operation.

(1) Both reactor coolant pumps may be deenergized provided:

a. No operations are permitted that would cause dilution of the reactor coolant system boron concentration,

b. Core outlet temperature is maintained at least 10°F below saturation temperature, and

c. The reactor trip breakers are open.

c. At least one reactor coolant pump or residual heat removal system shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

2. Steam Generator*

a. One steam generator shall be operable whenever the average reactor coolant temperature is above 350°F.

3. Components Required for Redundant Decay Heat Removal Capability*

a. Reactor coolant temperature less than 350°F and greater than 140°F.

(1) At least two of the decay heat removal methods listed shall be operable.

(a) Reactor Coolant Loop A, its associated steam generator and either reactor coolant pump

(b) Reactor Coolant Loop B, its associated steam generator and either reactor coolant pump

* Applicable only when one or more fuel assemblies are in the reactor vessel.

(c) Residual Heat Removal Loop (A)*

(d) Residual Heat Removal Loop (B)*

(2) If the conditions of specification (1) above cannot be met, corrective action to return a second decay heat removal method to operable status as soon as possible shall be initiated immediately

(3) At least one of the above decay heat removal methods shall be in operation except when required to be secured for testing.

(4) If no decay heat removal method is in operation, all operations causing an increase in the reactor decay heat load or a reduction in reactor coolant system boron concentration shall be suspended. Corrective actions to return a decay heat removal method to operation shall be initiated immediately.

b. Reactor Coolant Temperature Less Than 140°F

(1) Both residual heat removal loops shall be operable except as permitted in items (3) or (4) below.

(2) If no residual heat removal loop is in operation, all operations causing an increase in the reactor decay heat load or a reduction in reactor coolant system boron concentration shall be suspended. Corrective actions to return a decay heat removal method to operation shall be initiated immediately.

(3) One residual heat removal loop may be out of service when the reactor vessel head is removed and the refueling cavity flooded.

(4) One of the two residual heat removal loops may be temporarily out of service to meet surveillance requirements.

4. Pressurizer Safety Valves

a. At least one pressurizer safety valve shall be operable whenever the reactor head is on the vessel.

b. Both pressurizer safety valves shall be operable whenever the reactor is critical.

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*Mechanical design provisions of the residual heat removal system afford the necessary flexibility to allow an operable residual heat removal loop to consist of the RHR pump from one loop coupled with the RHR heat exchanger from the other loop. Electrical design provisions of the residual heat removal system afford the necessary flexibility to allow the normal or emergency power source to be inoperable or tied together when the reactor coolant temperature is less than 200°F.

Specification 15.3.1.A.1 requires that at least one reactor coolant pump must be operating whenever the average reactor coolant temperature is above 350°F unless the listed restrictions are established. This is required so that the FSAR zero power transients (rod withdrawal from subcritical and rod ejection) are addressed from conservative conditions. With the reactor subcritical, with required shutdown margin, and with the trip breakers open, a single rod ejection will not result in criticality being reached. With the reactor subcritical and the average reactor coolant temperature above 350°F, a single reactor coolant pump provides sufficient decay heat removal capability. Heat transfer analyses(1) show that reactor heat equivalent to 3.5% of the rated power can be removed with natural circulation only.

Items 15.3.1.A.1.a.(2) permits an orderly reduction in power if a reactor coolant pump is lost during operation between 3.5% and 50% of rated power.

Above 50% power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratio will be maintained equal to or less than 1.0, which ensures that the minimum DNB ratio increases at lower flow since the maximum enthalpy rise does not increase above its normal full-flow maximum value.(2)

Specification 15.3.1.A.3 provides limiting conditions for operation to ensure that redundancy in decay heat removal methods is provided. A single reactor coolant loop with its associated steam generator and a reactor coolant pump or a single residual heat removal loop provides sufficient heat removal capacity for removing the reactor core decay heat; however, single failure considerations require that at least two decay heat removal methods be available. Operability of a steam generator for decay heat removal includes two sources of water, water level indication in the steam generator, a vent path to atmosphere, and the Reactor Coolant System filled and vented so thermal convection cooling of the core is possible. If the steam generators are not available for decay heat removal, this Specification requires both residual heat removal loops to be operable unless the reactor system is in the refueling shutdown condition with the refueling cavity flooded and no operations in progress which could cause an increase in reactor decay heat load or a decrease

UNIT 1 CREVICE CLEANING
EXCESSIVE COOLDOWN
SEQUENCE OF EVENTS

Crevice cleaning conducted in accordance with RP-6B, "Steam Generator Crevice Cleaning."

Hot Soak Preparation Stage

- Successfully completed during day and mid shift on May 26

First Cleaning Cycle - May 27

0100	Cleaning cycle begins
0108	RCPs secured
0120-0155	Normal charging secured to leak test valve. Auxiliary charging placed in service
0133-0140	Addition of auxiliary feedwater to SGs
0211	Started "A" RCP
0211-0500	SGs drained and refilled in preparation for next crevice cleaning cycle

- Excessive cooldown transient occurred between 0108-0211 during first cleaning cycle. TS limit of 100°F/hr exceeded

UNIT 1 CREVICE CLEANING
EXCESSIVE COOLDOWN
SEQUENCE OF EVENTS (cont)

Second Cleaning Cycle - May 27

0840	Completed heatup to 295°F
0840	RCPs secured
0901	Addition of auxiliary feedwater to "A" SG
0905	CO/DOS observe cooldown rate of ≈40°F in 10 minutes. Auxiliary feedwater secured due to excessive cooldown
0909	DSS consults with Operations Manager. DSS is directed to terminate cleaning cycle
0909	Atmospheric steam dumps shut and RCPs started. Cooldown stopped
0910-1000	Crew reviews first cleaning cycle data to determine how mid shift performed crevice cleaning evolution. Identifies >100°F/hr cooldown
1000	Plant management and NRC notified. Further crevice cleaning cycles suspended

Temperature Indications Available

- RHR inlet and outlet temperature recorder
- Wide range loop temperature indicators
- Digital display selected to core exit thermocouple average
- CRT trends of RHR inlet and outlet and wide range cold leg temperature
- CRT display of heatup/cooldown curve and temperature rates of change for 20, 40 and 60-minute time intervals
- Strip charter recorder for wide range cold leg temperatures
- Digital display selected to 20-minute temperature rate of change: Gives higher of 20-minute rate; 5-minute rate, or 5-second rate if difference between any rates >10%

B. Pressure/Temperature Limits

Specification:

1. The Reactor Coolant System temperature and pressure shall be limited in accordance with the limit lines shown in Figure 15.3.1-1 and 15.3.1-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:
   a. A maximum heatup of 100°F in any one hour,
   b. A maximum cooldown of 100°F in any one hour, and
   c. An average temperature change of ≤10°F per hour during inservice leak and hydrostatic testing operations.
2. The secondary side of the steam generator will not be pressurized above 200 psig if the temperature of the steam generator vessel shell is below 70°F.
3. The pressurizer temperature shall be limited to:
   a. A maximum heatup of 100°F in any one hour and a maximum cooldown of 200°F in any one hour, and
   b. A maximum spray water temperature differential between the pressurizer and spray fluid of not greater than 320°F.
4. The reactor vessel irradiation surveillance specimens are removed and examined, according to NRC approved schedules, to determine changes in material properties. The results of these examinations shall be considered in the evaluation of the prediction method to be used to update Figures 15.3.1-1 and 15.3.1-2. Revised figures shall be provided to the Commission at least sixty (60) days before the calculated exposure of the applicable reactor vessel exceeds the exposure for which the figures apply.

ENGINEERING ANALYSIS

EXCESSIVE COOLDOWN

- Engineering analysis was performed to determine if the Unit 1 May 27, 1992, temperature transient had any impact on the integrity of the reactor vessel
- A linear elastic fracture mechanics (LEFM) analysis was performed using the B&W computer program PCRIT
- Evaluation was based on ASME Code Section XI, Appendix E, Article E-1300
- Cooldown rate was determined by incore thermocouple readings. ΔT was measured as $\approx 140°F$
- Engineering analysis resulted in a minimum ratio of K_{IC}/K_{ITOTAL} of 1.27
- Since minimum ratio was greater than 1.0, it was concluded that the structural integrity of the vessel was assured and that acceptable margins of safety would be maintained during subsequent operations

ENGINEERING ANALYSIS

EXCESSIVE COOLDOWN

- On July 3, 1992, a review of all RCS temperature monitoring data associated with the May 27 temperature transient event identified that the "B" cold leg side range temperature monitor recorded a ΔT of $\approx 167°F$

- The original engineering analysis assumed that the incore thermocouple readings represented a more accurate and conservative reactor vessel cooldown rate than that measured by other instrumentation

- A second engineering analysis was performed on July 4, 1992, which used the "B" cold leg wide range temperature monitor indications

- This analysis resulted in a minimum ratio of K_{IC}/K_{ITOTAL} of 1.21

UNIT 1 CREVICE CLEANING
EXCESSIVE COOLDOWN

Causes and Contributing Factors

Operator Performance

- Inadequate temperature monitoring
- Inadequate cooldown control
- Misuse of administrative limits
- Concurrent work activities

Causes and Contributing Factors (cont)

Procedure Issues

- Cooldown control methods not specified
- Temperature monitoring methods not specified
- Technical Specification cooldown limit not specified
- Administrative limit not stated before cooldown step
- PBNP 3.4.19, "Infrequently Performed Tests or Evolutions/Special Test Procedures," not applied
- Technical review of multi-discipline procedures
- Unit 2 operating experience not effectively factored into procedure RP-6B
- Cooldown was not anticipated and therefore, appropriate precautions not included in procedure

Causes and Contributing Factors (cont)

Control Room Indications

- Confusing Plant Process Computer System (PPCS) cooldown rate

- No cooldown rate alarms on [illegible]CS

Causes and Contributing Factors (cont)

Training

- Before-use procedure training
- Knowledge of system thermodynamics

UNIT 2 CREVICE CLEANING
SECURING DECAY HEAT REMOVAL

Causes and Contributing Factors

- Cooldown control method not specified in Wisconsin Michigan Test Procedure WMTP 11.19
- Methods required for maintaining decay heat removal not specified in WMTP 11.19
- Technical Specification interpretation of "test" exception
- Technical Specification Basis does not provide information to support interpretation of requirement

CORRECTIVE ACTIONS

Actions Taken for Excessive Cooldown Event

- Crevice cleaning discontinued
- Engineering evaluation performed
- Commitment not to pressurize above LTOP
- Reviewed Unit 2 evolution
- Conducted incident investigation
- Operator removed from primary licensed duties
- Disciplinary actions taken
- Operator knowledge and skills were reevaluated

CORRECTIVE ACTIONS TO BE TAKEN

Procedure Enhancements

- Assess need to continue to perform crevice cleaning procedure
- Revise RP-6B procedure
  - Specify cooldown control method
    - Evaluate change to temperature range
    - Evaluate increase in component cooling water temperature
  - Specify temperature monitoring method
  - Include Technical Specification requirements
  - Require implementation of PBNP 3.4.19
- Clarify heatup and cooldown administrative limit
- Evaluate joint technical review of multi-disciplinary procedures

Corrective Actions to be Taken (cont)

Technical Specifications

- Evaluate the need for Technical Specification 15.3.1.A.3 revision
  - Secure RHR pumps during crevice cleaning
  - Isolate RHR heat exchanger manual outlet valves
  - Clarify the "test" exception and Basis

Corrective Actions to be Taken (cont)

Plant Process Computer System Upgrades

- Add heatup and cooldown rate alarms
- Improve cooldown rate indication

<u>Corrective Actions to be Taken (cont)</u>

<u>Training</u>

- Conduct classroom and simulator training on RP-6B and related procedure RP-6A
- Thermodynamics in RP-6B and RP-6A configuration
- Heatup and cooldown monitoring

SUMMARY

GENERIC ISSUES

Conduct of Business

- Standards, expectations for conservative operation
- TS interpretations
- Administrative limits
- Operating experience/lessons learned
- Control operator key responsibilities

SUMMARY

GENERIC ISSUES

Procedures/Evolutions

- Clarify application of PBNP 3.4.19, "Infrequently Performed Tests or Evolutions/Special Test Procedures"
- Training requirements
- Technical review of new or revised procedures
- Simulator support
- Consistent management of evolutions