

Attachment 1

Core Spray Pipe Failure Detection Information

- Technical Specification 4.5.A.5
- Daily Surveillance Test
- Alarm Response Procedures

3.0 LIMITING CONDITION FOR OPERATION

3. One of the following conditions of inoperability may exist for the period specified:
- a. One Core Spray subsystem may be inoperable for 7 days, or
  - b. One RHR pump may be inoperable for 30 days, or
  - c. One low pressure pump or valve (Core Spray or RHR) may be inoperable with an ADS valve inoperable for 7 days, or
  - d. One of the two LPCI injection paths may be inoperable for 7 days, or
  - e. Two RHR pumps may be inoperable for 7 days, or
  - f. Both of the LPCI injection paths may be inoperable for 72 hours, or
  - g. HPCI may be inoperable for 14 days, provided RCIC is operable, or
  - h. One ADS valve may be inoperable for 14 days, or
  - i. Two or more ADS valves may be inoperable for 12 hours.
4. If the requirements or conditions of 3.5.A.1, 2 or 3 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be placed in a condition in which the affected equipment is not required to be operable within 24 hours.

3.5/4.5

4.0 SURVEILLANCE REQUIREMENTS

4. Perform the following tests:

<u>Item</u>	<u>Frequency</u>
Motor Operated Valve Operability	Pursuant to Specification 4.15.B
ADS Valve Operability	Each Operating Cycle

Note: Safety/relief valve operability is verified by cycling the valve and observing a compensating change in turbine bypass valve position.

ADS Inhibit Switch Operability	Each Operating Cycle
--------------------------------	----------------------

Perform a simulated automatic actuation test (including HPCI transfer to the suppression pool and automatic restart on subsequent low reactor water level)

5. Perform the following test on the Core Spray Ap Instrumentation:

Check	Once/day
Test	Once/month
Calibrate	Once/3 months

4.5.A.5

TITLE:

OPERATIONS DAILY LOG - PART A

Revision 59

Page 6 of 8

0097 (CORE SPRAY HEADER DP INSTRUMENTATION CHECK)

Check the Core Spray header DP indicating switches. The switches should indicate less than the limit indicated when the reactor is at power operation and full pressure if the Core Spray lines are satisfactory. Required when reactor water temperature is greater than 212°F.

RECORD VALUE

DPIS- 14-43A (Panel C-121)  $\leq$  -47" H<sub>2</sub>O \_\_\_\_\_

DPIS- 14-43B (Panel C-121)  $\leq$  +14" H<sub>2</sub>O \_\_\_\_\_

Completed By: \_\_\_\_\_

(1900 - 0700)

Completion Reviewed By: \_\_\_\_\_

Sys Engr

TECH SPEC SURVEILLANCE TEST  
PERFORMED DAILY.

CORE SPRAY I NOZZLE  
HI DIFF PRESS

<b>SETPOINT</b>	<b>ACTUATING DEVICE</b>
3.6 PSID	DPIS 14-43A on C-214

<b>CONFIRMATORY INDICATIONS</b>
Core Spray header DP indicated by DPIS 14-43A on C-214. (DEPENDENT)
<b>AUTOMATIC ACTIONS INITIATED</b>
None
<b>OPERATOR ACTIONS</b>
<p>1. <b>NOTE:</b> DPIS is calibrated in "Inches of water" and alarm setpoint is a PSID equivalent. Indication during normal operations is about -100". If a break of the Core Spray piping (internal to the RPV, external to the shroud) were to occur, indication will get "less" negative. Alarm occurs at approximately -58". Alarm is expected during CSD conditions due to error induced by water density increase.</p> <p>Determine Core Spray nozzle DP by locally checking DPIS 14-43A on C-214.</p> <p>2. Notify Shift Supervision and System Engineering, <u>IF</u> found to be a valid alarm, <u>THEN</u> follow the actions of Tech Spec 3.5/4.5 (REGARDING CORE SPRAY SYSTEM OPERABILITY) and begin an orderly plant shutdown.</p>



**CORE SPRAY II NOZZLE  
HI DIFF PRESS**

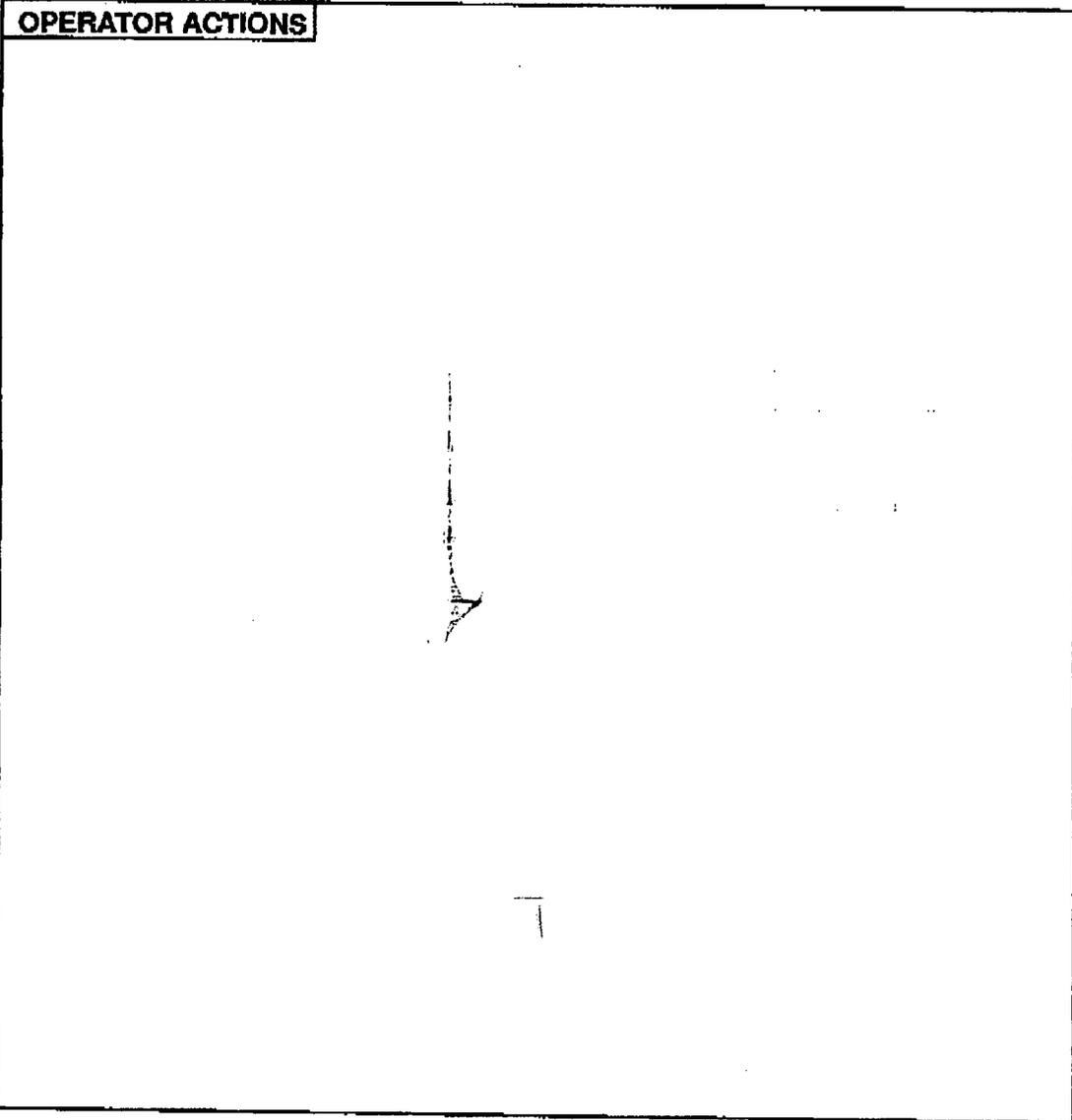
**WINDOW NUMBER**

<b>SETPOINT</b> 3.6 PSID	<b>ACTUATING DEVICE</b> DPIS 14-43B on C-215
-----------------------------	---

<b>CONFIRMATORY INDICATIONS</b>
Core Spray header DP Indicated by DPIS 14-43B on C-215. (DEPENDENT)
<b>AUTOMATIC ACTIONS INITIATED</b>
None
<b>OPERATOR ACTIONS</b>
<p>1.       <b>NOTE:</b> DPIS is calibrated in "Inches of water" and alarm setpoint is a PSID equivalent. Indication during normal operations is about -100". If a break of the Core Spray piping (Internal to the RPV, external to the shroud) were to occur, indication will get "less" negative. Alarm occurs at approximately +3". Alarm is expected during CSD conditions due to error induced by water density increase.</p> <p style="padding-left: 40px;">Determine Core Spray nozzle DP by locally checking DPIS 14-43B on C-215.</p> <p>2.       Notify Shift Supervision and System Engineering, IF found to be a valid alarm, <u>THEN</u> follow the actions of Tech Spec 3.5/4.5 (REGARDING CORE SPRAY SYSTEM OPERABILITY) and begin an orderly plant shutdown.</p>

3-B-14

WINDOW NUMBER

OPERATOR ACTIONS


REFERENCE OPERATIONS MANUAL - B.3.1

PREPARED BY: M. Hammer REVIEWED BY: A. Cassatens

OC REVIEW REQ'D: YES OC MEETING NUMBER: 1886 DATE: 3/9/93

APPROVED BY: [Signature] DATE: 3/9/93

THIS REVISION INCORPORATES VOLUME F MEMOS: None per Pete Kissinger

DOC. TYPE: 1060

Attachment 2

Peak Cladding Temperature Discussion

Nominal SAFER/GESTR ECCS analyses were performed for Monticello for a design basis recirculation suction line break along with failure of the LPCI injection valve and complete failure of the core spray line at the crack location. The peak cladding temperature calculated for this case was less than 1650F. GE believes that a peak cladding temperature (PCT) of 1650F is acceptable for this evaluation, because this is an extremely degraded and unlikely condition, which is well beyond the design basis for the plant. This analysis was not done to demonstrate compliance with the accepted licensing criteria, but it was done to show that there would not be significant core damage even with this highly degraded, unlikely event.

In addition, it should be noted that the 1600F qualification limit set by the NRC in 1984 was established based on the range of test and qualification data available at that time and the limit was set primarily because of concerns regarding metal-water reaction. Temperatures of 1650F should not result in a significant change in metal-water reaction compared to that at 1600F, since the metal-water reaction is a relatively insignificant heat source at temperatures below 1800F.

Also, the NRC has accepted SAFER/GESTR nominal analysis for BWR2's (which only have core spray systems for core cooling) for PCT values up to 1800F based on core spray heat transfer test data (NEDE-30996P-A). This acceptance should be directly applicable to this Monticello analysis, since the Monticello case which resulted in 1650F used only core spray for core cooling.

Finally, SAFER nominal calculations are recognized to give values for peak clad temperatures which are on the order of 100F greater than best-estimate TRAC results (NEDE-23785-1-PA and NEDE-30996P-A). Therefore, best estimate PCT for this very degraded event will most likely remain below 1600F.

Attachment 3

Water Hammer Evaluation

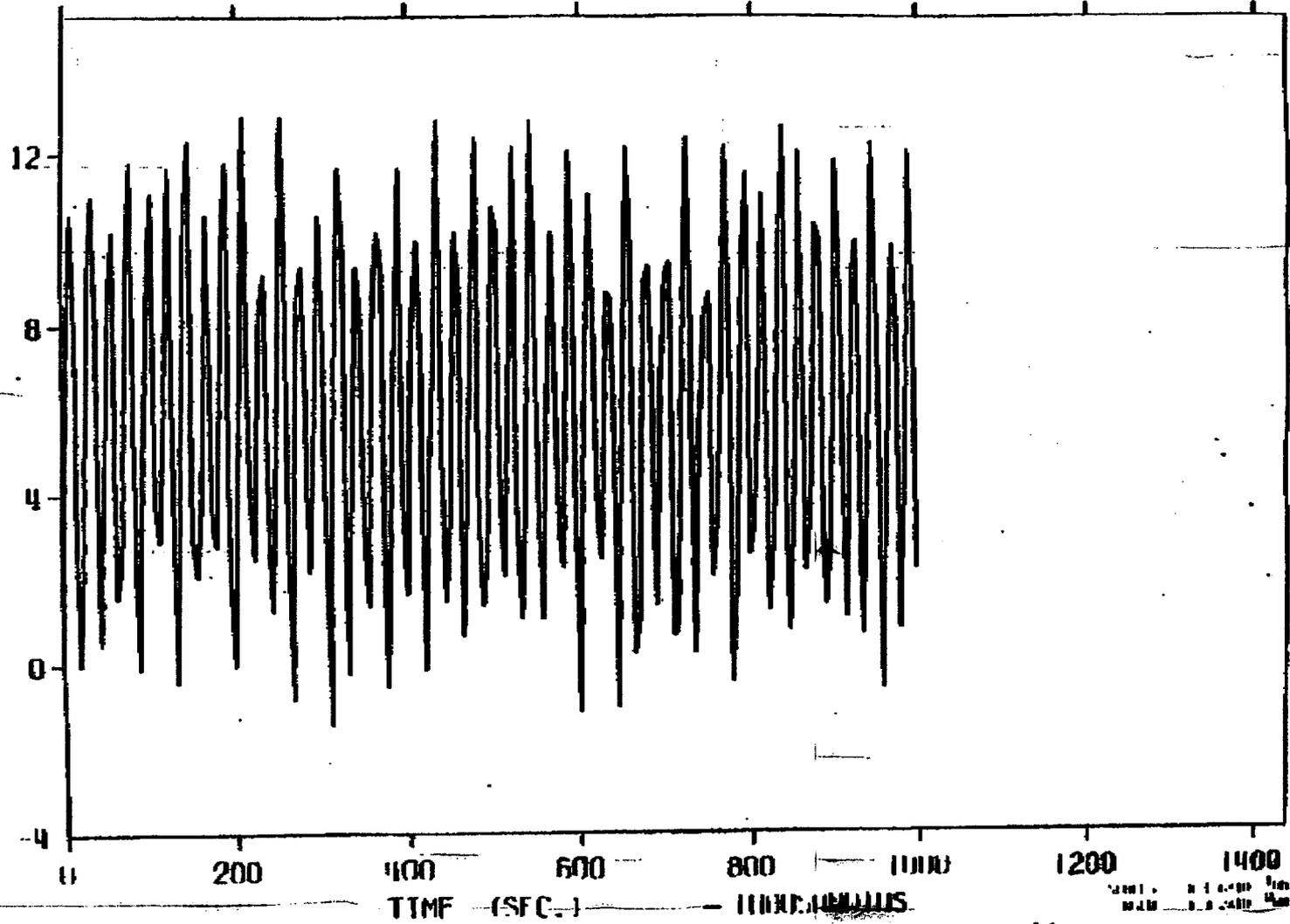
RDE-50-0692 Rev. 0

#### **3.3.4 Hydraulic Transient Loads**

The hydraulic transient is shown in Reference 3. The core spray flow is developed linearly in 20 seconds. The load on the piping system is very small. A hydraulic transient analysis was performed using 10 seconds instead of 20 seconds which is very conservative. The maximum pipe segment force amplitude is less than 20 lb. An example of pipe segment load is shown in Figure 1-3. The load is negligible for the piping analysis.

A-16

FORCE (POUNDS)



MAR 14 1993

RDE-50-0692 Rev. 0

FIGURE 1-3. EXAMPLE OF HYDRAULIC FORCE TIME HISTORY DUE TO FLOW TRANSIENT

Attachment 5

24 GPM Calculation

NUMBER \_\_\_\_\_

DATE 2/23/93

SUBJECT Monticello Core Spray Line

BY SEP

SHEET \_\_\_\_\_

OF \_\_\_\_\_

A second method of estimating the leakage is using orifice flow equations. The LEM method uses the following method to calculate the crack width,  $\delta$ :

$$\delta = \frac{4\sigma l}{E} \quad (\text{from EPRI Report \#NR-2472, Vol. 2, D-2})$$

$$l = \frac{1}{2} \text{ crack length} = \frac{3.5}{2} = 1.75''$$

$$\sigma = \text{Applied Stress} = \sigma_m + \sigma_g = 1306 + 1606 = 2912 \text{ psi}$$

$$E = \text{Young's Modulus} = 25.8 \times 10^6 \text{ psi} \quad (@ 550^\circ\text{F})$$

$$\delta = \frac{4(2912)(1.75)}{25.8 \times 10^6} = .00079'' \quad \checkmark$$

To be conservative, a crack width of .01'' will be assumed.

$$W = C_d A \sqrt{2 g_c \Delta P \rho} \quad (\text{pg 77 of DEF E21-00090})$$

$$W = .6 \frac{\pi (5.305)(.001)}{2 (144)} \sqrt{2(32.2)(150)(62.4)(144)} = 3.23 \text{ lb/sec} \quad \checkmark$$

180° Flow

$$Q = 3.23 \text{ lb/sec} * \left(\frac{60 \text{ sec}}{\text{min}}\right) * \left(\frac{1 \text{ ft}^3}{62.4 \text{ lb}}\right) * \left(\frac{7.48 \text{ gal}}{1 \text{ ft}^3}\right) = \underline{23.23 \text{ GPM}} \quad \checkmark$$

Attachment 4

References 2 & 3 of GE Analysis

Table 4-2

SINGLE-FAILURE EVALUATION

The following single, active failures were considered in the ECCS performance evaluation:

<u>Assumed Failure</u> <sup>(1)</sup>	<u>Systems Remaining</u> <sup>(2)(3)</sup>
Battery	ADS, 1 CS, 2 LPCI
LPCI Injection Valve (LPCI IV)	ADS, 2 CS, HPCI
Diesel Generator (D/G)	ADS, 1 CS, HPCI, 2 LPCI
HPCI	ADS, 2 CS, 4 LPCI

- (1) Other postulated failures are not specifically considered because they all result in at least as much ECCS capacity as one of the above assumed failures.
- (2) Systems remaining, as identified in this table, are applicable to all non-ECCS line breaks. For a LOCA from an ECCS line break, the systems remaining are those listed, less the ECC system in which the break is assumed.
- (3) Analyses performed with one ADS valve assumed unavailable in addition to the single failure (Table 4-1).

NEDC-31786P  
 GE PROPRIETARY INFORMATION  
 CLASS III

Table 5-1

SUMMARY OF RECIRCULATION LINE BREAK RESULTS<sup>(1)</sup>

<u>Break Size</u>	<u>Single Failure</u>	<u>BP/P8x8R</u>		<u>GE8x8EB/NB</u>	
		<u>PCT (°F)</u>	<u>Peak Local Oxidation %</u>	<u>PCT (°F)</u>	<u>Peak Local Oxidation %</u>
<u>NOMINAL:</u>					
DBA	Battery	1229	0.08	1205	<0.08
DBA	LPCI IV	1084	<0.08	1137	<0.08
80% DBA	Battery	1114	<0.08		
60% DBA	Battery	790	<0.08		
1.0 ft <sup>2</sup>	Battery	758	<0.08		
0.5 ft <sup>2</sup>	Battery	582 <sup>(2)</sup>	<0.08	585 <sup>(2)</sup>	<0.08
0.1 ft <sup>2</sup>	Battery	1016	<0.08	1045	<0.08
0.05 ft <sup>2</sup>	Battery	986	<0.08	1010	<0.08

APPENDIX K:

DBA	Battery	1730	1.00	1668	<1.00
DBA	LPCI IV	1769	1.81	1765	<1.81
80% DBA	Battery	1466	<1.00		
60% DBA	Battery	1427	<1.00		

Note: (1) Core-wide metal-water Reaction < 0.1% for all cases.

(2) PCT represents initial cladding temperature (i.e., no cladding heatup occurred).

Attachment 6

February 12, 1993 Amendment Request

TRANSMITTAL MANIFEST

NORTHERN STATES POWER COMPANY

NUCLEAR LICENSING DEPARTMENT

M1993022

MONTICELLO NUCLEAR GENERATING PLANT

MRE JR  
TR

License Amendment Request Dated February 12, 1993  
Revised Core Spray Pump Flow and Other Editorial Corrections

Manifest Date: February 15, 1993

USNRC		Monticello Plant	
*Attn: DCD	1	Oper Comm & File	1
*Tony Hsia	1	Document Control File	1
*Regional Admin III	1	A D Hennigan (MDH)	1
*Steve Ray	1	R O Anderson	1
State of Minnesota		D D Antony	1
Kris Sanda	1	K M Beadell	1
General Electric		B D Day	1
C N Gallit-Nuclear Serv	1	E B Earney	1
San Jose-Fuel Proj Mgr	1	External Comm Dept	1
San Jose-Licensing	1	G D Hudson	1
Safety Audit Committee		P A Johnson	1
D J Mendele	1	D M Musolf	1
A B Cutter	1	*NSS Monti Plant File	1
R L Hannen	1	L L Nolan	2
F W Hartley	1	L H Waldinger	1
W J Hill	1	SAC Secretary	1
D D Lanning	1	NSS History File (TS)	1
T M Parker	1	NSS File	1
M B Sellman	1	USAR File Yes <input checked="" type="checkbox"/> No <input type="checkbox"/>	
C R Steinhardt	1	NRC Commitment Yes <input type="checkbox"/> No <input checked="" type="checkbox"/>	
J A Thie	1		
* Pat Tobin	1		
* Dave Pennington	1		

Correspondence Date: February 12, 1993



Northern States Power Company

414 Nicollet Mall  
Minneapolis, Minnesota 55401  
Telephone (612) 330-5500

February 12, 1993

M1993022

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

10 CFR Part 50  
Section 50.90

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

License Amendment Request Dated February 12, 1993  
Revised Core Spray Pump Flow and Other Editorial Corrections

Attached is a request for a change in the Technical Specifications, Appendix A of the Operating License for the Monticello Nuclear Generating Plant. This request is submitted in accordance with the provisions of 10 CFR Part 50, Section 50.90.

The proposed amendment involves increasing the minimum Core Spray pump flow to more conservatively account for Emergency Core Cooling Systems bypass leakage paths. In addition, the proposed amendment involves editorial or administrative changes to correct branching errors, typographical errors, and similar discrepancies currently existing in various sections.

Exhibit A contains a description of the proposed change, the reasons for requesting the change, a Safety Evaluation, a Determination of Significant Hazards Consideration, and an Environmental Assessment. Exhibit B contains the current Technical Specification pages marked up with the proposed change. Exhibit C contains revised Monticello Technical Specification pages.

9302220250 2pp

USNRC  
February 12, 1993  
Page 2

NORTHERN STATES POWER COMPANY

Please contact Terry Coss at (612) 295-1449 if you require further information related to this request.



Thomas M Parker  
Director  
Nuclear Licensing

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

Attachments: Affidavit to the US Nuclear Regulatory Commission  
Exhibit A - Evaluation of Proposed Change  
Exhibit B - Technical Specification pages marked up with  
proposed change  
Exhibit C - Revised Monticello Technical Specification Pages

UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

REQUEST FOR AMENDMENT TO  
OPERATING LICENSE DPR-22

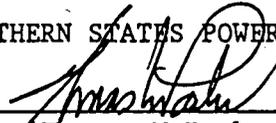
LICENSE AMENDMENT REQUEST DATED February 12, 1993

Northern States Power Company, a Minnesota corporation, requests authorization for changes to Appendix A of the Monticello Operating License as shown on the attachments labeled Exhibits A, B, and C. Exhibit A describes the proposed changes, describes the reasons for the changes, and contains a Safety Evaluation, a Determination of Significant Hazards Consideration and an Environmental Assessment. Exhibit B contains current Technical Specification pages marked up with the proposed changes. Exhibit C is a copy of the Monticello Technical Specifications incorporating the proposed changes.

This letter contains no restricted or other defense information.

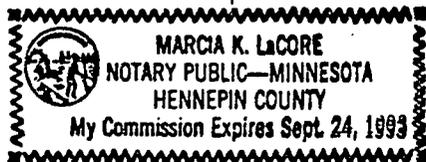
NORTHERN STATES POWER COMPANY

By



Thomas M Parker  
Director  
Nuclear Licensing

On this 12<sup>th</sup> day of February, 1993 before me a notary public in and for said County, personally appeared Thomas M Parker, Director, Nuclear Licensing, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company, that he knows the contents thereof, and that to the best of his knowledge, information, and belief the statements made in it are true and that it is not interposed for delay.



9302220253 9pp

Exhibit A

MONTICELLO NUCLEAR GENERATING PLANT

License Amendment Request Dated February 12, 1993

Evaluation of proposed changes to the Technical Specifications  
for Operating License DPR-22

Pursuant to 10 CFR Part 50, Section 50.59 and 50.90, the holders of Operating License DPR-22 hereby propose the following changes to the Monticello Technical Specifications:

<u>Page</u>	<u>Section</u>	<u>Proposed Changes</u>
52	Table 3.2.2	<p>Revise Core Spray and Low Pressure Coolant Injection trip function A.1.b.ii to delete the word "and" from the function description (Reactor Low Pressure Permissive Bypass Timer) and to refer to Required Condition "B" in lieu of Required Condition "C".</p> <p>Also, correct the spelling of the word "Channels" in the heading for the second column from the right.</p>
53	Table 3.2.2	<p>Revise HPCI System trip functions B.1 (High Drywell Pressure) and B.2 (Low-Low Reactor Water Level) to refer to Required Condition "A" in lieu of "B".</p> <p>Also, revise Automatic Depressurization System trip functions C.1 (Low-Low Reactor Water Level), C.2 (Auto Blowdown Timer), and C.3 (Low Pressure Core Cooling Pumps Discharge Pressure Interlock) to refer to Required Condition "B" in lieu of "C".</p>
54	Table 3.2.2	<p>Revise Diesel Generator trip functions D.2 (Low-Low Reactor Water Level) and D.3 (High Drywell Pressure) to refer to Required Condition "C" in lieu of "D".</p>
55	Table 3.2.2	<p>Delete the existing Required Condition "B", and re-identify remaining Required Conditions "C" and "D" as "B" and "C", respectively.</p>
60d	Table 3.2.8	<p>Revise Required Condition "B" to refer to Specification 3.5.D in lieu of 3.5.F.2.</p> <p>Also, delete redundant "status" from the description of Required Condition "C." near bottom of the page.</p>

101 4.5.A.1 Revise the minimum required flow rate of the Core Spray Pumps upwards from 2,700 gpm to 2,800 gpm.

107 3.5.F.1.a.2 The specification currently reads:

"The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) will be changed as noted in Table 3.11.1."

Revise this specification to read:

"The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) will be changed as noted in Table 1 of the Core Operating Limit Report."

110 3.5/4.5 Bases Part A The second sentence of the second paragraph of the ECCS Bases currently reads:

"The Core Spray pump is designed to deliver greater than or equal to 3,020 gpm (safety analysis assumed 2700 gpm) against a system head corresponding to a reactor pressure 130 psi greater than containment pressure."

Revise the above sentence to read:

"The Core Spray pump is designed to deliver greater than or equal to 3,020 gpm (the SAFER/GESTR-LOCA safety analysis assumed a Core Spray pump flow of 2,800 gpm, or 2,700 gpm flow into the core + 100 gpm to account for ECCS bypass leakage) against a system head corresponding to a reactor pressure 130 psi greater than containment pressure."

Also, in the fifth paragraph on this page, reference Specification 3.5.A.3 in lieu of 3.5.A.2.

113 3.5/4.5 Bases Part E Revise the last paragraph of this page to refer to Specification 3.5.E.2 in lieu of 3.5.E.4.

127 3.6.E.1 The specification currently reads:

"E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F:

- a. The safety valve function (self-actuation) of seven safety/relief valves shall be operable.
- b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.
- c. The Low-Low Set Function for three non-Automatic Pressure Relief Valves shall be operable as required by Specification 3.2.H."

Revise this specification to read as follows:

"E. Safety/Relief Valves

- 1. During Operating Conditions and whenever reactor coolant pressure is less than 100 psig and temperature is greater than 345°F, the safety valve function (self-actuation) of seven safety/relief valves shall be operable (Note: Low-Low Set and ADS requirements are located in Specifications 3.2.H and 3.5.A, respectively)."

3.6.E.2

Revise this specification to refer to Specification 3.6.E.1 in lieu of 3.6.E.1.a.

151 3.6/4.6 Bases

The first paragraph on this page currently reads:

"The safety/relief valves have two functions; i.e. power relief or self-actuated by high pressure. The solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate opening of the valves. This function is discussed in Specification 3.5.E. In addition, the valves can be operated manually."

Revise this paragraph to read:

"The safety/relief valves have two functions; 1) over pressure relief (self-actuated by high pressure), and 2) Depressurization/Pressure

Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation). The Low-Low Set and ADS functions are discussed further in Sections 3.2 and 3.5."

156 3.7.A.1 Revise this specification to refer to Specification 3.5.E.2 in lieu of 3.5.G.4.

Reason for Change:

The portion of the change related to increasing the required Core Spray Pump flow from 2,700 gpm to 2,800 gpm is intended to account for the flow losses (bypass leakage paths) inherent to the Emergency Core Cooling Systems (ECCS) design. Increasing the required flow rate for the Core Spray Pumps will assure that the total flow entering the core (ECCS pump flow minus bypass leakage) during a Loss of Coolant Accident (LOCA) is consistent with the value assumed in the Monticello SAFER/GESTR-LOCA Analysis.

The changes to the 3.6/4.6 Bases discussion on page 151 are intended to clarify and correct existing statements that are both confusing and misleading. The current wording states, incorrectly, that coincident high drywell pressure and low-low water level signals initiate automatic actuation of the safety relief valves. This is no longer true because of a modification performed in response to NUREG 0737, Item II.K.3.18 (Reference: License Amendment No. 62 dated March 31, 1989). The correct discussion of this function is provided in Section 3.2 of the Technical Specifications. The proposed change will address this discrepancy and reference the proper information.

Similarly, the changes to Specification 3.6.E are intended to clarify the intent of the specification with respect to Automatic Depressurization System (ADS) and Low-Low Set system requirements. As presently written, Specifications 3.2.H, 3.5.A and 3.6.E cross reference each other in a manner that could lead to misinterpretation of the governing requirements for these systems. The language of the proposed change is intended to alleviate this concern.

The remaining changes are editorial in nature and are intended primarily to correct branching errors that occurred in previous License Amendments. Most of these errors resulted from License Amendment 79 (SAFER/GESTR), dated April 9, 1991, in which Section 3.5/4.5 (Core and Containment Cooling Systems) was substantially rewritten and reorganized. Several specifications were either deleted or re-numbered at that time and related changes to associated cross-references were missed.

Safety Evaluation:

1. Increase in Core Spray Pump Required Flow Rate:

Technical Specification 4.5.A.1 currently requires that the Core Spray

Pumps develop a flow rate of 2,700 gpm against a system head corresponding to a reactor pressure of 130 psi greater than the containment pressure. Technical Specification 4.5.A.2 requires that the Low Pressure Coolant Injection (LPCI) pumps develop a flow rate of 3,870 gpm, corresponding to two pumps delivering 7,740 gpm, at a reactor pressure of 20 psi greater than containment pressure. The SAFER/GESTR-LOCA analysis prepared for Monticello by General Electric incorrectly utilized the above flow rates to represent actual flow into the core.

Due to the design of the Core Spray and LPCI Systems, there are minor flow losses (bypass leakage paths) that cause the actual flow rate into the core to be slightly less than the measured discharge flow rate of the pumps. The Core Spray System is assumed to have 20 gpm leakage from a 1/4 inch vent hole in the T-box which is located between the inner reactor vessel wall and the core shroud. The LPCI system is assumed to have 50 gpm leakage from slip joints on the jet pump assemblies. These flow diversions are treated as leakage paths because the associated coolant goes into the annulus region of the vessel and would flow out the postulated Design Basis Loss of Coolant Accident (DBA-LOCA) Recirculation System suction line break.

An evaluation was performed (Reference: Nonconforming Item Report 92-037) which confirmed that the actual flow rates for individual ECCS pumps minus assumed leakage was adequate to meet the flow rates assumed in the SAFER/GESTR-LOCA analysis, therefore there were no immediate operability concerns. However, the discrepancy between the flow rates required by the Technical Specifications and the values assumed in the SAFER/GESTR-LOCA analysis remains. To resolve this issue, we propose to increase the required Core Spray flow rate by 100 gpm (20 gpm to account for Core Spray leakage + 50 gpm to account for LPCI leakage + 30 gpm for margin) to account for all of the assumed ECCS bypass leakage paths. The LPCI flow rate currently required by the Technical Specification (3,870 gpm per pump/ 7,740 gpm total) would remain unchanged.

This issue has been discussed with General Electric, who performed the Monticello SAFER/GESTR-LOCA analysis. General Electric has concluded that with respect to the analysis, it is of no significance whether the assumed 70 gpm bypass leakage (increased to 100 gpm to provide 30 gpm margin) is accounted for by increasing Core Spray flow, LPCI flow, or both. However, when the trade-off between increasing Core Spray or LPCI flow is considered, increased Core Spray flow is preferred for the following reasons:

- a. In addition to replenishing vessel water inventory lost during the DBA-LOCA, Core Spray flow (which is injected into the vessel above the core) is more effective in collapsing any steam bubble that might form in the vessel.
- b. The Core Spray pumps deliver flow to the reactor vessel at higher reactor pressures than the Residual Heat Removal (RHR) pumps operating in the LPCI mode, which is beneficial in mitigating a postulated DBA-LOCA.

An additional factor in our decision to account for all ECCS assumed bypass leakage by increasing Core Spray flow involves the relative capacities of the Core Spray and RHR pumps. Each of the four RHR pumps (which provide LPCI flow) is currently capable of consistently meeting the existing Technical Specification flow rate requirement of 3,870 gpm. A review of recent surveillance test results has confirmed that the pumps are also capable of meeting the slightly higher flow rate assumed by the SAFER/GESTR-LOCA analysis (3,895 gpm, which equates to an additional 25 gpm per operating pump assuming only two pumps are running, to account for the total LPCI bypass leakage of 50 gpm). However, the higher value (3,895 gpm) is very near the upper limit of RHR pump capacity, and there is insufficient margin remaining to ensure the pumps would consistently achieve this higher flow in the future.

Conversely, Core Spray pump performance is such that the minimum required flow could be increased by 100 gpm to 2,800 gpm without difficulty. The current test criteria for the Core Spray pumps conservatively specifies an acceptance criteria of 3,020 gpm against a system head corresponding to 130 psi greater than containment pressure. Thus, the current test criteria provides a margin of more than 200 gpm over the proposed new Technical Specification criteria.

The combination of ECCS pumps available for each single failure evaluated for a DBA-LOCA by the SAFER/GESTR-LOCA analysis includes a Core Spray pump whenever two LPCI (RHR) pumps are available. Therefore, a Core Spray pump would always be available to provide the additional flow necessary to offset the assumed LPCI bypass leakage.

As discussed above, the proposed change will adequately resolve the discrepancy between the current Technical Specification Emergency Core Cooling Systems pump flow rates and the flow into the core assumed by the SAFER/GESTR-LOCA analysis. The change is primarily administrative and has no impact on plant safety, since the basic assumptions supporting the SAFER/GESTR-LOCA analysis, and therefore the conclusions of the analysis, remain unchanged.

2. Editorial Changes to Correct Branching Errors, Cross Reference Errors, and Clarify/Correct Bases Information:

The changes to the 3.6/4.6 Bases discussion should have been included as part of License Amendment No. 62, dated March 31, 1989. Amendment No. 62 reflected modifications to the Automatic Depressurization System logic that, among other things, removed the High Drywell Pressure interlock in response to NUREG-0737 Item II.K.3.18. Other portions of Technical Specifications affected by the modification were updated appropriately, but the necessary changes to page 151 were missed. Safety considerations associated with the Automatic Depressurization System logic change were fully addressed at the time Amendment No. 62 was processed and the proposed correction does not present any new safety questions or concerns. The proposed change is necessary to ensure the 3.6/4.6 Bases discussion is consistent with the intent of the remainder of the Technical Specifications.

The remaining changes are editorial in nature and do not change the intent of the existing Technical Specifications. Most of these changes serve to correct internal branching and cross reference errors that occurred during previous license amendments. The remaining changes clarify, but do not change, the intent of existing specifications. These changes have no impact on plant safety.

Determination of Significant Hazards Consideration:

This proposed change to the Operating License has been evaluated to determine if it constitutes a significant hazards consideration as required by 10 CFR Part 50, Section 50.91 using the standards provided in Section 50.92. This analysis is provided below:

- a. The proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Increasing the required Core Spray pump flow rate to 2,800 gpm will make the Technical Specification consistent with the SAFER/GESTR-LOCA analysis. The change is in the conservative direction (increased ECCS flow) and will not increase the probability or consequences of a DBA-LOCA or any other accident previously analyzed.

The remaining changes proposed are editorial or administrative in nature and have no impact on the probability or consequences of any accident previously evaluated.

- b. The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

The proposed changes, including the revised Core Spray pump flow rates and the 3.6/4.6 Bases changes, are primarily editorial or administrative in nature. No safety-related equipment, safety function, or plant operations will be altered as a result of the proposed changes. Therefore, the proposed amendment does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

- c. The proposed amendment will not involve a significant reduction in the margin of safety.

The proposed amendment will not reduce the margin of safety because the Core Spray pump flow is being conservatively increased so that total ECCS pump flow into the core is consistent with that assumed by the SAFER/GESTR-LOCA analysis. The remaining changes are either editorial in nature or are based on previously reviewed and approved Technical Specifications and have no impact on plant safety.

Based on the evaluation described above, and pursuant to 10 CFR Part 50, Section 50.91, Northern States Power Company has determined that operation of the Monticello Nuclear Generating Plant in accordance with the proposed

license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR Part 50, Section 50.92.

Environmental Assessment:

Northern States Power has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration,
2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or
3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR Part 51 Section 51.22(c)(9). Therefore, pursuant to 10 CFR Part 51 Section 51.22(b), an environmental assessment of the proposed changes is not required.

Exhibit B

Monticello Nuclear Generating Plant

License Amendment Request Dated February 12, 1993

Technical Specification Pages Marked Up  
with Proposed Wording Changes

Exhibit B consists of the existing Technical Specification pages marked up with the proposed changes. Existing pages affected by this change are listed below:

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~~9302220255~~ 26pp.

Table 3.2.2  
Instrumentation That Initiates Emergency Core Cooling Systems

<u>Function</u>	<u>Trip Setting</u>	<u>Minimum No. of Operable or Operating Trip Systems(3)</u>	<u>Total No. of Instru- ment Channels Per Trip System</u>	<u>Minimum No. of Oper- able or Operating Instrument Channels Per Trip System (3)</u>	<u>Required Conditions*</u>
A. <u>Core Spray and LPCI</u>					
1. Pump Start					
a. Low Low Reactor Water Level and	≥6'6"≤6'10"	2	4(4)	4	A.
b. i. Reactor Low Pressure Permissive	≥450 psig	2	2(4)	2	A.
or ii. Reactor Low Pressure Permissive—and Bypass Timer	20±1 min	2	1	1	CB.
c. High Drywell Pressure (1)	≤2 psig	2	4(4)	4	A.
2. Low Reactor Pressure (Valve Permissive)	≥450 psig	2	2(4)	2	A.
3. Loss of Auxiliary Power	-----	2	2(2)	2	A.

Table 3.2.2  
Instrumentation That Initiates Emergency Core Cooling Systems

<u>Function</u>	<u>Trip Setting</u>	<u>Minimum No. of Operable or Operating Trip Systems (3)</u>	<u>Total No. of Instrument Channels Per Trip System</u>	<u>Minimum No. of Operable or Operating Instrument Channels Per Trip System (3)</u>	<u>Required Conditions*</u>
<b>B. <u>HPCI System</u></b>					
1. High Drywell Pressure (1)	≤2 psig	1	4	4	BA.
2. Low-Low Reactor Water Level	≥6'6"≤6'10"	1	4	4	BA.
<b>C. <u>Automatic Depressurization</u></b>					
1. Low-Low Reactor Water Level and	≥6'6"≤6'10"	2	2	2	CB.
2. Auto Blowdown Timer and	≤120 seconds	2	1	1	CB.
3. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	≤100 psig	2	12(4)	12(4)	CB.

Table 3.2.2 - Continued  
Instrumentation That Initiates Emergency Core Cooling System

<u>Function</u>	<u>Trip Setting</u>	<u>Min. No. of Operable or Operating Trip Systems(3)</u>	<u>Total No. of Instrument Channels Per Trip System</u>	<u>Min. No. of Operable or Operating Instrument Channels Per Trip System (3)</u>	<u>Required Conditions*</u>
<u>D. Diesel Generator</u>					
1. Degraded or Loss of Voltage Essential Bus (5)					
2. Low Low Reactor Water Level	≥6'6"≤6'10"	2	4(4)	4	DC.
3. High Drywell Press	≤2 psig	2	4(4)	4	DC.

NOTES:

1. High drywell pressure may be bypassed when necessary only by closing the manual containment isolation valves during purging for containment inerting or de-inerting. Verification of the bypass condition shall be noted in the control room log. Also need not be operable when primary containment integrity is not required.
2. One instrument channel is a circuit breaker contact and the other is an undervoltage relay.

Table 3.2.2 - Continued

Notes:

3. Upon discovery that minimum requirements for the number of operable or operating trip systems, or instrument channels are not satisfied action shall be initiated to:
  - (a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or
  - (b) Place the plant under the specified required conditions using normal operating procedures.
4. All instrument channels are shared by both trip systems.
5. See table 3.2.6.
- \* Required conditions when minimum conditions for operation are not satisfied.
  - A. Comply with Specification 3.5.A.
  - ~~B. Comply with Specification 3.5.D.~~
  - ~~C. Reactor pressure  $\leq$ 150 psig.~~
  - ~~D. Comply with Specification 3.9.B.~~

Table 3.2.8  
Other Instrumentation

Function	Trip Setting	Minimum No. of Operable or Operating Trip System (1)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Conditions*
A. RCIC Initiation					
1. Low-Low Reactor Level	$\geq 6'6"$ & $\leq 6'10"$ above top of active fuel	1	2	2	B
B. HPCI/RCIC Turbine Shutdown					
a. High Reactor Level	$\leq 14'6"$ above top of active fuel	1	2	2	A
C. HPCI/RCIC Turbine Suction Transfer					
a. Condensate Storage Tank Low Level	$\geq 2'0"$ above tank bottom	1	2	2	C

**NOTE:**

1. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated to:

- a. Satisfy the requirements by placing the appropriate channels or systems in the tripped condition (Turbine/Feedwater Trip only), or
- b. Place the plant under the specified required condition using normal operating procedures.

\* Required conditions when minimum conditions for operation are not satisfied:

- A. Reactor in Startup, Refuel, or Shutdown Mode.
- B. Comply with Specification 3.5.2.
- C. Align HPCI and RCIC suction to the suppression pool. Restore channels to operable status ~~status~~ within 30 days or place the plant in Required Condition A.

### 3.0 LIMITING CONDITION FOR OPERATION

#### 3.5 CORE AND CONTAINMENT SPRAY/COOLING SYSTEMS

##### Applicability:

Applies to the operational status of the emergency cooling systems.

##### Objective:

To insure adequate cooling capability for heat removal in the event of a loss of coolant accident or isolation from the normal reactor heat sink.

##### Specification:

#### A. ECCS Systems

1. Except as specified in section 3.5.A.3, both Core Spray subsystems and the Low Pressure Coolant Injection (LPCI) Subsystem (LPCI Mode of RHR System) shall be operable whenever irradiated fuel is in the reactor vessel and the reactor water temperature is greater than 212°F.
2. Except as specified in section 3.5.A.3, the High Pressure Coolant Injection (HPCI) System and the Automatic Depressurization System (ADS) shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel except during reactor vessel hydrostatic or leakage tests.

### 4.0 SURVEILLANCE REQUIREMENTS

#### 4.5 CORE AND CONTAINMENT SPRAY/COOLING SYSTEMS

##### Applicability:

Applies to the periodic testing of the emergency cooling systems.

##### Objective:

To verify the operability of the emergency cooling systems.

##### Specification:

#### A. ECCS Systems

1. Demonstrate the Core Spray Pumps develop a ~~2,700~~ 3,800 gpm flow rate against a system head corresponding to a reactor pressure of 130 psi greater than containment pressure, when tested pursuant to Specification 4.15.B.
2. Demonstrate the LPCI Pumps develop a 3,870 gpm flow rate against a system head corresponding to two pumps delivering 7,740 gpm at a reactor pressure of 20 psi greater than containment pressure, when tested pursuant to Specification 4.15.B.
3. Demonstrate the HPCI Pump develops a 2700 gpm flow rate against a reactor pressure range of 1120 psig to 150 psig, when tested pursuant to Specification 4.15.B.

### 3.0 LIMITING CONDITION FOR OPERATION

#### F. Recirculation System

1. The reactor may be started and operated, or operation may continue with only one recirculation loop in operation provided that:
  - a. The following changes to setpoints and safety limit settings will be made within 24 hours after initiating operation with only one recirculation loop in operation.
    1. The Operating Limit MCPR (MCPR) will be changed per Specification 3.11.C.
    2. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) will be changed as noted in Table 1 of the Core Operating Limits Report 3.11.1.
    3. The APRM Neutron Flux Scram and APRM Rod Block setpoints will be changed as noted in Specification 2.3.A and Table 3.2.3.
  - b. Total core flow will be maintained greater than 39% when core thermal power is above the limit specified in Figure 3.5.1.

### 4.0 SURVEILLANCE REQUIREMENTS

#### F. Recirculation System

1. See Specification 4.6.G
2. The following baseline noise levels will be obtained prior to operation with only one recirculation pump in operation at a core thermal power greater than that specified in Figure 3.5.1 or with a core flow greater than 45% provided that baseline values have not been established since the last core refueling. Baseline values will be taken with only one recirculation pump running.
  - a. Establish a baseline core plate  $\Delta P$  noise level.
  - b. Establish a baseline APRM and LPRM neutron flux noise level.
3. With only one recirculation loop in operation at a core thermal power greater than that specified in Figure 3.5.1 or with a core flow greater than 45%, determine the following noise levels at least once per 8 hour period and within 30 minutes after a core thermal power increase of greater than 5% of rated thermal power.
  - a. Core plate  $\Delta P$  noise levels.
  - b. APRM and LPRM neutron noise levels.

## Bases 3.5/4.5

### A. ECCS Systems

The core spray system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and, together with the LPCI mode of the RHR system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the automatic depressurization system (ADS).

The Core Spray System is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining. The Core Spray pump is designed to deliver greater than or equal to 3020 gpm (~~safety analysis assumed 2700 gpm~~ the SAFER/CESTR LOCA safety analysis assumed a Core Spray Pump flow of 2,800 gpm, or 2,700 gpm flow into the core + 100 gpm to account for ECCS bypass leakage) against a system head corresponding to a reactor pressure of 130 psi greater than containment pressure.

The surveillance requirements provide adequate assurance that the Core Spray System will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Four pumps are available to provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS. LPCI Loop Selection Logic determines which Recirculation loop the four RHR pumps will pump into. Each RHR pump was designed to deliver greater than or equal to 4000 gpm (the safety analysis assumed two pumps delivering 7,740 gpm) against a system head corresponding to a reactor pressure of 20 psi greater than containment pressure.

The allowed out-of-service conditions (Section 3.5.A.3.2) are determined from ECCS analysis cases analyzed. Only one of these conditions is permitted to exist. If more than one condition exists, an orderly shutdown shall be initiated. A LPCI injection path consists of the two motor operated injection valves on that path.

The surveillance requirements provide adequate assurance that the LPCI system will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The

### Bases 3.5/4.5 Continued:

The RHR service water system provides cooling for the RHR heat exchangers and can thus maintain the suppression pool water within limits. With the flow specified, the pool temperature limits are maintained as specified in Specification 3.7.A.1.

#### D. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. The pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therefore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

The surveillance requirements provide adequate assurance that the RCIC system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

#### E. Cold Shutdown and Refueling Requirements

The purpose of Specification 3.5.E is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment spray/cooling subsystems may be out of service. This specification allows all core and containment spray/cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.E.4~~2~~ recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

### 3.0 LIMITING CONDITION FOR OPERATION

#### E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F the safety valve function (self actuation) of seven safety/relief valves shall be operable (note: Low-Low Set and ADS requirements are located in Specifications 3.2.H and 3.5.A, respectively) :-

- ~~a. The safety valve function (self-actuation) of seven safety/relief valves shall be operable.~~
- ~~b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.~~
- ~~c. The Low-Low Set Function for three non Automatic Pressure Relief Valves shall be operable as required by Specification 3.2.H.~~

2. If Specification 3.6.E.1-a is not met, initiate an orderly shutdown and have reactor coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.

### 4.0 SURVEILLANCE REQUIREMENTS

#### E. Safety/Relief Valves

1. a. A minimum of seven safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. The nominal self-actuation setpoints are specified in Section 2.4.B.
  - b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.
  - c. The integrity of the safety/relief valve bellows shall be continuously monitored.
  - d. The operability of the bellows monitoring system shall be demonstrated at least once every three months.
2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

Bases Continued 3.6 and 4.6:

~~The safety/relief valves have two functions; i.e. power relief or self actuated by high pressure. The solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low low water level initiate opening of the valves. This function is discussed in Specification 3.5.E. In addition, the valves can be operated manually.~~

The safety/relief valves have two functions: 1) over-pressure relief (self-actuated by high pressure), and 2) Depressurization/Pressure Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation). The Low-Low Set and ADS functions are discussed further in Sections 3.2 and 3.5.

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9 of the ASME Pressure Vessel Code Section III Nuclear Vessels requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system quarterly provisions assurance of bellows integrity.

When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

I. Deleted

3.0 LIMITING CONDITION FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment.

1. Suppression Pool Volume and Temperature

When irradiated fuel is in the reactor vessel and either the reactor water temperature is greater than 212°F or work is being done which has the potential to drain the vessel, the following requirements shall be met, except as permitted by Specification 3.5.E.2G.4:

- a. Water temperature during normal operating shall be  $\leq 90^{\circ}\text{F}$ .
- b. Water temperature during test operation which adds heat to the suppression pool shall be  $\leq 100^{\circ}\text{F}$  and shall not be  $> 90^{\circ}\text{F}$  for more than 24 hours.
- c. If the suppression chamber water temperature is  $> 110^{\circ}\text{F}$ , the reactor shall be scrammed immediately. Power operation shall not be resumed until the pool temperature is  $\leq 90^{\circ}\text{F}$ .

3.7/4.7

4.0 SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. Suppression Pool Volume and Temperature

- a. The suppression chamber water temperature shall be checked once per day.
- b. Whenever there is indication of relief valve operation which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged ever 5 minutes until the heat addition is terminated.
- c. A visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage

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Exhibit C

Monticello Nuclear Generating Plant

License Amendment Request Dated February 12, 1993

Revised Monticello Technical Specification Pages

Exhibit C consists of revised Technical Specification pages that incorporate the proposed changes. The pages included in this exhibit are:

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Table 3.2.2  
Instrumentation That Initiates Emergency Core Cooling Systems

<u>Function</u>	<u>Trip Setting</u>	<u>Minimum No. of Operable or Operating Trip Systems(3)</u>	<u>Total No. of Instrument Channels Per Trip System</u>	<u>Minimum No. of Operable or Operating Instrument Channels Per Trip System (3)</u>	<u>Required Conditions*</u>
A. <u>Core Spray and LPCI</u>					
1. Pump Start					
a. Low Low Reactor Water Level and	$\geq 6'6" \leq 6'10"$	2	4(4)	4	A.
b. i. Reactor Low Pressure Permissive	$\geq 450$ psig	2	2(4)	2	A.
or					
ii. Reactor Low Pressure Permissive Bypass Timer	$20 \pm 1$ min	2	1	1	B.
c. High Drywell Pressure (1)	$\leq 2$ psig	2	4(4)	4	A.
2. Low Reactor Pressure (Valve Permissive)	$\geq 450$ psig	2	2(4)	2	A.
3. Loss of Auxiliary Power	-----	2	2(2)	2	A.

Table 3.2.2  
Instrumentation That Initiates Emergency Core Cooling Systems

<u>Function</u>	<u>Trip Setting</u>	<u>Minimum No. of Operable or Operating Trip Systems (3)</u>	<u>Total No. of Instrument Channels Per Trip System</u>	<u>Minimum No. of Operable or Operating Instrument Channels Per Trip System (3)</u>	<u>Required Conditions*</u>
<b>B. <u>HPCI System</u></b>					
1. High Drywell Pressure (1)	≤2 psig	1	4	4	A.
2. Low-Low Reactor Water Level	≥6'6"≤6'10"	1	4	4	A.
<b>C. <u>Automatic Depressurization</u></b>					
1. Low-Low Reactor Water Level and	≥6'6"≤6'10"	2	2	2	B.
2. Auto Blowdown Timer and	≤120 seconds	2	1	1	B.
3. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	≤100 psig	2	12(4)	12(4)	B.

Table 3.2.2 - Continued  
Instrumentation That Initiates Emergency Core Cooling System

<u>Function</u>	<u>Trip Setting</u>	<u>Min. No. of Operable or Operating Trip Systems(3)</u>	<u>Total No. of Instrument Channels Per Trip System</u>	<u>Min. No. of Operable or Operating Instrument Channels Per Trip System (3)</u>	<u>Required Conditions*</u>
<u>D. Diesel Generator</u>					
1. Degraded or Loss of Voltage Essential Bus (5)					
2. Low Low Reactor Water Level	≥6'6"≤6'10"	2	4(4)	4	C.
3. High Drywell Press	≤2 psig	2	4(4)	4	C.

NOTES:

1. High drywell pressure may be bypassed when necessary only by closing the manual containment isolation valves during purging for containment inerting or de-inerting. Verification of the bypass condition shall be noted in the control room log. Also need not be operable when primary containment integrity is not required.
2. One instrument channel is a circuit breaker contact and the other is an undervoltage relay.

Table 3.2.2 - Continued

Notes:

3. Upon discovery that minimum requirements for the number of operable or operating trip systems, or instrument channels are not satisfied action shall be initiated to:
  - (a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or
  - (b) Place the plant under the specified required conditions using normal operating procedures.
4. All instrument channels are shared by both trip systems.
5. See table 3.2.6.
- \* Required conditions when minimum conditions for operation are not satisfied.
  - A. Comply with Specification 3.5.A.
  - B. Reactor pressure  $\leq 150$  psig.
  - C. Comply with Specification 3.9.B.

Table 3.2.8  
Other Instrumentation

Function	Trip Setting	Minimum No. of Operable or Operating Trip System (1)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Conditions*
A. RCIG Initiation					
1. Low-Low Reactor Level	$\geq 6'6"$ & $\leq 6'10"$ above top of active fuel	1	2	2	B
B. HPCI/RCIG Turbine Shutdown					
a. High Reactor Level	$\leq 14'6"$ above top of active fuel	1	2	2	A
C. HPCI/RCIG Turbine Suction Transfer					
a. Condensate Storage Tank Low Level	$\geq 2'0"$ above tank bottom	1	2	2	C

- NOTE:**
1. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated to:
    - a. Satisfy the requirements by placing the appropriate channels or systems in the tripped condition (Turbine/Feedwater Trip only), or
    - b. Place the plant under the specified required condition using normal operating procedures.
- \* Required conditions when minimum conditions for operation are not satisfied:
- A. Reactor in Startup, Refuel, or Shutdown Mode.
  - B. Comply with Specification 3.5.D.
  - C. Align HPCI and RCIG suction to the suppression pool. Restore channels to operable status within 30 days or place the plant in Required Condition A.

### 3.0 LIMITING CONDITION FOR OPERATION

#### 3.5 CORE AND CONTAINMENT SPRAY/COOLING SYSTEMS

##### Applicability:

Applies to the operational status of the emergency cooling systems.

##### Objective:

To insure adequate cooling capability for heat removal in the event of a loss of coolant accident or isolation from the normal reactor heat sink.

##### Specification:

#### A. ECCS Systems

1. Except as specified in section 3.5.A.3, both Core Spray subsystems and the Low Pressure Coolant Injection (LPCI) Subsystem (LPCI Mode of RHR System) shall be operable whenever irradiated fuel is in the reactor vessel and the reactor water temperature is greater than 212°F.
2. Except as specified in section 3.5.A.3, the High Pressure Coolant Injection (HPCI) System and the Automatic Depressurization System (ADS) shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel except during reactor vessel hydrostatic or leakage tests.

### 4.0 SURVEILLANCE REQUIREMENTS

#### 4.5 CORE AND CONTAINMENT SPRAY/COOLING SYSTEMS

##### Applicability:

Applies to the periodic testing of the emergency cooling systems.

##### Objective:

To verify the operability of the emergency cooling systems.

##### Specification:

#### A. ECCS Systems

1. Demonstrate the Core Spray Pumps develop a 2,800 gpm flow rate against a system head corresponding to a reactor pressure of 130 psi greater than containment pressure, when tested pursuant to Specification 4.15.B.
2. Demonstrate the LPCI Pumps develop a 3,870 gpm flow rate against a system head corresponding to two pumps delivering 7,740 gpm at a reactor pressure of 20 psi greater than containment pressure, when tested pursuant to Specification 4.15.B.
3. Demonstrate the HPCI Pump develops a 2700 gpm flow rate against a reactor pressure range of 1120 psig to 150 psig, when tested pursuant to Specification 4.15.B.

### 3.0 LIMITING CONDITION FOR OPERATION

#### F. Recirculation System

1. The reactor may be started and operated, or operation may continue with only one recirculation loop in operation provided that:
  - a. The following changes to setpoints and safety limit settings will be made within 24 hours after initiating operation with only one recirculation loop in operation.
    1. The Operating Limit MCPR (MCPR) will be changed per Specification 3.11.C.
    2. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) will be changed as noted in Table 1 of the Core Operating Limits Report.
    3. The APRM Neutron Flux Scram and APRM Rod Block setpoints will be changed as noted in Specification 2.3.A and Table 3.2.3.
  - b. Total core flow will be maintained greater than 39% when core thermal power is above the limit specified in Figure 3.5.1.

### 4.0 SURVEILLANCE REQUIREMENTS

#### F. Recirculation System

1. See Specification 4.6.G
2. The following baseline noise levels will be obtained prior to operation with only one recirculation pump in operation at a core thermal power greater than that specified in Figure 3.5.1 or with a core flow greater than 45% provided that baseline values have not been established since the last core refueling. Baseline values will be taken with only one recirculation pump running.
  - a. Establish a baseline core plate  $\Delta P$  noise level.
  - b. Establish a baseline APRM and LPRM neutron flux noise level.
3. With only one recirculation loop in operation at a core thermal power greater than that specified in Figure 3.5.1 or with a core flow greater than 45%, determine the following noise levels at least once per 8 hour period and within 30 minutes after a core thermal power increase of greater than 5% of rated thermal power.
  - a. Core plate  $\Delta P$  noise levels.
  - b. APRM and LPRM neutron noise levels.

## Bases 3.5/4.5

### A. ECCS Systems

The core spray system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and, together with the LPCI mode of the RHR system, provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the automatic depressurization system (ADS).

The Core Spray System is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining. The Core Spray pump is designed to deliver greater than or equal to 3020 gpm (the SAFER/GESTR-LOCA safety analysis assumed a Core Spray Pump flow of 2,800 gpm, or 2,700 gpm flow into the core + 100 gpm to account for ECCS bypass leakage) against a system head corresponding to a reactor pressure of 130 psi greater than containment pressure.

The surveillance requirements provide adequate assurance that the Core Spray System will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Four pumps are available to provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS. LPCI Loop Selection Logic determines which Recirculation loop the four RHR pumps will pump into. Each RHR pump was designed to deliver greater than or equal to 4000 gpm (the safety analysis assumed two pumps delivering 7,740 gpm) against a system head corresponding to a reactor pressure of 20 psi greater than containment pressure.

The allowed out-of-service conditions (Section 3.5.A.3) are determined from ECCS analysis cases analyzed. Only one of these conditions is permitted to exist. If more than one condition exists, an orderly shutdown shall be initiated. A LPCI injection path consists of the two motor operated injection valves on that path.

The surveillance requirements provide adequate assurance that the LPCI system will be operable when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The

Bases 3.5/4.5 Continued:

The RHR service water system provides cooling for the RHR heat exchangers and can thus maintain the suppression pool water within limits. With the flow specified, the pool temperature limits are maintained as specified in Specification 3.7.A.1.

D. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. The pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therefore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

The surveillance requirements provide adequate assurance that the RCIC system will be operable when required. All active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

E. Cold Shutdown and Refueling Requirements

The purpose of Specification 3.5.E is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment spray/cooling subsystems may be out of service. This specification allows all core and containment spray/cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.E.2 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

### 3.0 LIMITING CONDITION FOR OPERATION

#### E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F the safety valve function (self actuation) of seven safety/relief valves shall be operable (note: Low-Low Set and ADS requirements are located in Specifications 3.2.H and 3.5.A, respectively).
2. If Specification 3.6.E.1 is not met, initiate an orderly shutdown and have reactor coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.

### 4.0 SURVEILLANCE REQUIREMENTS

#### E. Safety/Relief Valves

1. a. A minimum of seven safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. The nominal self-actuation setpoints are specified in Section 2.4.B.
  - b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.
  - c. The integrity of the safety/relief valve bellows shall be continuously monitored.
  - d. The operability of the bellows monitoring system shall be demonstrated at least once every three months.
2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

Bases Continued 3.6 and 4.6;

The safety/relief valves have two functions; 1) over-pressure relief (self-actuated by high pressure), and 2) Depressurization/Pressure Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation). The Low-Low Set and ADS functions are discussed further in Sections 3.2 and 3.5.

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9 of the ASME Pressure Vessel Code Section III Nuclear Vessels requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system quarterly provisions assurance of bellows integrity.

When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

I. Deleted

### 3.0 LIMITING CONDITION FOR OPERATION

#### 3.7 CONTAINMENT SYSTEMS

##### Applicability:

Applies to the operating status of the primary and secondary containment systems.

##### Objective:

To assure the integrity of the primary and secondary containment systems.

##### Specification:

#### A. Primary Containment.

#### 1. Suppression Pool Volume and Temperature

When irradiated fuel is in the reactor vessel and either the reactor water temperature is greater than 212°F or work is being done which has the potential to drain the vessel, the following requirements shall be met, except as permitted by Specification 3.5.E.2:

- a. Water temperature during normal operating shall be  $\leq 90^{\circ}\text{F}$ .
- b. Water temperature during test operation which adds heat to the suppression pool shall be  $\leq 100^{\circ}\text{F}$  and shall not be  $> 90^{\circ}\text{F}$  for more than 24 hours.
- c. If the suppression chamber water temperature is  $> 110^{\circ}\text{F}$ , the reactor shall be scrammed immediately. Power operation shall not be resumed until the pool temperature is  $\leq 90^{\circ}\text{F}$ .

### 4.0 SURVEILLANCE REQUIREMENTS

#### 4.7 CONTAINMENT SYSTEMS

##### Applicability:

Applies to the primary and secondary containment integrity.

##### Objective:

To verify the integrity of the primary and secondary containment.

##### Specification:

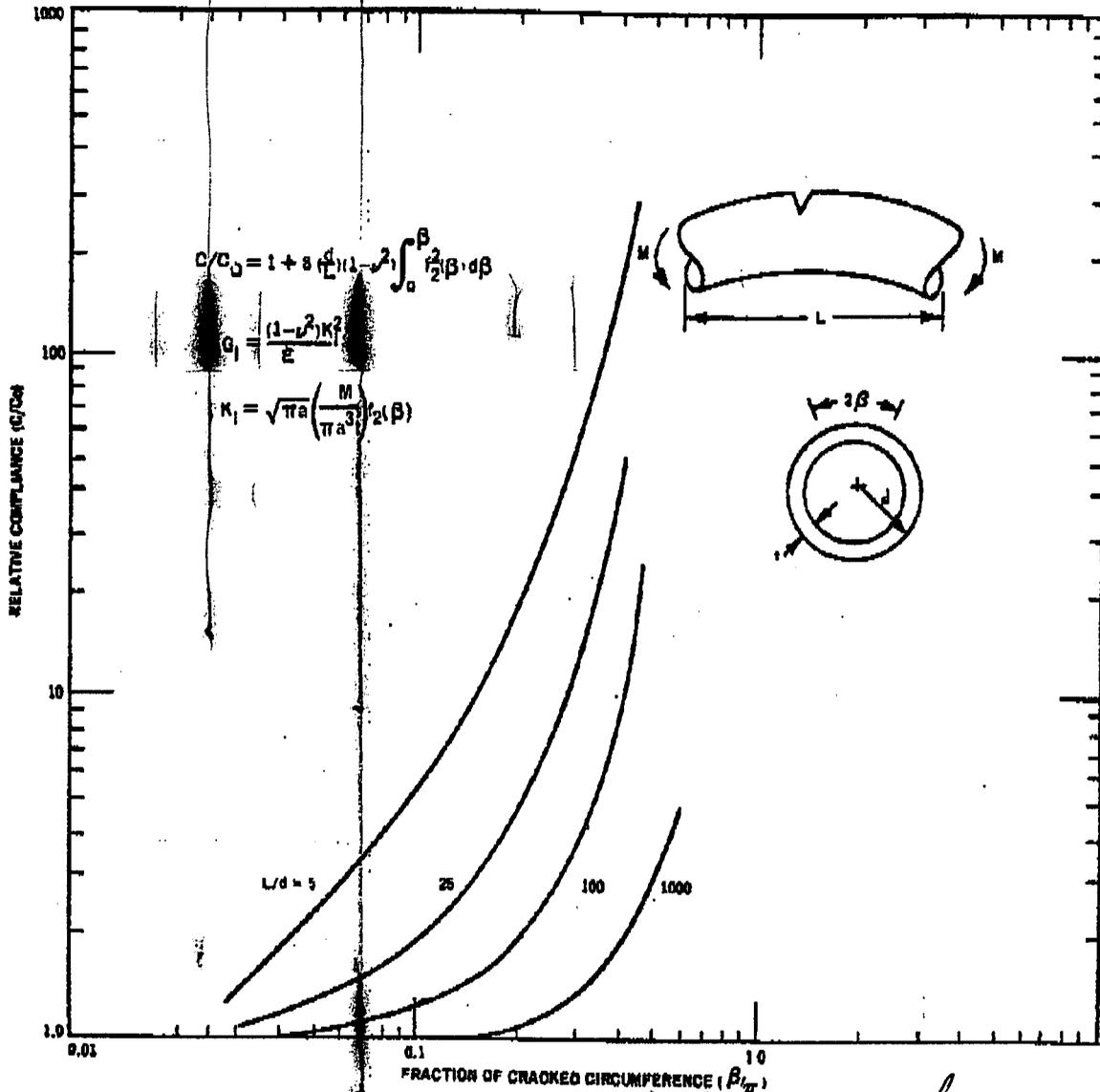
#### A. Primary Containment

#### 1. Suppression Pool Volume and Temperature

- a. The suppression chamber water temperature shall be checked once per day.
- b. Whenever there is indication of relief valve operation which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged ever 5 minutes until the heat addition is terminated.
- c. A visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage

Attachment 7

Page 3-7 of GE Analysis and Reference Equations



$$\frac{C}{C_0} = 1 + 8 \left(\frac{d}{L}\right) (1-\nu^2) \int_0^\beta f_2^2(\beta) d\beta$$

$$G_1 = \frac{(1-\nu^2) K_1^2}{E}$$

$$K_1 = \sqrt{\pi a} \left( \frac{M}{\pi a^3 t} \right) f_2(\beta)$$