Northern States Power Company

Monticello Nuclear Generating Plant
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NSP

July 20, 2000

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 July 20, 2000
Reactor Water Cleanup (RWCU) System Automatic Isolation
and Miscellaneous Instrumentation System Technical Specification Changes

By letter dated July 26, 1996, Northern States Power Company (NSP) requested changes to the Technical Specifications governing the performance of the Control Room Emergency Filtration System and the limit on reactor coolant dose equivalent I-131 (DEI). Our request included a reduction in the DEI limit from 5 to 2 μCi/gm to support a revised Main Steam Line Break (MSLB) accident analysis. Subsequently, by letter dated April 11, 1997, NSP submitted a revision to the July 26, 1996, request which further lowered the DEI limit from 2.0 to 0.25 μCi/gm due to RWCU high energy line break concerns described in our letter dated September 16, 1996. This was an interim measure pending installation of an automatic RWCU line break isolation system as described in our letter dated June 10, 1997.

Installation of the automatic RWCU pipe break isolation system has been completed. In our June 10, 1997, letter we stated that we would submit a request to revise the Monticello Nuclear Generating Plant (MNGP) Technical Specifications to include the automatic RWCU isolation feature. This License Amendment Request satisfies our June 10, 1997 commitment to incorporate the RWCU automatic isolation feature in the MNGP Technical Specifications.

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NSP believes that the DEI limit can now be safely restored to 2 μCi/gm in accordance with the analytical assumption used in the MSLB accident analysis contained in the MNGP Updated Safety Analysis Report (USAR). This License Amendment Request includes a proposed change to restore the DEI limit to 2 μCi/gm.

In addition to the automatic RWCU isolation instrumentation, this License Amendment Request also proposes several additional Technical Specification changes related to other instrumentation systems. These changes implement several desirable improvements and fulfill earlier commitments made to the NRC Staff for Technical Specification improvements. Included are the following requested changes:

a) Change RWCU reactor water level automatic isolation signal from Low to Low-Low reactor water level. Because a design change is required to implement this change, implementation will occur no later than the first refueling outage beginning 180 days or more following NRC approval of this change.

b) Add Technical Specifications for the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) low steam line pressure isolation instrumentation.

c) Delete HPCI 150,000 lb/hr low range high flow isolation instrumentation and add a time delay to 300,000 lb/hr upper range high flow isolation instrumentation. Because a design change is required to implement this change, implementation of the associated Technical Specifications will occur no later than the first refueling outage beginning 180 days or more following NRC approval of this change.

d) Change suppression chamber water allowable water level from volume units to level units.

This request is submitted in accordance with the provisions of 10 CFR Part 50, Section 50.90. Exhibit A contains a description of the proposed changes, the reasons for requesting the changes, a Safety Evaluation, a Determination of No Significant Hazards Consideration, and an Environmental Assessment. Exhibit B contains the current Technical Specification pages marked up with the proposed changes. Exhibit C contains the revised Monticello Technical Specification pages.

Please contact Mr Sam Shirey, Monticello Licensing, at (763) 295-1449 if additional information is required.

This letter does not contain any new NRC commitments and does not modify any prior commitments. This letter contains no restricted or other defense information.

NSP requests a period of up to 45 days following receipt of the approved License Amendment to implement the requested changes.

To the best of my knowledge, information, and belief, the statements made in this document are true and correct.

By Byron D. Day
Byron D. Day
Plant Manager
Monticello Nuclear Generating Plant

Signed before me on this 20TH day of JULY, 2000 by Byron D. Day, Plant Manager, Monticello Nuclear Generating Plant, and being first duly sworn acknowledged that he is authorized to execute this document on behalf of Northern States Power Company.

Samuel I. Shirey

SAMUEL I. SHIREY
NOTARY PUBLIC - MINNESOTA
My Comm. Exp. Jan. 31, 2005

c: Regional Administrator - III, NRC
NRR Project Manager, NRC
Sr. Resident Inspector, NRC
State of Minnesota Department of Commerce
J. E. Silberg, Esq.

Attachments: Exhibit A - Evaluation of Proposed Changes to the Monticello Technical Specifications

Exhibit B - Proposed Changes Marked Up on Existing Technical Specification Pages

Exhibit C - Revised Monticello Technical Specification Pages

EXHIBIT A

Evaluation of Proposed Changes to the Monticello Technical Specifications

License Amendment Request Dated July 20, 2000

Reactor Water Cleanup (RWCU) System Automatic Isolation
and Miscellaneous Instrumentation System Technical Specification Changes

I. Proposed Changes

I.1. Reactor Water Cleanup (RWCU) System Automatic Isolation Instrumentation

Add Limiting Conditions for Operation and Surveillance Requirements for the newly installed RWCU System automatic high energy line break isolation instrumentation. Separate Group 3 isolation entries are proposed in Technical Specification Tables 3.2.1 and 4.2.1. Refer to Exhibit B, pages 50, 62, 63, and 63a.

Increase the reactor coolant dose equivalent I-131 (DEI) limit from 0.25 μCi/gram to 2.0 μCi/gram. The 0.25 μCi/gram DEI limit was an interim measure which is no longer necessary following installation of the automatic RWCU line break instrumentation. Refer to Exhibit B, pages 123 and 148.

Also, revise and add a new paragraph to the Section 3.2 Bases to describe the new automatic RWCU isolation logic. Include a discussion of the basis for the redundant Group 3 closure signal provided for the recirc sample isolation valves. Refer to Exhibit B, pages 64, 65, and 66.

I.2. RWCU Automatic Isolation Reactor Water Level Setpoint

Change the RWCU reactor water level automatic isolation setpoint from Low Reactor Water Level to Low-Low Reactor Water Level to eliminate unnecessary isolations of this system following reactor scrams. This is a plant modification which will be made following NRC approval of this change. Refer to Exhibit B, pages 50, 51a (new page), and 62.

I.3. High Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System Low Steam Line Pressure Isolation Instrumentation

Add Limiting Conditions for Operation and Surveillance Requirements for HPCI and RCIC low steam line pressure isolation instrumentation to satisfy a commitment made to the NRC Staff. Refer to Exhibit B, pages 50, 50a (new page), 51, 51a (new page), 62, 67, and 70.

I.4. Deletion of 150,000 lb/hr HPCI High Flow Isolation Setpoint and Addition of Time Delay to 300,000 lb/hr HPCI High Flow Isolation Setpoint

In an effort to improve the overall reliability of the HPCI System, a modification to the HPCI high steam flow isolation instrumentation is proposed. The low range high flow isolation signal will be deleted and a time delay will be added to the existing high range isolation signal. This will change the high flow logic to be consistent with more recent boiling water reactor protective instrumentation designs and reduce the potential for spurious isolation signals on HPCI startup and testing. Refer to Exhibit B, pages 50, 51a (new page), and 67.

I.5. Suppression Pool Water Inventory Limiting Conditions for Operation (LCOs) and Surveillance Requirements

Change the suppression pool water inventory Limiting Conditions for Operation and Surveillance Requirements from volume measurement (cubic feet) to the equivalent level measurement (inches). Establishing suppression pool inventory limits in terms of level is preferred because it is unambiguous and is directly measured. Refer to Exhibit B, pages 157, 175a, 177, and 183.

II. Reasons for Proposed Changes

II.1. RWCU System Automatic Isolation Instrumentation

NSP is requesting changes to the MNGP Technical Specifications to add Limiting Conditions for Operation and Surveillance Requirements for the RWCU automatic isolation instrumentation that was installed during the 1998 refueling outage. A related change to the reactor coolant system DEI activity is also included. To the extent practical, the requested changes are consistent with those found in the Standard Technical Specifications for General Electric Plants, BWR/4 (NUREG-1433).

An automatic isolation of the RWCU system now occurs if a high RWCU system flow or a high RWCU area temperature is sensed by the new instrumentation. The need for this design change became clear during the Monticello Power Rerate program when it was discovered that significantly higher mass and energy release could result from a RWCU line break than previously assumed. Interim Technical Specification changes were approved by the NRC Staff which reduced the reactor coolant DEI limit during plant operation to 0.25 μCi/gram. This limited the radiological consequences of a RWCU line break to less than the consequences of the previously limiting main steam line break (MSLB).

The addition of automatic RWCU pipe break isolation instrumentation significantly reduces the mass release, and the corresponding release of radioactive material, from the break. The

radiological consequences of the MSLB accident once again bound the RWCU break consistent with MNGP's original licensing basis.

The new instrumentation is designed with redundant logic and initiates a safety related diverse Group 3 isolation. Automatic Group 3 isolation is provided for RWCU supply isolation valves MO-2397 and MO-2398. Recirc sample valves CV-2790 and CV-2791, as well as RWCU return valve MO-2399, also receive the Group 3 isolation signal. Refer to Table 5.2-3b of the Monticello Updated Safety Analysis Report (USAR).

II.1.1. Description of High RWCU Flow Instrumentation RWCU flow is sensed by a common flow element installed in the RWCU piping as near as practical to the outboard containment isolation valve and penetration. A break between the containment isolation valve and the flow element would be isolated by the high temperature instrumentation.

The high system flow logic uses four differential pressure transmitters located outside of the RWCU room and four trip units located in the cable spreading room and Emergency Filtration Treatment (EFT) building. High flow signals are combined in a one-out-of-two-twice logic scheme. The trip logic incorporates a time delay to prevent spurious trips caused by momentary flow disturbances in the RWCU system. A reversal of the differential pressure signal, indicative of a broken flow element sensing line, also initiates automatic RWCU system isolation.

Normal RWCU flow is approximately 140 gpm. The flow element, differential pressure transmitters, and trip units are calibrated to provide an isolation signal at $\leq$ 500 gpm with a time delay of $\leq$ 27 seconds. These flow and time delay setpoints were selected, as shown in Section II.1.3 below, to ensure that the radiological effects of a RWCU break are conservatively bounded by the MSLB accident.

Proposed instrument calibration surveillance requirements for the high flow instrumentation are consistent with NUREG-1433.

II.1.2. RWCU Room High Temperature Isolation Instrumentation High RWCU room temperature is sensed by four RTDs located in the hallway which connects the pump room and the heat exchanger room. This location would see high temperature from a line break in any part of the RWCU room. The RTDs provide signals to analog trip units located in the cable spreading room and EFT building. A one-out-of-two-twice logic is used to initiate a RWCU isolation signal.

The proposed high area temperature setpoint of 188 °F is set sufficiently low to initiate isolation on sensing a line break, but sufficiently high to avoid spurious isolation during normal operation. Calculations show that the temperature in this area would exceed 200 °F within one second of initiation of a design basis break and within 18 seconds of the initiation of a critical crack in the RWCU piping.

The proposed surveillance requirements for the high temperature isolation are consistent with the NUREG-1433 Standard Technical Specifications definition of channel calibration which states in part, "Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel."

II.1.3. High Flow Setpoint Determination and Radiological Consequence Analysis The RWCU high flow isolation and associated time delay setpoint are chosen to ensure that the offsite and control room doses resulting from a RWCU break are conservatively bounded by the MSLB accident.

The maximum mass flow rate resulting from a break in the RWCU system has been calculated by General Electric. A break in the 4-inch RWCU piping at 1880 MWt results in a maximum mass flow rate of 740 lbm/sec of reactor coolant.

The maximum total mass release from the design basis MSLB calculated by General Electric occurs at hot standby and is assumed to be 82,200 lbm in the USAR radiological analysis of this accident.

The radiological impact of a RWCU break or MSLB is directly proportional to the mass of reactor coolant released in the accident. To ensure that the MSLB is the bounding accident, the RWCU isolation logic delay time, T_D, cannot exceed:

$$T_D \leq (82{,}200\text{lbm} / 740\ \text{lbm/sec}) - (T_{VS}) - T_{ID} \qquad = 80\ \text{sec}$$

where T_{VS} = Isolation Valve Stroke Time $\qquad = 29$ sec

T_{ID} = Instrument Response Time $\qquad = 2$ sec

The proposed Technical Specification isolation delay of $\leq$ 27 seconds is sufficiently high to prevent spurious isolation of the RWCU system during normal operation while providing about a factor of three margin to the maximum allowable time delay of 80 seconds.

The maximum permissible RWCU high flow setpoint, F_M, is established by assuming the RWCU break flow is just below the high flow isolation setpoint and the accident is bounded by the MSLB. In this case, it is assumed that isolation of the break is performed manually after a delay of 10 minutes:

$F_M \leq 82{,}200\text{lbm} / (T_{MO} + T_{ID} + T_{VS})$ = 130 lbm/sec

where T_{MO} = Operator Response Time = 600 sec

The high flow setpoint in gal/min is:

$F_M \leq (130 \text{ lbm/sec})(60 \text{ sec}/1 \text{ min})(7.48 \text{ gal}/1 \text{ ft}^3)V_F = 969$ gal/min

where V_F = Specific Volume of Reactor Coolant @ 200 °F (Minimum) specific volume under high energy conditions) = 0.0166 ft^3/lbm

The proposed Technical Specification setpoint of ≤ 500 gal/min is high enough to prevent spurious RWCU isolation while providing about a factor of two margin to the maximum allowable high flow setpoint of 969 gal/min.

The maximum offsite and control room doses resulting from a RWCU line break would occur from a break of slightly less than 500 gpm requiring manual isolation. The total mass, W, of reactor coolant released under these conditions is:

$$W = (500 \text{ gal/min})(T_{MO} + T_{VS})(1 \text{ min}/60 \text{ sec})(1 \text{ ft}^3 / 7.48 \text{ gal}) / V_F$$

$$= 42{,}200 \text{ lbm}$$

The radiological consequences of a RWCU break with a total mass release of 42,200 lbm of reactor coolant with DEI at the proposed Technical Specification limit of 2.0 μCi/gram is presented in Table A-1. The methodology and assumptions used in the analysis in Table A-1 are identical to those used in the analysis of the design basis MSLB accident described in Section 14.7 of the Monticello USAR. The resulting offsite and control room doses are:

RWCU Break Radiological Consequences

Locations	Whole Body Dose (Rem)	Thyroid Dose (Rem)
Exclusion Area Boundary (EAB)	0.204	12.4
Low Population Zone (LPZ)	0.018	1.07
Control Room	0.002	5.26

The corresponding radiological consequences of the design basis MSLB with DEI at the proposed Technical Specification limit of 2.0 μCi/gram results in the following doses as reported in Section 14.7 of the USAR:

MSLB Radiological Consequences (Ref: USAR Tables 14.7-18 & 14.7-19)

Locations	Whole Body Dose (Rem)	Thyroid Dose (Rem)
EAB	0.40	24.2
LPZ	0.03	2.08
Control Room	0.004	10.1

Therefore, offsite and control room doses resulting from a RWCU break with the proposed flow and time delay setpoints will be limited to about 50% of the doses resulting from the design basis MSLB and are considered acceptable.

II.1.4. DEI Limit By letter dated April 11, 1997, NSP proposed to incorporate its DEI administrative limit of 0.25 μCi/gram into the MNGP Technical Specifications. This change became effective with Amendment No. 101 to the Monticello Operating License issued August 28, 1998. This limit was intended to serve as an interim limit until the RWCU automatic isolation design change was completed.

Since the RWCU automatic isolation significantly reduces the radioactivity released from a postulated RWCU break, it is no longer necessary to incorporate this DEI interim limit into the MNGP Technical Specifications.

The interim DEI limit of 0.25 μCi/gram can be safely replaced with the value of 2.0 μCi/gram used in the USAR design basis MSLB radiological analysis, as demonstrated above. The more restrictive interim DEI limit is no longer necessary. The USAR MSLB accident radiological consequences analysis demonstrates that the resulting dose consequences are well within the guidelines of 10 CFR Part 100 and of 10 CFR Part 50, Appendix A, GDC 19 assuming 2.0 μCi/gram steady state DEI.

Changes are included to the Section 3.2 Bases to describe the design basis for new RWCU automatic break detection and isolation logic. A description of the redundant Group 3 automatic isolation signal provided to the recirc sample valves is also added. Group 3 isolation of the recirc sample valves was implemented several years ago as part of a design change to install a continuous intergranular stress corrosion monitoring system. The design change was later determined to be unnecessary and the remainder of the project was cancelled. LCOs and

surveillance requirements are not needed for this redundant isolation feature since it currently serves no essential function.

II.2. RWCU Automatic Isolation Reactor Water Level Setpoint

We proposed changing the RWCU System water level isolation signal from Low Reactor Water Level (≥7-inches in the reactor vessel annulus, which corresponds to ≥10 ft-4 inches above the top of the active fuel) to Low Low Reactor Water Level (>6'-6", <6'-10" above the top of the active fuel).

General Electric Service Information Letter (SIL) 131 issued on April 16, 1975 provides the basis for this requested change. As discussed in SIL 131, reactor scrams during full power operation usually result in a reactor vessel water level transient due to void collapse. This currently can result in the isolation of the RWCU System when the Low Reactor Water Level setpoint is reached. The result is an unnecessary complication in recovery from the scram because operators must restart the RWCU System to control reactor vessel water inventory immediately following the scram.

The SIL recommends the installation of a RWCU automatic break isolation system to provide the primary isolation function for the RWCU and limit the radiological consequences of a RWCU System break. With the installation of an automatic RWCU break isolation system, as described in Section II.1 above, the reactor water level isolation signal can be safely changed from Low Reactor Water Level (Level 3) to Low Low Reactor Water Level (Level 2). Low Low Reactor Water Level corresponds to the level at which emergency core cooling systems receive an automatic initiation signal. RWCU isolation on Low Low Reactor Water Level conforms to NUREG-1433.

An automatic RWCU break detection and isolation system is now installed at Monticello as described above. The RWCU reactor water level isolation signal can be safely lowered to Low Low Reactor Vessel Water Level to prevent unnecessary isolation of this system following reactor scrams.

II.3. HPCI System and RCIC System Low Steam Line Pressure Isolation Instrumentation

We propose adding the low steam line pressure automatic isolation functions for HPCI and RCIC to the Technical Specifications. These automatic isolation features are original plant design features, but they have not previously been included in the Technical Specifications.

This request satisfies the commitment made in our February 3, 1998, letter to incorporate the HPCI and RCIC System low steam line pressure isolation signals in the Technical Specifications. This will achieve consistency with the NRC Standard Technical Specifications for this instrumentation.

The proposed setpoints and allowable deviations were selected, based on engineering judgment, to be low enough to allow for extended non-design basis low pressure HPCI and RCIC system operation, yet high enough to provide system isolation when the function is no longer required and turbine stalling and possible damage could occur.

II.4. Deletion of 150,000 lb/hr HPCI High Flow Isolation Setpoint and Addition of Time Delay to 300,000 lb/hr HPCI High Flow Isolation Setpoint

NSP is requesting changes to the MNGP Technical Specifications to eliminate the existing low range 150,000 lb/hr delayed HPCI high steam flow trip and add a 7-second time delay to the existing high range 300,000 lb/hr high steam flow trip. The purpose of this change is to improve HPCI reliability by reducing the potential for spurious closure of the steam supply valves on HPCI startup. The existing 150,000 lb/hr isolation is too close to the nominal rated HPCI steam flow of 112,000 lb/hr.

A review of the HPCI high steam line flow isolation designs for several other boiling water reactor plants similar to Monticello was performed. Only Monticello has the dual-range HPCI high steam flow isolation logic. A single high range delayed HPCI steam flow isolation signal, nominally 300% of rated flow, is the most common logic provided in other boiling water reactors.

II.4.1. Radiological Consequences of Proposed Changes to HPCI High Steam Flow Isolation Instrumentation

The maximum mass flow rate resulting from a break in the HPCI system has been calculated by General Electric. A break in the 8-inch HPCI steam supply line results in a maximum mass flow rate of 767 lbm/sec.

The maximum total mass release from the design basis MSLB calculated by General Electric occurs at hot standby and is assumed to be 82,200 lbm in the USAR radiological analysis of this accident.

The radiological impact of a HPCI or MSLB break is directly proportional to the mass of reactor coolant released in the accident. To ensure that the MSLB is the bounding accident, the HPCI isolation logic delay time, T_D, cannot exceed:

$$T_D \leq (82{,}200\text{lbm} / 767\ \text{lbm/sec}) - T_{VS} - T_{ID} \qquad = 65\ \text{sec}$$

where T_{VS} = Isolation Valve Stroke Time $\qquad = 40$ sec

T_{ID} = Instrument Response Time $\qquad = 2$ sec

The proposed Technical Specification isolation delay of $\leq$ 7 seconds is sufficiently high to prevent spurious isolation of the HPCI system during normal operation while providing a factor of about nine margin to the maximum allowable time delay of 65 seconds.

The maximum permissible HPCI high flow setpoint, F_M, is established by assuming the HPCI break flow is just below the high flow isolation setpoint and the accident is bounded by the MSLB. In this case, it is assumed that isolation of the break is performed manually after a delay of 10 minutes:

$$F_M \leq 82{,}200\text{lbm} / (T_{MO} + T_{VS}) \qquad = 128 \text{ lbm/sec}$$

$$= 462{,}000 \text{ lbm/hr}$$

where T_{MO} = Operator Response Time $\qquad = 600$ sec

The Technical Specification setpoint of $\leq$ 300,000 lbm/hr is high enough to prevent spurious HPCI isolation, while providing a factor of about 1.5 margin to the maximum allowable high flow setpoint of 462,000 lbm/hr.

The maximum offsite and control room doses resulting from a HPCI line break would occur from a break of just under 300,000 lbm/hr requiring manual isolation. The total mass, W, of reactor coolant released under these conditions is:

$$W = (300{,}000 \text{ lbm/hr})(T_{MO} + T_{VS})(1 \text{ hr}/3600 \text{ sec})$$

$$= 53{,}300 \text{ lbm}$$

The radiological consequences of a HPCI break with a total mass release of 53,300 lbm of reactor coolant with DEI at the proposed Technical Specification limit of 2.0 μCi/gram is presented in Table A-2. The methodology and assumptions used in the analysis in Table A-2 are identical to those used in the analysis of the design basis MSLB accident described in Section 14.7 of the Monticello USAR. The resulting offsite and control room doses are:

HPCI Break Radiological Consequences

Locations	Whole Body Dose (Rem)	Thyroid Dose (Rem)
Exclusion Area Boundary (EAB)	0.258	15.7
Low Population Zone (LPZ)	0.022	1.35
Control Room	0.003	6.65

The corresponding radiological consequences of the design basis MSLB are presented in Section II.1.3, above. The offsite and control room doses resulting from a HPCI break with the proposed isolation logic changes will be limited to about two thirds of the doses resulting from the design basis MSLB and are considered acceptable.

II.4.2. Other Considerations Related to Proposed Changes to HPCI High Steam Flow Isolation Instrumentation

The proposed HPCI isolation high flow setpoint of ≤ 300,000 lb/hr and time delay of ≤ 7 seconds, combined with valve closure times, have been evaluated to demonstrate that the core remains covered in the event of the limiting HPCI steam line break.

The environmental time history curves for a HPCI steam supply line break are bounded by the existing composite profiles used for the Environmental Qualification Program. Additionally, the pressures that could be generated in the steam chase or the Reactor Building are a small fraction of the bounding pressures used to evaluate Reactor Building block wall and structural integrity.

II.5. Suppression Pool Water Inventory LCOs and Surveillance Requirements

It is proposed that allowable Technical Specification suppression pool inventory be expressed in terms of water level instead of cubic feet.

Suppression pool level at Monticello is measured directly by installed instrumentation. Suppression pool volume is determined indirectly from pool level measurements. Level is therefore a more direct and non-ambiguous measure of suppression pool inventory. Expressing pool inventory directly in terms of level is consistent with NUREG-1433.

The proposed band for suppression pool water level is –4.0 to +3.0 inches with respect to the instrumentation zero reference point (elevation 910'-0"). These levels correspond to the maximum downcomer submergence of 3'-7" and the minimum downcomer submergence of 3'-0" used in the Mark I Containment Long Term Program analyses.

The 3'-7" maximum downcomer submergence, with zero drywell-wetwell dp, formed the basis of the loads development for the Mark I Long Term Program at Monticello, except for pool swell impact, drag, froth, and froth fallback loads, where 3'-0" downcomer submergence was also considered. The minimum level corresponding to a downcomer submergence of 3'-0" yields a minimum suppression pool water volume of approximately 68,000 cubic feet. This minimum water inventory is used in the various Monticello accident analyses that involve suppression pool heatup.

The Technical Specification Bases is revised to clearly explain the design basis for the allowable suppression pool water level range.

III. Safety Evaluation

III.1. RWCU System Automatic Isolation Instrumentation

To support the Monticello Power Rerate Project, a thorough review of the mass and energy release assumptions used for all postulated line breaks was conducted, including breaks in the RWCU System. An instantaneous, double-ended, rupture of the RWCU line is assumed for purposes of determining the mass flux. The limiting Monticello RWCU break mass flux was assumed to be 244 lbm/sec based on information originally supplied to the plant. The break was assumed to occur at 100% power and a break isolation time of 120 seconds was assumed to be initiated automatically on Low Reactor Water Level. In August, 1996 GE provided revised mass release calculations for the RWCU break which showed that the limiting RWCU break would occur at power levels $\leq$77%, and would be significantly larger than previously assumed due to a combination of increased mass flux and a 10-minute manual isolation time. The Low Reactor Water Level isolation would be ineffective at reduced power levels due to the ability of the feedwater system to maintain reactor water level.

In response to this concern, the reactor water DEI limit was reduced to 0.25 μCi/gram on an interim basis until an automatic RWCU break isolation system could be designed and installed at Monticello.

Installation of an automatic RWCU break isolation system is now complete. Setpoints for the new isolation instrumentation have been selected which provide for rapid isolation of any significant break, while at the same time preventing spurious isolation of the RWCU System during normal plant operation.

The new RWCU automatic isolation system will limit offsite and control room doses resulting from a RWCU pipe break to a fraction of the dose consequences of the design basis MSLB accident. The interim restriction on reactor coolant steady state activity of 0.25 μCi/gram I-131 dose equivalent is no longer necessary and the value can be safely increased to the USAR value of 2.0 μCi/gram.

III.2. RWCU Automatic Isolation Reactor Water Level Setpoint

A Technical Specification change is requested to change the reactor water level RWCU isolation setpoint from Low Reactor Water Level to Low Low Reactor Water Level consistent with NUREG-1433 and General Electric SIL 131.

The primary RWCU break isolation protection is now provided by the high flow and temperature sensors of the automatic RWCU break isolation system. Low Reactor Water Level RWCU isolation is no longer necessary. Changing the reactor water level RWCU isolation setpoint to Low Low Reactor Water Level will eliminate unnecessary isolation of the RWCU on reactor scrams, while initiating isolation of the RWCU at the same reactor water level setpoint used for initiation of emergency core cooling systems. The high drywell pressure isolation signal will remain unchanged and will continue to provide protection for RWCU line breaks inside the drywell.

This change also revised the Technical Specification Bases to include discussion of the automatic isolation of the reactor water sample lines on receipt of an RWCU isolation signal. This isolation feature was added in conjunction with a planned on-line coolant intergranular stress corrosion cracking (IGSCC) monitoring system.

III.3. HPCI System and RCIC System Low Steam Pressure Isolation Instrumentation

Low steam line isolation is an original plant design feature of the HPCI and RCIC Systems. The proposed Technical Specification changes will simply add LCOs and Surveillance Requirements for the HPCI and RCIC low steam supply pressure isolation logic.

These changes will bring the Monticello Technical Specifications into conformance with NUREG-1433 and satisfy a commitment made earlier to the NRC Staff to submit this change request.

III.4. Deletion of 150,000 lb/hr HPCI High Flow Isolation Setpoint and Addition of Time Delay to 300,000 lb/hr HPCI High Flow Isolation Setpoint

Proposed changes to the HPCI high flow steam line isolation instrumentation will significantly improve HPCI reliability by reducing the potential for spurious steam line isolations on startup of the system.

The ability of the 300,000 lb/hr high flow isolation and associated 7-second time delay to provide an adequate level of protection for a HPCI steam supply line break has been evaluated as described in Section II.4, above. Our evaluation has shown that:

- The offsite and control room doses resulting from a HPCI steam line break remain conservatively bounded by the MSLB accident.

- The proposed HPCI isolation delay, combined with valve closure times, are such that the core remains covered in the event of the limiting steam supply line break.

- HPCI steam line break remain bounded by the environmental conditions established in the Environmental Qualification Program.

III.5. Suppression Pool Water Inventory LCOs and Surveillance Requirements

This is an administrative change which changes the method of specifying allowable suppression pool water inventory from volume measurement to level measurement.

Specifying allowable suppression pool water inventory in terms of level is preferred since it is unambiguous and directly measured.

There are no changes proposed to the actual permissible water inventory or to the permissible submergence of the vent downcomers, simply a change in the method of specifying the acceptable limits for suppression pool inventory.

IV. No Significant Hazards Consideration Evaluation

A new Reactor Water Cleanup (RWCU) pipe break isolation system has been installed at the Monticello Nuclear Generating Plant (MNGP). This License Amendment Request incorporates Limiting Conditions for Operation and Surveillance Requirements for the RWCU automatic pipe break isolation feature in the MNGP Technical Specifications. Installation of this system allows the Dose Equivalent Iodine 131 (DEI) Technical Specification limit to be safely restored to 2 μCi/gm in conformance with the analytical assumption used in the main steam line break (MSLB) accident analysis contained in the MNGP Updated Safety Analysis Report (USAR).

In addition to the automatic RWCU isolation instrumentation, this License Amendment Request also proposes several additional Technical Specification changes related to other instrumentation systems. These changes implement several desirable improvements and fulfill earlier commitments made to the NRC Staff for Technical Specification improvements. Included are the following requested changes:

- Change RWCU reactor water level automatic isolation signal from Low to Low-Low reactor water level.

- Add Technical Specifications for the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) low steam line pressure isolation instrumentation.

- Delete HPCI 150,000 lb/hr low range high flow isolation instrumentation and add a time delay to 300,000 lb/hr upper range high flow isolation instrumentation.

- Change suppression chamber water allowable water level from volume units to level units.

By 10 CFR 50.92, the Commission provided standards for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

NSP believes that the proposed Technical Specification changes do not involve a significant hazards consideration. Each of the 50.92 criterion is discussed below.

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

The proposed new limiting conditions for operation and surveillance requirements for the RWCU high system flow and room temperature signals and the increase in the allowable reactor coolant DEI are administrative in nature and do not involve an increase in the probability or consequences of a previously evaluated accident. The RWCU automatic isolation instrumentation will reduce the consequences of a break in the RWCU System allowing the reactor coolant DEI to be increased, consistent with the value assumed in the Monticello USAR for the design basis MSLB safety analysis.

Changing the reactor water level RWCU automatic isolation setpoint to Low Low Reactor Water Level will not increase the probability or consequences of a break in the RWCU system because new high flow and high area temperature instrumentation provides an improved capability for isolating a RWCU break independent of changes in reactor water level.

Changes to the HPCI steam supply line automatic isolation instrumentation will not increase the probability or consequences of a break in the HPCI steam line. Elimination of the 150,000 lb/hr isolation signal will improve the reliability of the HPCI system. The remaining 300,000 lb/hr delay high flow isolation signal provides more than adequate protection for a steam line break in this system. The consequences of a break in the HPCI remain bounded by the MSLB safety analysis.

Adding a description of the Group 3 logic and recirc sample valves isolation in the Bases; adding LCOs and Surveillance Requirements for the HPCI and RCIC low steam line pressure isolation logic; and changing the method of describing the allowable suppression

pool water inventory from volume to level are all administrative changes than cannot adversely affect the consequences of any evaluated accident.

The proposed changes do not present the opportunity for a new release path for radioactive material.

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

No system, structure, or component (SSC) described in the USAR as important to safety is adversely affected by these changes. No new type of credible event could be identified which would be created by the proposed Technical Specification changes. Nothing was identified in these changes which could create the possibility for a new or different kind of accident.

The RWCU line break accident was previously analyzed. However, non-conservative values for mass and energy release were used. New, more conservative, calculations prompted the prudent installation of an automatic RWCU break isolation system at Monticello. The RWCU line break outside of containment is once again bounded by previously analyzed design basis MSLB accident.

Changes to the HPCI steam line high flow instrumentation will improve the reliability of the HPCI system, while continuing to provide a high degree of protection for a break in the HPCI steam supply line.

Addition of LCOs and Surveillance Requirements for the HPCI and RCIC low steam line pressure isolation and changing the way in which suppression pool level is specified in the Technical Specifications are administrative changes that cannot result in a new or different kind of accident than any previously analyzed.

3) *The proposed amendment will not involve a significant reduction in the margin of safety.*

The margin of safety for the RWCU line break can be expressed in terms of the offsite and control room doses which could result from this event. The previous RWCU line break analysis relied on operator action to isolate the line break after an assumed delay of 10 minutes. The new RWCU automatic isolation instrumentation initiates isolation of a RWCU line break in less than 27 seconds. The new isolation logic limits the potential offsite radiological consequences from a RWCU line break to a fraction of the bounding MSLB accident. The proposed Technical Specification would incorporate LCOs and Surveillance Requirements for the new isolation logic and change the reactor water level setpoint to a value which helps prevent unnecessary isolation of the system following reactor scrams. Margins of safety are improved by these changes.

Proposed changes to the HPCI steam line high flow instrumentation will improve the reliability of the HPCI system and increase existing margins of safety for accidents in which HPCI operation is credited. The margin of safety for a HPCI steam line break can also be expressed in terms of the offsite and control room doses which could result from this event. The modified HPCI isolation logic limits the potential offsite radiological consequences from a HPCI steam line break to a fraction of the bounding MSLB accident.

Adding a description of the Group 3 recirc sample isolation to the Bases, adding LCOs and Surveillance Requirements for the HPCI and RCIC low steam line pressure isolation, and changing the method of describing the allowable suppression pool water inventory are administrative changes. No significant changes in plant equipment or plant operation will occur and no equipment important to safety is affected as a result of these changes. No margin of safety is therefore affected.

V. Environmental Assessment

Northern States Power Company has evaluated the proposed change and believes 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, we believe the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Therefore pursuant to 10 CFR Section 51.22(b), an environmental assessment of the proposed changes is not required.

TABLE A-1
RWCU LINE BREAK RADIOLOGICAL EVALUATION

RWCU LINE BREAK RADIOLOGICAL ANALYSIS — RWCUGEDF.XLS
MONTICELLO NUCLEAR GENERATING PLANT — 28-Jun-00 11:43 AM

ASSUMPTIONS:
1. USAR CHAPTER 14 ASSUMPTIONS - CASE 1 DEI-131
2. GE DOSE FACTORS (REF: OG93-781-09) FOR DOSE COMPUTATION
3. 2 uCi/g DEI131 SOURCE TERM BASED ON RG 1.109
4. 250 CFM UNFILTERED CONTROL ROOM LEAKAGE
98% FILTER EFFICIENCY, 900 CFM FILTER FLOW

EQUIVALENT LIQUID RELEASE (lbm): 4.220E+04

EAB DOSE

LIQUID RELEASED (LBM)	ISOTOPE	COOLANT (uCi/g)	CURIES RELEASED	WH BODY DF (Rem*M³/Ci*sec)	THYROID DF (Rem/Ci)	X/Q (sec/m³)	BR (m³/sec)	WHOLE BODY DOSE (Rem)	THYROID DOSE (Rem)
4.22E+04	I131	1.08E+00	2.07E+01	9.37E-02	1.08E+06	9.20E-04	3.47E-04	1.78E-03	7.13E+00
4.22E+04	I132	4.72E+00	9.03E+01	5.72E-01	6.44E+03	9.20E-04	3.47E-04	4.75E-02	1.86E-01
4.22E+04	I133	3.98E+00	7.62E+01	1.59E-01	1.80E+05	9.20E-04	3.47E-04	1.11E-02	4.38E+00
4.22E+04	I134	1.02E+01	1.94E+02	6.74E-01	1.07E+03	9.20E-04	3.47E-04	1.21E-01	6.64E-02
4.22E+04	I135	3.58E+00	6.85E+01	3.64E-01	3.13E+04	9.20E-04	3.47E-04	2.29E-02	6.85E-01
4.22E+04	Kr83m	3.56E-03	6.82E-02	7.28E-04	0.00E+00	9.20E-04	3.47E-04	4.57E-08	0.00E+00
4.22E+04	Kr85m	6.22E-03	1.19E-01	3.88E-02	0.00E+00	9.20E-04	3.47E-04	4.25E-06	0.00E+00
4.22E+04	Kr85	2.43E-05	4.65E-04	5.65E-04	0.00E+00	9.20E-04	3.47E-04	2.42E-10	0.00E+00
4.22E+04	Kr87	1.94E-02	3.71E-01	2.14E-01	0.00E+00	9.20E-04	3.47E-04	7.31E-05	0.00E+00
4.22E+04	Kr88	1.99E-02	3.81E-01	5.00E-01	0.00E+00	9.20E-04	3.47E-04	1.75E-04	0.00E+00
4.22E+04	Kr89	8.28E-02	1.58E+00	5.54E-01	0.00E+00	9.20E-04	3.47E-04	8.08E-04	0.00E+00
4.22E+04	Xe131m	1.99E-05	3.80E-04	5.75E-03	0.00E+00	9.20E-04	3.47E-04	2.01E-09	0.00E+00
4.22E+04	Xe133m	2.97E-04	5.69E-03	5.95E-03	0.00E+00	9.20E-04	3.47E-04	3.12E-08	0.00E+00
4.22E+04	Xe133	8.31E-03	1.59E-01	1.13E-02	0.00E+00	9.20E-04	3.47E-04	1.65E-06	0.00E+00
4.22E+04	Xe135m	2.44E-02	4.66E-01	1.10E-01	0.00E+00	9.20E-04	3.47E-04	4.72E-05	0.00E+00
4.22E+04	Xe135	2.25E-02	4.30E-01	6.16E-02	0.00E+00	9.20E-04	3.47E-04	2.44E-05	0.00E+00
4.22E+04	Xe137	1.09E-01	2.09E+00	4.79E-02	0.00E+00	9.20E-04	3.47E-04	9.22E-05	0.00E+00
4.22E+04	Xe138	8.28E-02	1.58E+00	2.34E-01	0.00E+00	9.20E-04	3.47E-04	3.41E-04	0.00E+00
	TOTAL	2.39E+01	4.57E+02					2.04E-01	1.24E+01

LPZ DOSE

LIQUID RELEASED (LBM)	ISOTOPE	COOLANT (uCi/g)	CURIES RELEASED	WH BODY DF (Rem*M³/Ci*sec)	THYROID DF (Rem/Ci)	X/Q (sec/m³)	BR (m³/sec)	WHOLE BODY DOSE (Rem)	THYROID DOSE (Rem)
4.22E+04	I131	1.08E+00	2.07E+01	9.37E-02	1.08E+06	7.93E-05	3.47E-04	1.54E-04	6.14E-01
4.22E+04	I132	4.72E+00	9.03E+01	5.72E-01	6.44E+03	7.93E-05	3.47E-04	4.10E-03	1.60E-02
4.22E+04	I133	3.98E+00	7.62E+01	1.59E-01	1.80E+05	7.93E-05	3.47E-04	9.61E-04	3.77E-01
4.22E+04	I134	1.02E+01	1.94E+02	6.74E-01	1.07E+03	7.93E-05	3.47E-04	1.04E-02	5.73E-03
4.22E+04	I135	3.58E+00	6.85E+01	3.64E-01	3.13E+04	7.93E-05	3.47E-04	1.98E-03	5.90E-02
4.22E+04	Kr83m	3.56E-03	6.82E-02	7.28E-04	0.00E+00	7.93E-05	3.47E-04	3.94E-09	0.00E+00
4.22E+04	Kr85m	6.22E-03	1.19E-01	3.88E-02	0.00E+00	7.93E-05	3.47E-04	3.66E-07	0.00E+00
4.22E+04	Kr85	2.43E-05	4.65E-04	5.65E-04	0.00E+00	7.93E-05	3.47E-04	2.08E-11	0.00E+00
4.22E+04	Kr87	1.94E-02	3.71E-01	2.14E-01	0.00E+00	7.93E-05	3.47E-04	6.30E-06	0.00E+00
4.22E+04	Kr88	1.99E-02	3.81E-01	5.00E-01	0.00E+00	7.93E-05	3.47E-04	1.51E-05	0.00E+00
4.22E+04	Kr89	8.28E-02	1.58E+00	5.54E-01	0.00E+00	7.93E-05	3.47E-04	6.96E-05	0.00E+00
4.22E+04	Xe131m	1.99E-05	3.80E-04	5.75E-03	0.00E+00	7.93E-05	3.47E-04	1.73E-10	0.00E+00
4.22E+04	Xe133m	2.97E-04	5.69E-03	5.95E-03	0.00E+00	7.93E-05	3.47E-04	2.69E-09	0.00E+00
4.22E+04	Xe133	8.31E-03	1.59E-01	1.13E-02	0.00E+00	7.93E-05	3.47E-04	1.42E-07	0.00E+00
4.22E+04	Xe135m	2.44E-02	4.66E-01	1.10E-01	0.00E+00	7.93E-05	3.47E-04	4.07E-06	0.00E+00
4.22E+04	Xe135	2.25E-02	4.30E-01	6.16E-02	0.00E+00	7.93E-05	3.47E-04	2.10E-06	0.00E+00
4.22E+04	Xe137	1.09E-01	2.09E+00	4.79E-02	0.00E+00	7.93E-05	3.47E-04	7.95E-06	0.00E+00
4.22E+04	Xe138	8.28E-02	1.58E+00	2.34E-01	0.00E+00	7.93E-05	3.47E-04	2.94E-05	0.00E+00
	TOTAL	2.39E+01	4.57E+02					1.76E-02	1.07E+00

CONTROL ROOM DOSE

LIQUID RELEASED (LBM)	ISOTOPE	COOLANT (uCi/g)	CURIES RELEASED	WH BODY DF (Rem*M³/Ci*sec)	THYROID DF (Rem/Ci)	X/Q (sec/m³)	BR (m³/sec)	GEOMETRY FACTOR	IODINE PROT FACTOR	WHOLE BODY DOSE (Rem)	THYROID DOSE (Rem)
4.22E+04	I131	1.08E+00	2.07E+01	9.37E-02	1.08E+06	1.67E-03	3.47E-04	3.73E+01	4.29E+00	2.02E-05	3.02E+00
4.22E+04	I132	4.72E+00	9.03E+01	5.72E-01	6.44E+03	1.67E-03	3.47E-04	3.73E+01	4.29E+00	5.39E-04	7.86E-02
4.22E+04	I133	3.98E+00	7.62E+01	1.59E-01	1.80E+05	1.67E-03	3.47E-04	3.73E+01	4.29E+00	1.26E-04	1.85E+00
4.22E+04	I134	1.02E+01	1.94E+02	6.74E-01	1.07E+03	1.67E-03	3.47E-04	3.73E+01	4.29E+00	1.37E-03	2.81E-02
4.22E+04	I135	3.58E+00	6.85E+01	3.64E-01	3.13E+04	1.67E-03	3.47E-04	3.73E+01	4.29E+00	2.60E-04	2.90E-01
4.22E+04	Kr83m	3.56E-03	6.82E-02	7.28E-04	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	2.22E-09	0.00E+00
4.22E+04	Kr85m	6.22E-03	1.19E-01	3.88E-02	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	2.07E-07	0.00E+00
4.22E+04	Kr85	2.43E-05	4.65E-04	5.65E-04	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	1.18E-11	0.00E+00
4.22E+04	Kr87	1.94E-02	3.71E-01	2.14E-01	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	3.56E-06	0.00E+00
4.22E+04	Kr88	1.99E-02	3.81E-01	5.00E-01	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	8.54E-06	0.00E+00
4.22E+04	Kr89	8.28E-02	1.58E+00	5.54E-01	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	3.93E-05	0.00E+00
4.22E+04	Xe131m	1.99E-05	3.80E-04	5.75E-03	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	9.79E-11	0.00E+00
4.22E+04	Xe133m	2.97E-04	5.69E-03	5.95E-03	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	1.52E-09	0.00E+00
4.22E+04	Xe133	8.31E-03	1.59E-01	1.13E-02	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	8.05E-08	0.00E+00
4.22E+04	Xe135m	2.44E-02	4.66E-01	1.10E-01	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	2.30E-06	0.00E+00
4.22E+04	Xe135	2.25E-02	4.30E-01	6.16E-02	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	1.19E-06	0.00E+00
4.22E+04	Xe137	1.09E-01	2.09E+00	4.79E-02	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	4.49E-06	0.00E+00
4.22E+04	Xe138	8.28E-02	1.58E+00	2.34E-01	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	1.66E-05	0.00E+00
	TOTAL	2.39E+01	4.57E+02							2.39E-03	5.26E+00

IODINE PROTECTION FACTOR (IPF):			GEOMETRY FACTOR:	
	EFT FLOW (cfm)	900	MURPHY & CAMPE	
	INFILT (cfm)	250	VOL(CF)=	27000
	FILTER EFF (%)	98	GF=	3.73E+01
	IPF	4.29		

TABLE A-2
HPCI STEAM LINE BREAK RADIOLOGICAL EVALUATION

HPCI LINE BREAK RADIOLOGICAL ANALYSIS — HPCIGEDF.XLS
MONTICELLO NUCLEAR GENERATING PLANT — 28-Jun-00 — 12:04 PM

ASSUMPTIONS:
1. USAR CHAPTER 14 ASSUMPTIONS - CASE 1 DEI-131
2. GE DOSE FACTORS (REF: OG93-781-09) FOR DOSE COMPUTATION
3. 2 uCi/g DEI131 SOURCE TERM BASED ON RG 1.109
4. 250 CFM UNFILTERED CONTROL ROOM LEAKAGE
98% FILTER EFFICIENCY, 900 CFM FILTER FLOW

EQUIVALENT LIQUID RELEASE (lbm): 5.330E+04

EAB DOSE

LIQUID RELEASED (LBM)	ISOTOPE	COOLANT (uCi/g)	CURIES RELEASED	WH BODY DF (Rem*M³/Ci*sec)	THYROID DF (Rem/Ci)	X/Q (sec/m³)	BR (m³/sec)	WHOLE BODY DOSE (Rem)	THYROID DOSE (Rem)
5.33E+04	I131	1.08E+00	2.61E+01	9.37E-02	1.08E+06	9.20E-04	3.47E-04	2.25E-03	9.00E+00
5.33E+04	I132	4.72E+00	1.14E+02	5.72E-01	6.44E+03	9.20E-04	3.47E-04	6.01E-02	2.35E-01
5.33E+04	I133	3.98E+00	9.62E+01	1.59E-01	1.80E+05	9.20E-04	3.47E-04	1.41E-02	5.53E+00
5.33E+04	I134	1.02E+01	2.46E+02	6.74E-01	1.07E+03	9.20E-04	3.47E-04	1.52E-01	8.39E-02
5.33E+04	I135	3.58E+00	8.66E+01	3.64E-01	3.13E+04	9.20E-04	3.47E-04	2.90E-02	8.65E-01
5.33E+04	Kr83m	3.56E-03	8.62E-02	7.28E-04	0.00E+00	9.20E-04	3.47E-04	5.77E-08	0.00E+00
5.33E+04	Kr85m	6.22E-03	1.50E-01	3.88E-02	0.00E+00	9.20E-04	3.47E-04	5.37E-06	0.00E+00
5.33E+04	Kr85	2.43E-05	5.88E-04	5.65E-04	0.00E+00	9.20E-04	3.47E-04	3.05E-10	0.00E+00
5.33E+04	Kr87	1.94E-02	4.69E-01	2.14E-01	0.00E+00	9.20E-04	3.47E-04	9.23E-05	0.00E+00
5.33E+04	Kr88	1.99E-02	4.81E-01	5.00E-01	0.00E+00	9.20E-04	3.47E-04	2.21E-04	0.00E+00
5.33E+04	Kr89	8.28E-02	2.00E+00	5.54E-01	0.00E+00	9.20E-04	3.47E-04	1.02E-03	0.00E+00
5.33E+04	Xe131m	1.99E-05	4.80E-04	5.75E-03	0.00E+00	9.20E-04	3.47E-04	2.54E-09	0.00E+00
5.33E+04	Xe133m	2.97E-04	7.19E-03	5.95E-03	0.00E+00	9.20E-04	3.47E-04	3.94E-08	0.00E+00
5.33E+04	Xe133	8.31E-03	2.01E-01	1.13E-02	0.00E+00	9.20E-04	3.47E-04	2.09E-06	0.00E+00
5.33E+04	Xe135m	2.44E-02	5.89E-01	1.10E-01	0.00E+00	9.20E-04	3.47E-04	5.96E-05	0.00E+00
5.33E+04	Xe135	2.25E-02	5.44E-01	6.16E-02	0.00E+00	9.20E-04	3.47E-04	3.08E-05	0.00E+00
5.33E+04	Xe137	1.09E-01	2.64E+00	4.79E-02	0.00E+00	9.20E-04	3.47E-04	1.16E-04	0.00E+00
5.33E+04	Xe138	8.28E-02	2.00E+00	2.34E-01	0.00E+00	9.20E-04	3.47E-04	4.31E-04	0.00E+00
	TOTAL	2.39E+01	5.78E+02					2.58E-01	1.57E+01

LPZ DOSE

LIQUID RELEASED (LBM)	ISOTOPE	COOLANT (uCi/g)	CURIES RELEASED	WH BODY DF (Rem*M³/Ci*sec)	THYROID DF (Rem/Ci)	X/Q (sec/m³)	BR (m³/sec)	WHOLE BODY DOSE (Rem)	THYROID DOSE (Rem)
5.33E+04	I131	1.08E+00	2.61E+01	9.37E-02	1.08E+06	7.93E-05	3.47E-04	1.94E-04	7.76E-01
5.33E+04	I132	4.72E+00	1.14E+02	5.72E-01	6.44E+03	7.93E-05	3.47E-04	5.18E-03	2.02E-02
5.33E+04	I133	3.98E+00	9.62E+01	1.59E-01	1.80E+05	7.93E-05	3.47E-04	1.21E-03	4.77E-01
5.33E+04	I134	1.02E+01	2.46E+02	6.74E-01	1.07E+03	7.93E-05	3.47E-04	1.31E-02	7.23E-03
5.33E+04	I135	3.58E+00	8.66E+01	3.64E-01	3.13E+04	7.93E-05	3.47E-04	2.50E-03	7.45E-02
5.33E+04	Kr83m	3.56E-03	8.62E-02	7.28E-04	0.00E+00	7.93E-05	3.47E-04	4.97E-09	0.00E+00
5.33E+04	Kr85m	6.22E-03	1.50E-01	3.88E-02	0.00E+00	7.93E-05	3.47E-04	4.62E-07	0.00E+00
5.33E+04	Kr85	2.43E-05	5.88E-04	5.65E-04	0.00E+00	7.93E-05	3.47E-04	2.63E-11	0.00E+00
5.33E+04	Kr87	1.94E-02	4.69E-01	2.14E-01	0.00E+00	7.93E-05	3.47E-04	7.96E-06	0.00E+00
5.33E+04	Kr88	1.99E-02	4.81E-01	5.00E-01	0.00E+00	7.93E-05	3.47E-04	1.91E-05	0.00E+00
5.33E+04	Kr89	8.28E-02	2.00E+00	5.54E-01	0.00E+00	7.93E-05	3.47E-04	8.79E-05	0.00E+00
5.33E+04	Xe131m	1.99E-05	4.80E-04	5.75E-03	0.00E+00	7.93E-05	3.47E-04	2.19E-10	0.00E+00
5.33E+04	Xe133m	2.97E-04	7.19E-03	5.95E-03	0.00E+00	7.93E-05	3.47E-04	3.39E-09	0.00E+00
5.33E+04	Xe133	8.31E-03	2.01E-01	1.13E-02	0.00E+00	7.93E-05	3.47E-04	1.80E-07	0.00E+00
5.33E+04	Xe135m	2.44E-02	5.89E-01	1.10E-01	0.00E+00	7.93E-05	3.47E-04	5.14E-06	0.00E+00
5.33E+04	Xe135	2.25E-02	5.44E-01	6.16E-02	0.00E+00	7.93E-05	3.47E-04	2.66E-06	0.00E+00
5.33E+04	Xe137	1.09E-01	2.64E+00	4.79E-02	0.00E+00	7.93E-05	3.47E-04	1.00E-05	0.00E+00
5.33E+04	Xe138	8.28E-02	2.00E+00	2.34E-01	0.00E+00	7.93E-05	3.47E-04	3.71E-05	0.00E+00
	TOTAL	2.39E+01	5.78E+02					2.22E-02	1.35E+00

CONTROL ROOM DOSE

LIQUID RELEASED (LBM)	ISOTOPE	COOLANT (uCi/g)	CURIES RELEASED	WH BODY DF (Rem*M³/Ci*sec)	THYROID DF (Rem/Ci)	X/Q (sec/m³)	BR (m³/sec)	GEOMETRY FACTOR	IODINE PROT FACTOR	WHOLE BODY DOSE (Rem)	THYROID DOSE (Rem)
5.33E+04	I131	1.08E+00	2.61E+01	9.37E-02	1.08E+06	1.67E-03	3.47E-04	3.73E+01	4.29E+00	2.55E-05	3.81E+00
5.33E+04	I132	4.72E+00	1.14E+02	5.72E-01	6.44E+03	1.67E-03	3.47E-04	3.73E+01	4.29E+00	6.81E-04	9.92E-02
5.33E+04	I133	3.98E+00	9.62E+01	1.59E-01	1.80E+05	1.67E-03	3.47E-04	3.73E+01	4.29E+00	1.60E-04	2.34E+00
5.33E+04	I134	1.02E+01	2.46E+02	6.74E-01	1.07E+03	1.67E-03	3.47E-04	3.73E+01	4.29E+00	1.73E-03	3.55E-02
5.33E+04	I135	3.58E+00	8.66E+01	3.64E-01	3.13E+04	1.67E-03	3.47E-04	3.73E+01	4.29E+00	3.29E-04	3.66E-01
5.33E+04	Kr83m	3.56E-03	8.62E-02	7.28E-04	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	2.81E-09	0.00E+00
5.33E+04	Kr85m	6.22E-03	1.50E-01	3.88E-02	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	2.61E-07	0.00E+00
5.33E+04	Kr85	2.43E-05	5.88E-04	5.65E-04	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	1.49E-11	0.00E+00
5.33E+04	Kr87	1.94E-02	4.69E-01	2.14E-01	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	4.50E-06	0.00E+00
5.33E+04	Kr88	1.99E-02	4.81E-01	5.00E-01	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	1.08E-05	0.00E+00
5.33E+04	Kr89	8.28E-02	2.00E+00	5.54E-01	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	4.97E-05	0.00E+00
5.33E+04	Xe131m	1.99E-05	4.80E-04	5.75E-03	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	1.24E-10	0.00E+00
5.33E+04	Xe133m	2.97E-04	7.19E-03	5.95E-03	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	1.92E-09	0.00E+00
5.33E+04	Xe133	8.31E-03	2.01E-01	1.13E-02	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	1.02E-07	0.00E+00
5.33E+04	Xe135m	2.44E-02	5.89E-01	1.10E-01	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	2.90E-06	0.00E+00
5.33E+04	Xe135	2.25E-02	5.44E-01	6.16E-02	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	1.50E-06	0.00E+00
5.33E+04	Xe137	1.09E-01	2.64E+00	4.79E-02	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	5.67E-06	0.00E+00
5.33E+04	Xe138	8.28E-02	2.00E+00	2.34E-01	0.00E+00	1.67E-03	3.47E-04	3.73E+01	0.00E+00	2.10E-05	0.00E+00
	TOTAL	2.39E+01	5.78E+02							3.02E-03	6.65E+00

IODINE PROTECTION FACTOR (IPF):			GEOMETRY FACTOR:	MURPHY & CAMPE	
	EFT FLOW (cfm)	900			
	INFILT (cfm)	250		VOL(CF)=	27000
	FILTER EFF (%)	98		GF=	3.73E+01
	IPF	4.29			

EXHIBIT B

Proposed Changes Marked Up on Existing Technical Specification Pages

License Amendment Request Dated July 20, 2000

Reactor Water Cleanup (RWCU) System Automatic Isolation
and Miscellaneous Instrumentation System Technical Specification Changes

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

Pages

50
50a (new page)
51
51a (new page)
62
63
63a
64
65
66
67
70
123
148
157
175a
177
183

Table 3.2.1 (Continued)

Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1, 2)	Required Conditions *
b. High Drywell Pressure (5)	≤ 2 psig	2	2	D
3. Reactor Cleanup System (Group 3)				
~~a. Low Reactor Water Level~~	~~≥ 7" (annulus)~~	~~2~~	~~2~~	~~E~~
a. ~~b.~~ High Drywell Pressure	≤ 2 psig	2	2	E
b. Low Low Reactor Water Level ***	≥ 6'-6", ≤ 6'10"	2	2	E
c. High RWCU Room Temperature Allowable Value	≤ 188 °F	2	2	E
d. High RWCU System Flow Allowable Value	≤ 500 gpm, ≤ 27 sec delay	2	2	E
4. HPCI Steam Lines (Group 4)				
a. HPCI High Steam Flow ***	≤ ~~150,000~~ 300,000 lb/hr with ≤ ~~60~~ 7 second time delay	2(4)	2	F
~~b. HPCI High Steam Flow~~	~~≤ 300,000 lb/hr~~	~~2(4)~~	~~2~~	~~F~~
b. ~~c.~~ HPCI Steam Line Area High Temp.	≤ 200°F	16(4)	16	F
c. Low Pressure in HPCI Steam Supply Line	≥ 85 psig	4(6)	4(6)	F

Table 3.2.1 (Continued)

Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1, 2)	Required Conditions*
5. <u>RCIC Steam Lines</u> (Group 5)				
a. RCIC High Steam Flow	≤ 45,000 lb/hr with 5 ± 2 sec time delay	2(4)	2	G
b. RCIC Steam Line Area	≤ 200°F	16(4)	16	G
c. Low Pressure in RCIC Steam Supply Line	≥ 55 psig	4 (7)	4 (7)	G
6. <u>Shutdown Cooling Supply Isolation</u>				
a. Reactor Pressure Interlock	≤ 75 psig at the reactor steam dome	2(4)	2	C

Table 3.2.1 (Continued)

NOTES:

(1) There shall be two operable or tripped trip systems for each function. A channel (a shared channel is considered one channel) may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other operable channel in the same trip system is monitoring that parameter.

(2) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied action shall be initiated as follows:

(a) With one required instrument channel inoperable in one or more trip functions, place the inoperable channel(s) or trip system in the tripped condition within 12 hours, or

(b) With more than one instrument channel inoperable for one or more trip functions, immediately satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

(c) Place the plant under the specified required conditions using normal operating procedures.

(3) Low pressure in main steam line only need to be available in the RUN position.

(4) All instrument channels are shared by both trip systems.

(5) 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.

(6) The four pressure switches are arranged in a one-out-of-two-twice logic, the output of the logic providing a trip signal to a single trip system for isolation.

(7) The four pressure switches are arranged in a one-out-of-two-twice logic, the output of the logic providing a trip signal to each of two separate trip systems. Each trip system is able, by itself, to initiate isolation.

Table 3.2.1 (Continued)

NOTES (continued):

* Required conditions when minimum conditions for operation are not satisfied.

A. Group 1 isolation valves closed.
B. Reactor Power on IRM range or below and reactor in startup, refuel, or shutdown mode.
C. Isolation Valves closed for: Shutdown Cooling System, and Reactor Head Cooling Line.
D. Comply with Condition C. above.
E. Isolation Valves closed for: Reactor Cleanup System.
F. HPCI steam line isolated. (See specification 3.5 for additional requirements.)
G. RCIC steam line isolated.

** Function changed from Low Reactor Water Level to Low Low Reactor Water Level following completion of design change.

*** Function changed from ≤ 150,000 lb /hr, ≤ 60 sec delay, and ≤ 300,000 lb/hr, instantaneous, isolation to ≤300,000 lb /hr, ≤ 7 sec delay, isolation following completion of design change.

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
3. Steam Line Low Pressure	Once/3 months	Once/3 months	None
4. Reactor Low Low Water Level	Once/3 months (Note 5)	Every Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/12 hours
CONTAINMENT ISOLATION (GROUP 2)			
1. Reactor Low Water Level (Note 10)	-	-	-
2. Drywell High Pressure (Note 10)	-	-	-
RWCU ISOLATION (GROUP 3)			
1. High RWCU Room Temperature	Once/3 months	Once Operating Cycle - RTD Input (Note 12) Once/3 months - Trip Unit	Once/12 hours
2. High RWCU System Flow	Once/3 months	Once Operating Cycle - Transmitter Once/3 Months - Trip Unit	Once/12 hours
* 3. Reactor Low Low Water Level (Note 11)	-	-	
4. Drywell High Pressure (Note 10)	-	-	
HPCI (GROUP 4) ISOLATION			
1. Steam Line High Flow	Once/3 months	Once/3 months	None
2. Steam Line High Temperature	Once/3 months	Once/3 months	None
3. Steam Line Low Pressure	Once/3 months	Once/3 months	None
RCIC (GROUP 5) ISOLATION			
1. Steam Line High Flow	Once/3 months	Once/3 months	None
2. Steam Line High Temperature	Once/3 months	Once/3 months	None
3. Steam Line Low Pressure	Once/3 months	Once/3 months	None

* Function changed from Low Reactor Water Level to Low Low Reactor Water Level following completion of design change

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation

Moved

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
REACTOR BUILDING VENTILATION & STANDBY GAS TREATMENT			
1. Reactor Low Low Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Once/3 months - Trip Unit	Once/12 hours
2. Drywell High Pressure (Note 10)	-	-	-
3. Radiation Monitors (Plenum)	Once/3 months	Once/3 months	Once/day
4. Radiation Monitors (Refueling Floor)	Once/3 months	Once/3 months	Note 4
RECIRCULATION PUMP TRIP AND ALTERNATE ROD INJECTION			
1. Reactor High Pressure	Once/3 months (Note 5)	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/Day
2. Reactor Low Low Water Level	Once/3 months (Note 5)	Once/Operating Cycle- Transmitter Once/3 Months-Trip Unit	Once/12 hours
SHUTDOWN COOLING SUPPLY ISOLATION			
1. Reactor Pressure Interlock	Once/3 months	Once/3 Months	None
SAFEGUARDS BUS VOLTAGE			
1. Degraded Voltage Protection	Once/month	Quarterly	Not applicable
2. Loss of Voltage Protection	Once/month	Once/Operating Cycle	Not applicable
SAFETY/RELIEF VALVE LOW-LOW SET LOGIC			
1. Reactor Scram Sensing	Once/Shutdown (Note 8)	-	-
2. Reactor Pressure - Opening	Once/3 months (Note 5)	Once/Operating Cycle	Once/day
3. Reactor Pressure - Closing	Once/3 months (Note 5)	Once/Operating Cycle	Once/day
4. Discharge Pipe Pressure	Once/3 months (Note 5)	See Table 4.14.1	See Table 4.14.1
5. Inhibit Timer	Once/3 months (Note 5)	Once/Operating Cycle	-
CONTROL ROOM HABITABILITY PROTECTION			
1. Radiation	Monthly (Note 5)	18 months	Daily

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

NOTES:

(1) (Deleted)

(2) Calibrate prior to normal shutdown and start-up and thereafter check once per 12 hours and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed, as necessary, in the APRM/IRM overlap region.

(3) Functional tests, calibrations and sensor checks are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

(4) Whenever fuel handling is in process, a sensor check shall be performed once per 12 hours.

(5) A functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action.

(6) (Deleted)

(7) (Deleted)

(8) Once/shutdown if not tested during previous 3 month period.

(9) Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position.

(10) ~~Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2.~~

(11) Uses contacts from Group 1 Isolation logic. Tested and calibrated in accordance with Group 1 Low Low Water Level instrumentation.

(12) Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

Bases 3.2:

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminate a single operator error before it results in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, and other safety related functions. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 7″ on the instrument. This corresponds to a lower water level above the top of active fuel at 100% power due to the pressure drop across the dryer/separator. This has been accounted for in the affected transient analysis. This trip initiates closure of ~~group 2, and 3~~ primary containment isolation valves. Reference Section 7.7.2.2 FSAR. The trip setting provides assurance that the valves will be closed before perforation of the clad occurs even for the maximum break in that line and therefore the setting is adequate.

The low low reactor water level instrumentation is set to trip when reactor water level is 6′6″ above the top of the active fuel. This trip initiates closure of the Group 1 Primary containment isolation valves, Reference Section 7.7.2.2 FSAR, and also activates the ECC systems and starts the emergency diesel generator.

and Group 3

Group 2

Bases 3. 2 (Continued):

This trip setting level was chosen to be low enough to prevent spurious operation but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Section 6.2.7 and 14.6.3 FSAR. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria. Reference Section 6.2.7 FSAR.

The high drywell pressure instrumentation is a back-up to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 and Group 3 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 and Group 3 isolation valves include the drywell vent, purge, sump isolation, ~~and~~ RWCU, ~~system valves.~~ and recirc sample valves.

Two pressure switches are provided on the discharge of each of the two core spray pumps and each of the four RHR pumps. Two trip systems are provided in the control logic such that either trip system can permit automatic depressurization. Each trip system consists of two trip logic channels such that both trip logic channels are required to permit a system trip.

Division I core spray and RHR pump discharge pressure permissives will interlock one trip system and Division II permissives will interlock the other trip system. One pressure switch on each pump will interlock one of the trip channels and the other pressure switch will interlock the other trip channel within their respective trip system.

The pump pressure permissive control logic is designed such that no single failure (short or open circuit) will prevent auto-blowdown or allow auto-blowdown when not required. The trip setting for the low pressure ECCS pump permissive for ADS is set such that it is less than the pump discharge pressure when a pump is operating in a full flow condition and also high enough to avoid any condition that results in a discharge pressure permissive when the pumps are not operating.

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow,

Bases 3.2 (Continued):

instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst case accident, main steamline break outside the drywell, this trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steamline valve closure, limit the mass inventory loss such that fuel is not uncovered, fuel clad temperatures remain less than 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Sections 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a back-up to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

Pressure instrumentation is provided which trips when main steamline pressure drops below 825 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "refuel" and "Startup" mode this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 825 psig inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500°F; thus, there are no fission products available for release other than those in the reactor water. Reference License Amendment Request Dated December 1, 1975 from L. O. Mayer (NSP) to R. S. Boyd (USNRC).

ADD

The RWCU high flow and temperature instrumentation is provided to detect a break in the RWCU piping. Tripping of this instrumentation results in actuation of the RWCU isolation valves; i.e. Group 3 valves. The trip settings have been established so that the radiological consequences of a high energy line break in this system are bounded by a break in the main steam system. The recirc sample isolation valves, which receive a Group 1 isolation signal, also receive a redundant Group 3 isolation signal.

Bases 3.2 (Continued):

REPLACE

~~The HPCI and/or RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves; i.e., Group 4 and/or Group 5 valves. The trip settings of 200°F and 150% of HPCI and 300% of RCIC design flows and valve closure times are such that the core will not be uncovered and fission product release will not exceed 10 CFR 100 guidelines.~~

The HPCI and RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI or RCIC piping. The trip settings of 200 °F and approximately 300% of HPCI and RCIC design steam flow are such that the core will not be uncovered and the radiological consequences are bounded by the main steam line break accident. The HPCI and RCIC low steam line pressure instrumentation protects the HPCI and RCIC turbines when system operation is no longer useful or possible. Tripping of the high flow, high temperature, or low steam line pressure instrumentation results in actuation of the HPCI or RCIC steam supply valves; i.e. a Group 4 or Group 5 isolation.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification prese[illegible] or testing is being performed.

The control rod bloc[illegible] the Safety Limit (T.S.2.1.A). [illegible] SRM's will result in a rod b[illegible] short period of time t[illegible]

The APRM roc[illegible] ecially during operati[illegible] withdrawal of [illegible] ter than that stat[illegible] or recirculation c[illegible] an 50% of design due [illegible]

The RBM provi[illegible] ngle rod withdrawal error fro[illegible] ed by the APRMs. A statistical analysis [illegible] level the results show that with the specified trip settings, [illegible] Limit, thus allowing adequate margin. This analysis assumes a steady state MCPR prior to the postulated rod withdrawal error. The RBM functions are required when core thermal power is greater than 30% and a Limiting Control Rod Pattern exists. When both RBM channels are operating either channel will assure required withdrawal blocks occur even assuming a single failure of one channel. With one RBM channel inoperable for no more than 24 hours, testing of the RBM prior to withdrawal of control rods assures that improper control rod withdrawal will be blocked. Requiring at least half of the normal LPRM inputs to be operable assures that the RBM response will be adequate to protect against rod withdrawal errors, as shown by a statistical failure analysis.

Bases 3.2 (Continued):

	Trip Function	Deviation
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Specification 3.2.E.3 and Table 3.2.4	Reactor Building Vent Plenum Monitors	+5 mR/hr
	Refueling Floor Radiation Monitors	+5 mR/hr
	* Low Low Reactor Water Level High Drywell Pressure	-3 inches +1 psi
Primary Containment Isolation Functions Table 3.2.1	* Low Low Water Level	-3 inches
	High Flow in Main Steam Line	+2%
	High Temp. in Main Steam Line Tunnel	+10°F
	Low Pressure in Main Steam Line	-10 psi
	High Drywell Pressure	+1 psi
	* Low Reactor Water Level	-6 inches
	HPCI High Steam Flow	+7,500 lb/hr
	HPCI Steam Line Area High Temp.	+2°F
	RCIC High Steam Flow	+2250 lb/hr
	RCIC Steam Line Area High Temp	+2°F
	Shutdown Cooling Supply ISO	+7 psi

RCIC Steam Line Pressure -5 psi

HPCI Steam Line Pressure -5 psi

3.0 LIMITING CONDITIONS FOR OPERATION

4. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head are ≥70°F.

C. Coolant Chemistry

1. (a) The steady state radioiodine concentration in the reactor coolant shall not exceed 0.25 microcuries of I-131 dose equivalent per gram of water.

(b) The steady state radioiodine concentration in the reactor coolant shall not exceed 0.02 microcuries of I-131 dose equivalent per gram of water when the reactor coolant temperature is > 212°F, the reactor is not critical, and primary containment integrity has not been established.

2.0

4.0 SURVEILLANCE REQUIREMENTS

4. When the reactor vessel head studs are under tension and the reactor is in the Cold Shutdown Condition, the reactor vessel shell flange temperature shall be permanently recorded.

C. Coolant Chemistry

1. (a) A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactive iodines of I-131 through I-135 during power operation.

(b) A sample of reactor coolant shall be taken and analyzed for radioactive iodines of I-131 through I-135 within 24 hours prior to raising the reactor coolant temperature ≥ 212°F, with the reactor not critical, and with primary containment integrity not established.

consequences are bounded by the steam line rupture.

Replace

Bases 3.6/4.6 (Continued):

C. Coolant Chemistry

In the event of a main steam line break outside primary containment, calculations show the resultant radiological dose at the exclusion area boundary to be less than 10% of the dose guidelines of 10 CFR 100. This dose was calculated on the basis of the radioiodine concentration limit of 2 μCi of I-131 dose equivalent per gram of water. In the event of a postulated high energy line break in the RWCU system outside the drywell, calculations show the resultant radiological ~~dose at the exclusion area boundary to be less than 10% of the dose guidelines of 10 CFR 100. This dose was calculated on the basis of the radioiodine concentration limit of 0.25 μCi of I-131 dose equivalent per gram of water.~~ In the event of a large primary system break in primary containment during a reactor vessel hydrostatic or leakage test with the reactor coolant temperature >212°F, the reactor not critical, and primary containment integrity not established, calculations show the resultant radiological dose at the exclusion area boundary to be conservatively bounded by the dose calculated for a main steam line break outside primary containment. This dose was calculated on the basis of the radioiodine concentration limit of 0.02 μCi of I-131 dose equivalent per gram of water

The reactor coolant sample will be used to assure that the limit of Specification 3.6.C.1(a) is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

Whenever an isotopic analysis is performed, a reasonable effort will be made to determine a significant percentage of those contributors representing the total radioactivity in the reactor coolant sample. Usually at least 80 percent of the total gamma radioactivity can be identified by the isotopic analysis.

It has been observed that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations such as reactor shutdown, reactor power changes, and reactor startup if failed fuel is present. As specified, additional reactor coolant samples shall be taken and analyzed for reactor operations in which steady state radioiodine concentrations in the reactor coolant indicate various levels of iodine releases from the fuel. Since the radioiodine concentration in the reactor coolant is not continuously measured, reactor coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam line.

Materials in the primary system are primarily 304 stainless steel and zircaloy. The reactor water chemistry limits are established to prevent damage to these materials. The limit placed on chloride concentration is to prevent stress corrosion cracking of the stainless steel.

3.0 LIMITING CONDITIONS FOR OPERATION

d. During reactor isolation conditions the reactor pressure vessel shall be depressurized to <200 psig at ~~normal cooldown rates~~ if the suppression pool temperature exceed 120°F.

e. The suppression chamber water ~~volume~~ level shall be ~~≥68,000 and ≤72,910 cubic feet.~~ ≥ −4.0 and ≤ +3.0 inches.

f. Two channels of torus water level instrumentation shall be operable. From and after the ~~date that~~ one channel is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 30 days unless such channel is sooner made operable. If both channels are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding six hours unless at least one channel is sooner made operable.

4.0 SURVEILLANCE REQUIREMENTS

d. Whenever there is indication of relief valve operation with a suppression pool temperature of ≥160°F and the primary coolant system pressure >200 psig, an extended visual examination of the suppression chamber shall be conducted before resuming power operation.

e. The suppression chamber water ~~volume~~ level shall be checked once per day.

f. The suppression chamber water ~~volume~~ level indicators shall be calibrated semiannually.

Bases 3.7/4.7 (Continued):

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the maximum allowable primary containment pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. See USAR Section 5.2.3.2.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 42 psig which is below the allowable pressure of 62 psig.

INSERT

The maximum and minimum suppression chamber water levels specified correspond to the values of downcomer submergence assumed in the development of loads for the Monticello Mark I Long Term Program. The minimum specified level corresponds to a downcomer submergence of 3 feet, or an indicated water level of –4 inches. The maximum specified water level corresponds to a downcomer submergence of 3 feet 7 inches, or an indicated water level of + 3 inches. The minimum suppression chamber water level also assures an adequate volume of water is available so that, following a loss of coolant accident, the maximum suppression chamber water temperature limits will not be exceeded.

Replace level

Bases 3.7 (Continued):

If a loss of coolant accident were to occur when the reactor water temperature is below 330°F, the containment pressure will not exceed the 62 psig design pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water ~~volume~~ temperature requirements applicable for reactor water temperatures above 212°F provides additional margin above that available at 330°F.

The large amount of water that must be added or removed to cause a significant change in the suppression chamber water inventory is not likely to go un-noticed. With a daily check of water ~~volume~~, there is an extremely low probability that a loss of coolant accident will occur simultaneously with water ~~volume~~ being outside of the specified range. Two indicators provide redundant readings for comparison (with no automatic action initiation). The provisions allowing one or both indicators out of service are consistent with the need for a redundant indicator and the frequency for checking the ~~volume~~, respectively.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping.

Bases 4.7:

A. Primary Containment

level

replace

The water in the suppression chamber is used only for cooling in the event of an accident. Daily checks are specified of pool temperature and ~~volume~~ to ensure that these parameters are within their allowable ranges.

The interiors of the drywell and suppression chamber are painted to prevent corrosion. The inspection of the paint during each refueling outage, approximately once per year, assures the paint is intact and is not deteriorating. Experience with this type of paint indicates that the inspection interval specified is adequate.

Because of the large volume and thermal capacity of the suppression pool, the ~~volume~~ and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be points of highest stress. Visual inspection of the suppression chamber including water line regions each refueling outage is adequate to detect any changes in the suppression chamber structures.

The design basis loss of coolant accident was analyzed at the primary containment maximum allowable accident leak rate of 1.2% and has been evaluated by the NRC Staff[1]. Computed offsite doses are well below the guidelines of 10 CFR Part 100.

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(1) Safety Evaluation by the Division of Reactor Licensing, US Atomic Energy Commission, in the Matter of Northern States Power Company Monticello Nuclear Generating Plant, Unit 1, Docket No. 50-263, March 18, 1970, Section 4.1.

EXHIBIT C

Revised Monticello Technical Specification Pages

License Amendment Request Dated July 20, 2000

Reactor Water Cleanup (RWCU) System Automatic Isolation
and Miscellaneous Instrumentation System Technical Specification Changes

Exhibit C consists of the existing Technical Specification pages with the proposed changes incorporated. Existing and new pages affected by this change are listed below:

Pages

50
50a (new page)
51
51a (new page)
62
63
63a
64
65
66
67
70
123
148
157
175a
177
183

Table 3.2.1 (Continued)				
Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1, 2)	Required Conditions*
b. High Drywell Pressure (5)	≤2 psig	2	2	D
3. Reactor Cleanup System (Group 3)				
a. High Drywell Pressure	≤2 psig	2	2	E
b. Low Low Reactor Water Level**	≥6′ 6″, ≤6′ 10″	2	2	E
c. High RWCU Room Temperature Allowable Value	≤188°F	2	2	E
d. High RWCU System Flow Allowable Value	≤500 gpm with ≤27 second time delay	2	2	E
4. HPCI Steam Lines (Group 4)				
a. HPCI High Steam Flow***	≤300,000 lb/hr with ≤7 second time delay	2(4)	2	F
b. HPCI Steam Line Area High Temp.	≤200°F	16(4)	16	F
c. Low Pressure in HPCI Steam Supply Line	≥85 psig	4(6)	4(6)	F

Table 3.2.1 (Continued)				
Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1, 2)	Required Conditions*
5. RCIC Steam Lines (Group 5)				
a. RCIC High Steam Flow	≤45,000 lb/hr with 5 ± 2 sec time delay	2(4)	2	G
b. RCIC Steam Line Area	≤200°F	16(4)	16	G
c. Low Pressure in RCIC Steam Supply Line	≥55 psig	4(7)	4(7)	G
6. Shutdown Cooling Supply Isolation				
a. Reactor Pressure Interlock	≤75 psig at the reactor steam dome	2(4)	2	C

Table 3.2.1 (Continued)

NOTES:

(1) There shall be two operable or tripped trip systems for each function. A channel (a shared channel is considered one channel) may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided that at least one other operable channel in the same trip system is monitoring that parameter.

(2) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied action shall be initiated as follows:

(a) With one required instrument channel inoperable in one or more trip functions, place the inoperable channel(s) or trip system in the tripped condition within 12 hours, or

(b) With more than one instrument channel inoperable for one or more trip functions, immediately satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

(c) Place the plant under the specified required conditions using normal operating procedures.

(3) Low pressure in main steam line only need to be available in the RUN position.

(4) All instrument channels are shared by both trip systems.

(5) 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.

(6) The four pressure switches are arranged in a one-out-of-two-twice logic, the output of the logic providing a trip signal to a single trip system for isolation.

(7) The four pressure switches are arranged in a one-out-of-two-twice logic, the output of the logic providing a trip signal to each of two separate trip systems. Each trip system is able, by itself, to initiate isolation.

Table 3.2.1 (Continued)

NOTES: (Continued)

* Required conditions when minimum conditions for operation are not satisfied.

A. Group 1 isolation valves closed.

B. Reactor Power on IRM range or below and reactor in startup, refuel, or shutdown mode.

C. Isolation Valves closed for: Shutdown Cooling System, and Reactor Head Cooling Line.

D. Comply with Condition C. above.

E. Isolation Valves closed for: Reactor Cleanup System.

F. HPCI steam line isolated. (See specification 3.5 for additional requirements.)

G. RCIC steam line isolated.

** Function changed from Low Reactor Water Level to Low Low Reactor Water Level following completion of design change.

*** Function changed from ≤150,000 lb/hr, ≤60 second delay, and ≤300,000 lb/hr, instantaneous, isolation to ≤300/000 lb/hr, ≤7 second delay, isolation following completion of design change.

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling, Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
3. Steam Line Low Pressure	Once/3 months	Once/3 months	None
4. Reactor Low Low Water Level	Once/3 months (Note 5)	Every Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/12 hours
CONTAINMENT ISOLATION (GROUP 2)			
1. Reactor Low Water Level (Note 10)	-	-	-
2. Drywell High Pressure (Note 10)	-	-	-
RWCU ISOLATION (GROUP 3)			
1. High RWCU Room Temperature	Once/3 months	Once/Operating Cycle-RTD Input (Note 12) Once/3 months-Trip Unit	Once/12 hours
2. High RWCU System Flow	Once/3 months	Once/Operating Cycle-Transmitter Once/3 months-Trip Unit	Once/12 hours
* 3. Reactor Low Low Water Level (Note 11)	-	-	
4. Drywell High Pressure (Note 10)	-	-	
HPCI (GROUP 4) ISOLATION			
1. Steam Line High Flow	Once/3 months	Once/3 months	None
2. Steam Line High Temperature	Once/3 months	Once/3 months	None
3. Steam Line Low Pressure	Once/3 months	Once/3 months	None
RCIC (GROUP 5) ISOLATION			
1. Steam Line High Flow	Once/3 months	Once/3 months	None
2. Steam Line High Temperature	Once/3 months	Once/3 months	None
3. Steam Line Low Pressure	Once/3 months	Once/3 months	None

* Function changed from Low Reactor Water Level to Low Low Reactor Water Level following completion of design change.

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
REACTOR BUILDING VENTILATION & STANDBY GAS TREATMENT			
1. Reactor Low Low Water Level	Once/3 months (Note 5)	Every Operating Cycle - Transmitter Once/3 months - Trip Unit	Once/12 hours
2. Drywell High Pressure (Note 10)	-	-	-
3. Radiation Monitors (Plenum)	Once/3 months	Once/3 months	Once/day
4. Radiation Monitors (Refueling Floor)	Once/3 months	Once/3 months	Note 4
RECIRCULATION PUMP TRIP AND ALTERNATE ROD INJECTION			
1. Reactor High Pressure	Once/3 months (Note 5)	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/Day
2. Reactor Low Low Water Level	Once/3 months (Note 5)	Once/Operating Cycle- Transmitter Once/3 Months-Trip Unit	Once/12 hours
SHUTDOWN COOLING SUPPLY ISOLATION			
1. Reactor Pressure Interlock	Once/3 months	Once/3 Months	None
SAFEGUARDS BUS VOLTAGE			
1. Degraded Voltage Protection	Once/month	Quarterly	Not applicable
2. Loss of Voltage Protection	Once/month	Once/Operating Cycle	Not applicable
SAFETY/RELIEF VALVE LOW-LOW SET LOGIC			
1. Reactor Scram Sensing	Once/Shutdown (Note 8)	-	-
2. Reactor Pressure - Opening	Once/3 months (Note 5)	Once/Operating Cycle	Once/day
3. Reactor Pressure - Closing	Once/3 months (Note 5)	Once/Operating Cycle	Once/day
4. Discharge Pipe Pressure	Once/3 months (Note 5)	See Table 4.14.1	See Table 4.14.1
5. Inhibit Timer	Once/3 months (Note 5)	Once/Operating Cycle	-
CONTROL ROOM HABITABILITY PROTECTION			
1. Radiation	Monthly (Note 5)	18 months	Daily

Table 4.2.1 Continued
Minimum Test and Calibration Frequency for Core Cooling,
Rod Block and Isolation Instrumentation

NOTES:

(1) (Deleted)

(2) Calibrate prior to normal shutdown and start-up and thereafter check once per 12 hours and test once per week until no longer required. Calibration of this instrument prior to normal shutdown means adjustment of channel trips so that they correspond, within acceptable range and accuracy, to a simulated signal injected into the instrument (not primary sensor). In addition, IRM gain adjustment will be performed, as necessary, in the APRM/IRM overlap region.

(3) Functional tests, calibrations and sensor checks are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

(4) Whenever fuel handling is in process, a sensor check shall be performed once per 12 hours.

(5) A functional test of this instrument means the injection of a simulated signal into the instrument (not primary sensor) to verify the proper instrument channel response alarm and/or initiating action.

(6) (Deleted)

(7) (Deleted)

(8) Once/shutdown if not tested during previous 3 month period.

(9) Testing of the SRM Not-Full-In rod block is not required if the SRM detectors are secured in the full-in position.

(10) Uses contacts from scram system. Tested and calibrated in accordance with Tables 4.1.1 and 4.1.2.

(11) Uses contacts from Group 1 Isolation logic. Tested and calibrated in accordance with Group 1 Low Low Water Level Instrumentation.

(12) Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

Bases 3.2:

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminate a single operator error before it results in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, and other safety related functions. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 7″ on the instrument. This corresponds to a lower water level above the top of active fuel at 100% power due to the pressure drop across the dryer/separator. This has been accounted for in the affected transient analysis. This trip initiates closure of Group 2 primary containment isolation valves. Reference Section 7.7.2.2 FSAR. The trip setting provides assurance that the valves will be closed before perforation of the clad occurs even for the maximum break in that line and therefore the setting is adequate.

The low low reactor water level instrumentation is set to trip when reactor water level is 6′6″ above the top of the active fuel. This trip initiates closure of the Group 1 and Group 3 Primary containment isolation valves, Reference Section 7.7.2.2 FSAR, and also activates the ECC systems and starts the emergency diesel generator.

Bases 3. 2 (Continued):

This trip setting level was chosen to be low enough to prevent spurious operation but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. Reference Section 6.2.7 and 14.6.3 FSAR. The instrumentation also covers the full range or spectrum of breaks and meets the above criteria. Reference Section 6.2.7 FSAR.

The high drywell pressure instrumentation is a back-up to the water level instrumentation and in addition to initiating ECCS it causes isolation of Group 2 and Group 3 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. Group 2 and Group 3 isolation valves include the drywell vent, purge, sump isolation, RWCU, and recirc sample valves.

Two pressure switches are provided on the discharge of each of the two core spray pumps and each of the four RHR pumps. Two trip systems are provided in the control logic such that either trip system can permit automatic depressurization. Each trip system consists of two trip logic channels such that both trip logic channels are required to permit a system trip.

Division I core spray and RHR pump discharge pressure permissives will interlock one trip system and Division II permissives will interlock the other trip system. One pressure switch on each pump will interlock one of the trip channels and the other pressure switch will interlock the other trip channel within their respective trip system.

The pump pressure permissive control logic is designed such that no single failure (short or open circuit) will prevent auto-blowdown or allow auto-blowdown when not required. The trip setting for the low pressure ECCS pump permissive for ADS is set such that it is less than the pump discharge pressure when a pump is operating in a full flow condition and also high enough to avoid any condition that results in a discharge pressure permissive when the pumps are not operating.

Venturis are provided in the main steamlines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steamline break accident. In addition to monitoring steam flow,

Bases 3.2 (Continued):

instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steamline, thus only Group 1 valves are closed. For the worst case accident, main steamline break outside the drywell, this trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steamline valve closure, limit the mass inventory loss such that fuel is not uncovered, fuel clad temperatures remain less than 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines. Reference Sections 14.6.5 FSAR.

Temperature monitoring instrumentation is provided in the main steamline tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of 200°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a back-up to high steam flow instrumentation discussed above, and for small breaks with the resultant small release of radioactivity, gives isolation before the guidelines of 10 CFR 100 are exceeded.

Pressure instrumentation is provided which trips when main steamline pressure drops below 825 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the "refuel" and "Startup" mode this trip function is bypassed. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open. With the trip set at 825 psig inventory loss is limited so that fuel is not uncovered and peak clad temperatures are much less than 1500°F; thus, there are no fission products available for release other than those in the reactor water. Reference License Amendment Request Dated December 1, 1975 from L. O. Mayer (NSP) to R. S. Boyd (USNRC).

The RWCU high flow and temperature instrumentation is provided to detect a break in the RWCU piping. Tripping of this instrumentation results in actuation of the RWCU isolation valves, i.e., Group 3 valves. The trip settings have been established so that the radiological consequences of a high energy line break in this system are bounded by a break in the main steam system. The recirc sample isolation valves, which receive a Group 1 isolation signal, also receive a redundant Group 3 isolation signal.

Bases 3.2 (Continued):

The HPCI and RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI or RCIC piping. The trip settings of 200°F and approximately 300% of HPCI and RCIC design steam flow are such that the core will not be uncovered and the radiological consequences are bounded by the main steam line break accident. The HPCI and RCIC low steam line pressure instrumentation protects the HPCI and RCIC turbines when system operation is no longer useful or possible. Tripping of the high flow, high temperature, or low steam line pressure instrumentation results in actuation of the HPCI or RCIC steam supply valves; i.e., a Group 4 or Group 5 isolation.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR remains above the Safety Limit (T.S.2.1.A). The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, eight IRM's, or four SRM's will result in a rod block. The minimum instrument channel requirements for the IRM and RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. See Section 7.3 FSAR.

The APRM rod block trip is referenced to flow and prevents operation significantly above the licensing basis power level especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The operator will set the APRM rod block trip settings no greater than that stated in Table 3.2.3. However, the actual setpoint can be as much as 3% greater than that stated in Table 3.2.3 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 39.

The RBM provides local protection of the core; i.e., the prevention of critical power in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is referenced to power. This power signal is provided by the APRMs. A statistical analysis of many single control rod withdrawal errors has been performed and at the 95/95 level the results show that with the specified trip settings, rod withdrawal is blocked at MCPRs greater than the Safety Limit, thus allowing adequate margin. This analysis assumes a steady state MCPR prior to the postulated rod withdrawal error. The RBM functions are required when core thermal power is greater than 30% and a Limiting Control Rod Pattern exists. When both RBM channels are operating either channel will assure required withdrawal blocks occur even assuming a single failure of one channel. With one RBM channel inoperable for no more than 24 hours, testing of the RBM prior to withdrawal of control rods assures that improper control rod withdrawal will be blocked. Requiring at least half of the normal LPRM inputs to be operable assures that the RBM response will be adequate to protect against rod withdrawal errors, as shown by a statistical failure analysis.

Bases 3.2 (Continued):

	Trip Function	Deviation
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Specification 3.2.E.3 and Table 3.2.4	Reactor Building Vent Plenum Monitors	+5 mR/hr
	Refueling Floor Radiation Monitors	+5 mR/hr
	* Low Low Reactor Water Level High Drywell Pressure	-3 inches +1 psi
Primary Containment Isolation Functions Table 3.2.1	* Low Low Water Level	-3 inches
	High Flow in Main Steam Line	+2%
	High Temp. in Main Steam Line Tunnel	+10°F
	Low Pressure in Main Steam Line	-10 psi
	High Drywell Pressure	+1 psi
	* Low Reactor Water Level	-6 inches
	HPCI High Steam Flow	+7,500 lb/hr
	HPCI Steam Line Area High Temp.	+2°F
	HPCI Steam Line Pressure	-5 psi
	RCIC High Steam Flow	+2250 lb/hr
	RCIC Steam Line Area High Temp	+2°F
	RCIC Steam Line Pressure	-5 psi
	Shutdown Cooling Supply ISO	+7 psi

3.0 LIMITING CONDITIONS FOR OPERATION

4. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head are ≥70°F.

C. Coolant Chemistry

1. (a) The steady state radioiodine concentration in the reactor coolant shall not exceed 2.0 microcuries of I-131 dose equivalent per gram of water.

   (b) The steady state radioiodine concentration in the reactor coolant shall not exceed 0.02 microcuries of I-131 dose equivalent per gram of water when the reactor coolant temperature is >212°F, the reactor is not critical, and primary containment integrity has not been established.

4.0 SURVEILLANCE REQUIREMENTS

4. When the reactor vessel head studs are under tension and the reactor is in the Cold Shutdown Condition, the reactor vessel shell flange temperature shall be permanently recorded.

C. Coolant Chemistry

1. (a) A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactive iodines of I-131 through I-135 during power operation.

   (b) A sample of reactor coolant shall be taken and analyzed for radioactive iodines of I-131 through I-135 within 24 hours prior to raising the reactor coolant temperature >212°F, with the reactor not critical, and with primary containment integrity not established.

Bases 3.6/4.6 (Continued):

C. Coolant Chemistry

In the event of a main steam line break outside primary containment, calculations show the resultant radiological dose at the exclusion area boundary to be less than 10% of the dose guidelines of 10 CFR 100. This dose was calculated on the basis of the radioiodine concentration limit of 2 μCi of I-131 dose equivalent per gram of water. In the event of a postulated high energy line break in the RWCU system outside the drywell, calculations show the resultant radiological consequences are bounded by the steam line rupture. In the event of a large primary system break in primary containment during a reactor vessel hydrostatic or leakage test with the reactor coolant temperature >212°F, the reactor not critical, and primary containment integrity not established, calculations show the resultant radiological dose at the exclusion area boundary to be conservatively bounded by the dose calculated for a main steam line break outside primary containment. This dose was calculated on the basis of the radioiodine concentration limit of 0.02 μCi of I-131 dose equivalent per gram of water.

The reactor coolant sample will be used to assure that the limit of Specification 3.6.C.1(a) is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

Whenever an isotopic analysis is performed, a reasonable effort will be made to determine a significant percentage of those contributors representing the total radioactivity in the reactor coolant sample. Usually at least 80 percent of the total gamma radioactivity can be identified by the isotopic analysis.

It has been observed that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations such as reactor shutdown, reactor power changes, and reactor startup if failed fuel is present. As specified, additional reactor coolant samples shall be taken and analyzed for reactor operations in which steady state radioiodine concentrations in the reactor coolant indicate various levels of iodine releases from the fuel. Since the radioiodine concentration in the reactor coolant is not continuously measured, reactor coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system and on the main steam line.

Materials in the primary system are primarily 304 stainless steel and zircaloy. The reactor water chemistry limits are established to prevent damage to these materials. The limit placed on chloride concentration is to prevent stress corrosion cracking of the stainless steel.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
d. During reactor isolation conditions the reactor pressure vessel shall be depressurized to <200 psig at normal cooldown rates if the suppression pool temperature exceed 120°F.	d. Whenever there is indication of relief valve operation with a suppression pool temperature of ≥160°F and the primary coolant system pressure >200 psig, an extended visual examination of the suppression chamber shall be conducted before resuming power operation.
e. The suppression chamber water level shall be ≥ -4.0 and ≤ +3.0 inches.	e. The suppression chamber water level shall be checked once per day.
f. Two channels of torus water level instrumentation shall be operable. From and after the date that one channel is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 30 days unless such channel is sooner made operable. If both channels are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding six hours unless at least one channel is sooner made operable.	f. The suppression chamber water level indicators shall be calibrated semiannually.

Bases 3.7/4.7 (Continued):

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the maximum allowable primary containment pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. See USAR Section 5.2.3.2.

The maximum and minimum suppression chamber water levels specified correspond to the values of downcomer submergence assumed in the development of loads for the Monticello Mark I Long Term Program. The minimum specified level corresponds to a downcomer submergence of 3 feet, or an indicated water level of -4 inches. The maximum specified water level corresponds to a downcomer submergence of 3 feet 7 inches, or an indicated water level of +3 inches. The minimum suppression chamber water level also assures an adequate volume of water is available so that, following a loss of coolant accident, the maximum suppression chamber water temperature limits will not be exceeded.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 42 psig which is below the allowable pressure of 62 psig.

Bases 3.7 (Continued):

If a loss of coolant accident were to occur when the reactor water temperature is below 330°F, the containment pressure will not exceed the 62 psig design pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor water temperatures above 212°F provides additional margin above that available at 330°F.

The large amount of water that must be added or removed to cause a significant change in the suppression chamber water inventory is not likely to go un-noticed. With a daily check of water level, there is an extremely low probability that a loss of coolant accident will occur simultaneously with water level being outside of the specified range. Two indicators provide redundant readings for comparison (with no automatic action initiation). The provisions allowing one or both indicators out of service are consistent with the need for a redundant indicator and the frequency for checking the level, respectively.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping.

Bases 4.7:

A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident. Daily checks are specified of pool temperature and level to ensure that these parameters are within their allowable ranges.

The interiors of the drywell and suppression chamber are painted to prevent corrosion. The inspection of the paint during each refueling outage, approximately once per year, assures the paint is intact and is not deteriorating. Experience with this type of paint indicates that the inspection interval specified is adequate.

Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be points of highest stress. Visual inspection of the suppression chamber including water line regions each refueling outage is adequate to detect any changes in the suppression chamber structures.

The design basis loss of coolant accident was analyzed at the primary containment maximum allowable accident leak rate of 1.2% and has been evaluated by the NRC Staff[1]. Computed offsite doses are well below the guidelines of 10 CFR Part 100.

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(1) Safety Evaluation by the Division of Reactor Licensing, US Atomic Energy Commission, in the Matter of Northern States Power Company Monticello Nuclear Generating Plant, Unit 1, Docket No. 50-263, March 18, 1970, Section 4.1.