NMC
Committed to Nuclear Excellence

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

Operated by Nuclear Management Company LLC

June 18, 2001

US 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 June 18, 2001
Changes to the Technical Specifications
Revised Reference Point for Reactor Vessel Level Setpoints,
Simplification of Safety Limits, and Improvements to the Bases

Attached is a request for changes to the Technical Specifications of the Operating License for the Monticello Nuclear Generating Plant. The proposed changes would:

- Revise the reference point for reactor vessel level instrumentation specifications to use instrument "zero" instead of "top of active fuel (TAF)." Use of TAF is no longer an unambiguous reference point because of variations in the length of active fuel in modern fuel element designs.

- Simplify the Safety Limits and Limiting Safety System Settings to eliminate specifications that are unnecessary, outdated, or redundant to portions of Section 3 of the Technical Specifications.

- Change the reactor coolant system pressure Safety Limit from 1335 psig to 1332 psig to correct a minor error in the calculation of this limit.

- Change the Bases for the upper limit on reactor coolant system safety/relief valve self actuation setpoint to correct the description of this specification.

These changes have been combined and included in one submittal because they affect many of the same Technical Specification pages.

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This request is submitted in accordance with the provisions of 10 CFR 50, Section 50.90. This submittal contains no new NRC commitments, nor does it modify any prior commitments.

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

Please contact Mr. Doug Neve, Sr. Licensing Engineer, at 763-295-1353 if you require additional information related to this request.

by ______________________
Jeff S. Forbes
Plant Manager
Monticello Nuclear Generating Plant

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

Subscribed to and sworn before me this 18 day of June, 2001.

Notary

Attachments: Exhibit A – Evaluation of Proposed Change to the Monticello Technical Specifications

Exhibit B: – Current Monticello Technical Specification Pages Annotated With Proposed Changes

Exhibit C – Revised Monticello Technical Specification Pages

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

License Amendment Request dated June 18, 2001
Changes to the Technical Specifications
Revised Reference Point for Reactor Vessel Level Setpoints,
Simplification of Safety Limits, and Improvements to the Bases

Pursuant to 10 CFR Part 50, Section 50.90, Nuclear Management Company, LLC (NMC) hereby proposes the following changes to Appendix A of Facility Operating License DPR-22, "Technical Specifications," for the Monticello Nuclear Generating Plant:

I. Proposed Changes

I.1 Revise the Reference Point for Reactor Vessel Level Specifications From "Top of Active Fuel" to Instrument "Zero"

In several locations in the Monticello Technical Specifications reactor vessel level is referenced to the top of the active fuel (TAF). Use of TAF as a reference point can be ambiguous because of the variety of active fuel lengths used in modern core designs. As a practical matter, however, TAF has always been established as 351.5" above the inner clad bottom of the reactor vessel based on the original core fuel design.

It is proposed that all reactor vessel level Technical Specifications currently referenced to TAF be changed to be referenced to reactor vessel instrument "zero." Instrument zero is defined unambiguously as 477.5 inches above the inner clad bottom of the reactor vessel. TAF referenced to instrument "zero" is –126".

Also, where both a lower and upper limit are currently specified for a reactor vessel level, it is proposed that the upper limit be dropped. For example, the low-low reactor water level setpoint is currently specified as:

≥ 6'6" & ≤ 6'10" above top of active fuel

(or some variation of this wording)

This setpoint will be changed, wherever appearing, to:

≥ -48"

where -48" = -126" + 6'6"

Other specifications referenced to TAF are corrected to instrument "zero" in a similar manner.

Changes are also proposed to the Bases to define the reference point for reactor vessel level specifications and the region of the vessel where this level is sensed (annulus). The term "annulus", where appearing in reactor water level specifications, therefore becomes unnecessary and may be deleted.

These changes are also consistent with the NRC Standard Technical Specifications for General Electric Plants, NUREG-1433. Refer to Exhibit B pages 29, 38, 49, 50, 52, 53, 54, 59, 60, 60d, and 64 and the corresponding pages in Exhibit C.

Corrections of three minor typographical errors are also included in the proposed changes:

1. The numbering of items "B.a" and "C.a" are corrected to "B.1" and C.1" on Table 3.2.8 on page 60d.

2. The word generator is corrected to "generators" on page 64.

I.2 Simplify the Safety Limits and Limiting Safety System Settings

Section 2 of the current Monticello Technical Specifications, "Safety Limits and Limiting Safety System Settings," currently contains many limits and setpoints which are redundant to limits and setpoints specified in Section 3 of the Technical Specifications. In addition, several of the specifications in Section 2 are unnecessary or outdated. We propose to simplify Section 2 with the following proposed changes:

- Setpoints for the intermediate range monitor scram, the flow referenced average power range monitor scram, and safety/relief valves will be relocated from Section 2 to Section 3.

- Setting which are specified in both Section 2 and Section 3 will be eliminated from Section 2.

- Specification 2.1.C is being deleted. It has been superseded by commitments made to the NRC following the Salem ATWS event.

- The reactor vessel water level Safety Limit is reduced from 12 inches above TAF to above TAF to conform to current standard Technical Specifications.

- Pressure units are changed from psia to psig by subtracting 15 psi to conform to current standard Technical Specifications.

- The reactor coolant system pressure Safety Limit is reduced from 1335 to 1332 psig to correct a minor error. Refer to Section I.3 below.

- Section 6.4 is being replaced by new Section 2.2 and its associated Bases. Administrative requirements contained in Section 6.4, which are covered in the Operational Quality Assurance Program, are deleted.

Corresponding changes to the Bases for Section 2 and Section 3 are proposed to support the simplification of Section 2 and relocation of setpoints for intermediate range monitor scram, flow referenced average power range monitor scram, and safety/relief valves to Section 3. The Bases changes consist largely of relocation of text.

Refer to Exhibit B pages i, 6, 8, 10 – 16, 18 – 25, 108, 127, 150, 151, 217, 243, and 249b and the corresponding pages in Exhibit C.

I.3 <u>Reduce the Reactor Coolant System Pressure Safety Limit from 1335 psig to 1332 psig.</u>

Change the value of the reactor coolant system pressure Safety Limit from 1335 psig to 1332 psig. This minor correction in the value of this Safety Limit was identified during the design basis reconstitution program at Monticello. Also revise the Section 2 Bases to correctly explain the derivation of this Safety Limit.

Refer to Exhibit B pages 6 and 13 and the corresponding pages in Exhibit C.

I.4 <u>Correct the Basis for the Upper Limit on Reactor Coolant System Safety/Relief Valve Self Actuation Setpoint</u>

Change the basis for the upper limit on safety/relief valve setpoint to read:

> The upper limit on safety/relief valve setpoint is established by the *operating limit* of the HPCI and RCIC systems of 1120 psig.

The current Bases is incorrect by stating that the upper limit is based on the *design pressure* of the HPCI and RCIC systems. The need for this change was identified during the design basis reconstitution program at Monticello.

Refer to Exhibit B page 150 and the corresponding page in Exhibit C.

II. Reasons for Proposed Changes

II.1 Reactor Vessel Level Instrumentation Reference Point

Top of active fuel was originally an unambiguous term when the plant was designed and all of the fuel bundles in the core had the same active fuel length of 144" of enriched uranium. Since then, however, fuel bundle designs have evolved and there is now a spectrum of active fuel lengths present in the core, ranging from approximately 90" to 145.24".

Table II.1-1 shows the active fuel lengths for fuel pins loaded in the core in the current fuel cycle (Cycle 20).

Table II.1-1

Number of Pins	Height of Active Fuel
3,640	145.24"
21,340	141.24"
752	139.24"
5,852	133.24"
56	93.00"
3,296	90.00"

The values in Table II.1-1 demonstrate that the phrase "top of active fuel" does not have a precise meaning with the fuel designs that are now in service.

Table II.1-2 shows instances where the phrase "top of active fuel" is currently used or implied in the Technical Specifications.

Table II.1-2

Page	Section	Wording
7	2.3.C 2.3.D	≥10' 6" Above top of active fuel (ATOAF) ≥6' 6", ≤6' 10" ATOAF
8	2.1.D	not be less than .... 12" ATOAF
12	Bases 2.1	12 inches above the top of the fuel
18	Bases 2.3	no lower than 10' 6" ATOAF
19	Bases 2.3	≥6' 6", ≤6' 10" ATOAF
49	Table 3.2.1 # 1.a	Low Low Reactor Water Level ≥6' 6", ≤6' 10"
50	Table 3.2.1 # 3.b	Low Low Reactor Water Level ≥6' 6", ≤6' 10"
52	Table 3.2.2 # A.1.a	Low Low Reactor Water Level ≥6' 6", ≤6' 10"
53	Table 3.2.2 # B.2	Low Low Reactor Water Level ≥6' 6", ≤6' 10"

Page	Section	Wording
	# C.1	Low Low Reactor Water Level ≥6' 6", ≤6' 10"
54	Table 3.2.2 # D.2	Low Low Reactor Water Level ≥6' 6", ≤6' 10"
59	Table 3.2.4 #1	Low Low Reactor Water Level ≥6' 6", ≤6' 10"
60	Table 3.2.5, #2	≥6' 6" ATOAF
60d	Table 3.2.8, #A.1 #B.a	≥6' 6" & ≤6' 10" ATOAF ≤14' 6" ATOAF
64	Bases 3.2, para. 4 para. 5	ATOAF 6' 6" ATOAF

Use of reactor vessel level specifications referenced to the defined instrument "zero" level is a superior method of specifying vessel level since the ambiguity with respect to active fuel length is removed.

An upper limit specification on reactor vessel level is unnecessary since the actual instrument calibration setpoint will conform to standard setpoint methodology and will be established to both:

- Prevent a violation of the Technical Specification limit due to instrument drift, and
- Minimize the possibility of an unnecessary automatic actuation of engineered safeguards equipment.

Specification of only the lower limit is also consistent with the NRC Standard Technical Specifications, NUREG-1433.

II.2 Simplify the Safety Limits and Limiting Safety System Settings

As noted in Table II.2, correction of the reference point for reactor vessel level specifications would affect six pages of Section 2 of the Monticello Technical Specifications, "Safety Limits and Limiting Safety System Settings." In considering the Technical Specification changes necessary to address the reactor vessel water level issue, it was observed that the format and content of Section 2 contained many redundancies and several cross references with instrument settings specified in Section 3 of the Technical Specifications.

To eliminate these objectionable redundancies and cross references, as well as other outdated and unnecessary requirements, it appeared appropriate to adopt the format of the Safety Limits section of the NRC's Standard Technical Specifications for General Electric Plants, NUREG-1433, as part of this request.

Significant differences between the new proposed Safety Limits section and current Safety Limits section include:

1. Conversion of psia units to psig units. The existing pressures specified in psia units have been converted to pressures expressed in psig by subtracting 15 psi.

2. Deletion of Section 2.1.C, Power Transients. This specification requires confirmation that each scram is initiated by its primary source signal as indicated by the plant process computer. The requirements of Section 2.1.C have been superseded by commitments made in response to NRC Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," dated July 8, 1983. Refer to Section 14.8.1.2 of the Monticello Updated Safety Analysis Report (USAR).

3. The reactor water level Safety Limit has been reduced from "...12 inches above the top of the active fuel when it is seated in the core" to "...shall be greater than the top of active irradiated fuel." Fuel integrity is assured when all of the fuel is covered with water and the additional 12 inches is not necessary to meet the objectives of this Safety Limit. In addition, the requirement for continuous level monitoring when the recirculation pumps are not operating has been deleted. The requirement for continuous level monitoring is redundant to the requirements for operability of reactor vessel level instrumentation in Section 3 of the Technical Specifications.

4. The reactor coolant system pressure Safety Limit is reduced slightly. Refer to Section II.3, below, for a discussion of the reason for this change.

5. Specification 6.4, "Action to Be Taken If a Safety Limit is Exceeded," has been deleted and replaced by proposed Specification 2.2 and the accompanying proposed Bases Section. Actions taken by corporate officers, the Operations Committee, and Safety Audit Committee when a Safety Limit is exceeded have been deleted. These actions are adequately covered in the following sections of the Operational Quality Assurance Plan found in Appendix C of the Monticello USAR:

Section 18.2.1	Operating Occurrences and Events
Section 21.6	Safety Audit Committee Responsibilities
Section 22.4	Operations Committee Responsibilities

Table II.3-1 shows the new location of the existing Section 2 specifications or, if deleted, the reason or location where a redundant specification is located.

Table II.3-1

Page	Section	New Location or Basis for Deletion
6	2.1.A	Revised and relocated to Specification 2.1.A.2
6	2.3.A.1	Relocated to Table 3.1.1, Item 4
7	2.1.B	Revised and relocated to Specification 2.1.A.1
7	2.1.C	Deleted. Requirements satisfied by commitments to NRC Generic Letter 83-28 described in USAR Section 14.8.1.2.
7	2.3.A.2	Relocated to Table 3.3.1, Item 3
7	2.3.C	Deleted – Currently duplicated in Table 3.1.1
7	2.3.D	Deleted – Currently duplicated in Table 3.2.2
8	2.1.D	Revised and relocated to Specification 2.1.A.3
8	2.3.E	Deleted – Currently duplicated in Table 3.1.1
8	2.3.F	Deleted – Currently duplicated in Table 3.1.1
8	2.3.G	Deleted – Currently duplicated in Table 3.1.1
8	2.3.H	Deleted – Currently duplicated in Table 3.2.1
21	2.2	Revised and relocated to Specification 2.1.B
21	2.4.A	Deleted – Currently duplicated in Table 3.1.1
21	2.4.B	Relocated to Specification 3.6.E
243	6.4	Existing Technical Specification Section 6.4, "Action to Be Taken If a Safety Limit is Exceeded," is replaced by proposed Section 2.2, "Safely Limit Violations" and associated Bases. Corporate notification and onsite and offsite review committee actions are adequately described in the Operational Quality Assurance Program and have been deleted from the Technical Specifications.
249b	6.7.A.7.a	Reference to Bases Section 2.3.A has been changed to reference Bases Section 3.1

The proposed changes to Section 2 represent an overall improvement in this section and brings the Monticello Technical Specifications into closer conformance with the NRC Standard Technical Specifications for BWRs, NUREG-1433.

II.3 <u>Reduce the Reactor Coolant System Pressure Safety Limit from 1335 psig to 1332 psig</u>

The 1335 psig reactor coolant system pressure Safety Limit *currently* specified in Section 2.2 of the Technical Specifications is based on the ASME Code, Section III, 1965 Edition, and was derived as follows:

Reactor Pressure Vessel (RPV) design pressure at lowest elevation of reactor coolant system	= 1250 psig
Overpressurization at lowest elevation of reactor coolant system	= 1250 psig x 1.1 = 1375 psig
Assumed static head	= 40 psi
Overpressurization at steam dome	= 1375 psig – 40 psi = 1335 psig

This calculation incorrectly applies the 40 psi static head correction value and ignores the design pressure of piping attached to the reactor vessel. The 40 psi static head correction was originally used only to independently establish the steam dome design pressure of:

1250 psig – 40 psi = 1210 psig

Piping attached to the reactor vessel steam space, including the main steam, RCIC, and HPCI systems, is designed for a pressure of 1110 psig. ANSI B31.1, Power Piping code, allows for overpressurization of 120% of design for up to 1% of the operating time of the piping. Therefore:

RPV overpressurization Limit (steam dome)	= 1.1x1210 = 1331 psig
Design pressure for piping attached to vessel steam space	= 1331 psig/1.2 = 1109.2

The calculated value of 1109.2 psig was rounded up to yield a piping design pressure of 1110 psig. Using a steam space attached piping design pressure of 1110 psig yields:

Steam dome pressure limit	= 1110 psig x 1.2 = 1332 psig

The design pressure for piping attached to the vessel water space was derived using the steam dome pressure limit of 1332 psig and a conservative value for static head of 31 psi as follows:

Design pressure for piping attached to vessel water space	= (1332 psig + 31 psi)/1.2 = 1136 psig

Based on the above discussion, it is proposed that the reactor coolant system pressure boundary Technical Specification Safety Limit be reduced slightly from its current value of 1335 psig to 1332 psig. This is a conservative change.

II.4 Correct the Basis for the Upper Limit on Reactor Coolant System Safety/Relief Valve Self Actuation Setpoint

The current Bases section of for Technical Specification 2.4 states:

> The upper limit on safety/relief valve setpoint is established by the design pressure of the HPCI and RCIC systems [1120 psig].

This statement states that the rated design pressure for the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems is 1120 psig. As discussed in Section II.3, above, the design pressure for the HPCI and RCIC systems, as well as other piping attached to the reactor vessel steam space, is 1110 psig.

The upper SRV setpoint limit of 1120 psig is not based on the HPCI or RCIC *design pressure*, but is based on the HPCI and RCIC *operating limit* of 1120 psig. The 1120 psig design operating limit for the HPCI and RCIC systems is documented in the General Electric HPCI and RCIC system design specification data sheets and in the following locations in the Monticello USAR:

Tables	6.1-1	Sections	6.2.4.2.2
	6.2-3		7.1.1.2.2.2
	10.2-3		14.7.2.3.1.2

The maximum transient pressure for HPCI and RCIC piping is 1332 psig which bounds the SRV setpoint of 1120 psig *and* the operating limit for these systems.

Based on the above discussion, it is requested that the discussion in the Bases of the SRV setpoint limit be corrected to refer to system "operating limit" instead of "design pressure."

While it was not necessary to include this proposed Bases change in a License Amendment Request, it was included here because it is associated with other proposed changes to Sections 2 and 3 of the Technical Specifications.

III. Safety Evaluation

III.1 Reactor Vessel Level Instrumentation Reference Point

The trip settings for reactor vessel water level have been historically defined in terms of distance above the top of the active fuel. The top of active fuel, however, no longer has a precise physical meaning. The length of the fuel pellet column in individual fuel pins has changed over time with improvements in fuel design.

The General Electric SAFER-GESTR loss of coolant accident (LOCA) analysis conservatively assumes that the emergency core cooling systems (ECCS) initiate at a absolute level well below the current Technical Specification low low reactor water level setpoint of -48" relative to instrument level zero. SAFER-GESTR also models the actual active fuel lengths present in all of the fuel designs utilized in the Monticello core. The Group 1 and Group 2 containment isolation setpoints are the same as the ECCS initiation set point to ensure that 10 CFR 100 dose guidelines are not exceeded during a LOCA and thus are anticipatory trips. The ATWS level set point uses the low-low reactor level as an initiation point which indicates that an ATWS event may be occurring and thus is insensitive to the small variations in active fuel length discussed here. Therefore the safety analysis incorporates the effect of all actual fuel column lengths and they are not sensitive to the reference point used in the Technical Specifications for reactor vessel water level.

Changing all of the Technical Specification reactor water level setpoints to be referenced to instrument "zero" will remove all possible ambiguity that exists due to the current practice of referencing levels to the top of the active fuel. The actual absolute reactor vessel water levels specified in the Technical Specifications will not change. Therefore this change cannot adversely affect plant safety.

III.2 Simplify the Safety Limits and Limiting Safety System Settings

Safety System Settings which are redundant to requirements in Section 3 of the Technical Specifications are being deleted. Three Limiting Safety System Settings which do not currently appear in Section 3, but are referenced there, are being moved to Section 3. The actual instrumentation settings remain unchanged.

Section 2.1.C, Power Transients, is being deleted. This is acceptable since the requirements in this section have been superseded by commitments made in response to NRC Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," dated July 8, 1983.

It is proposed that the reactor water level Safety Limit be reduced from 12 inches above the top of the active fuel to greater than the top of the active fuel consistent with NUREG-1433. This is acceptable because fuel integrity is assured when all of the fuel is covered with water. While the change in reactor water level represents a less restrictive limit, the proposed numerical value still ensures an adequate margin for core cooling and provides an adequate margin for effective action. The requirement for continuous level monitoring in the Safety Limits may be deleted because it is redundant to level instrumentation requirements contained in Section 3 of the Technical Specifications.

The reactor coolant system pressure Safety Limit reduction is evaluated in Section III.3, below.

Specification 6.4, "Action to Be Taken If a Safety Limit is Exceeded," has been deleted and replaced by proposed Specification 2.2 and the accompanying proposed Bases Section. Actions required by corporate officers, the Operations Committee, and Safety Audit Committee when a Safety Limit is exceeded currently contained in Section 6.4 may be deleted. These actions are covered in the Operational Quality Assurance Plan, Appendix C, of the Monticello USAR:

III.3 Reduce the Reactor Coolant System Pressure Safety Limit from 1335 psig to 1332 psig

This proposed change will correct the reactor coolant system pressure Safety Limit by reducing its value to 1332 psig. The proposed change in the Safety Limit is small and represents a conservative action taken to correct a minor error in the derivation of the reactor coolant pressure limit.

III.4 Correct the Basis for the Upper Limit on Reactor Coolant System Safety/Relief Valve Setpoints

This proposed change will correct the explanation in the Bases of the numerical upper limit on safety/relief valve self-actuation setpoint. No change in any actual setpoint is involved in this proposed change.

IV. Significant Hazards Consideration Evaluation

NMC has proposed changes to the Monticello Technical Specifications in the following four areas:

- Revise the Reference Point for Reactor Vessel Level Specification From "Top of Active Fuel" to Instrument "Zero."

  The proposed changes eliminate possible ambiguities in reactor vessel level specifications which are currently referenced to the top of the active fuel in the core. No actual physical changes affecting level instrumentation setpoints are proposed.

- Simplify the Safety Limits and Limiting Safety System Settings.

  Changes are proposed to Section 2 of the Technical Specifications which will simplify this section and eliminate redundancies with Section 3 of the Technical Specifications. Outdated specifications are revised and administrative requirements which are duplicated in the Operational Quality Assurance Plan and USAR are deleted. A change in the reactor vessel water level Safety Limit is included which eliminates an unnecessary 12-inch margin.

- Reduce the Reactor Coolant System Pressure Safety Limit from 1335 psig to 1332 psig.

  A small reduction in the reactor coolant system pressure Safety Limit is proposed to correct an error in the original derivation of this numerical limit.

- Correct the Basis for the Upper Limit on Reactor Coolant System Safety/Relief Valve Self Actuation Setpoint.

  This change corrects the description in the Bases of the derivation of the upper limit on safety/relief valve self actuation setpoint. No actual Technical Specification change is involved.

The proposed changes have been evaluated to determine whether they constitute a significant hazards consideration as required by 10 CFR Part 50, Section 50.91 using the standards provided in Section 50.92. This analysis is provided below:

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

The requested changes are administrative in nature in that they change instrumentation reference points, reformat sections to conform to current NRC guidance, or correct minor errors.

One change involves a small conservative reduction in the reactor coolant system pressure limit. This change corrects a long standing minor discrepancy in this numerical limit.

Another change eliminates the extra 12 inches above the top of active fuel currently specified in the reactor water level Safety Limit. It is sufficient to require that all active fuel is covered by water to satisfy the objective of the Safety Limit and assure the integrity of the fuel cladding.

None of these changes affect the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in any of the accident analyses. Since the accident analyses remain bounding, their radiological consequences are not adversely affected.

Therefore, the probability or consequences of an accident previously evaluated are not affected.

2. <u>The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.</u>

The proposed changes do not involve a change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do they affect any assumptions or conditions in any of the accident analyses. Accordingly, no new failure modes have been created for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes.

Therefore the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. <u>The proposed amendment will not involve a significant reduction in the margin of safety.</u>

One change involves a small conservative reduction in the reactor coolant system pressure limit. This change corrects a long standing minor discrepancy in the derivation of the numerical value of this limit of less than 0.3%. The correction is conservative.

Another change eliminates the extra 12 inches above the top of active fuel currently specified in the reactor water level Safety Limit. The additional 12 inches of water does not significantly contribute to fuel cooling under plant conditions for which the Safety Limit would be applicable. While the change in reactor water level represents a less restrictive limit, the proposed numerical value still ensures an adequate margin for core cooling and provides an adequate margin for effective action. The benefits gained from achievement of uniformity with the reactor water level Safety Limit established by the NRC for plants similar to Monticello outweigh any negative aspects of this change.

The remainder of the requested changes are administrative in nature or correct minor errors.

Therefore, a significant reduction in the margin of safety is not involved in the proposed changes.

Based on the above evaluation, and pursuant to 10 CFR 50.91, the operation of Monticello in accordance with the proposed license amendment request does not involve any significant hazards considerations as defined by NRC regulations in 10 CFR 50.92.

V. Environmental Assessment

NMC has evaluated the proposed changes and determined that:

1. The changes do not involve a significant hazards consideration, or

2. The changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or

3. The changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

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

Exhibit B

License Amendment Request dated June 18, 2001
Changes to the Technical Specifications
Revised Reference Point for Reactor Vessel Level Setpoints,
Simplification of Safety Limits, and Improvements to the Bases

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

Pages
i
6 - 8
10 - 16
18 – 25
28 - 29
36 - 40
49 - 50
52 - 54
59 - 60
60d
64
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127
150 - 151
217
243
249b

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2.0 SAFETY LIMITS

~~2.1 FUEL CLADDING INTEGRITY~~

Applicability:

2.1 SAFETY LIMITS

A. Reactor Core Safety Limits

1. **With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:**

   Thermal Power shall be ≤ 25% Rated Thermal Power

2. **With the reactor steam dome pressure ≥785 psig and core flow ≥ 10% rated core flow:**

   MCPR shall be ≥ 1.11 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

3. **Reactor vessel water level shall be greater than the top of active irradiated fuel.**

B. Reactor Coolant System Pressure Safety Limit

Reactor steam dome pressure shall be ≤ 1332 psig.

LIMITING SAFETY SYSTEM SETTINGS

~~2.3 FUEL CLADDING INTEGRITY~~

Applic

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Limiting Safety System Settings are incorporated into Section 3 of the Technical Specifications.

C

To c at which
automat to prevent the safety
limits from being exceeded.

Specification:

The Limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM - The APRM flux scram trip setting shall be:

   a. For two recirculation loop operation (TLO):

   $$S \leq 0.66W + 65.6\%$$

   where

   S = Setting in percent of rated thermal power, rated power being 1775 MWt

   W = Percent of recirculation drive flow required to produce a core flow of 57.6×10^6 lb/hr

   b. For single recirculation loop operation (SLO):

   $$S \leq 0.66(W - 5.4) + 65.6\%$$

   c. No greater than 120%.

~~...re <800 Psia~~

2.2 SAFETY LIMIT VIOLATIONS

With any Safety Limit violation, the following actions shall be completed within 2 hours:

2.2.A **Restore compliance with all Safety Limits; and**

2.2.B **Insert all insertable control rods**

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...be
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~~2. IRM - Flux Scram setting shall be ≤20% of rated neutron flux~~

~~B. (DELETED)~~

~~C. Reactor Low Water Level Scram setting shall be ≥ 10′6″ above the top of the active fuel.~~

~~D. Reactor Low Low Water Level ECCS initiation shall be ≥6′ 6″, ≤6′ 10″ above the top of the active fuel.~~

2.0 SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

D. Reactor Water Level (Shutdown Condition)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core. This level shall be continuously monitored whenever the recirculation pumps are not operating.

E. Turbine Control Valve Fast Closure Scram shall initiate upon loss of presure at the acceleration relay with turbine first stage pressure ≥30%.

F. Turbine Stop Valve Scram shall be ≤10% valve closure from full open with turbine first stage pressure ≥30%.

G. Main Steamline Isolation Valve Closure Scram shall be ≤10% valve closure from full open.

H. Main Steamline Pressure initiation of main steamline isolation valve closure shall be ≥825 psig.

Bases 2.1:

A. The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is no less than the values specified in Technical Specification 2.1.A. This limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection systems safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling. (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The concept of MCPR, as used in the GETAB/GEXL critical power analyses, is discussed in Reference 1.

<785 psig <10% 785 psig

1. ~~B.~~ Core Thermal Power Limit (Reactor Pressure ≤~~800 psia~~ or Core Flow ~~≤10%~~ of Rated) At pressure below ~~800 psia~~, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and all core flows, this pressure differential is maintained in the bypass region of the core.

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and all flows will always be greater than 4.56 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than 28×10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from ~~14.7 psia to 800 psia~~ indicate that the fuel assembly critical power at 28×10^3 lbs/hr is approximately 3.35 MWt. With the design peaking factors this corresponds ~~to a~~ core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below ~~800 psia~~ or core flow less than 10% is conservative.

0 to 785 psig 785 psig

Bases 2.1:

2. ~~A.~~ Core Thermal Power Limit (Reactor Pressure ≥ 785 psig ~~≥ 800 psia~~ and Core Flow ≥ 10% ~~≥ 10%~~ of Rated.) Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. The Safety Limit ~~(T.S.2.1.A)~~ has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the Operating MCPR Limit (T.S.3.11.C) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the Safety Limit are provided at the beginning of each fuel cycle.

Because the boiling transition correlation is based on a large quantity of full scale data, there is a very high confidence that operation of a fuel assembly at the MCPR Safety Limit would not produce boiling transition. Thus, although it is not required to establish the Safety Limit, additional margin exists between the Safety Limit and the actual occurrence of loss of cladding integrity.

However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) ~~where~~ fuel similar in design to Monticello operated above the boiling transition for a significant period of time (30 minute~~s~~) without clad perforation.

If reactor pressure should ever exceed 1385 psig ~~1400 psia~~ during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the ~~fuel~~ cladding integrity Safety Limit has been violated.

In addition to the MCPR Safety Limit, operation is constrained to a maximum design linear heat generation rate for any fuel type in the core.

~~6.~~ Power Transient Plant safety analyses have shown that the scrams initiated by exceeding safety system setting will assure that the Safety Limit of 2.1.A or 2.1.B will not be exceeded. Control rod scram times and safety systems settings are checked periodically to assure that a scram will proceed as analyzed. As a further check, the plant process computer will be used as a fast data-acquisition system, when available during a scram, to verify that the scram was initiated by the primary source signal. The computer is normally available for this function. However, it is recognized that the plant may operate without the computer in service, in which event the confirmatory data will not be available and the verification specified by 2.1.C will not be required. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. For this specification, when a scram is only accomplished by means of a backup feature of the plant design, a specific analysis is required to determine whether or not a Safety Limit has been violated. The concept of not approaching a Safety Limit, providing scram signals are operable, is supported by the extensive plant safety analysis.

3. ~~D~~. Reactor Water Level (Shutdown Condition) During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. ~~The core will be cooled sufficiently to prevent clad melting should the water level be reduced two-thirds the core height.~~ Establishment of the safety limit ~~at 12 inches~~ above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

Bases 2.1 (Continued):

B. The pressure safety limit of 1332 psig as measured in the vessel steam space was derived from the design pressures of the reactor pressure vessel, steam space piping, water space piping, and recirculation pump casing. The respective design pressures are 1250 psig, 1110 psig, 1136 psig, and 1380 psig. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and USAS Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10% over the vessel design pressure (110% x 1250 = 1375 psig) and the USAS Code permits pressure transients up to 20% over the piping design pressure (120% x 1110 = 1332 psig for piping communicating with the vessel steam space and 120% x 1136 = 1363 psig at the bottom of the vessel). The pressure limit is 1332 psig based on reactor coolant system steam piping.

References

1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDO 10958.

Bases 2.2

Exceeding a Safety Limit may cause~~s~~ fuel damage and create~~d~~ a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," guidelines. Therefore, it is required to insert all insertable control rods and restore compliance with the Safety Limits within 2 hours. The 2 hour completion time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal. Other required actions are delineated in 10 CFR 50.36, 10 CFR 50.72, and 10 CFR 50.73.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect ~~to their effect on the applicable transient results as determined by the current analysis model.~~

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Bases 2.3 (Continued):

For analyses of the thermal consequences of the transients, the Operating MCPR Limit (T.S.3.11.C) is conservatively assumed to exist prior to initiation of the transients.

This choice of using conservative values of controlling parameters and initiating transients at the design power level, produces more pessimistic answers than would result by using expected values of control parameters and analyzing at higher power levels.

Deviations from as-left settings of setpoints are expected due to inherent instrument error, operator setting error, drift of the setpoint, etc. Allowable deviations are assigned to the limiting safety system settings for this reason. The effect of settings being at their allowable deviation extreme is minimal with respect to that of the conservatisms discussed above. Although the operator will set the setpoints within the trip settings specified, the actual values of the various setpoints can vary from the specified trip setting by the allowable deviation.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting or when a sufficient number of devices have been affected by any means such that the automatic function is incapable of preventing a safety limit from being exceeded while in a reactor mode in which the specified function must be operable. Sections 3.1 and 3.2 list the reactor modes in which the functions listed above are required.

A. Neutron Flux Scram The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1775 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that, with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Also, the flow biased neutron flux scram (specification 2.3.A.1) provides protection to the fuel safety limit in the unlikely event of a thermal-hydraulic instability.

Bases 2.3 (Continued):

Maximum Extended Load Line Limit Analyses (MELLLA) have been performed to allow operation at higher powers at flows below 87%. The flow referenced scram (and rod block line) have increased (higher slope and y-intercept) for two loop operation (See Core Operating Limits Report). The supporting analyses are discussed in GE NEDC-31849P report (Reference: Letter from NSP to NRC dated September 16, 1992).

Increased Core Flow (ICF) analyses have been performed to allow operating at flows above 100% for powers equal to or less than 100% (See Core Operating Limit Report). The supporting analyses are discussed in General Electric NEDC-31778P report (Reference: Letter from NSP to NRC dated September 16, 1992).

Evaluations discussed in NEDC-32546P, July 1996, demonstrate the operability of MELLLA and ICF for rerate conditions. In addition, the evaluation demonstrated the acceptability of MELLLA for single loop operation.

For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 20% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position and the associated APRM is not downscale. This switch occurs when reactor pressure is greater than 850 psig.

The operator will set the APRM neutron flux trip setting no greater than that stated in Specification 2.3.A.1. However, the actual setpoint can be as much as 3% greater than that stated in Specification 2.3.A.1 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.

B. Deleted

Bases 2.3 (Continued):

C. Reactor Low Water Level Scram The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained.

The operator will set the low water level trip setting no lower than 10'6" above the top of the active fuel. However, the actual setpoint can be as much as 6 inches lower due to the deviations discussed on page 39.

D. Reactor Low Low Water Level ECCS Initiation Trip Point The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. The design of the ECCS components to meet the above criterion was dependent on three previously set parameters; the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS could prevent the ECCS components from meeting their criterion. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

The operator will set the low low water level ECCS initiation trip setting ≥6'6" ≤6'10" above the top of the active fuel. However, the actual setpoint can be as much as 3 inches lower than the 6'6" setpoint and 3 inches greater than the 6'10" setpoint due to the deviations discussed on page 39.

E. Turbine Control Valve Fast Closure Scram The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass. This transient is less severe than the turbine stop valve closure with bypass failure and therefore adequate margin exists. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% thermal power as indicated by turbine first stage pressure. This takes into account the possibility of 14% power being passed directly to the condenser through the bypass valves.

F. Turbine Stop Valve Scram The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the Safety Limit (T.S.2.1.A) even during the worst case transient that assumes the turbine bypass is closed. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% thermal power as indicated by turbine first stage pressure. This takes into account the possibility of 14% power being passed directly to the condenser through the bypass valves.

G. Main Steam Line Isolation Valve Closure Scram The main steam line isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation closure. With the scram set at 10% valve closure there is no increase in neutron flux.

H. Main Steam Line Low Pressure Initiates Main Steam Isolation Valve Closure The low pressure isolation of the main steam lines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation at steamline pressures lower than 825 psig requires

Bases 2.3 (Continued):

that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of the neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

The operator will set this pressure trip at greater than or equal to 825 psig. However, the actual trip setting can be as much as 10 psi lower due to the deviations discussed on page 39.

~~2.2 REACTOR COOLANT SYSTEM~~

~~Applicability:~~

~~Applies to limits on reactor coolant system pressure.~~

~~Objective:~~

~~To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.~~

~~Specification:~~

~~The reactor vessel pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.~~

~~2.4 REACTOR COOLANT SYSTEM~~

~~Applicability:~~

~~Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.~~

~~Objective:~~

~~To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.~~

~~Specification:~~

~~A. Reactor Coolant High Pressure Scram shall be ≤ 1075 psig.~~

~~B. The self-actuation function of at least seven Reactor Coolant System safety relief valves shall be operable. Valves shall be set as follows:~~

~~8 valves at ≤ 1120 psig~~

Bases 2.2:

The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1335 psig as measured in the vessel steam space is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value was derived from the design pressures of the reactor pressure vessel, coolant piping, and recirculation pump casing. The respective design pressures are 1250 psig at 575°F, 1148 psig at 562°F, and 1380 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and the USAS Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10 percent over the vessel design pressure (110% x 1250 = 1375 psig) and the USAS Code permits pressure transients up to 20 percent over the piping design pressure (120% x 1148 = 1378 psig).
The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig and temperature of 575°F; this is more than a factor of 1.5 below the yield strength of 42,300 psi at this temperature. At the pressure limit of 1375 psig, the general membrane stress increases to 29,400 psi, still safely below the yield strength.
The reactor coolant system piping provides a comparable margin of protection at the established pressure safety limit.

Bases 2.2 (Continued):

The normal operating pressure of the reactor coolant system is approximately 1010 psig. Evaluations have determined that the most severe pressure transient is bounded by the closure of all MSIVs, followed by a reactor scram on high neutron flux (failure of the direct scram associated with MSIV position is assumed). The USAR discusses the analysis of this event. The analysis results demonstrate the safety/relief valve capacity is capable of maintaining pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). The safety limit ensures that the acceptance limit of 1375 psig is met during the design basis event at the vessel location with the highest pressure.

Bases 2.4:

The settings on the reactor high pressure scram, reactor coolant system safety/relief valves, turbine control valve fast closure scram, and turbine stop valve closure scram have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. The APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits. In addition to preventing power operation above 1075 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients.

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, with at least one safety/relief valve set to open at a pressure no greater than design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1775 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure remains below the 1375 psig ASME Code limit. Only five of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 3% above their setpoint of 1109 psig with a 0.4 second delay. The upper limit on safety/relief valve setpoint is established by the design pressure of the HPCI and RCIC systems. The design capability of the HPCI and RCIC systems has been conservatively demonstrated to be acceptable at pressures 3% greater than the safety/relief valve setpoint of 1109 psig. HPCI and RCIC pressures required for system operation are limited by the Low-Low Set SRV System to well below these values.

The operator will set the reactor coolant high pressure scram trip setting at 1075 psig or lower. However, the actual setpoint can be as much as 10 psi above the 1075 psig indicated set point due to the deviations discussed in the basis of Specification 3.1. In a like manner, the operator will set the reactor coolant system safety/relief valve initiation trip setting at 1120 psig (1109 psig + 1%) or lower. However, the as-found set point can be as much as 22.3 psi above the 1120 psig indicated set point due to the deviations discussed in the basis of Specification 3.6.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or when a sufficient number of devices have been affected by any means.

Bases 2.4 (Continued):

such that the automatic function is incapable of preventing a safety limit from being exceeded while in a reactor mode in which the specified function must be operable. The functions listed in this specification are required in all modes except cold shutdown.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Limiting Trip Settings	Modes in which function must be Operable or Operating** Refuel (3)	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Condition*
1. Mode Switch in Shutdown		X	X	X	1	1	A
2. Manual Scram		X	X	X	1	1	A
3. Neutron Flux IRM (See Note 2) a. High-High b. Inoperative	≤ 120/125 of full scale and < 20% of Rated Thermal Power	X	X		4	3	A
4. Flow Referenced Neutron Flux APRM (See Note 5) a. High-High b. Inoperative	≤ [0.66W+65.6] %Rated Thermal Power for two loop operation or ≤ [0.66(W-5.4)+65.6] % Rated Thermal Power for single loop operation Where: W = percent of recirculation drive flow to produce a core flow of 57.6×10^6 lbm/hr			X	3	2	A or B
c. High Flow Clamp	≤ 120 %						
5. High Reactor Pressure (See Note 9)	≤ 1075 psig	X	X(f)	X(f)	2	2	A

TABLE 3.1.1 - CONTINUED

Trip Function	Limiting Trip Settings	Modes in which function must be Operable or Operating** Refuel (3)	Startup	Run	Total No. of Instrument Channels per Trip System	Min. No. Operable or Operating Instrument Channels Per Trip System (1)	Required Conditions*
6. High Drywell Pressure (See Note 4)	$\leq$2 psig	X	X(e, f)	X(e, f)	2	2	A
7. Reactor Low Water Level	$\geq$ 7 in. ~~(annulus)~~	X	X(f)	X(f)	2	2	A
8. Scram Discharge Volume High Level							
a. East	$\leq$56 gal. (8)	X(a)	X(f)	X(f)	2	2	A
b. West	$\leq$56 gal. (8)	X(a)	X(f)	X(f)	2	2	A
9. Turbine Condenser Low Vacuum	$\geq$22 in. Hg	X(b)	X(b,f)	X(f)	2	2	A or C
10. Main Steamline Isolation Valve Closure	$\leq$10% Valve Closure	X(b)	X(b)	X	8	8	A or C
11. Turbine Control Valve Fast Closure	(See Note 7)			X(d, f)	2	2	D
12. Turbine Stop Valve Closure	$\leq$10% Valve Closure			X(d)	4	4	D

NOTES:

1. There shall be two operable or tripped trip systems for each function. A 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. For an IRM channel to be considered operable, its detector shall be fully inserted.

3. In the refueling mode with the reactor subcritical and reactor water temperature less than 212°F, only the following trip functions need to be operable: (a) Mode Switch in Shutdown, (b) Manual Scram, (c) High Flux IRM, (d) Scram Discharge Volume High Level.

4. Not required to be operable when primary containment integrity is not required.

5. To be considered operable, an APRM must have at least 2 LPRM inputs per level and at least a total of 14 LPRM inputs, except that channels 1, 2, 5, and 6 may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

RELOCATED

1. Mode Switch in Shutdown

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.6.1 of the USAR.

2. Manual Scram

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

3. Neutrom Flux IRM Scram

For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 20% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position and the associated APRM is not downscale. This switch occurs when reactor pressure is greater than 850 psig.

The IRMs are calibrated by the heat balance method such that 120/125 of full scale on the highest IRM range is below 20% of rated neutron flux ~~(see Specification 2.3.A.2)~~. The requirement that the IRM detectors be inserted in the core assures that the heat balance calibration is not invalidated by the withdrawal of the detector.

RELOCATED

4. Flow Referenced Neutron Flux APRM Scram

Neutron Flux Scram The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1775 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that, with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Also, the flow biased neutron flux scram ~~(specification 2.3.A.1)~~ provides protection to the fuel safety limit in the unlikely event of a thermal-hydraulic instability.

Maximum Extended Load Line Limit Analyses (MELLLA) have been performed to allow operation at higher powers at flows below 87%. The flow referenced scram (and rod block line) have increased (higher slope and y-intercept) for two loop operation (See Core Operating Limits Report). The supporting analyses are discussed in GE NEDC-31849P report (Reference: Letter from NSP to NRC dated September 16, 1992).

Increased Core Flow (ICF) analyses have been performed to allow operating at flows above 100% for powers equal to or less than 100% (See Core Operating Limit Report). The supporting analyses are discussed in General Electric NEDC-31778P report (Reference: Letter from NSP to NRC dated September 16, 1992).

Evaluations discussed in NEDC-32546P, July 1996, demonstrate the operability of MELLLA and ICF for rerate conditions. In addition, the evaluation demonstrated the acceptability of MELLLA for single loop operation.

RELOCATED

5. High Reactor Pressure Scram

The settings on the reactor high pressure scram, reactor coolant system safety/relief valves, turbine control valve fast closure scram, and turbine stop valve closure scram have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. The APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits. In addition to preventing power operation above 1075 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients.

6. High Drywell Pressure Scram

Instrumentation (pressure switches) in the drywell are provided to detect a loss of coolant accident and initiate the emergency core cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

7. Reactor Low Water Level Scram

The low reactor water level instrumentation is set to trip when reactor water level is ~~7"~~ ≥7" on the instrument. This corresponds to a lower water level ~~above the top of active fuel~~ inside the shroud at 100% power due to the pressure drop across the dryer/separator. This has been accounted for in the affected ~~trancient analysis.~~ safety analyses. All Technical Specification reactor water level setpoints are specified as inches measured in the reactor annulus and referenced to instrument "zero." Instrument "zero" is a point 477.5" above the inner clad surface on the bottom of the reactor vessel.

8. Scram Discharge Volume Scram

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by the scram can be accommodated in the discharge piping. Part of this piping consists of two instrument volumes which accommodate in excess of 56 gallons of water each and is the low point in the piping. During normal operation the discharge volumes are empty; however, should they fill with water, the water discharge to the piping from the reactor could not be accommodated which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volumes which alarm and scram the reactor when the volume of water in either of the discharge volume receiver tanks reaches 56 gallons. At this point there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

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9. Turbine Condenser Low Vacuum

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. ~~To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass. Reference FSAR Section 14.5.1.2.2 and supplemental information submitted February 13, 1973.~~ The condenser low vacuum scram is a back-up to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 22" Hg vacuum, stop valve closure occurs at 20" Hg vacuum, and bypass closure at 7" Hg vacuum.

10. Main Steamline Isolation Valve Closure

The main steamline isolation valve closure scram is set to scram when the isolation valves are ≤10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting the resultant transient is insignificant. ~~Reference Section 14.5.1.3.1 FSAR and supplemental information submitted February 13, 1973.~~

11. Turbine Control Valve Fast Closure

The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass. This transient is less severe than the turbine stop valve closure with bypass failure and therefore adequate margin exists. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% thermal power as indicated by turbine first stage pressure. This takes into account the possibility of 14% power being passed directly to the condenser through the bypass valves.

12. Turbine Stop Valve Closure

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the Safety Limit (T.S.2.1.A) even during the worst case transient that assumes the turbine bypass is closed. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% thermal power as indicated by turbine first stage pressure. This takes into account the possibility of 14% power being passed directly to the condenser through the bypass valves.

Although the operator will set the set points within the trip settings specified on Table 3.1-1, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. For power rerate, GE setpoint methodology provided in NEDC 31336, "General Electric Setpoint Methodology," is used in establishing setpoints. The deviations could be caused by inherent instrument error, operator setting error, drift of the set point, etc. Therefore, such deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the following amounts:

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Trip Function	Deviation	Trip Function	Deviation
3. High Flux IRM	+2/125 of scale	*7. Reactor Low Water Level	-6 inches
5. High Reactor Pressure	+10 psi	8. Scram Discharge Volume High Level	+1 gallon
6. High Drywell Pressure	+1 psi	9. Turbine Condenser Low Vacuum	-1/2 in. Hg

* This indication is reactor coolant temperature sensitive. The calibration is thus made for rated conditions. The level error at low pressures and temperatures is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap, and not the indicated level.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or a sufficient number of devices have been affected by any means such that the automatic function is incapable of operating within the allowable deviation while in a reactor mode in which the specified function must be operable, or the actions specified in 3.1.B are not initiated as specified.

If an unsafe failure is detected during surveillance testing, it is desirable to determine as soon as possible if other failures of a similar type have occurred and whether the particular function involved is still operable or capable of meeting the single failure criterion. To meet the requirements of Table 3.1.1, it is necessary that all instrument channels in one trip system be operable to permit testing in the other trip system. Thus, when failures are detected in the first trip system tested, they would have to be repaired before testing of the other system could begin. In the majority of cases, repairs or replacement can be accomplished quickly. If repair or replacement cannot be completed in a reasonable time, operation could continue with one tripped trip system until the surveillance testing deadline.

The ability to bypass one instrument channel when necessary to complete surveillance testing will preclude continued operation with scram functions which may be either unable to meet the single failure criterion or completely inoperable. It also eliminates the need for an unnecessary shutdown if the remaining channels are found to be operable. The conditions under which the bypass is permitted require an immediate determination that the particular function is operable. However, during the time a bypass is applied, the function will not meet the single failure criterion; therefore, it is prudent to limit the time the bypass is in effect by requiring that surveillance testing proceed on a continuous basis and that the bypass be removed as soon as testing is completed.

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Table 3.2.1 Instrumentation That Initiates Primary Containment Isolation Functions				
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 *
1. Main Steam and Recirc <u>Sample Line (Group 1)</u>				
a. Low Low Reactor Water Level	≥ -48" ~~≥6'-6" ≤6'-10"~~	2	2	A
b. High Flow In Main Steam Line	≤140% rated	8	8	A
c. High temp. in Main Steam Line Tunnel	≤200°F	8	2 of 4 in each of 2 sets	A
d. Low Pressure in Main Steam Line (3)	≥825 psig	2	2	B
2. RHR System, Head Cooling, Drywell, Sump, TIP (Group 2)				
a. Low Reactor Water Level	≥7" ~~(annulus)~~	2	2	C

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. <u>Reactor Cleanup System (Group 3)</u>				
a. High Drywell Pressure	≤2 psig	2	2	E
b. Low Low Reactor Water Level**	~~≥6' 6", ≤6' 10"~~ ≥-48"	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. <u>HPCI Steam Lines</u> (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 3.2.2 Instrumentation That Initiates Emergency Core Cooling Systems					
Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (3) (6)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (3) (6)	Required Conditions *
A. Core Spray and LPCI					
1. Pump Start					
a. Low Low Reactor Water Level and	≥ -48" ~~≥8' 6" ≤8' 10"~~	2	4(4)	4	A.
b. i. Reactor Low Pressure Permissive or	~~≥450 psig~~	2	2(4)	2	A.
ii. Reactor Low Pressure Permissive Bypass Timer	20 ± 1 min	2	1	1	B.
c. High Drywell Pressure (1)	≤2 psig	2	4(4)	4	A.
2. Low Reactor Pressure (Valve Permissive)	≥450 psig	2	2(4)	2	A.
3. Loss of Auxiliary Power	- - - - - -	2	2(2)	2	A.

Table 3.2.2 Instrumentation That Initiates Emergency Core Cooling Systems					
Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (3) (6)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (3) (6)	Required Conditions *
B. HPCI System					
1. High Drywell Pressure (1)	≤ 2 psig	1	4	4	A.
2. Low-Low Reactor Water Level	≥ -48" ~~≥ 6' 6" ≤ 6' 10"~~	1	4	4	A.
C. Automatic Depressurization					
1. Low-Low Reactor Water Level and	≥ -48" ~~≥ 6' 6" ≤ 6' 10"~~	2	2	2	B.
2. Auto Blowdown Timer and	≤ 120 seconds	2	1	1	B.
3. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	≥ 60 psig ≤ 150 psig	2	12(4)	12(4)	B.

Table 3.2.2 - (Continued)
Instrumentation That Initiates Emergency Core Cooling Systems

Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (3) (6)	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (3) (6)	Required Conditions *
D. Diesel Generator					
1. Degraded or Loss of Voltage Essential Bus (5)					
2. Low Low Reactor Water Level	≥ -48" ~~≥6'6" ≤6'10"~~	2	4(4)	4	C.
3. High Drywell Press	≤2 psig	2	4(4)	4	C.

NOTES:

1. High drywell pressure may be bypassed when necessary only by closing the manual containment isolation valves during purging for containment inerting or de-inerting. Verification of the bypass condition shall be noted in the control room log. Also need not be operable when primary containment integrity is not required.

2. One instrument channel is a circuit breaker contact and the other is an undervoltage relay.

Table 3.2.4 Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment System Initiation				
Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System	Required Conditions *
1. Low Low Reactor Water Level	~~≥ 6'-6", ≤ 6'-10"~~ ≥ -48"	2	2 (Notes 1, 3, 5, 6)	A. or B.
2. High Drywell Pressure	≤ 2 psig	2	2 (Notes 1, 3, 5, 6)	A. or B.
3. Reactor Building Plenum Radiation Monitors	≤ 100 mR/hr	1	1 (Notes 1, 2, 4)	A. or B.
4. Refueling Floor Radiation Monitors	≤ 100 mR/hr	1	1 (Notes 1, 2, 4)	A. or B.

Notes:

(1) There shall be two operable or tripped trip systems for each function with two instrument channels per trip system and there shall be one operable or tripped trip system for each function with one instrument channel per trip system.

(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 to:

(a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

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

(3) Need not be operable when primary containment integrity is not required.

(4) One of the two monitors may be bypassed for maintenance and/or testing.

Table 3.2.5 Instrumentation That Initiates a Recirculation Pump Trip and Alternate Rod Injection					
Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (1)	Total No. of Instrument Channels per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Conditions *
1. High Reactor Dome Pressure	≤1150 psig	2	2	2	A
2. Low-Low Reactor Water Level	≥ -48" ~~≥6'6" above the top of the active fuel~~	2	2	2	A

NOTE:

1. When one of the two trip systems is made or found to be inoperable, restore the inoperable trip system to operable status within 14 days or place the plant in the specified required condition within the next eight hours. When both trip systems are inoperable, place the plant in the specified required condition within eight hours unless at least one trip system is sooner made operable.

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

A. Reactor in Startup, Refuel, or Shutdown Mode.

Table 3.2.8
Other Instrumentation

Function	Trip Setting	Minimum No. of Operable or Operating Trip System (1) (2)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1) (2)	Required Conditions*
A. RCIC Initiation					
1. Low-Low Reactor Level	≥ -48" ~~≥6' 6" & ≤6' 10" above top of active fuel~~	1	4	4	B
B. HPCI/RCIC Turbine Shutdown					
1. ~~a.~~ High Reactor Level	≤ 48" ~~≤14' 6" above top of active fuel~~	1	2	2	A
C. HPCI/RCIC Turbine Suction Transfer					
1. ~~a.~~ Condensate Storage Tank Low Level Allowable Values	≥2' 3" above tank bottom (Two Tank Operation)	1	2	2	C
	≥6' 9" above tank bottom (One Tank Operation)	1	2	2	C

NOTE:

1. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated as follows:
   a. With one required instrument channel inoperable per trip function, place the inoperable channel or trip system in the tripped condition within 12 hours, or
   b. With more than one instrument channel per trip system inoperable, immediately satisfy the requirements by placing the appropriate channels or systems in the tripped condition, or
   c. Place the plant under the specified required condition using normal operating procedures.
2. A 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.

* Required conditions when minimum conditions for operation are not satisfied:
   A. Comply with Specification 3.5.A.
   B. Comply with Specification 3.5.D.
   C. Align HPCI and RCIC suction to the suppression pool.

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" ~~7"~~ on the instrument. This corresponds to a lower water level ~~above the top of active fuel~~ inside the shroud 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 ≥ -48" ~~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.~~ generators.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

F. Recirculation System

3. The reactor may be started and operated, or operation may continue with only one recirculation loop in operation provided that:

   a. The following changes to setpoints and safety limit settings will be made within 24 hours after initiating operation with only one recirculation loop in operation.

      1. The Operating Limit MCPR (MCPR) will be changed per Specification 3.11.C.
      2. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) will be changed per Specification ~~3.11.A.~~
      3. The APRM ~~Neutron~~ Flux Scram and APRM Rod Block setpoints will be changed as noted in ~~Specification 2.3.A and Table~~ Tables 3.1.1 and 3.2.3.

   b. Technical Specifications 3.5.F.1 and 3.5.F.2 are met.

4. With no reactor coolant system recirculation loops in operation:

   a. Comply with Technical Specifications 3.5.F.1 and 3.5.F.2 by inserting control rods and then comply with specifications 3.6.A.2 and 3.5.F.3 for operation with only one recirculation loop in operation,

   OR

   b. The reactor shall be placed in hot shutdown within 12 hours.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
E. Safety/Relief Valves 1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F the safety valve function (self actuation) of seven safety/relief valves shall be operable (note: Low-Low Set and ADS requirements are located in Specification 3.2.H. and 3.5.A, respectively). Valves shall be set as follows: 8 valves at $\leq$ 1120 psig 2. If Specification 3.6.E.1 is not met, initiate an orderly shutdown and have reactor coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.	E. Safety/Relief Valves 1. a. Safety/relief valves shall be tested or replaced each refueling outage pursuant to Specification 4.15.B. ~~The nominal self-actuation setpoints are specified in Section 2.4.B.~~ b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage. c. The integrity of the safety/relief valve bellows shall be continuously monitored. d. The operability of the bellows monitoring system shall be demonstrated at least once every three months. 2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

Bases 3.6/4.6 (Continued):

D. Coolant Leakage

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached. Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. Systems connected to the reactor coolant systems boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period. Other leakage detection methods provide warning of abnormal leakage and are not directly calibrated to provide leak rate measurements.

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E. Safety/Relief Valves

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, with at least one safety/relief valve set to open at a pressure no greater than design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1775 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure remains below the 1375 psig ASME Code limit. Only five of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 3% above their setpoint of 1109 psig with a 0.4 second delay. The upper limit on safety/relief valve setpoint is established by the ~~design pressure~~ operating limit of the HPCI and RCIC systems of 1120 psig. The design capability of the HPCI and RCIC systems has been conservatively demonstrated to be acceptable at pressures 3% greater than the safety/relief valve setpoint of 1109 psig. HPCI and RCIC pressures required for system operation are limited by the Low-Low Set SRV System to well below these values.

Bases 3.6/4.6 (Continued):

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

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

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

Testing of the safety/relief valves in accordance with ANSI/ASME OM-1-1981 each refueling outage ensures that any valve deterioration is detected. An as-found tolerance value of 3% for safety/relief valve setpoints is specified in ANSI/ASME OM-1-1981. Analyses have been performed with the valves assumed to open at 3% above their setpoint of 1109 psig. ~~As discussed in the Section 2.2 Bases,~~ The 1375 psig Code limit is not exceeded in any case. When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

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

I. Deleted

Bases 3.11 (Continued):

MCPR Limit is determined from the analysis of transients discussed in Bases Sections 2.1 ~~and 2.3.~~ By maintaining an operating MCPR above these limits, the Safety Limit (T.S. 2.1.A) is maintained in the event of the most limiting abnormal operational transient.

At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_F$ and $MCPR_P$ at the existing core flow and power state. The required MCPR is a function of flow in order to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

Flow runout events are analyzed with the purpose of establishing a flow dependent MCPR limit that would prevent the Safety Limit CPR from being reached during a flow runout. A flow runout event is a slow flow and power increase which is not terminated by a scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. Initial conditions for the transient are set such that the limiting CPR is near the Safety Limit. MCPR values are determined from the resulting change in CPR when core flow is increased to a possible maximum. Several combinations of initial power, flow, and exposure are analyzed to cover the range of operability defined by the power/flow map. The calculated flow dependent MCPR limit ($MCPR_f$) for a given core flow is provided in the Core Operating Limits Report.

For operation above 45% of rated thermal power, the core power dependent MCPR operating limit is the rated MCPR limit, MCPR(100), multiplied by the factor, provided in the Core Operating Limits Report. For operation below 45% of rated thermal power (turbine control valve fast closure and turbine stop valve closure scrams can be bypassed) MCPR limits are provided in the Core Operating Limits Report. This protects the core from plant transients other than core flow increase, including a localized event such as rod withdrawal error.

6.2 (Deleted)

6.3 (Deleted)

6.4 (Deleted) ~~Action to be Taken if a Safety Limit is Exceeded~~

If a Safety Limit is exceeded, the reactor shall be shut down immediately. An immediate report shall be made to the Commission and to the corporate officer with direct responsibility for the plant or his designated alternate in his absence. A complete analysis of the circumstances leading up to and resulting from the situation, together with recommendations by the Operations Committee, shall also by prepared. This report shall be submitted to the Commission, to the corporate officer with direct responsibility for the plant and the Chairman of the Safety Audit Committee within 14 days of the occurrence.

Reactor operation shall not be resumed until authorized by the U.S. Nuclear Regulatory Commission.

7. Core Operating Limits Report

a. Core operating limits shall be established and documented in the Core Operating Limits Report before each reload cycle or any remaining part of a reload cycle for the following:

Rod Block Monitor Operability Requirements (Specification 3.2.C.2a)
Rod Block Monitor Upscale Trip Settings (Table 3.2.3, Item 4.a)
Recirculation System Power to Flow Map Stability Regions (Specification 3.5.F)
Maximum Average Planar Linear Heat Generation Rate Limits (Specification 3.11.A)
Linear Heat Generation ~~Rate~~ Limits (Specification 3.11.B)
Minimum Critical Power Ratio Limits (Specification 3.11.C)
Power to Flow Map ~~(Bases 2.3.A)~~ (Bases 3.1)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (the approved version at the time the reload analyses are performed)*

NSPNAD-8608-A, "Reload Safety Evaluation Methods for Application to the Monticello Nuclear Generating Plant" (the approved version at the time the reload analyses are performed)

NSPNAD-8609-A, "Qualification of Reactor Physics Methods for Application to Monticello" (the approved version at the time the reload analyses are performed)

ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors-EXEM BWR Evaluation Model," Siemens Power Corporation (the approved version at the time the reload analyses are performed)

NEDO-31960, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," June 1991 (the approved version at the time the reload analyses are performed)

NEDO-31960, Supplement 1, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," March 1992 (the approved version at the time the reload analyses are performed)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

d. The Core Operating Limits Report, including any mid-cycle revisions or supplements, shall be supplied upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

* For cycle 19 only as approved in SE dated April 20, 1998.

Exhibit C

License Amendment Request dated June 18, 2001
Changes to the Technical Specifications
Revised Reference Point for Reactor Vessel Level Setpoints,
Simplification of Safety Limits, and Improvements to the Bases

Exhibit C consists of pages of the Monticello Technical Specifications retyped to incorporate the proposed changes. Revised or renumbered pages are:

Pages
i
6 - 12
28 - 29
36 - 40
49 - 50
52 - 54
59 - 60
60d
64
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150 - 151
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Deleted pages are:

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2.0 SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

A. Reactor Core Safety Limits

1. With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

   Thermal power shall be ≤ 25% Rated Thermal Power

2. With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

   MCPR shall be ≥ 1.11 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

3. Reactor vessel water level shall be greater than the top of active irradiated fuel.

B. Reactor Coolant System Pressure Safety Limit

Reactor steam dome pressure shall be ≤ 1332 psig.

Limiting Safety System Settings are incorporated into Section 3 of the Technical Specifications.

2.2 SAFETY LIMIT VIOLATIONS

With any Safety Limit violation, the following actions shall be completed within 2 hours:

A. Restore compliance with all Safety Limits; and

B. Insert all insertable control rods.

Bases 2.1:

A. The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is no less than the values specified in Technical Specification 2.1.A. This limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection systems safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling. (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The concept of MCPR, as used in the GETAB/GEXL critical power analyses, is discussed in Reference 1.

1. Core Thermal Power Limit (Reactor Pressure < 785 psig or Core Flow < 10% of Rated) At pressure below 785 psig, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and all core flows, this pressure differential is maintained in the bypass region of the core.

   Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and all flows will always be greater than 4.56 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than 28×10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 0 to 785 psig indicate that the fuel assembly critical power at 28×10^3 lbs/hr is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig or core flow less than 10% is conservative.

Bases 2.1 (Continued):

2. Core Thermal Power Limit (Reactor Pressure ≥785 psig and Core Flow ≥10% of Rated.) Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. The Safety Limit has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the Operating MCPR Limit (T.S.3.11.C) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the Safety Limit are provided at the beginning of each fuel cycle.

   Because the boiling transition correlation is based on a large quantity of full scale data, there is a very high confidence that operation of a fuel assembly at the MCPR Safety Limit would not produce boiling transition. Thus, although it is not required to establish the Safety Limit, additional margin exists between the Safety Limit and the actual occurrence of loss of cladding integrity.

   However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Monticello operated above the boiling transition for a significant period of time (30 minutes) without clad perforation.

   If reactor pressure should ever exceed 1385 psig during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

   In addition to the MCPR Safety Limit, operation is constrained to a maximum design linear heat generation rate for any fuel type in the core.

Bases 2.1 (Continued):

3. Reactor Water Level (Shutdown Condition) During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. Establishment of the safety limit above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

Bases 2.1 (Continued):

B. The pressure safety limit of 1332 psig as measured in the vessel steam space was derived from the design pressures of the reactor pressure vessel, steam space piping, water space piping, and recirculation pump casing. The respective design pressures are 1250 psig, 1110 psig, 1136 psig, and 1380 psig. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and USAS Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10% over the vessel design pressure (110% x 1250 = 1375 psig) and the USAS Code permits pressure transients up to 20% over the piping design pressure (120% x 1110 = 1332 psig for piping communicating with the vessel steam space and 120% x 1136 = 1363 psig at the bottom of the vessel). The pressure limit is 1332 psig based on reactor coolant system steam piping.

References

1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDO 10958.

Bases 2.2:

Exceeding a Safety Limit may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," guidelines. Therefore, it is required to insert all insertable control rods and restore compliance with the Safety Limits within 2 hours. The 2 hour completion time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal. Other required actions are delineated in 10 CFR 50.36, 10 CFR 50.72, and 10 CFR 50.73

TABLE 3.1.1 REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS							
Trip Function	Limiting Trip Settings	Modes in which function must be Operable or Operating**			Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Condition*
		Refuel (3)	Startup	Run			
1. Mode Switch in Shutdown		X	X	X	1	1	A
2. Manual Scram		X	X	X	1	1	A
3. Neutron Flux IRM (See Note 2) a. High-High b. Inoperative	≤120/125 of full scale AND <20% of Rated Thermal Power	X	X		4	3	A
4. Flow Referenced Neutron Flux APRM (See Note 5) a. High-High b. Inoperative	≤[0.66W+65.6] %Rated Thermal Power for two loop operation OR ≤[0.66(W-5.4)+65.6] %Rated Thermal Power for single loop operation Where: W=percent of recirculation drive flow to produce a core flow of $57.6x10^6$ lbm/hr			X	3	2	A or B
c. High Flow Clamp	≤120 %						
5. High Reactor Pressure (See Note 9)	≤1075 psig	X	X(f)	X(f)	2	2	A

TABLE 3.1.1 - CONTINUED							
Trip Function	Limiting Trip Settings	Modes in which function must be Operable or Operating**			Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Condition*
		Refuel (3)	Startup	Run			
6. High Drywell Pressure (See Note 4)	≤2 psig	X	X(e, f)	X(e, f)	2	2	A
7. Reactor Low Water Level	≥7 in.	X	X(f)	X(f)	2	2	A
8. Scram Discharge Volume High Level							
a. East	≤56 gal. (8)	X(a)	X(f)	X(f)	2	2	A
b. West	≤56 gal. (8)	X(a)	X(f)	X(f)	2	2	A
9. Turbine Condenser Low Vacuum	≥22 in. Hg	X(b)	X(b,f)	X(f)	2	2	A or C
10. Main Steamline Isolation Valve Closure	≤10% Valve Closure	X(b)	X(b)	X	8	8	A or C
11. Turbine Control Valve Fast Closure	(See Note 7)			X(d, f)	2	2	D
12. Turbine Stop Valve Closure	≤10% Valve Closure			X(d)	4	4	D

NOTES:

1. There shall be two operable or tripped trip systems for each function. A 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. For an IRM channel to be considered operable, its detector shall be fully inserted.
3. In the refueling mode with the reactor subcritical and reactor water temperature less than 212°F, only the following trip functions need to be operable: (a) Mode Switch in Shutdown, (b) Manual Scram, (c) High Flux IRM, (d) Scram Discharge Volume High Level.
4. Not required to be operable when primary containment integrity is not required.
5. To be considered operable, an APRM must have at least 2 LPRM inputs per level and at least a total of 14 LPRM inputs, except that channels 1, 2, 5, and 6 may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

Bases 3.1 (Continued):

1. Mode Switch in Shutdown

   A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.6.1 of the USAR.

2. Manual Scram

   The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

3. Neutron Flux IRM Scram

   For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 20% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position and the associated APRM is not downscale. This switch occurs when reactor pressure is greater than 850 psig.

   The IRMs are calibrated by the heat balance method such that 120/125 of full scale on the highest IRM range is below 20% of rated neutron flux. The requirement that the IRM detectors be inserted in the core assures that the heat balance calibration is not invalidated by the withdrawal of the detector.

Bases 3.1 (Continued):

4. Neutron Flux IRM Scram

Neutron Flux Scram The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1775 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that, with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Also, the flow biased neutron flux scram provides protection to the fuel safety limit in the unlikely event of a thermal-hydraulic instability.

Maximum Extended Load Line Limit Analyses (MELLLA) have been performed to allow operation at higher powers at flows below 87%. The flow referenced scram (and rod block line) have increased (higher slope and y-intercept) for two loop operation (See Core Operating Limits Report). The supporting analyses are discussed in GE NEDC-31849P report (Reference: Letter from NSP to NRC dated September 16, 1992).

Increased Core Flow (ICF) analyses have been performed to allow operating at flows above 100% for powers equal to or less than 100% (See Core Operating Limit Report). The supporting analyses are discussed in General Electric NEDC-31778P report (Reference: Letter from NSP to NRC dated September 16, 1992).

Evaluations discussed in NEDC-32546P, July 1996, demonstrate the operability of MELLLA and ICF for rerate conditions. In addition, the evaluation demonstrated the acceptability of MELLLA for single loop operation.

5. High Reactor Pressure Scram

The settings on the reactor high pressure scram, reactor coolant system safety/relief valves, turbine control valve fast closure scram, and turbine stop valve closure scram have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. The APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits. In addition to preventing power operation above 1075 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients.

6. High Drywell Pressure Scram

Instrumentation (pressure switches) in the drywell are provided to detect a loss of coolant accident and initiate the emergency core cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

7. Reactor Low Water Level Scram

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 inside the shroud at 100% power due to the pressure drop across the dryer/separator. This has been accounted for in the affected safety analyses. All Technical Specification reactor water level setpoints are specified as inches measured in the reactor annulus and referenced to instrument "zero." Instrument "zero" is a point 477.5″ above the inner clad surface on the bottom of the reactor vessel.

8. Scram Discharge Volume Scram

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by the scram can be accommodated in the discharge piping. Part of this piping consists of two instrument volumes which accommodate in excess of 56 gallons of water each and is the low point in the piping. During normal operation the discharge volumes are empty; however, should they fill with water, the water discharge to the piping from the reactor could not be accommodated which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volumes which alarm and scram the reactor when the volume of water in either of the discharge volume receiver tanks reaches 56 gallons. At this point there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

9. Turbine Condenser Low Vacuum

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. The condenser low vacuum scram is a back-up to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 22″ Hg vacuum, stop valve closure occurs at 20″ Hg vacuum, and bypass closure at 7″ Hg vacuum.

Bases 3.1 (Continued):

10. Main Steamline Isolation Valve Closure

The main steamline isolation valve closure scram is set to scram when the isolation valves are ≤10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting the resultant transient is insignificant.

11. Turbine control Valve Fast Closure

The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass. This transient is less severe than the turbine stop valve closure with bypass failure and therefore adequate margin exists. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% thermal power as indicated by turbine first stage pressure. This takes into account the possibility of 14% power being passed directly to the condenser through the bypass valves.

12. Turbine Stop Valve Closure

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the Safety Limit (T.S.2.1.A) even during the worst case transient that assumes the turbine bypass is closed. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% thermal power as indicated by turbine first stage pressure. This takes into account the possibility of 14% power being passed directly to the condenser through the bypass valves.

Although the operator will set the set points within the trip settings specified on Table 3.1.1, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. For power rerate, GE setpoint methodology provided in NEDC 31336, "General Electric Setpoint Methodology," is used in establishing setpoints. The deviations could be caused by inherent instrument error, operator setting error, drift of the set point, etc. Therefore, such deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the following amounts:

Bases 3. 1 (Continued):

Trip Function	Deviation	Trip Function	Deviation
3. High Flux IRM	+2/125 of scale	*7. Reactor Low Water Level	-6 inches
5. High Reactor Pressure	+10 psi	8. Scram Discharge Volume High Level	+1 gallon
6. High Drywell Pressure	+1 psi	9. Turbine Condenser Low Vacuum	-1/2 in. Hg

* This indication is reactor coolant temperature sensitive. The calibration is thus made for rated conditions. The level error at low pressures and temperatures is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap, and not the indicated level.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or a sufficient number of devices have been affected by any means such that the automatic function is incapable of operating within the allowable deviation while in a reactor mode in which the specified function must be operable, or the actions specified in 3.1.B are not initiated as specified.

If an unsafe failure is detected during surveillance testing, it is desirable to determine as soon as possible if other failures of a similar type have occurred and whether the particular function involved is still operable or capable of meeting the single failure criterion. To meet the requirements of Table 3.1.1, it is necessary that all instrument channels in one trip system be operable to permit testing in the other trip system. Thus, when failures are detected in the first trip system tested, they would have to be repaired before testing of the other system could begin. In the majority of cases, repairs or replacement can be accomplished quickly. If repair or replacement cannot be completed in a reasonable time, operation could continue with one tripped trip system until the surveillance testing deadline.

The ability to bypass one instrument channel when necessary to complete surveillance testing will preclude continued operation with scram functions which may be either unable to meet the single failure criterion or completely inoperable. It also eliminates the need for an unnecessary shutdown if the remaining channels are found to be operable. The conditions under which the bypass is permitted require an immediate determination that the particular function is operable. However, during the time a bypass is applied, the function will not meet the single failure criterion; therefore, it is prudent to limit the time the bypass is in effect by requiring that surveillance testing proceed on a continuous basis and that the bypass be removed as soon as testing is completed.

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Table 3.2.1 Instrumentation That Initiates Primary Containment Isolation Functions				
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*
1. Main Steam and Recirc <u>Sample Line (Group 1)</u>				
a. Low Low Reactor Water Level	≥-48"	2	2	A
b. High Flow In Main Steam Line	≤140% rated	8	8	A
c. High temp. in Main Steam Line Tunnel	≤200°F	8	2 of 4 in each of 2 sets	A
d. Low Pressure in Main Steam Line (3)	≥825 psig	2	2	B
2. RHR System, Head Cooling, Drywell, Sump, TIP (Group 2)				
a. Low Reactor Water Level	≥7"	2	2	C

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. <u>Reactor Cleanup System (Group 3)</u>				
a. High Drywell Pressure	≤2 psig	2	2	E
b. Low Low Reactor Water Level**	≥-48"	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. <u>HPCI Steam Lines</u> (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 3.2.2 Instrumentation That Initiates Emergency Core Cooling Systems					
Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (3) (6)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (3) (6)	Required Conditions*
A. <u>Core Spray and LPCI</u>					
1. Pump Start					
a. Low Low Reactor Water Level and	≥-48″	2	4(4)	4	A.
b. i. Reactor Low Pressure Permissive or	≥450 psig	2	2(4)	2	A.
ii. Reactor Low Pressure Permissive Bypass Timer	20±1 min	2	1	1	B.
c. High Drywell Pressure (1)	≤2 psig	2	4(4)	4	A.
2. Low Reactor Pressure (Valve Permissive)	≥450 psig	2	2(4)	2	A.
3. Loss of Auxiliary Power	------	2	2(2)	2	A.

Table 3.2.2 Instrumentation That Initiates Emergency Core Cooling Systems					
Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (3) (6)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (3) (6)	Required Conditions*
B. <u>HPCI System</u>					
1. High Drywell Pressure (1)	≤2 psig	1	4	4	A.
2. Low-Low Reactor Water Level	≥-48″	1	4	4	A.
C. <u>Automatic Depressurization</u>					
1. Low-Low Reactor Water Level and	≥-48″	2	2	2	B.
2. Auto Blowdown Timer and	≤120 seconds	2	1	1	B.
3. Low Pressure Core Cooling Pumps Discharge Pressure Interlock	≥60 psig ≤150 psig	2	12(4)	12(4)	B.

Table 3.2.2 - (Continued)
Instrumentation That Initiates Emergency Core Cooling Systems

Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (3) (6)	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (3) (6)	Required Conditions*
D. Diesel Generator					
1. Degraded or Loss of Voltage Essential Bus (5)					
2. Low Low Reactor Water Level	≥-48″	2	4(4)	4	C.
3. High Drywell Press	≤2 psig	2	4(4)	4	C.

NOTES:

1. High drywell pressure may be bypassed when necessary only by closing the manual containment isolation valves during purging for containment inerting or de-inerting. Verification of the bypass condition shall be noted in the control room log. Also need not be operable when primary containment integrity is not required.

2. One instrument channel is a circuit breaker contact and the other is an undervoltage relay.

Table 3.2.4 Instrumentation That Initiates Reactor Building Ventilation Isolation And Standby Gas Treatment System Initiation				
Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System	Required Conditions*
1. Low Low Reactor Water Level	≥-48″	2	2 (Notes 1, 3, 5, 6)	A. or B.
2. High Drywell Pressure	≤2 psig	2	2 (Notes 1, 3, 5, 6)	A. or B.
3. Reactor Building Plenum Radiation Monitors	≤100 mR/hr	1	1 (Notes 1, 2, 4)	A. or B.
4. Refueling Floor Radiation Monitors	≤100 mR/hr	1	1 (Notes 1, 2, 4)	A. or B.

Notes:

(1) There shall be two operable or tripped trip systems for each function with two instrument channels per trip system and there shall be one operable or tripped trip system for each function with one instrument channel per trip system.

(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 to:

(a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

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

(3) Need not be operable when primary containment integrity is not required.

(4) One of the two monitors may be bypassed for maintenance and/or testing.

Table 3.2.5 Instrumentation That Initiates a Recirculation Pump Trip and Alternate Rod Injection					
Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (1)	Total No. of Instrument Channels per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Conditions*
1. High Reactor Dome Pressure	≤1150 psig	2	2	2	A
2. Low-Low Reactor Water Level	≥-48″	2	2	2	A

NOTE:

1. When one of the two trip systems is made or found to be inoperable, restore the inoperable trip system to operable status within 14 days or place the plant in the specified required condition within the next eight hours. When both trip systems are inoperable, place the plant in the specified required condition within eight hours unless at least one trip system is sooner made operable.

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

A. Reactor in Startup, Refuel, or Shutdown Mode.

Table 3.2.8 Other Instrumentation					
Function	Trip Setting	Minimum No. of Operable or Operating Trip System (1) (2)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1) (2)	Required Conditions*
A. RCIC Initiation					
1. Low-Low Reactor Level	≥-48″	1	4	4	B
B. HPCI/RCIC Turbine Shutdown					
1. High Reactor Level	≤48″	1	2	2	A
C. HPCI/RCIC Turbine Suction Transfer					
1. Condensate Storage Tank Low Level Allowable Values	≥2′ 3″ above tank bottom (Two Tank Operation)	1	2	2	C
	≥6′ 9″ above tank bottom (One Tank Operation)	1	2	2	C

<u>NOTE</u>:

1. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated as follows:
   a. With one required instrument channel inoperable per trip function, place the inoperable channel or trip system in the tripped condition within 12 hours, or
   b. With more than one instrument channel per trip system inoperable, immediately satisfy the requirements by placing the appropriate channels or systems in the tripped condition, or
   c. Place the plant under the specified required condition using normal operating procedures.
2. A 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.

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

A. Comply with Specification 3.5.A.
B. Comply with Specification 3.5.D.
C. Align HPCI and RCIC suction to the suppression pool.

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 inside the shroud 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 ≥-48″. 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 generators.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

F. Recirculation System

3. The reactor may be started and operated, or operation may continue with only one recirculation loop in operation provided that:

   a. The following changes to setpoints and safety limit settings will be made within 24 hours after initiating operation with only one recirculation loop in operation.

      1. The Operating Limit MCPR (MCPR) will be changed per Specification 3.11.C.
      2. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) will be changed per Specification 3.11.A.
      3. The APRM Neutron Flux Scram and APRM Rod Block setpoints will be changed as noted in Tables 3.1.1 and 3.2.3.

   b. Technical Specifications 3.5.F.1 and 3.5.F.2 are met.

4. With no reactor coolant system recirculation loops in operation:

   a. Comply with Technical Specifications 3.5.F.1 and 3.5.F.2 by inserting control rods and then comply with specifications 3.6.A.2 and 3.5.F.3 for operation with only one recirculation loop in operation,

      OR

   b. The reactor shall be placed in hot shutdown within 12 hours.

3.0 LIMITING CONDITIONS FOR OPERATION

E. Safety/Relief Valves

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

   Valves shall be set as follows:

   8 valves at $\leq$1120 psig

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

4.0 SURVEILLANCE REQUIREMENTS

E. Safety/Relief Valves

1. a. Safety/relief valves shall be tested or replaced each refueling outage in accordance with the Inservice Testing Program.

   b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.

   c. The integrity of the safety/relief valve bellows shall be continuously monitored.

   d. The operability of the bellows monitoring system shall be demonstrated at least once every three months.

2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

D. Coolant Leakage

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached. Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. Systems connected to the reactor coolant systems boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period. Other leakage detection methods provide warning of abnormal leakage and are not directly calibrated to provide leak rate measurements.

E. Safety/Relief Valves

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, with at least one safety/relief valve set to open at a pressure no greater than design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1775 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure remains below the 1375 psig ASME Code limit. Only five of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 3% above their setpoint of 1109 psig with a 0.4 second delay. The upper limit on safety/relief valve setpoint is established by the operating limit of the HPCI and RCIC systems of 1120 psig. The design capability of the HPCI and RCIC systems has been conservatively demonstrated to be acceptable at pressures 3% greater than the safety/relief valve setpoint of 1109 psig. HPCI and RCIC pressures required for system operation are limited by the Low-Low Set SRV System to well below these values.

Bases 3.6/4.6 (Continued):

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

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

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

Testing of the safety/relief valves in accordance with ANSI/ASME OM-1-1981 each refueling outage ensures that any valve deterioration is detected. An as-found tolerance value of 3% for safety/relief valve setpoints is specified in ANSI/ASME OM-1-1981. Analyses have been performed with the valves assumed to open at 3% above their setpoint of 1109 psig. The 1375 psig Code limit is not exceeded in any case. When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

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

I. Deleted

Bases 3.11 (Continued):

MCPR Limit is determined from the analysis of transients discussed in Bases Section 2.1. By maintaining an operating MCPR above these limits, the Safety Limit (T.S. 2.1.A) is maintained in the event of the most limiting abnormal operational transient.

At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_F$ and $MCPR_P$ at the existing core flow and power state. The required MCPR is a function of flow in order to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

Flow runout events are analyzed with the purpose of establishing a flow dependent MCPR limit that would prevent the Safety Limit CPR from being reached during a flow runout. A flow runout event is a slow flow and power increase which is not terminated by a scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. Initial conditions for the transient are set such that the limiting CPR is near the Safety Limit. MCPR values are determined from the resulting change in CPR when core flow is increased to a possible maximum. Several combinations of initial power, flow, and exposure are analyzed to cover the range of operability defined by the power/flow map. The calculated flow dependent MCPR limit ($MCPR_f$) for a given core flow is provided in the Core Operating Limits Report.

For operation above 45% of rated thermal power, the core power dependent MCPR operating limit is the rated MCPR limit, MCPR(100), multiplied by the factor, provided in the Core Operating Limits Report. For operation below 45% of rated thermal power (turbine control valve fast closure and turbine stop valve closure scrams can be bypassed) MCPR limits are provided in the Core Operating Limits Report. This protects the core from plant transients other than core flow increase, including a localized event such as rod withdrawal error.

6.2 (Deleted)

6.3 (Deleted)

6.4 (Deleted)

7. Core Operating Limits Report

a. Core operating limits shall be established and documented in the Core Operating Limits Report before each reload cycle or any remaining part of a reload cycle for the following:

Rod Block Monitor Operability Requirements (Specification 3.2.C.2a)
Rod Block Monitor Upscale Trip Settings (Table 3.2.3, Item 4.a)
Recirculation System Power to Flow Map Stability Regions (Specification 3.5.F)
Maximum Average Planar Linear Heat Generation Rate Limits (Specification 3.11.A)
Linear Heat Generation Rate Limits (Specification 3.11.B)
Minimum Critical Power Ratio Limits (Specification 3.11.C)
Power to Flow Map (Bases 3.11)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (the approved version at the time the reload analyses are performed)*

NSPNAD-8608-A, "Reload Safety Evaluation Methods for Application to the Monticello Nuclear Generating Plant" (the approved version at the time the reload analyses are performed)

NSPNAD-8609-A, "Qualification of Reactor Physics Methods for Application to Monticello" (the approved version at the time the reload analyses are performed)

ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors-EXEM BWR Evaluation Model," Siemens Power Corporation (the approved version at the time the reload analyses are performed)

NEDO-31960, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," June 1991 (the approved version at the time the reload analyses are performed)

NEDO-31960, Supplement 1, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," March 1992 (the approved version at the time the reload analyses are performed)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

d. The Core Operating Limits Report, including any mid-cycle revisions or supplements, shall be supplied upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

* For cycle 19 only as approved in SE dated April 20, 1998.