NRC FORM 195 (2-70)

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TO: Mr. Victor Stello

FROM: Northern States Power Company, Minneapolis, Minnesota, L. O. Mayer

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ENCLOSURE

Amdt. to OL/change to Appendix A tech specs concerning the containment water volume & differential pressure instrumentation .....

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NRC FORM 195 (2-70)

NSP

NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

April 15, 1977

Mr Victor Stello, Director
Division of Operating Reactors
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Stello:

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

License Amendment Request Dated April 15, 1977

Attached are three originals and 37 conformed copies of a request for a change of Technical Specifications, Appendix A of the Provisional Operating License for the Monticello Nuclear Generating Plant. This change request has been reviewed by the Monticello Operations Committee and the Safety Audit Committee.

These proposed changes revise the Technical Specifications covering the containment water volume and differential pressure instrumentation in response to a February 4, 1977 letter from Mr K R Goller, USNRC. Changes on a related subject proposed in our License Amendment Request Dated November 5, 1976 have been incorporated so as to supercede that document.

We have determined that these changes do not involve an unreviewed safety question.

Yours very truly,

L. O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/MHV/deh

cc: J G Keppler
G Charnoff
MPCA
Attn: J W Ferman
MECCA
Attn: H J Vogel
S J Gadler

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UNITED STATES NUCLEAR REGULATORY COMMISSION

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT

Docket No. 50- 263

REQUEST FOR AMENDMENT TO
OPERATING LICENSE NO. DPR- 22

(License Amendment Request Dated April 15, 1977)

Northern States Power Company, a Minnesota corporation, requests authorization for changes to the Technical Specifications as shown on the attachments labeled Exhibit A and Exhibit B. Exhibit A describes the proposed changes along with reasons for the change. Exhibit B is a set of Technical Specification pages incorporating the proposed changes.

This request contains no restricted or other defense information.

NORTHERN STATES POWER COMPANY

By L. J. Wachter
L J Wachter
Vice President, Power Production &
System Operation

On this 15th day of April, 1977, before me a notary public in and for said County, personally appeared L J Wachter, Vice President, Power Production & System Operation, and first being duly sworn acknowledged that he is authorized to execute this document in behalf of Northern States Power Company, that he knows the contents thereof and that to the best of his knowledge, information and belief, the statements made in it are true and that it is not interposed for delay.

Denise E. Halvorson

DENISE E. HALVORSON
NOTARY PUBLIC - MINNESOTA
HENNEPIN COUNTY
My Commission Expires Oct. 10, 1981

EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

LICENSE AMENDMENT REQUEST
DATED APRIL 15, 1977

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS
APPENDIX A OF
PROVISIONAL OPERATING LICENSE DPR-22

Pursuant to 10CFR50.59, the holders of Provisional Operating License DPR-22 hereby propose the following changes to the Appendix A Technical Specifications. Related changes proposed in our License Amendment Request Dated November 5, 1976 have been incorporated in this submittal, superceding the November request; however, the November document must be consulted for supporting information for those changes. Certain pages identified below as containing proposed changes are also subject to unrelated changes by License Amendment Request Dated January 30, 1976 as revised May 4, 1976. The latter changes are not incorporated into this submittal; they should be treated independently.

PROPOSED CHANGES

1. Specifications 3.7.A.1 and 4.7.A.1 (pages 139 and 140) - These specifications have been re-structured as shown in Exhibit B without changing the requirements in any way. This provides consistency with the format of other specifications, makes it easier to associate each surveillance requirement with the respective limiting condition for operation and replaces the undefined term "nuclear system" with alternate wording supported by the existing bases.

2. Specifications 3.7.A.1.f and 4.7.A.1.f (page 140) and Bases (page 157A) - The proposed wording shown in Exhibit B has been added to specify and discuss suppression chamber water volume indication operability and calibration requirements.

3. Specification 3.7.A.5.a (page 147A) - Remove the obsolete phrase, "After completion of the startup test program and demonstration of plant electrical output".

4. Specification 3.7.A.7.e (page 147B) and Bases (page 159) - A provision has been added which would remove the drywell to suppression chamber differential pressure requirement for planned safety/relief valve testing.

5. Specifications 3.7.A.7.d and 4.7.A.7.b (page 147B) and Bases (page 159) - The proposed wording shown in Exhibit B has been added to specify and discuss drywell to suppression chamber differential pressure indicator operability and calibration requirements.

6. Bases 3.7.A (page 156) - The reference to a control rod worth of 1.5% delta k has been replaced with an increment of rod worth of 1.3% delta k. (This Bases change was inadvertently omitted when the same change was issued for Specification 3.3.B.3 (a) on November 27, 1973.)

7. Bases 3.7.A (page 158A) - Change the words on the top line from "3/4 inch opening of any one valve or .08 in opening of all ten valves" to read, "one inch opening of any valve or 0.1 inch opening for all ten valves." (This proposed change was inadvertently omitted from an earlier change request. It should have been included in changes issued on February 26, 1975 to make the Bases consistent with the analysis supporting the change to the specifications.)

8. Specification 3.7/4.7 and Bases (pages 139 through 167) - There are numerous minor format changes proposed throughout these pages as shown in Exhibit B, in addition to the more significant changes identified above, to make the specifications easier to use through application of a consistent format.

REASONS FOR CHANGE

Proposed Changes 1, 3 and 8 - These changes re-structure and/or re-format the existing wording for easier readability and application. There are no inherent changes in the requirements of the specifications.

Proposed Changes 2 and 5 - These changes propose limiting conditions for operation and surveillance requirements for operation and surveillance requirements for instrumentation as requested in Mr Karl R Goller's February 4, 1977 letter. We do not share in the NRC opinion that Technical Specifications are appropriate for this type of instrumentation. The parameters monitored, suppression chamber water volume and drywell to suppression chamber differential pressure, do not experience rapid changes during normal operation. They represent the condition of stagnant, confined masses of water and gas. Standard practice for similar situations is to require that a parameter periodically be verified to be within the Technical Specification limit. Since the parameters in question are not used to initiate automatic action, it is irrelevant what method or instrument is used, as long as the parameter can be shown to be within the specified limit. Requiring the periodic check to verify conformance to limits, by itself, inherently requires that some acceptable means must exist to monitor a parameter. We also question the NRC staff technical position that instrument sticking or drifting should be addressed by requiring operability of redundant channels of instrumentation. If these problems do in fact exist, we believe that they should be addressed directly through improved surveillance techniques and a review of the application of instrumentation. Nevertheless, we have proposed limiting conditions for operation and surveillance requirements should you remain convinced that such requirements are appropriate.

A question has also arisen whether the suppression chamber water volume should be specified in terms of volume or level. The proposed Specification maintains a volume criteria, reflecting our position that volume is more correct than level. The initial plant design specified the minimum water volume required as a heat sink in the event of a design basis loss of coolant accident and a minimum volume to be consistent with test data available at the time. Subsequent testing and re-calculations have verified the suppression chamber integrity for the range of volumes specified, including the effect of the drywell to suppression chamber differential pressure. Our objection to converting from a volume to a level specification is as follows:

1. The change would simply be a matter of "optics"; the water volume would only be expressed in different units.
2. The volume is the basic requirement; secondary analyses use that volume expressed in terms of level, downcomer submergence, etc. to verify that other conditions are acceptable. Specifying level would lose sight of the basic requirement.
3. Volume is not affected by containment differential pressure while level is affected. Including this effect would unnecessarily complicate the Technical Specifications. Omitting the effect would either allow operation within Technical Specification limits but outside of the analyzed range or prevent operation over the entire analyzed range through unduly restrictive Technical Specification limits.
4. Changing the units in the Specification does not result in an improvement in safety. Making such a change introduces the potential for confusion and error as the change is incorporated into surveillance procedures, operating instructions, operational checklists and operator training.

Proposed Change 4 - An addition has been proposed, beyond the November 5, 1976 change request, to conveniently allow for special planned tests of safety/relief valves without processing a formal Technical Specification change.

Proposed Changes 6 and 7 - These changes affect the Bases only. In both cases the reason for the change was presented earlier when the corresponding Specifications were changed.

SAFETY EVALUATION

Proposed Changes 1, 3, 6, 7 and 8 - These proposed changes only make editorial changes and correct previous omissions. They do not affect the intent of any Specifications. Therefore, they have no affect on plant safety.

Proposed Changes 2 and 5 - Instrumentation is presently in place to monitor suppression chamber water volume and drywell to suppression chamber differential pressure. These proposed changes add limiting conditions for operation and surveillance requirements on the instrumentation. As stated in the proposed Specification, one of two redundant indicators may be out of service for 30 days and both indicators may be out of service for surveillance and maintenance for a period of time equal to the interval between surveillance checks of the parameter. These indicators do not initiate any automatic action; they are of an application different from most other instrumentation required by the Technical Specifications. Being redundant only provides the benefit of comparing one channel against another. Inoperability of one channel for up to 30 days is therefore reasonable. The parameters in question do not change rapidly during operation. This fact, coupled with the low probability that a loss of coolant accident occurs during a time when the parameter is outside of the specified limit, supports the fact that a check of the two parameters is required only once per day and once per shift, respectively. Allowing both channels out of service for the same interval is commensurate with the application. An annual calibration frequency is consistent with industry standards applied to similar application of analog transmitter devices.

Proposed Change 4 - Insufficient details of planned safety/relief valve tests are available for a safety evaluation at this time. Sufficient information will be provided for an NRC Staff safety evaluation supporting the differential pressure exemption in sufficient time prior to each such exemption.

EXHIBIT B

LICENSE AMENDMENT REQUEST DATED APRIL 15, 1977

This exhibit consists of the following pages revised or added to incorporate all of the proposed Technical Specification Changes:

139
140
147A
147B (Newly created in 11/5/77 License Amendment Request)
148
156
157
157A
158
158A
158B
159
160
161
161A
165
167

3.0 LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

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

Objective:

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

Specification:

A. Primary Containment.

1. Suppression Pool Volume and Temperature

   At any time that the reactor water temperature exceeds 212°F or work is being done which has the potential to drain the vessel, except as permitted by specification 3.5.G.4, the following requirements shall be met:

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

4.0 SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment.

1. Suppression Pool Volume and Temperature

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

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
d. During reactor isolation conditions the reactor pressure vessel shall be depressurized to $<$200 psig at normal cooldown rates if the suppression pool temperature exceeds 120°F.	d. Whenever there is indication of relief valve operation with a suppression pool temperature $\geq$160°F and the primary coolant system pressure $>$200 psig, an extended visual examination of the suppression chamber shall be conducted before resuming power operation.
e. The suppression chamber water volume shall be $\geq$68,000 and $\leq$77,970 cubic feet.	e. The suppression chamber water volume shall be checked once per day.
f. Two channels of suppression chamber water volume indication shall be available at all times except that one of the two channels may be out of service for $\leq$30 days or both channels may be out for $\leq$24 hours for service and maintenance.	f. The suppression chamber water volume indicators shall be calibrated annually.
2. Primary Containment Integrity Primary containment integrity, as defined in Section 1, shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 Mw(t).	2. Primary Containment Integrity The primary containment integrity shall be demonstrated as follows: a. Integrated Primary Containment Leak Test (IPCLT) (1) An integrated leak rate test shall be performed prior to initial unit operation at an initial test pressure (Pt) of 41 psig. (2) Subsequent leak rate tests shall be performed without preliminary leak detection surveys or leak repairs immediately prior to or during the test, at an initial pressure of approximately 41 psig. (3) Leak repairs, if necessary to permit integrated leak rate testing, shall be preceded by local leak rate measurements where possible. The leak rate differ-

3.0 LIMITING CONDITIONS FOR OPERATION

d. One position alarm circuit can be inoperable providing that the redundant position alarm circuit is operable. Both position alarm circuits may be inoperable for a period not to exceed seven days provided that all vacuum breakers are operable.

5. Oxygen Concentration

a. The primary containment atmosphere shall be reduced to less than 5% oxygen with nitrogen gas whenever the reactor is in the run mode, except as specified in 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the run mode following shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 5% by weight, and maintained in this condition. Deinerting may commence 24 hours prior to leaving the run mode for a reactor shutdown.

4.0 SURVEILLANCE REQUIREMENTS

b. When the position of any drywell-suppression chamber vacuum breaker valve is indicated to be not fully closed at a time when such closure is required, the drywell to suppression chamber differential pressure decay shall be demonstrated to be less than that shown on Figure 3.7.1 immediately and following any evidence of subsequent operation of the inoperable valve until the inoperable valve is restored to a normal condition.

c. When both position alarm circuits are made or found to be inoperable, the control panel indicator light status shall be recorded daily to detect changes in the vacuum breaker position.

5. Oxygen Concentration

Whenever inerting is required, the primary containment oxygen concentration shall be measured and recorded on a weekly basis.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
6. If specifications 3.7.A.1 through 3.7.A.5 cannot be met, the reactor shall be placed in the cold shutdown condition within 24 hours.	
7. DRYWELL-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE a. Drywell pressure shall be maintained $\geq$1.0 psi above the suppression chamber pressure except as specified in 3.7.A.7.b and c. b. Within the 26 hour period subsequent to placing the reactor in the run mode following a shutdown, the drywell pressure must be raised to $\geq$1.0 psi above the suppression chamber pressure and maintained in this condition. The differential pressure need not be maintained during the 26 hour period prior to leaving the run mode for a reactor shutdown. c. The differential pressure may be decreased to $<$1.0 psi for a maximum of 3 hours during required operability testing of the HPCI system pump, the RCIC system pump, and the drywell-pressure suppression chamber vacuum breakers. On receipt of written permission of the Nuclear Regulatory Commission, the differential pressure may also be decreased to $<$1.0 psi for periods necessary for planned safety/relief valve testing. d. Two channels of drywell suppression chamber differential pressure indication shall be available at all times except that one of the two channels may be out of service for $\leq$30 days or both channels may be out for $\leq$8 hours for service and maintenance. e. If specification 3.7.A.7 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the hot shutdown condition within 12 hours and the cold shutdown condition within the following 24 hours.	7. DRYWELL-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE a. The differential pressure between the drywell and suppression chamber shall be logged once per shift. b. The differential pressure indicators shall be calibrated annually.

3.0 LIMITING CONDITIONS FOR OPERATION

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment system circuits shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1.(a) and (b).

   a. After one of the standby gas treatment system circuits is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment system shall be demonstrated to be operable within 2 hours and daily thereafter. Within 36 hours following the 7 days, the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.1.(a) through (d).

4.0 SURVEILLANCE REQUIREMENTS

B. Standby Gas Treatment System

1. At least once per month, initiate from the control room 3500 cfm (± 10%) flow through both circuits of the standby gas treatment system. In addition:

   a. Within 2 hours from the time that one standby gas treatment system circuit is made or found to be inoperable for any reason and daily thereafter for the next succeeding seven days, initiate from the control room 3500 cfm (± 10%) flow through the operable circuit of the standby gas treatment system.

Bases:

3.7 A. Primary Containment

The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the off-site doses to values less than 10 CFR 100 guideline values in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit incremental control worth to less than 1.3% Δk. A drop of a 1.3% Δk increment of a rod does not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offers a sufficient barrier to keep off-site doses well within 10 CFR 100 guide line values.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the maximum allowable primary containment pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. Reference Section 5.2.3 FSAR.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 41 psig which is below the allowable pressure of 62 psig. The nominal downcomer submergence for the Monticello wetwell design is 4 feet which is in conformance with most of the Bodega tests. The majority of Bodega tests (1) were run with a submerged

(1) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

length of four feet, which resulted in complete condensation. Thus with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humboldt Bay [1] and Bodega Bay[2] tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

For an initial maximum suppression chamber water temperature of 90°F and assuming the normal complement of containment cooling pumps (2 LPCI pumps and 2 containment cooling service water pumps), containment pressure is not required to maintain adequate net positive suction head (NPSH) for the core spray, LPCI and HPCI pumps. However, during an approximately one-day period starting a few hours after a loss-of-coolant accident, should one RHR loop be inoperable and should the containment pressure be reduced to atmospheric pressure through any means, adequate NPSH would not be available. Since an extremely degraded condition must exist, the period of vulnerability to this event is restricted by Specification 3.7.A.1.b by limiting the suppression pool initial temperature and the period of operation with one inoperable RHR loop.

(1) Robbins, C. H., "Tests of Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

Bases Continued:

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

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

Bases Continued:

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and between the suppression chamber and reactor building during loss of coolant accident so that structural integrity of the containment is maintained.

The vacuum relief system between the pressure suppression chamber and reactor building consist of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 1 psig. The external design pressure is 2 psig. One valve may be out of service for repairs for a period of seven days. This period is based on the low probability that system redundancy would be required during this time. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

The capacity of the ten (10) drywell vacuum relief valves is sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to less than the design limit of 2 psi. The relief valves are sized on the basis of the Bodega Bay pressure suppression system tests. Since they are in series with the reactor building to suppression chamber vacuum relief valves pressure drop across these valves must be included in the evaluation of drywell negative pressures, even though there does not appear to be a mechanism for causing negative pressures in excess of the 2 psi design pressure. With eight of the ten valves in service, the differential pressure across the valves for maximum flow conditions would increase. With this additional pressure drop the total differential pressure would still be less than the 2 psi design valve. Containment integrity would therefore not be impaired.

In addition to the above considerations, postulated leakage through the vacuum breaker to the suppression chamber air space could result in a partial bypass of pressure suppression in the event of a LOCA or a small or intermediate steam leak. This effect could potentially result in exceeding containment design pressure. As a result of the leakage potential, the containment response has been analyzed for a number of postulated conditions. It was found that the maximum allowable bypass area for any postulated break size was equivalent to a six-inch diameter opening.[1] This bypass corresponds to a

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[1] Report on Torus to Drywell Vacuum Breaker Tests and Modifications for Monticello Nuclear Generating Plant, dated March 12, 1973, submitted to Mr. D. J. Skovholt, AEC-DL, from Mr. L. O. Mayer, NSP

Bases Continued:

One inch opening of any one valve or 0.1 inch opening for all ten valves, measured at the bottom of the disc with the top of the disc at the seat. The position indication system is designed to detect closure within 1/8 inch at the bottom of the disc.

At each refueling outage and following any sigificant maintenance on the vacuum breaker valves, positive seating of the vacuum breakers will be verified by leak test. The leak test is conservatively designed to demonstrate that leakage is less than that equivalent to leakage through a one-inch orifice which is about 3% of the maximum allowable. This test is planned to establish a baseline for valve performance at the start of each operating cycle and to ensure that vacuum breakers are maintained as nearly as possible to their design condition. This test is not planned to serve as a limiting condition for operation.

During reactor operation, an exercise test of the vacuum breakers will be conducted monthly. This test will verify that disc travel is unobstructed and will provide verification that the valves are closing fully through the position indication system. If one or more of the vacuum breakers do not seat fully as determined from the indicating system, a leak test will be conducted to verify that leakage is within the maximum allowable. Since the extreme lower limit of switch detection capability is approximately 1/16", the planned test is designed to strike a balance between the detection switch capability to verify closure and the maximum allowable leak rate. A special test was performed to establish the basis for this limiting condition. During the first refueling outage all ten vacuum breakers were shimmed 1/16" open at the bottom of the disc. The bypass area associated with the shimming corresponded to 63% of the maximum allowable.[1] The results of this test are shown in Figure 3.7.1.

When a drywell-suppression chamber vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights at the remote test panels are designed to function as follows:

Full Closed	2 Green - On 2 Red - Off
Intermediate Position	2 Green - Off 2 Red - Off
Full Open	2 Green - Off 2 Red - On

The remote test panel consists of a push button to actuate the air cylinder for testing, two red lights,

Bases Continued:

and two green lights for each of the ten valves. There are four independent limit switches on each valve. The two switches controlling the green lights are adjusted to provide an indication of disc opening of less than 1/8" at the bottom of the disc. These switches are also used to activate the valve position alarm circuits. The two switches controlling the red lights are adjusted to provide indication of the disc very near the full open position.

The control room alarm circuits are redundant and fail safe. This assures that no simple failure will defeat alarming to the control room when a valve is open beyond allowable and when power to the switches fails. The alarm is needed to alert the operator that action must be taken to correct a malfunction or to investigate possible changes in valve position status, or both. If the alarm cannot be cleared due to the inability to establish indication of closure of one or more valves, additional testing is required. The alarm system allows the operator to make this evaluation on a timely basis. The frequency of the testing of the alarms is the same as that required for the position indication system.

Operability of a vacuum breaker valve and the four associated indicating light circuits shall be established by cycling the valve. The sequence of the indicating lights will be observed to be that previously described. If both green light circuits are inoperable, the valve shall be considered inoperable and a pressure test is required immediately and upon indication of subsequent operation. If both red light circuits are inoperable, the valve shall be considered inoperable, however, no pressure test is required if positive closure indication is present.

The 5% oxygen concentration minimizes the possibility of hydrogen combustion following a loss of coolant accident. Significant quantities of hydrogen could be generated if the core cooling systems failed to sufficiently cool the core. The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is more probable than the occurrence of the loss of coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration. The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least once a week the oxygen concentration will be determined as added assurance.

Bases Continued:

Calculations of the forces on the suppression chamber and its support system indicate that the dynamic loads during a postulated design basis loss of coolant accident are dependent on the drywell to suppression chamber differential pressure and the suppression chamber water volume. The specifications require that the conditions assumed in the stress analysis be met after allowing sufficient time to first inert containment and then to establish the differential pressure. Similar provisions are allowed for the purpose of de-inerting. Provisions are included for allowing special tests without a Technical Specification change, pending special review and authorization by the Nuclear Regulatory Commission. The drywell to suppression chamber differential pressure is checked each shift to assure that equipment failure has not affected the differential pressure. With a check each shift, there is an extremely low probability that a loss of coolant accident will occur simultaneously with the differential pressure being below the specified limit. Two monitors provide redundant readings for comparison (with no automatic action initiation). The provisions allowing one or both indicators out of service are consistent with the need for a redundant indicator and the frequency for checking the differential pressure, respectively.

B. Standby Gas Treatment System and C. Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required except, however, for initial fuel loading prior to initial power testing.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. One standby gas treatment system circuit is designed to automatically start upon containment isolation and to maintain the reactor building pressure at the design negative pressure so that all leakage should be in-leakage. Should one circuit fail to start, the redundant alternate standby gas treatment circuit is designed to start automatically. Each of the two circuits has 100% capacity. Only one of the two standby gas treatment system circuits is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is placed in a condition that does not require a standby gas treatment system.

Bases Continued:

While only a small amount of particulates are released from the primary containment as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates. Laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

D. Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Sections 5.2 and 7.2 of the FSAR.

Bases:

4.7 A. Primary Containment

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a weekly check of the temperature and volume is adequate to assure that adequate heat removal capability is present. For additional margin, these will be checked once per day.

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

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

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss of coolant accident. The peak drywell pressure would be about 41 psig, which would rapidly reduce to 25 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 25 psig within 10 seconds, equalizes with drywell pressure and thereafter rapidly decays with the drywell pressure decay. See Section 5.2.3 FSAR.

The design pressure of the drywell and absorption chamber is 56 psig. See Section 5.2.3 FSAR. The design leak rate is 0.5%/day at a pressure of 56 psig. As indicated above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

Bases Continued:

The design basis loss of coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5% day at 41 psig. The analysis showed that with this leak

Bases Continued:

B. Standby Gas Treatment System, and C. Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain the design negative pressure within the secondary containment provides an adequate test of the reactor building isolation valves and the standby gas treatment system. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system operational capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Standby gas treatment system inplace testing procedures will be established utilizing applicable sections of ANSI N510-1975 standard as a procedural guideline only. Redundant heaters in the standby gas treatment system room prevent moisture buildup on the adsorbent. If painting, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed as required for operational use. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52 Revision 1 (June 1976). The charcoal adsorber efficiency test procedures will allow for the removal of one representative sample cartridge. The sample will be at least two inches in diameter and a length equal to the thickness of the bed. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system will be replaced. High efficiency particulate filters are installed before and after the charcoal filters to prevent clogging of the carbon adsorbers and to minimize potential release of particulates to the environment. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by inplace testing with DOP as the testing medium. Any HEPA filters found defective will be replaced with filters qualified pursuant to regulatory guide position C.3.d of Regulatory Guide 1.52 Revision 1 (June 1976). Once per operating cycle demonstration of HEPA filter pressure drop, operability of inlet heaters at rated power, automatic initiation of each standby gas treatment system circuit, and leakage tests after maintenance or testing which could affect leakage, is necessary to assure system performance capability.

Bases Continued:

The containment is penetrated by a large number of small diameter instrument lines. A program for the periodic testing (see Specification 4.7.D) and examination of the valves in these lines has been developed and a report covering this program was submitted to the AEC on July 27, 1973.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.