10/30/78

Docket No. 50-263

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Mr. L. O. Mayer, Manager
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Mayer:

In response to your request for license amendment dated November 5, 1976 and as supplemented by submittals dated April 15, 1977 and August 29, 1977, the Commission has issued the enclosed Amendment No. 36 to Provisional Operating License No. DPR-22 for the Monticello Nuclear Generating Plant.

This amendment incorporates provisions into the facility Technical Specifications which establish limiting conditions for operation and surveillance requirements for suppression pool water level.

These requirements provide assurance that facility operation will be in accordance with the assumptions utilized in your facility's plant-unique analysis which was performed in conjunction with the Mark I Containment Short Term Program evaluation.

The enclosed license amendment reflects those changes to your original request for license amendment which have been agreed to in discussions with your staff. These changes have been made to provide consistent requirements for all Mark I containment facilities.

Copies of the related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

7811220138

Enclosures:
1. Amendment No. 36 to DPR-22
2. Safety Evaluation
3. Notice

cc w/enclosures:
see next page

~~SEE PREVIOUS YELLOW FOR CONCURRENCE~~

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NRC FORM 318 (9-76) NRCM 0240

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 36
License No. DPR-22

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The submittal by Northern States Power Company (the licensee) dated November 5, 1976 as supplemented April 15 and August 29, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-22 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A as revised ised through Amendment No. 36, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

7811220140

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 30, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 36

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Pages

139
140
147A
156
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157A
158
158A
158B
159
160
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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.

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.

e. The suppression chamber water volume shall be $\geq$68,000 and $\leq$77,970 cubic feet.

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

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).

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 checked once per day.

f. The suppression chamber water volume indicators shall be calibrated semi-annually.

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-

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.

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.

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.

Bases Continued:

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.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a suppression chamber water level corresponding to a downcomer submergence range of 4.54 to 5.62 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

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

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:

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

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 36 TO LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

I. INTRODUCTION

In conjunction with the Short Term Program (STP) evaluation of Boiling Water Reactor facilities with the Mark I containment system, the Northern States Power Company (the licensee) submitted a Plant Unique Analysis (PUA) for the Monticello Nuclear Generating Plant. This analysis was performed to confirm the structural and functional capability of the containment suppression chamber and attached piping, to withstand newly-identified suppression pool hydrodynamic loading conditions which had not been explicitly considered in the original design analysis for the plant. As part of the STP evaluation, specific loading conditions were developed for each Mark I facility, to account for the change in the magnitude of the loads due to plant-specific variations from the reference plant design for which the basic loading conditions were developed.

The results of the NRC staff's review of the hydrodynamic load definition techniques and the Mark I containment plant unique analyses are described in the "Mark I Containment Short Term Program Safety Evaluation Report," NUREG-0408, December 1977. As discussed in this report, the NRC staff has concluded that each Mark I containment system would maintain its integrity and functional capability in the unlikely event of a design basis loss-of-coolant accident (LOCA) and, therefore, that licensed Mark I BWR facilities can continue to operate safely, without undue risk to the health and safety of the public, during an interim period of approximately two years, while a methodical, comprehensive Long Term Program is conducted.

As discussed in Section III.C of NUREG-0408, of all of the plant parameters that were considered in the development of the hydrohynamic loads for the STP, only two parameters are expected to vary during normal plant operation; these are (1) the drywell-wetwell differential pressure; and (2) the suppression chamber (torus) water level. Subsequent to the submittal of the PUA, the licensee was requested to submit proposed Technical Specifications which assure that the allowable range of these of these two parameters during facility operation would be in accordance with the values utilized in the PUA.

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The licensee has been operating this facility with differential pressure control to enhance the safety margins of the containment structure since early 1976. This evaluation provides a more detailed basis for establishing the allowable range of drywell-wetwell differential pressure and torus water level, in order to quantify containment safety margins. This amendment incorporates these parameters into the Technical Specifications with the associated limiting conditions for operation and surveillance requirements.

By letters dated November 5, 1976 and April 15, 1977, as supplemented August 29, 1977, the licensee proposed changes to the facility Technical Specifications to incorporate limiting conditions for operation and surveillance requirements for differential pressure control and torus water level. Our evaluation of these proposed changes follows:

II. EVALUATION

The licensee has proposed certain Technical Specification requirements for the purpose of assuring that the normal plant operating conditions are within the envelope of conditions considered in their PUA. These Technical Specification changes establish (1) limiting condition for operation (LCOs) for drywell to torus differential pressure and torus water level, and (2) associated surveillance requirements. All other initial conditions utilized in the PUA are either presently included in the Technical Specifications or are configurational conditions which have been confirmed by the licensee and will not change during normal operation.

As indicated on Table III-1 of NUREG-0408, the licensee has submitted a request to allow operation without drywell/torus differential pressure control . In this document, Northern States Power Company (NSP) provided information on the peak stresses in the torus ring and shell, and the peak compressive forces in the torus support columns, without a differential pressure being maintained between the drywell and the wetwell. The peak stress intensities in the torus ring and shell, and the peak compressive forces in the torus support columns are presented for both the minimum and the maximum torus water levels. They are presented with and without the 33 percent increase in the hydrodynamically induced portion of these stresses and forces, as stipulated for the Short Term Program (STP). The justification for dropping the 33 percent increase is that the pool swell loads resulting from the recent GE one quarter scale tests are approximately 80 percent of the values reported in the Short Term Program Report. Pool swell, dead and seismic loadings are considered in the calculation of the stress intensities and loads and these are presented for the torus ring and shell, and torus support columns, respectively, along with the corresponding strength ratios and the comparisons with the Code allowables.

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[1]Removal of Drywell-Wetwell Differential Pressure Controls

Two additional modifications have been performed to the existing structures. The first modification consisted of the reinforcement of the torus support column to shell connections. These capacities have increased the code allowable loads from 765K to 940K, while the ultimate capacity was lowered from 3150K to 2820K. The second modification consisted of adding reinforcement to the vent header support columns to increase their code allowable capacities from 74K to 132K and their ultimate capacities from 276K to 413K.

With the reinforcing of the vent header support columns, the code allowable capacity is greater than the STP loading of 131K. In addition, the strength ratio is 0.32 which is less than the 0.50 ratio allowed for the STP.

For the case of dead and seismic loadings superimposed upon the pool swell loadings without the drywell-wetwell ΔP and increased by 33 percent, at the maximum torus water level (the case which results in the highest pool swell loads), the inside column loads, and the local primary plus the secondary stresses in the ring and shell meet the code allowables. All the strength ratios are less than the 0.50 permitted under the STP.

For the case of dead and seismic loadings superimposed upon the pool swell loadings without ΔP and not increased by 33 percent, at the maximum water level the code allowables are satisfied everywhere except for the loads on the outside column shell and pin connections, and for the local primary stresses in the torus ring. In these locations, code allowables are only exceeded by 14, 8 and 6 percent, respectively, which is within the error band of the conservatively estimated pool swell loading function. However, all strength ratios are less than 0.50.

Based upon the above information, we find that sufficient margins of safety against failure currently exist in the torus and torus support columns to withstand the effects of dead, seismic, and pool swell loadings without the drywell-wetwell pressure differential. Therefore, the staff concludes that the controls on the maintenance of the drywell-wetwell pressure differential at the Monticello Nuclear Generating Plant may be safely removed.

The torus water level is not expected to vary significantly during normal operation, unless certain systems connected to the suppression pool are activated. The torus water level would normally be monitored whenever such systems are in use. Therefore, we find that inclusion of periodic torus water level surveillance requirements in the Technical Specifications is not required.

We have reviewed the torus water level monitoring instrumentation system proposed by the licensee with regard to the number of available channels and the instrumentation accuracy. This type of instrumentation is typically calibrated at six-month intervals. To assure proper operation during such intervals, two monitoring channels for torus water level have been provided, such that a comparison of the readings will indicate when one of the channels is inoperative or drifting. The errors in the instrumentation are sufficiently small relative to the magnitude of the measurement (i.e., a maximum torus water level measurement error of 10% of the difference between the maximum and minimum torus water level) that they may be neglected, based on the expected load variation with torus water level.

III. ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

IV. CONCLUSION

The proposed Technical Specifications will provide the necessary assurance that the plant's operating conditions remain within the envelope of the conditions assumed in the Plant Unique Analysis (PUA) performed in conjunction with the Mark I Containment Short Term Program. The PUA supplements the facility's Final Safety Analysis Report (FSAR) in that it demonstrates the plant's capability to withstand the suppression pool hydrodynamic loads which were not explicitly considered in the FSAR. We therefore conclude that the proposed changes to the Technical Specifications are acceptable.

We further conclude, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 30, 1978