FEB 26 1975

DISTRIBUTION
Docket
NRC ~~AEC~~ PDR
Local PDR
ORB #2 Reading
Attorney, OGC
OI&E ~~RO~~ (3)
NDube
BJones (4)
~~RVollmer~~
JSaltzman
RMDiggs
BBuckley
DLZiemann
SKari
WOMiller
BScharf (15)
TJCarter
PCollins
SVarga
CHebron
RSchemel
ACRS (16)
OIS
HJMcAlduff
JRBuchanan
TBAbernathy

Docket No. 50-263

Northern States Power Company
ATTN: Mr. L. O. Mayer
Director of Nuclear
Support Services
414 Nicollet Mall
Minneapolis, Minnesota 55401

Gentlemen:

The Commission has issued the enclosed Amendment No. 8 to Provisional Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment includes Change No. 17 to the Technical Specifications, and is in response to your request dated July 10, 1973.

The amendment specifies new in-service inspection, surveillance and testing requirements for the torus-to-drywell vacuum breakers to increase performance reliability and provides added assurance that the breakers will operate as required under accident conditions.

Copies of our Safety Evaluation and the Federal Register Notice relating to this action also are enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Enclosures:
1. Amendment No. 8
   w/Change No. 17
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
See next page

o/P.2

Subject to:
(1) Add Dzugan to service list.
(2) Make additional finding indicated on attached ~~of SE in att letter to J. Gallo~~
K. Goller.
LB

OK to D. Ross 2/12/75
OK per Ross 2/12/75 - no SE attached 2/20/75

OFFICE ➤	L:ORB #2	L:ORB #2	L:ORB #2	OGC	L:AD/OR	
SURNAME ➤	RDiggs	BBuckley/tc	DLZiemann	S H Lewis	KRGoller	
DATE ➤	1/21/75	1/27/75	1/30/75	2/11/75	2/26/75	

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166 for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at The Environmental Conservation Library, Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A copy of items (2) and (3) may be obtained upor request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention, Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 10th day of April 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

OFFICE ➤						
SURNAME ➤						
DATE ➤						

FEB 26 1975

cc w/enclosures:
Arthur Renquist, Esquire
Vice President - Law
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Gerald Charnoff
Shaw, Pittman, Potts, Trowbridge
and Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

Howard J. Vogel, Esquire
Legal Counsel
2750 Dean Parkway
Minneapolis, Minnesota 55416

Steve Gadler, P. E.
2120 Carter Avenue
St. Paul, Minnesota 55108

Mr. Daniel L. Ficker
Assistant City Attorney
638 City Hall
St. Paul, Minnesota 55102

Sandra S. Gardebring
Special Assistant Attorney General
Counsel for Minnesota Pollution
Control Agency
1935 W. County Road B2
Roseville, Minnesota 55113

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, N. W.
Washington, D. C. 20036

The Environmental Conservation
Library
Minneapolis Public Library
300 Nicollet Mall
Minneapolis, Minnesota 55401

Ken Dzugan, Director
City of St. Paul Pollution
Control Services
100 East 10th Street
St. Paul, Minnesota 55108

Mr. D. S. Douglas, Auditor
Wright County Borad of Commissioners
Buffalo, Minnesota 55313

cc w/enclosures and copy of
NSP's filings dtd. 3/12/73,
7/10/73 and 9/17/73:
Warren R. Lawson, M. D.
Secretary & Executive Officer
State Department of Health
University Campus
Minneapolis, Minnesota 55440

Mr. Gary Williams
Federal Activities Branch
Environmental Protection Agency
1 N. Wacker Drive, Room 822
Chicago, Illinois 60606

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 8
License No. DPR-22

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

   A. The application for amendment by the Northern States Power Company (the licensee) dated July 10, 1973, 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. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

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

OFFICE➤						
SURNAME➤						
DATE➤						

"B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 17."

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Karl R. Goller
Karl R. Goller, Assistant Director
for Operating Reactors
Division of Reactor Licensing

Attachment:
Change No. 17 to the
Technical Specifications

Date of Issuance: FEB 26 1975

OFFICE➤						
SURNAME➤						
DATE➤						

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

ATTACHMENT TO LICENSE AMENDMENT NO. 8

CHANGE NO. 17 TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

The Technical Specifications contained in Appendix A, attached to Provisional Operating License No. DPR-22, are hereby changed by replacing pages 147, 153 and 158 with revised pages bearing the same numbers and additional pages 147A, 158A, 158B and 167A. Changed areas on the revised pages are reflected by marginal lines.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

3.0 LIMITING CONDITIONS FOR OPERATION

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

   a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be operable and positioned in the closed position as indicated by the position indication system, except during testing and except as specified in 3.7.A. 4.b and c below.

   b. Any drywell-suppression chamber vacuum breaker may be nonfully closed as indicated by the position indication and
17 alarm systems provided that drywell to suppression chamber differential pressure decay does not exceed that shown on Figure 3.7.1.

   c. Up to two drywell-suppression chamber vacuum breakers may be inoperable provided that: (1) the vacuum breakers are determined to be fully closed and at least one position alarm circuit is operable or (2) the vacuum breaker is secured in the closed position.

4.0 SURVEILLANCE REQUIREMENTS

4. Pressure Suppression Chamber-Drywell Vacuum Breakers

   a. Operability and full closure of the drywell-suppression chamber vacuum breakers shall be verified by performance of the following:

      (1) Monthly each operable drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle.

      (2) Once each operating fuel cycle,
17 drywell to suppression chamber leakag shall be demonstrated to be less than that equivalent to a one-inch diameter orifice and each vacuum breaker shall be visually inspected. (Containment access required)

      (3) Once each operating cycle, vacuum breaker position indication and alarm systems shall be calibrat and functionally tested. (Containment access required)

      (4) Once each operating cycle, the vacuum breakers shall be tested to determine that the force required to open each valve from fully closed to fully open does not exceed that equivalent to 0.5 psi acting on the suppression chamber face of the valve disc. (Containment access required)

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
17 circuits may be inoperable for a period not to exceed seven days provided that all vacuum breakers are operable.

5. Oxygen Concentration

a. After completion of startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 5% oxygen with nitrogen gas whenever the reactor coolant pressure is above 110 psig in the power operating condition, 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
17 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.

TABLE 3.7.1

PRIMARY CONTAINMENT ISOLATION

Isolation Group	Valve Identification	Number of Valves		Maximum Operating Time (Sec)	Normal Position
		Inboard	Outboard		
1	Main Steam Line Isolation	4	4	$3 \leq T \leq 5$	Open
1	Main Steam Line Drain	1	1	60	Closed
1	Recirculation Loop Sample Line	1	1	60	Closed
2	Drywell Floor Drain		2	60	Open
2	Drywell Equipment Drain		2	60	Open
2	Drywell Vent		2	60	Closed
2	Drywell Vent Bypass		1	60	Closed
2	Drywell Purge Inlet		2	60	Closed
2	Drywell and Suppression Chamber Air Makeup		1	60	Closed
2	Suppression Chamber Vent		2	60	Closed
2	Suppression Chamber Vent Bypass		1	60	Closed
2	Shutdown Cooling System	1	1	120	Closed

17

Bases Continued:

3.7 A. Primary Containment

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
17 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

[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

3/4 inch opening of any one valve or .08 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
17 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, 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.

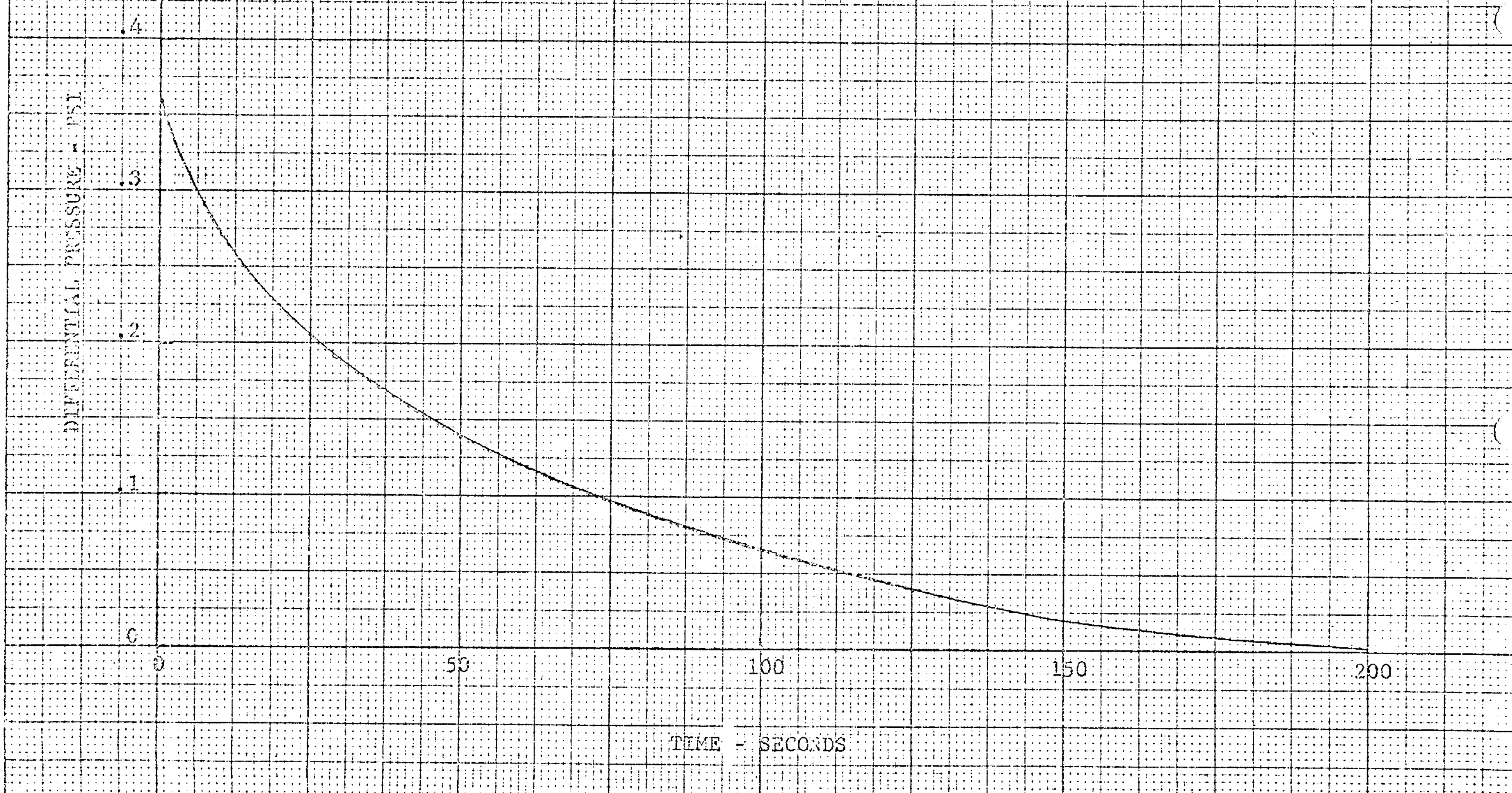
The control room alarm circuits are redundant and fail safe. This assures that no simple failure will defeat alarming to the control room when a valve is open beyond allowable and when power to the switches fails. The alarm is needed to alert the operator that action must be taken to correct a malfunction
17 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
(Continued on page 159)

FIGURE 3.7.1

DIFFERENTIAL PRESSURE DECAY BETWEEN THE DRYWELL AND WETWELL WITH A SHIM HOLDING EACH VACUUM BREAKER 1/16 INCH OPEN AT THE BOTTOM. TEST PERFORMED 5-10-73.



SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NO. 8 TO LICENSE DPR-22

(CHANGE NO. 17 TO THE TECHNICAL SPECIFICATIONS)

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

TORUS-TO-DRYWELL VACUUM BREAKER LEAKAGE

INTRODUCTION

A letter[1] from the Directorate of Licensing requested that the Northern States Power Company (NSP) provide torus-to-drywell vacuum breaker design and test information. The letter also requested: (1) a reevaluation of breaker performance, (2) additional equipment and systems and/or modification to the vacuum breakers, and (3) technical specification changes or additions related to limiting conditions of operation and surveillance. Northern States Power Company provided the requested information[2] within 60 days. Based on their response,[2] NSP later requested[3] a change to the Technical Specifications, Appendix A, of the Provisional Operating License, DPR-22 for the Monticello Nuclear Generating Plant. We met[4] with NSP representatives on September 6, 1973, to discuss details of the torus-to-drywell vacuum breaker valve position indicators and alarm circuits and to review the adequacy of proposed leak test requirements for the Technical Specifications. As a result of the meeting, NSP described an additional circuit modification[5] to the original proposal[2,3] such that torus-drywell vacuum breaker switches will cause an audible alarm as well as panel light indication in the control room if any of the 10 vacuum breakers open. Additional changes to the NSP proposal, which have evolved from discussions with NSP representatives, have been incorporated. These changes relate to surveillance and test requirements, specification 4.7.A.4a (3) and (4), 4.7.A.4c and limiting conditions for operation, specification 3.7.A.4(d).

OFFICE ➤						
SURNAME ➤						
DATE ➤						

EVALUATION

The drywell suppression chamber (torus) vacuum breakers protect the drywell from damage by a drywell negative pressure differential that could result with most of the non-condensible gas collected in the torus above the suppression pool after the water vapor in the drywell condenses following a design basis loss-of-coolant accident. Tests of vacuum breakers in some BWR plants revealed that some breakers were not in the fully closed position as they were designed to be during normal reactor operating conditions. A partly open vacuum breaker would permit steam to bypass the suppression pool following loss-of-coolant accidents causing higher-than-design containment pressure.

In response to our request(1), NSP presented the calculated drywell-to-torus leak rates that could be tolerated for primary system break areas as large as the design basis accident (DBA) break. The results showed the variation in allowable drywell-to-torus leakage with the primary system break area. For primary system breaks greater than 0.3 ft^2, the allowable drywell-to-torus leakage increases, i.e. the drywell-to-torus equivalent bypass increases from about 0.2 ft^2 to more than 1 ft^2. For primary system breaks less than 0.3 ft^2, the allowable drywell-to-torus leakage is less than 0.2 ft^2. We have reviewed the calculational method and assumptions used by NSP and have concluded that there is sufficient conservatism in the calculations. Therefore, the calculated drywell-to-torus bypass leakage equivalent to that from a 0.2 ft^2 (6 inch diameter) equivalent orifice is a justifiable limit for the entire range of core coolant breaks up to the design basis accident.

If the vacuum breaker opens during normal operation, drywell-to torus leak rate tests must be performed to show that the leakage is not excessive. NSP has demonstrated that all breaker discs wedged open 1/16 inch will result in less than the allowable leakage, i.e. leakage through 0.2 ft^2 equivalent orifice. We have concluded that the requirements for leak tests during reactor operation proposed by NSP are adequate to assure a 37% margin to the 0.2 ft^2 equivalent orifice limit value (maximum allowable leak for continuous operation is 63% of the calculated limit - 6" diameter opening). Figure 3.7.1 of the proposed technical specification change defines, therefore, an acceptable test limit for continued reactor operation. Moreover, at least once during each fuel cycle, while the reactor is shut down and the containment is accessible, it must be demonstrated that the combined drywell-to-torus leakage is less than the leakage through an equivalent 1-inch orifice (less than 3% of the calculated leakage limit). At the same time all vacuum breakers must be visually inspected to assure proper valve disc seating. These and other proposed surveillance requirements (Proposed Technical Specification 4.7.4) we have concluded, further reduce the probability that containment pressure will exceed design limits following any break in the core coolant piping systems.

We have reviewed the change in the vacuum breaker closure switch design. We agree that replacement of Snap Lock switches attached to the valve shaft with Micro Switches located on the bottom of the valve seat improves sensitivity in detecting valve disc movement from the closed position. Two Micro Switches have been installed on each valve and wired to separate valve position indicator panels in the reactor building and control room. These modifications permit verification of valve closure and satisfy requirements for system redundancy. The modifications including the alarm, which sounds if a valve disc leaves the seat, represent a substantial improvement in torus-to-drywell vacuum breaker performance reliability.

All of the proposed modifications have been installed and the systems are now operational. The proposed technical specifications as modified by NSP and the NRC staff will enhance reactor safety and should be approved.

Although tests have shown that leakage from the valve disc shaft is negligible with the teflon packing removed, we understand that NSP will repack one of the vacuum breaker shafts with new material during the scheduled refueling outage in January 1975. The remaining valve shafts may be repacked after sufficient operating time has elapsed to confirm the adequacy of the new packing.

CONCLUSION

Based on calculated drywell-to-torus leak rate limits provided by NSP for the spectrum of design basis core coolant breaks, we have concluded that the limit curve for leak tests during reactor operation is acceptable. We also have concluded that the leak rate test to be performed at the end of each operating cycle will reasonably assure that the torus-to drywell vacuum breakers are properly seated because measured leakage from the drywell to the torus must be less than 3% of the safety limit before returning to reactor operation, i.e., 97% margin to a safety limit. We agree with NSP that:

OFFICE ➤						
SURNAME ➤						
DATE ➤						

1. redundant audible alarms, as now provided at Monticello, assure that a valve opening more than 1/16 inch will be detected promptly during normal plant operation.

2. redundant indicator panels will show which valve has opened

3. surveillance and test requirements are adequate considering the vacuum breaker performance history, test results, and the recently completed vacuum breaker system modifications.

For these reasons we have concluded that the proposed changes, as modified with mutual consent, will enhance reactor safety by assuring that containment pressure following loss of core coolant accidents will not exceed the 62 psig design limit because of faulty torus-to-drywell vacuum breaker(s). The steam released during blowdown will condense in the suppression pool as described in the FSAR.

We have concluded, based on the considerations discussed above, that: (1) because the change 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 change 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, and 4) an environmental impact statement, negative declaration, or environmental impact appraisal are not necessary, since this ~~change~~ amendment will not ~~inc~~ change the types or increase the amounts of effluents from this facility, ~~nor increase the authorized power level.~~ per Myron Karman 3/12/75

~~Date:~~

FEB 26 1975

See later language in note to K. Goller fm. J. Gallo

REFERENCES

(1) Directorate of Licensing letter dated January 12, 1973. Request for design and test information related to torus-drywell vacuum breakers, plus evaluation of improvements and submittal of new technical specifications.

(2) NSP letter dated March 12, 1973 response to AEC letter dated January 12, 1973. "Report on Torus to Drywell Vacuum Breakers Tests and Modifications for Monticello Nuclear Generating Plant."

(3) NSP letter dated July 10, 1973. Request for changes to the Monticello Technical Specifications to improve torus-to-drywell vacuum breaker reliability.

(4) Directorate of Licensing Memo to Files dated September 11, 1973. Minutes of September 6, 1973 meeting with NSP representatives to discuss Monticello Torus-to-Drywell Vacuum Breakers.

(5) NSP letter dated September 17, 1973. "Further Vacuum Breaker Modifications."

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 8 to Provisional Operating License No. DPR-22 issued to the Northern States Power Company (the licensee) which revised the Technical Specifications for operation of the Monticello Nuclear Generating Plant located in Wright County, Minnesota. The amendment is effective as of its date of issuance.

The amendment incorporates new in-service inspection, surveillance, and testing requirements for the torus-to-drywell vacuum breakers to increase performance reliability and provide added assurance that the breakers will operate as required under accident conditions.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

For further details with respect to this action, see (1) the application for amendment dated July 10, 1973, and related filings by

OFFICE ➤						
SURNAME ➤						
DATE ➤						

the licensee dated March 12, 1973, and September 17, 1973, (2) Amendment No. 8 to License No. DPR-22, with Change No. 17, (3) the Commission's concurrently issued Safety Evaluation and (4) the Commission's letter to the licensee dated January 12, 1973, and memorandum to Files dated September 11, 1973. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at The Environmental Conservation Library, 300 Nicollet Mall, Minneapolis, Minnesota 55414. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention, Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 26th day of February, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann
Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

OFFICE ➤						
SURNAME ➤						
DATE ➤						

February 26, 1975

NOTE TO KARL GOLLER, ASSISTANT DIRECTOR
FOR OPERATING REACTORS, DRL

You will note that our concurrence on this package refers to adding language set forth in a memorandum from me to you. As discussed yesterday, my memorandum, which will be sent in the next day or so, will discuss the problem on a generic basis and I agree that this particular action may be released without adding any language.

Joseph Gallo /jls

Joseph Gallo
Chief Hearing Counsel

May be issued now w/out any addition per KR Goller 2/26/75.

Reba D.

2/26/75

2/1?/75

NOTE TO R. REIL AND K. GOLLER:

Stephen Lewis (ELD) has suggested that we add a general statement (regarding Part 51 consideration) to the conclusion paragraph of our SER on the Monticello torus-to-drywell vacuum breakers TS change. Perhaps we should be doing this in all SERs, but this is the first time that I know about. Should I just add it??? If I do, does this become a part of our future standard conclusion paragraph for SERs that do not require environmental reviews or negative statements??? S. Lewis and his boss (Fred Grey) are on travel (until Tuesday) so only Myron Karman is aware that we have not been putting this in SERs.

Reba

cc: Bart Buckley