MAY 14 1974

Docket No. 50-263

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

Gentlemen:

Your request dated November 19, 1973, as supplemented by filings dated December 14, 1973, January 15, 1974, February 8, 27 and 28, 1974, and April 1, 1974, requested authorization to operate your Monticello Nuclear Generating Plant using a partial loading of 8x8 fuel, including a fuel assembly containing segmented test rods, and also proposed changes to the Technical Specifications related to limiting conditions for operation associated with fuel densification for the 8x8 and 7x7 fuels.

The use of 8x8 fuel in reloads has been reviewed on a generic basis by the Licensing staff and the Advisory Committee on Reactor Safeguards (ACRS). The reports based on these reviews were transmitted to you by letters dated February 11 and 20, 1974. The staff Safety Evaluation for the use of 8x8 fuel assemblies in the Monticello facility was transmitted to you by our letter dated April 8, 1974. Based on these reviews and the analysis of abnormal core transients and the effects of fuel densification considered in the enclosed Safety Evaluation, we have concluded that the health and safety of the public will not be endangered by operation of the Monticello facility with the 8x8 fuel and with the associated proposed changes to the Technical Specifications. This conclusion is based on a reactor power level reduction to 95% of rated power beyond a 4200 MWd/T average fuel exposure during fuel cycle 3 if additional analysis has not been submitted and approved prior to reaching 4200 MWd/T average fuel exposure.

Accordingly, Amendment No. 3 to Provisional Operating License No. DPR-22 with Change No. 14 to the Technical Specifications is enclosed authorizing you to operate the Monticello facility using 8x8 fuel including one fuel assembly containing segmented test rods and changing the limiting conditions for operations associated with fuel densification for the 8x8 and 7x7 fuels.

A copy of the Atomic Safety and Licensing Board's "Memorandum and Order Ruling on the Petition for Leave to Intervene" dated April 30, 1974, also is enclosed for your information.

The above Amendment No. 3 to License No. DPR-22 with Change No. 14 to the Technical Specifications also includes most of the changes you requested January 23, 1974, and March 1, 1974, as supplemented by your filings dated March 8 and 19 and April 10 and 26, 1974, relating to pressure relief, control rod scram times, standby gas treatment system and reactor vessel temperature measurements. A discussion of these changes and our findings related thereto, and a discussion of those items not approved follows.

By letter dated April 10, 1974, you requested changes to the Technical Specifications to allow operation of fuel cycle 3 without dependence on the prompt relief trip system (PRT) or reduction in calculational conservatisms. Our review of the PRT system proposed by your letter dated January 23, 1974, is in progress. At this time we have not determined that the PRT system can be made operational without affecting the existing engineered safety systems or the reactor protection system.

According to your April 10, 1974 proposal and the analytical results that were included, core design and plant safety margins are adequate to permit normal plant operations for 4200 MWd/T fuel depletion during fuel cycle 3. Thereafter a power reduction would be necessary unless the PRT system has been approved and placed in service. Approval of the PRT will require amendment of the license for the Monticello Nuclear Facility.

We have reviewed your April 10, 1974 reanalysis for operation with fuel cycle 3 without the PRT system and the technical specification changes that you have selected from earlier submittals dated February 27, 28 and March 1, 1974, when it was anticipated that the PRT system would be approved for the initial reactor operation during cycle 3. We have concluded that with four new safety relief valves installed in place of the four spring-loaded safety valves that have been removed (in addition to the four initially installed safety relief valves), there are no significant safety considerations and safety margins during the first 4200 MWd/T of cycle 3 fuel exposure and, thereafter as long as MCHFRs are greater than the technical specification limit of 1, are acceptable. Because we have not completed our evaluation of your analytical methods and in accordance with telephone conversations with your representatives, we have changed the technical specifications to require that a minimum of seven safety/relief valves be operative during reactor operation instead of the six that you had proposed. The reason for this change is that seven safety/relief valves set at a nominal trip pressure of 1080 psig provide a total steam relief flow capacity equivalent to the four safety/relief valves and four spring-loaded safety valves that were installed prior to the spring 1974 plant outage. On an interim basis or until we have completed our evaluations related to your calculational methods, to prevent overstressing in the primary system, the relief capacity equal to or greater than it has been in the past is being maintained.

MAY 14 1974

Those proposed changes listed as items 4, 5, 16, 18, 19a, 19b, 20a, 20b, and 24 of your March 1, 1974 letter also have been approved.

Based on our review of your proposed Technical Specifications, as modified, and the analysis that you have provided, we have determined that there are no significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation of the plant in the manner proposed, as modified. A copy of our related Safety Evaluation is enclosed.

A copy of the notice relating to the issuance of Amendment No. 3 that we are forwarding to the Office of the Federal Register for publication is enclosed.

Sincerely,

Orig. [illegible]
Dennis [illegible]

for Karl R. Goller
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures:
1. Amendment No. 3 to DPR-22
2. ASLB Memorandum and Order dtd 4/30/74
3. Safety Evaluation
4. Federal Register Notice

cc w/enclosures: See next page

Applicant copy sent 5-15-74.

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DATE ➤	5/14/74	5/14/74	5/14/74	5/14/74	5/14/74	5/14/74

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

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

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

Howard J. Vogel, Esquire
Knittle & Vogel
1154 East Grain Exchange Building
412 South 4th Street
Minneapolis, Minnesota 55414

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

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

Sandra S. Gardebring, Esquire
Minnesota Pollution Control Agency
1935 W. County Road B2
Roseville, Minnesota 55113

Ken Dzugan
Minnesota Pollution Control Agency
1935 W. County Road B2
Roseville, Minnesota 55113

Mr. B. S. Douglas, Auditor
Wright County Board of Commissioners
Buffalo, Minnesota 55313

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

Environmental Library of Minnesota
1222 S. E. 4th Street
Minneapolis, Minnesota 55414

cc w/encls of NSP filings dtd.
1/15/74, 1/23/74, 2/8, 27, 28/74,
3/1, 8, 19/74, and 4/1, 10, 26/74:
Mr. Gary Williams
Federal Activities Branch
Environmental Protection Agency
1 N. Wacker Drive, Room 822
Chicago, Illinois 60606

Warren R. Lawson, M. D.
Secretary & Executive Officer
State Department of Health
717 Delaware Street, S. E.
Minneapolis, Minnesota 55440

OFFICE ▶						
SURNAME ▶						
DATE ▶						

ATTACHMENT TO LICENSE AMENDMENT NO. 3

CHANGE NO. 14 TO APPENDIX A TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-22

The following pages of Appendix A to Provisional Operating License have been revised to incorporate the changes related to safety valve modifications, control rod scram times, standby gas treatment system, 8 x 8 reload fuel, and fuel densification. Except as otherwise indicated, the enclosed revised pages supersede pages bearing the same number. The revised pages have marginal lines indicating where the changes appear.

Pages:
6
7
10
12
14
16
19
21
23
79
108A
108B - Replacing one unnumbered page issued 8/24/73 (Change 9)
108C - Addition
112
113A
113B } Replacing four of the unnumbered pages issued 8/24/73 (Ch. 9)
115
118
119
120
134
148
149
166
190

OFFICE						
SURNAME						
DATE						

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 3
License No. DPR-22

1. The Atomic Energy Commission (the Commission) has found that:

   A. The applications for amendment by the Northern States Power Company (the licensee) dated November 19, 1973, January 23, 1974, and March 1, 1974, as supplemented, comply 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 license, 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;

   E. The request for a hearing and petition for leave to intervene (by the Minnesota Pollution Control Agency) on the proposed action of those items relating to operation with 8 x 8 fuels and limiting conditions for operation associated with fuel densification for 8 x 8 and 7 x 7 fuels has been withdrawn and the proceeding dismissed based upon the consolidation of these issues with the licensing proceeding involving the conversion of the provisional operating license of the Monticello facility to a full term license (see: Atomic Safety and Licensing Board's Memorandum and Order Ruling on Petition for Leave to Intervene dated April 30, 1974), and

   F. Prior public notice of those items relating to pressure relief, control rod scram times, standby gas treatment, and reactor vessel temperature measurements is not required since they do not involve a significant hazards consideration.

2. Accordingly, paragraph 3.B of Facility License No. DPR-22 is hereby amended to read as follows:

"B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A attached to Facility Operating License No. DPR-22 are revised as indicated in the attachment to this license amendment. The Technical Specifications, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised."

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

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
Dennis L. Ziemann

for Karl R. Goller
Assistant Director
for Operating Reactors
Directorate of Licensing

Attachment:
Change No. 14 to Appendix A
Technical Specifications

Date of Issuance: MAY 14 1974

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DATE ➤						

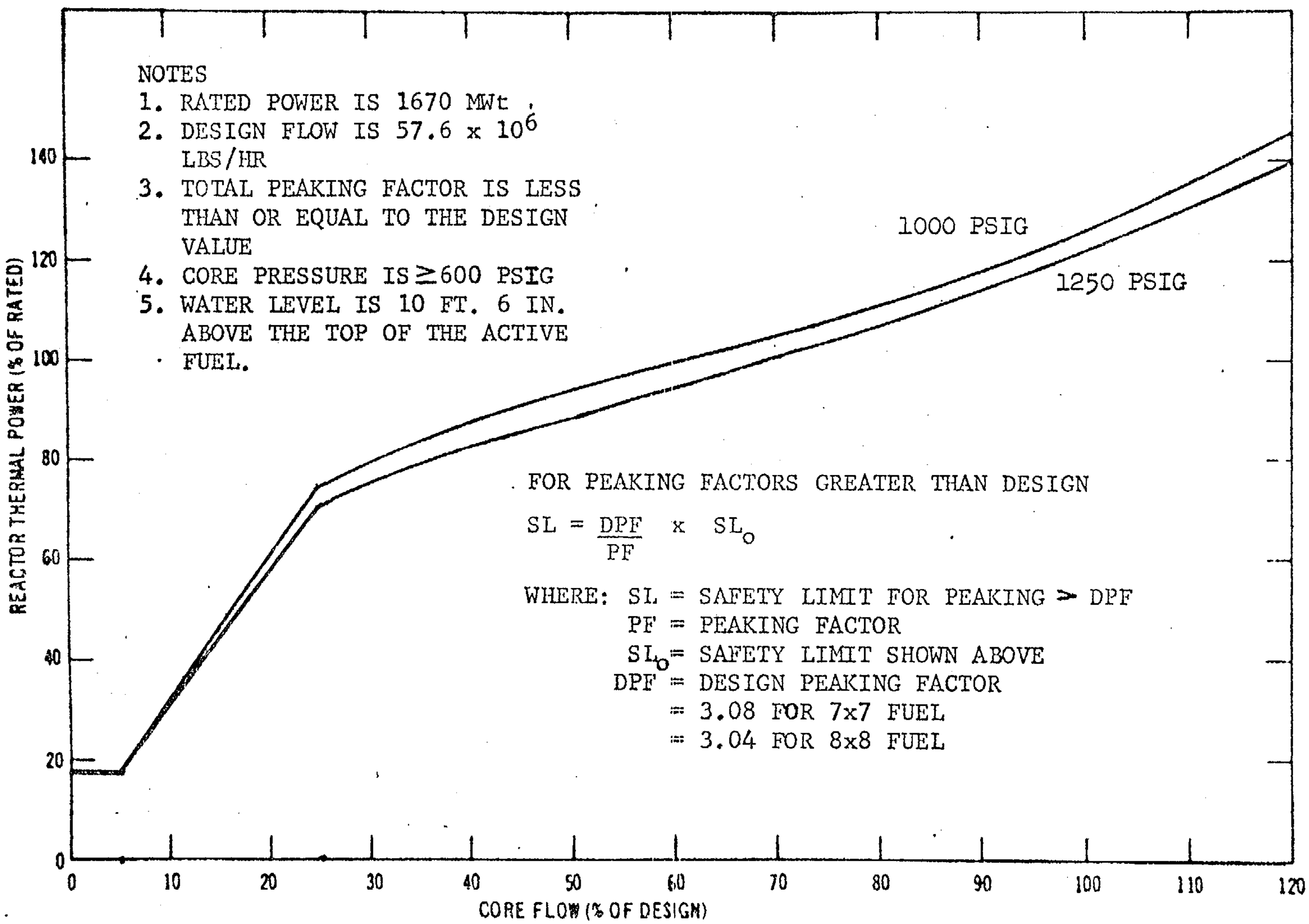
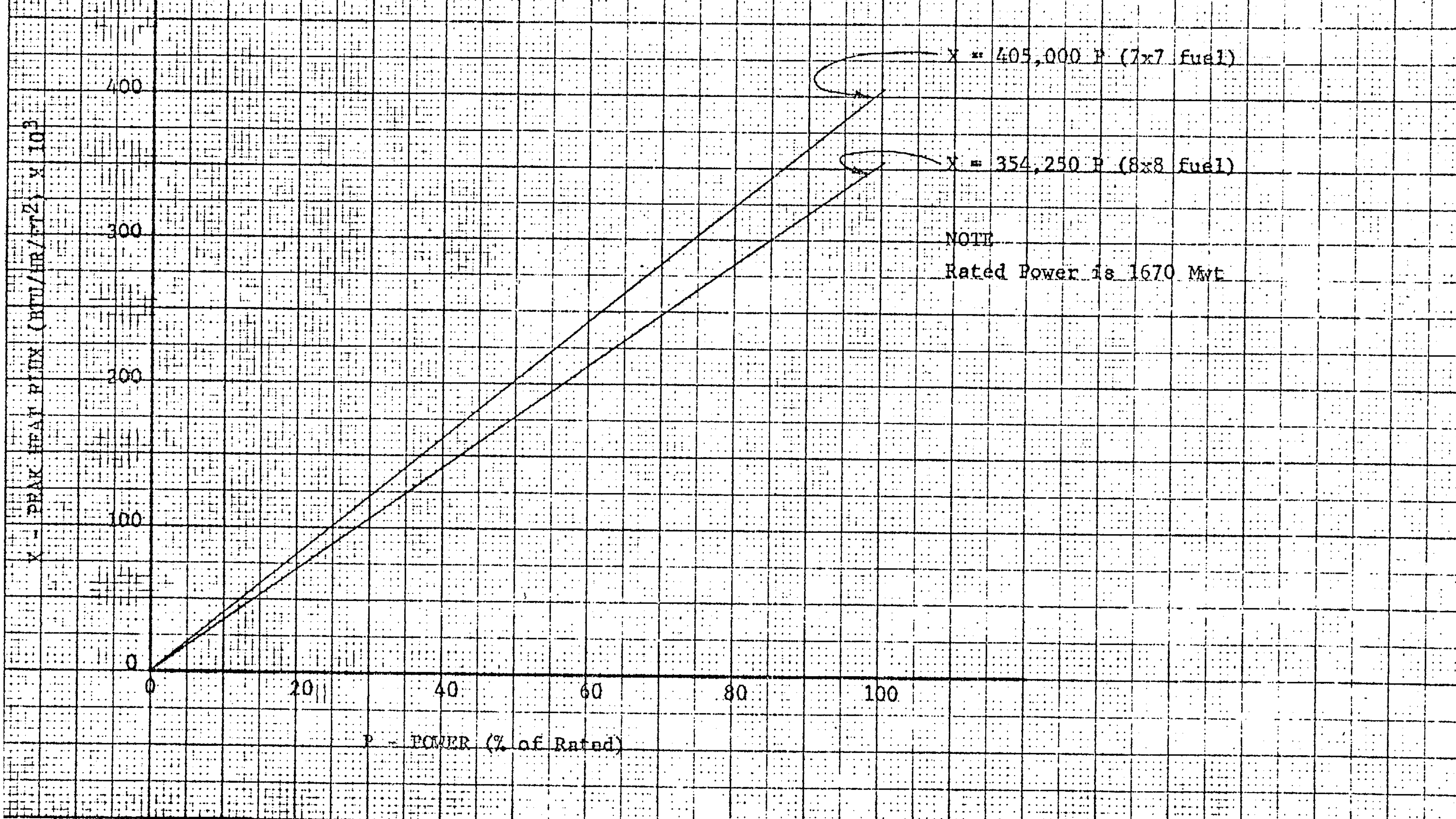


FIGURE 2.1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT

FIGURE 2.3.2

RELATIONSHIP BETWEEN PEAK HEAT FLUX AND POWER
FOR PEAKING FACTORS OF 3.08 (7x7 fuel)
AND 3.04 (8x8 fuel)



Bases Continued:

2.1 The design basis critical heat flux is based on an interrelationship of reactor coolant flow and steam quality. Steam quality is determined by reactor power, pressure, and coolant inlet enthalpy, which in turn, is a function of feedwater temperature and to a lesser degree reactor water level. This correlation is based upon experimental data taken over the pressure range of interest in a BWR, and the correlation line was very conservatively drawn below all the available data. Since the correlation line was drawn below the data, there is a very high probability that operation at the calculated safety limit would not result in a critical heat flux occurrence. In addition, if a critical heat flux were to occur, clad perforation would not necessarily be expected. Cladding temperature would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Monticello operated above the critical heat flux for significant period of time (30 minutes) without clad perforation. (1)

Curves are presented for two different pressures in Figure 2.1.1. The upper curve is based on a nominal operating pressure 1000 psig. The lower curve is based on a pressure 1250 psig. In no case is reactor pressure ever expected to exceed 1250 psig, and therefore, the curves will cover all operating conditions with interpolation. If reactor pressure should ever exceed 1250 psig during power operation, it would be assumed that the safety limit has been violated. For pressures between 600 psig, which is the lowest pressure used in the critical heat flux data, and 1000 psig, the upper curve is applicable with increased margin.

The power shape assumed in the calculation of these curves was based on design limits and results in a total peaking factor of 3.08 for 7x7 fuel and 3.04 for 8x8 fuel. For any peaking of smaller magnitude, the curves are conservative. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core Local Power Range Monitor (LPRM) System. To maintain applicability of the safety limit curve, the safety limit will be lowered according to the equation given in Figure 2.1.1 in the rare event of power operation with a total peaking factor in excess of the design value.

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(1) T. Sorlie, et. al. - "Experiences with Operating BWR Fuel Rods above the Critical Heat Flux" - Nucleonics, Vol. 23, No. 4, April, 1965.

Bases Continued:

2.1 During transient operation, the heat flux would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail (4,5,6,7). In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times of each control rod are checked each refueling outage to assure the insertion times are adequate. Exceeding a neutron flux scram setting and a delay in the control rod action to reduce neutron flux to less than the scram setting within 0.95 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed anytime a neutron flux scram setting of the APRM's is exceeded for longer than 0.95 seconds.

Analysis within the nominal uncertainty range of all appropriate significant parameters, show that if the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 0.95 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected.

The computer provided with Monticello has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram set point is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 2.1.C.2 will be relied on to determine if a safety limit has been violated.

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

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(4) FSAR Volume I, Section III-2.2.3
(5) FSAR Volume III, Sections XIV-5
(6) Supplement on Transient Analyses submitted by NSP to the AEC February 13, 1973
(7) Letter from NSP to the AEC, "Planned Reactor Operation from 2,000 MWD/T to end of cycle 2", dated August 21, 1973

Bases Continued:

2.3 A. Neutron Flux Scram - The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in per cent power. Since fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during transients induced by disturbances and with an APRM scram setting as shown in Figure 2.3.1, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analysis reported in the FSAR demonstrates that, even with a fixed 120% scram trip setting, none of the postulated transients result in violation of the fuel safety limit and there is a substantial margin from fuel damage. Therefore, use of a flow-biased scram provides additional margin.

An increase in the APRM scram setting to greater than that shown in Figure 2.3.1, would decrease the margin present before the thermal hydraulic safety limit is reached. The APRM scram setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. A reduction in this operating margin would increase the frequency of spurious scrams which have an adverse affect on reactor safety because of unnecessary thermal stress which it causes. Thus, the former 120% APRM setting was selected because it provides adequate margin from the thermal-hydraulic safety limit, yet allows operating margin which minimizes the possibility of unnecessary scrams. Therefore, it is intended to ultimately replace (with prior AEC approval) the automatic flow referenced scram with a fixed 120 percent scram setting, providing that initial power operation confirms the nuclear behavior characteristics used in these transient analysis.

The thermal hydraulic safety limit of Specification 2.1 was based on a total peaking factor of 3.08 for 7x7 fuel and 3.04 for 8x8 fuel. A factor has been included on Figure 2.1.1 to adjust the safety limit in the event peaking factor exceeds the design value. Likewise, the scram setting should also be adjusted to assure MCHFR does not become less than 1.0 in this degraded situation. This has been accomplished by use of Figure 2.3.2. If the combination of power and heat flux is greater than shown by the curve; i.e., a peaking factor greater than the design value exists, the APRM scram setting is adjusted downward by the formula given in the specification. The scram setting as given by the equation will prevent MCHFR from becoming less than 1.0 for the given heat flux condition for the worst expected transients. If the APRM scram setting should require a change due to an abnormal peaking condition, it will be done by increasing the APRM gain and thus reducing the slope and intercept point of the flow - biased scram curve by the reciprocal of the APRM gain change.

Bases Continued:

2.3 the worst case MCHFR during steady state operation is at 110% of rated power. Peaking factors as specified in Section 3.2 of the FSAR were considered. The total peaking factor was 3.08 for 7x7 fuel and 3.04 for 8x8 fuel. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the in-core LPRM system. As with the APRM scram setting, the APRM rod block setting is adjusted downward if peaking factors greater than the design value exist. This assures a rod block will occur before MCHFR becomes less than 1.0 even for this degraded case. The rod block setting is changed by increasing the APRM gain and thus reducing the slope and intercept point of the flow-biased rod block curve by the reciprocal of the APRM gain change.

The operator will set the APRM rod block trip settings no greater than that shown in Figure 2.3.1. However, the actual set point can be as much as 3% greater than that shown on Figure 2.3.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on Page 18.

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

The operator will set the low water level trip setting no lower than 10'6" above the top of the active fuel. However, the actual set point can be as much as 6 inches lower due to the deviations discussed on Page 18.

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

2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

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

Specification:

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

2.4 REACTOR COOLANT SYSTEM

Applicability:

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

Objective:

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

Specification:

A. Reactor Coolant High Pressure Scram shall be $\leq$ 1075 psig.

B. Reactor Coolant System Safety/Relief Valves shall be set as follows:

8 valves at $\leq$ 1080 psig.

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Percent of Rod Length Inserted	Seconds
5	0.398
20	0.954
50	2.120
90	3.80

C. Scram Insertion Times

During each operation cycle, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished during reactor power operation, the measured scram insertion times shall be extrapolated to the reactor power operation condition utilizing previously determined correlations.

3.0 LIMITING CONDITIONS FOR OPERATION

I. Recirculation System

1. Except as specified in 3.5.1.2 below, whenever irradiated fuel is in the reactor, with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be operable.

2. The recirculation system cross tie valve interlocks may be inoperable if at least one cross tie valve is maintained fully closed.

J. Average Planar LHGR

During steady state power operation, the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5.1.

4.0 SURVEILLANCE REQUIREMENTS

I. Recirculation System

1. Once per month, when irriated fuel is in the reactor with reactor coolant temperature great than 212°F and both reactor recirculation pumps operating, the recirculation system cros tie valve interlocks shall be demonstrated to be operable by verifying that the cros tie valves cannot be opened using the normal contr switch.

2. When a recirculation system cross tie valve interlock is inoperable, the position of at least one fully closed cross tie valve shall be recorded daily.

J. Average Planar LHGR

Daily during power operation, the average planar LHGR shall be checked.

K. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_{d} \left[1 - \left(\frac{\Delta P}{P}\right)_{max} \left(\frac{L}{LT}\right)\right]$$

$LHGR_d$ = Design LHGR

= 17.5 kw/ft for 7x7 fuel

= 13.4 kw/ft for 8x8 fuel

$\left(\frac{\Delta P}{P}\right)_{max}$ = Maximum power spiking penalty

= 0.033 for 7x7 fuel

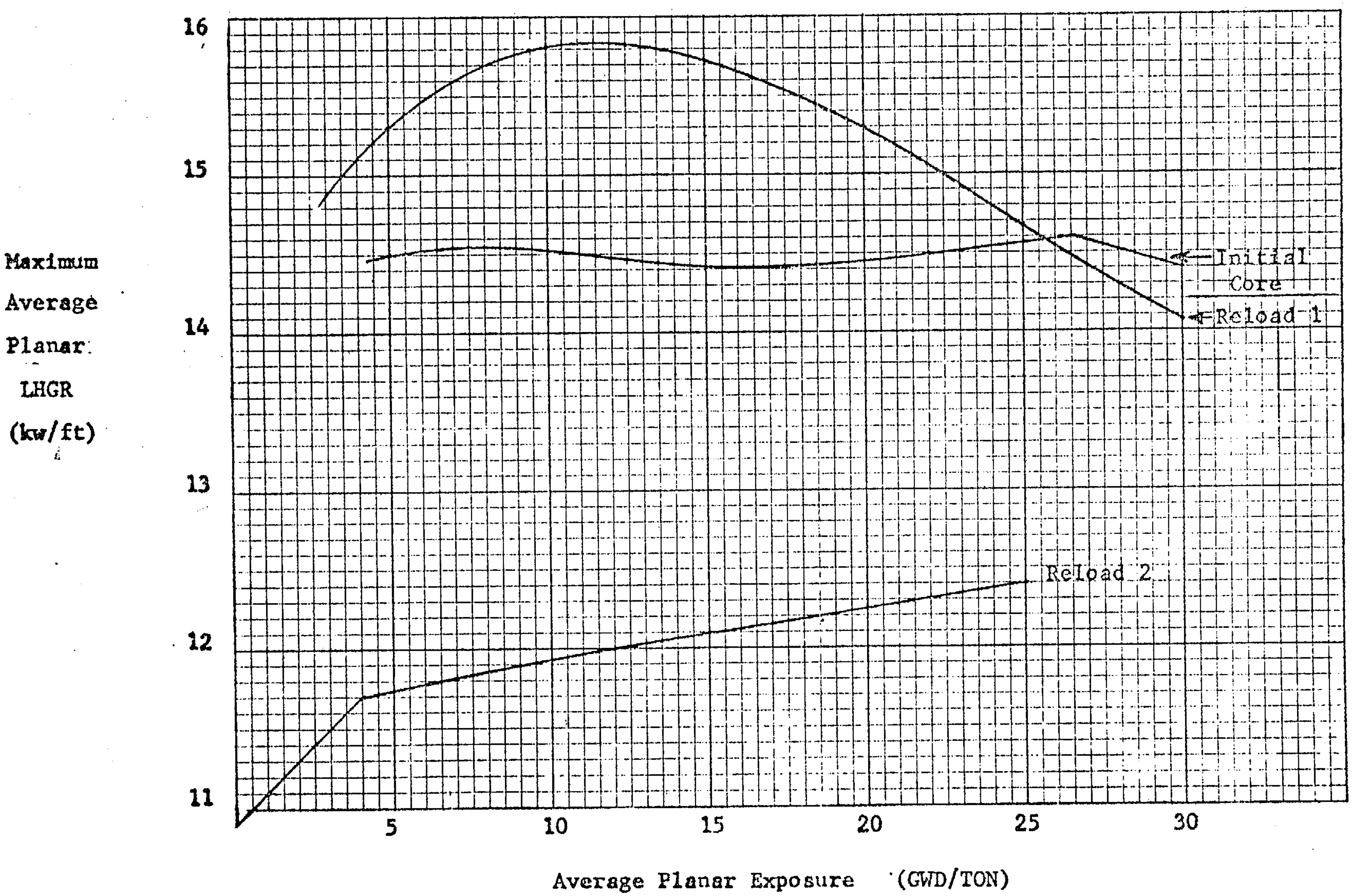
= 0.024 for 8x8 fuel

LT = Total core length = 12 ft

L = Axial position above bottom core

K. Local LHGR

Daily during reactor power operation, the local LHGR shall be checked.



Maximum
Average
Planar
LHGR
(kw/ft)
16
15
14
13
12
11
5
10
15
20
25
30
Initial Core
Reload 1
Reload 2
Average Planar Exposure (GWD/TON)

Figure 3.5.1

Bases Continued:

margin, the RCIC system (a non-safeguard system) has been required to be operable during this time, since the RCIC system is capable of supplying significant water makeup to the reactor (400 gpm).

E. Automatic Pressure Relief

The relief valves of the automatic pressure relief subsystem are a backup to the HPCI subsystem. They enable the core spray system or LPCI to provide protection against the small pipe break in the event of HPCI failure, by depressurizing the reactor vessel rapidly enough to actuate the core sprays or LPCI. Either of the two core spray systems or LPCI provide sufficient flow of coolant to limit fuel clad temperatures to well below clad melt and to assure that core geometry remains intact. Three safety/relief valves are included in the automatic pressure relief system. Of these three, only two are required to provide sufficient capacity for the automatic pressure reliefsystem. See section 4.4 and 6.2.5.3 FSAR.

F. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. The pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therefore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

Bases Continued 3.5:

J. Average Planar LHGR

This Specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2300^{o}F limit specified in the Interim Acceptance Criteria (IAC) issued in June 1971 considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm$ 20^{o}F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are below the IAC limit.

The maximum average planar LHGR curves shown in Figure 3.5.1 were calculated for the various Monticello fuel types in the manner discussed in Section 4.3 of General Electric topical report, "GECAP-III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods", NEDO-20181, Revision 1, November 1973. These curves show the composite limitation based on the design LHGR of the fuel and the peak cladding temperature in the event of a LOCA. Calculations based on the AEC "Modified GE Model for Fuel Densification" attached to a December 5, 1973 letter from D J Skovholt (USAEC) to L O Mayer (NSP).

The possible effects of fuel pellet densification were: (1)creep collapse of the caldding due to axial gap formation; (2)increase in the LHGR because of pellet column shortening; (3)power spikes due to axial gap formation; and (4)changes in stored energy due to increased radial gap size. Calculations show that clad collapse is conservatively predicted not to occur currently or prior to September 1974. Therefore, clad collapse is not considered in the analyses. Since axial thermal expansion of the fuel pellets is greater than axial shrinkage due to densification, the analyses of peak clad temperature do not consider any change in LHGR due to pellet column shortening. Although, the formation of axial gaps might produce a local power spike at one location on any one rod in a fuel assembly, the increase in local power density would be less than 2% at the axial midplane. Since small local variations in power distribution have a small effect on peak clad temperature, power spikes were not considered in the analysis of loss-of-coolant accidents. Changes in radial gap size affect the peak clad temperature by their effect on pellet clad thermal conductance and fuel pellet stored energy. The pellet-clad thermal conductance assumed for each rod is dependent on the steady state operating linear heat generation rate and gap size. As discussed in NEDO-20181, Revision 1, the gap size was calculated with the assumption that the pellet densified from its measured value to 96.5% of theoretical density.

The curves used to determine pellet-clad thermal conductance as a function of linear heat generation are based on experimental data and predict with a 95% confidence that 90% of the population exceed the predictions.

K. Local LHGR

This Specification assures that the linear heat generation rate in any rod is less than the design linear heat generation even if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis reported in NEDO-20181, Revision 1, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

3.0 LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr. when averaged over a one-hour period.

2. The pump in an idle recirculation loop shall not be started unless the temperature of the coolant within the idle recirculation loop is within 50°F of the reactor coolant temperature.

4.0 SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal Limitations

1. During heatups and cooldowns recirculation loops A and B temperatures shall be permanently recorded at 15 minutes intervals.

2. The temperatures listed in 4.6.A.1 shall be permanently recorded subsequent to a heatup or cooldown at 15 minute intervals until three consecutive readings are within 5 degrees of each other.

3.0 LIMITING CONDITIONS FOR OPERATION

4. If Specification 3.6.C.1, 3.6.C.2, and 3.6.C.3 are not met, normal orderly shutdown shall be initiated.

D. Coolant Leakage

Any time irradiated fuel is in the reactor vessel, and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met, initiate an orderly shutdown and have the reactor placed in the cold shutdown condition within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

(b) When the continuous conductivity monitor is inoperable, a reactor coolant sample should be taken at least once per shift and analyzed for conductivity and chloride ion content.

D. Coolant Leakage

Reactor coolant system leakage into the drywell shall be checked and recorded at least once per day.

3.0 LIMITING CONDITIONS FOR OPERATION

E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F.

   a. The safety valve function (self-actuation) of seven safety/relief valves shall be operable.

   b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.

4.0 SURVEILLANCE REQUIREMENTS

E. Safety/Relief Valves

1. a. A minimum of six safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. The nominal setpoint of all operational safety/relief valves shall be $\leq$ 1080 psig.

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

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

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

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

G. Jet Pumps

Whenever the reactor is in the Startup or Run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the plant shall be placed in a cold shutdown condition within 24 hours.

F. Structural Integrity

The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the AEC.

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the Startup or Run modes, jet pump operability shall be checked daily by verifying that all the following conditions do not occur simultaneously:

1. The two recirculation loop flows are unbalanced by 15% or more when the recirculation pumps are operating at the same speed.
2. The indicated value of core flow rate is 10% or more less than the value derived from loop flow measurements.

D. Coolant Leakage

The former 15 gpm limit for leaks from unidentified sources was established assuming such leakage was coming from the primary system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin. A leakage of 5 gpm is detectable and measurable. The 24 hour period allowed for determination of leakage is also based on the low probability of the crack propagating.

The capacity of the drywell sump pumps is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

An annual report will be prepared and submitted to the AEC summarizing the primary coolant to drywell leakage measurements. Other techniques for detecting leaks and the applicability of these techniques to the Monticello Plant will be the subject of continued study.

E. Safety/Relief Valves

Testing of all safety/relief valves each refueling outage ensures that any valve deterioration is detected. A tolerance value of 1% for safety/relief valve setpoints is specified in Section III of the ASME Boiler and Pressure Vessel Code. Analyses have been performed with all valves assumed set 1% higher (1080 psig + 1%) than the nominal setpoint; the 1375 psig code limit is not exceeded in any case.

The safety/relief valves are used to limit reactor vessel overpressure and fuel thermal duty.

The required safety/relief valve steam flow capacity is determined by analyzing the transient accompanying the mainsteam flow stoppage resulting from a postulated MSIV Closure from a power of 1670 MW_t. The analysis assumes a multiple-failure wherein direct scram (valve position) is neglected. Scram is assumed to be from indirect means (high flux). In this event, the safety/relief valve capacity is assumed to be 71% of the full power steam generation rate.

3.0 LIMITING CONDITIONS FOR OPERATION

6. If the specifications of 3.7.A cannot be met, the reactor shall be placed in a cold shutdown condition within 24 hours.

B. Standby Gas Treatment System

1. Except as specified in 3.7.B.3 below, both circuits of the standby gas treatment system shall be operable at all times when secondary containment integrity is required.

4.0 SURVEILLANCE REQUIREMENTS

B. Standby Gas Treatment System

1. Standby gas treatment system surveillance shall be performed as indicated below:

   a. At least once per operating cycle it shall be demonstrated that:

      (1) Pressure drop across the combined high-efficiency and charcoal filters is less than 7.0 inches of water, and

      (2) Inlet heater output is at least 15 kw.

   b. During each refueling outage prior to refueling, whenever a filter is changed, whenever work is performed that could affect filter systems efficiency, and at intervals not to exceed six months between refueling outages, it shall be demonstrated that:

      (1) The removal efficiency of the installed particulate filters for particles having a mean diameter of 0.7 microns shall be

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
	equal to or greater than 99% based on an in-place dioctyl phthalate (DOP) test. (2) The removal efficiency of the charcoal filters is not less than 99% for freon based on a freon test. c. At least once each five years removable charcoal cartridges shall be removed and adsorption shall be demonstrated. d. At least once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.
2. From and after the date that one circuit of the standby gas treatment system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such circuit is sooner made operable, provided that during such seven days all active components of the other standby gas treatment circuit including its emergency power source shall be operable. 3. If this condition cannot be met, procedures shall be initiated immediately to establish the conditions listed in 3.7.C.1. (a) through (d), and compliance shall be completed within 24 hours thereafter.	2. When one circuit of the standby gas treatment system becomes inoperable, the operable circuit including its emergency power source shall be demonstrated to be operable immediately. The operable circuit of the Standby Gas Treatment System shall be demonstrated to be operable daily thereafter.

Bases Continued:

4.7 High efficiency particulate filters are installed before and after the charcoal filters to minimize potential release of particulates to the environment and to prevent clogging of the iodine filters. 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 in-place testing with DOP as testing medium. Individual filter units will be tested and certified to have a removal efficiency of equal to or greater than 99% for particles having a mean diameter of 0.3 microns at the time of purchase.

The test interval for filter efficiency was selected to minimize plugging of the filters. In addition, retention capacity in terms of microcuries of iodine per gram of charcoal will be demonstrated. This will be done by removing small test cartridges of the same charcoal filter material. These cartridges complement the charcoal filter system and will be available for withdrawal and testing. These tests will normally be performed every five years unless filter efficiency seriously deteriorates. Since shelf lives greater than five years have been demonstrated, the test interval is reasonable.

D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system whose failure could result in uncovering the reactor core are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein are adequate to prevent loss of more coolant from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, this isolation valve closure time is sufficient to prevent uncovering the core.

In order to assure that the doses that may result from a steam line break do not exceed the 10 CFR 100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses suggest that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds. However, for added margin the Technical Specifications require a valve closure time of not greater than 5 seconds.

The primary containment isolation valves are highly reliable, have low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. Reference Section 5.2.2.4.3 and Table 5-2-3 FSAR. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a more reliable system.

5.0 DESIGN FEATURES

5.1 Site

A. The reactor center line is located at approximately 850,810 feet North and 2,038,920 feet East as determined on the Minnesota State Grid, South Zone. The nearest site boundary is approximately 1630 feet S 30° W of the reactor center line and the exclusion area is defined by the minimum fenced area shown in FSAR Figure 2.2.2a. Due to the prevailing wind pattern, the direction of maximum integrated dosage is SSE. The southern property line follows the northern boundary of the right-of-way for the Burlington Northern Railway.

5.2 Reactor

A. The reactor core shall consist of not more than 484 fuel assemblies with fuel rods in either a 7 x 7 or 8 x 8 array.

B. The reactor core shall contain 121 cruciform-shaped control rods. The control rod material shall be boron carbide powder (B_4C) compacted to approximately 70% of theoretical density.

5.3 Reactor Vessel

A. The pressure vessel shall be designed for a pressure of 1250 psig and a temperature of 575°F. The coolant recirculation system shall be designed for a pressure of 1148 psig on suction side of pump and 1248 psig at pump discharge. Both the pressure vessel and recirculation system shall be designed in accordance with the ASME Boiler and Pressure Vessel Code Sections III and IX.

5.4 Containment

A. The primary containment shall be of the pressure suppression type having a drywell and an absorption chamber constructed of steel. The drywell shall have a volume of approximately 134,200 ft^3 and is designed to conform to ASME Boiler and Pressure Vessel Code Section III Class B for an internal pressure of 56 psig at 281°F and an external pressure of 2 psig at 281°F. The absorption chamber shall have a total volume of approximately 176,250 ft^3.

2.0 SAFETY LIMITS

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. When the reactor pressure is greater than 600 psig, the combination of reactor coolant core flow and reactor thermal power transferred to the coolant shall not exceed the limit shown in Figure 2.1.1. The safety limit is exceeded when the reactor coolant. core flow and thermal power transferred to the coolant results in a point above or to the left of the limit line.

LIMITING SAFETY SYSTEM SETTINGS

2.3 FUEL CLADDING INTEGRITY

Applicability:

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

Objective:

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

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM -- The APRM flux scram trip setting shall be as shown in Figure 2.3.1 unless the combination of power and peak heat flux is above the applicable curve in Figure 2.3.2. When the combination of power and peak heat flux is above the curve in Figure 2.3.2, a scram setting (S) as given by:

B. When the reactor pressure is less than 600 psig or core flow is less than 5% of design, the reactor thermal power transferred to the coolant shall not exceed 300 MW.

C. 1. The neutron flux shall not exceed the scram setting established in Specification 2.3.A for longer than 0.95 seconds as indicated by the process computer.

$$S = \frac{486{,}000 \quad P}{X} \quad \text{(7x7 fuel)}$$

$$S = \frac{425{,}000 \quad P}{X} \quad \text{(8x8 fuel)}$$

Where:
P = percent of rated power
X = peak heat flux - ($BTU/HR/FT^2$)
shall be used.

2. IRM--Flux Scram setting shall be $\leq$ 15% of rated neutron flux.

B. APRM Rod Block - The APRM rod block setting shall be as shown in Figure 2.3.1 unless the combination of power and peak heat flux is above the applicable curve in Figure 2.3.2. When the combination of power and peak flux is above the curve in Figure 2.3.2, a rod block trip setting (RB) as given by:

$$RB = \frac{437{,}400 \quad P}{X} \quad \text{(7x7 fuel)}$$

$$RB = \frac{382{,}400 \quad P}{X} \quad \text{(8x8 fuel)}$$

where:
P = percent of rated power
X = peak heat flux ($BTU/HR/FT^2$)
shall be used.

C. Reactor Low Water Level Scram setting shall be $\geqq$ 10'6 above the top of the active fuel.

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

Robert M Lazo, Chairman
John C. Geyer, Member
Walter H. Jordan, Member

In the Matter of	)
	)
NORTHERN STATES POWER COMPANY	)Docket No. 50-263
(Monticello Nuclear Generating Plant)	)

MEMORANDUM AND ORDER

RULING ON PETITION FOR LEAVE TO INTERVENE

On February 13, 1974, the Atomic Energy Commission (the Commission) published in the Federal Register a Notice of Proposed Changes to Technical Specifications of Facility Operating License with respect to Northern States Power Company's (Applicant) Monticello Nuclear Generating Plant (39 F.R. 5529). The proposed change in the technical specifications of Provisional Operating License No. DPR-22, would permit the use of a partial laoding of 8 x 8 fuel (containing U-235), including a fuel assembly containing segmented test rods, and also would authorize changes to the limiting conditions for operations associated with fuel densification for the 8 x 8 and 7 x 7 fuels. The Applicant is presently licensed to possess and operate its facility located in Wright County, Minnesota, at power levels up to 1670 MWt using a full core of 7 x 7 fuel (containing U-235). The Notice provided that any person whose interest might be affected by the proceeding may file a request for hearing and petition for leave to intervene.

On March 15, 1974, the Minnesota Pollution Control Agency (MPCA), an agency of the State of Minnesota, filed a timely request that the Commission hold a public hearing on the proposed change to Technical Specifications and petitioned for leave to intervene as a party in such a proceeding. Both the Applicant and Staff have urged that the petition for leave to intervene be granted.

On March 21, 1974, Petitioner, by telegram, requested the Commission to defer ruling on its request for hearing and petition to intervene until Petitioner had an opportunity to review the Staff's Safety evaluation relating to 8 x 8 fuel assemblies. Subsequently, during a conference telephone call on April 15, 1974, counsel for MPCA, the Regulatory Staff and the Applicant, advised the Chairman of the Atomic Safety and Licensing Board (the Board) established to rule on petitions for leave to intervene in the instant proceeding, that an agreement had been reached whereby the request by MPCA for hearing on the proposed change in technical specifications would be withdrawn provided that the Board would permit the contentions asserted in that petition to be raised in the presently pending proceeding on the conversion of Applicant's provisional operating license for the Monticello plant to a full term operating license. A formal motion entitled "Withdrawal of Request for Hearing and Petition to Intervene" was filed by MPCA on April 16, 1974.

The Board hereby grants the motion by MPCA for withdrawal of its request for hearing and petition for leave to intervene and accepts the agreement entered into by MPCA, the Staff and the Applicant. Said agreement is further evidenced by a document dated April 16, 1974, to which neither the Staff nor Applicant has objected, entitled "Additional Contentions with Respect to 8 x 8 Fuel Assemblies," which MPCA has filed in the proceeding on the application for conversion of the Monticello provisional operating license to a full term operating license.

WHEREFORE, IT IS ORDERED, that in accordance with the Atomic Energy Act, as amended, and the Rules of Practice of the Commission, the petition of the Minnesota Pollution Control Agency (MPCA) for a hearing and for leave to intervene is withdrawn and no other petition having been filed, the proceeding designated in the Commission's February 13, 1974 notice of proposed changes to technical specifications of Provisional Operating License No. DPR-22, is dismissed.

FOR THE ATOMIC SAFETY AND LICENSING BOARD
designated to rule on petitions
for leave to intervene

Robert M Lazo

Robert M. Lazo, Chairman

Issued at Bethesda, Maryland
this 30th day of April 1974.

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NO. 3 TO LICENSE NO. DPR-22

(CHANGE NO. 14 TO APPENDIX A TECHNICAL SPECIFICATIONS)

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

INTRODUCTION

In letters dated November 19, 1973, as supplemented by filings dated December 14, 1973, January 15, 1974, February 8, 27, and 28, 1974, and April 1, 1974, Northern States Power (NSP) requested authorization to operate the Monticello Nuclear Generating Plant using a partial loading of 8x8 fuel, including a fuel assembly containing segmented test rods, and also proposed changes to the Technical Specifications related to limiting conditions for operation associated with fuel densification for the 8x8 and 7x7 fuels.

The use of 8x8 fuel in reloads has been reviewed on a generic basis by the Licensing staff and the Advisory Committee on Reactor Safeguards (ACRS). The reports based on these reviews were transmitted to NSP by letters dated February 11 and 20, 1974. The staff Safety Evaluation for the use of 8x8 fuel assemblies in the Monticello facility was transmitted to NSP by letter dated April 8, 1974.

NSP requested review of changes in letters dated January 23, 1974 and March 1, 1974, as supplemented by filings dated March 8 and 19 and April 10 and 26, 1974, relating to pressure relief, control rod scram times, standby gas treatment system and reactor vessel temperature measurements. By letter dated April 10, 1974, NSP requested approval of proposed changes(1,4) that had been previously described(11,12,13) in time to return to power operation on May 5, 1974. A preliminary response to our April 4, 1974 letter on the Prompt Relief Trip System(8) was submitted by NSP letter dated April 26, 1974.

We have previously reviewed(7) and approved(10) the 8x8 fuel rod array including a fuel assembly containing segmented test rods as "reload 2" fuel for the Monticello core and authorized(9) insertion of the 8x8 reload-2 fuel assemblies into the Monticello core for reactor tests up to 1% power level[5]. In addition, we have met with NSP representatives on two separate occasions[2,6] to review reactor operation during Monticello fuel cycle 3 with and without reliance on the proposed prompt relief trip[1,4] system (PRT) to preserve acceptable design margins to a fuel damage threshold (i.e., MCHFR > 1).

In response to the NSP proposal[1,4] to install a prompt relief system (PRT) "to compensate for equilibrium core scram reactivity insertion functions by minimizing the peak pressure and fuel thermal effects resulting from pressurization type abnormal transients", we stated[2] that the Directorate of Licensing safety review of the proposed PRT system will be completed by September 1974. We requested[8] additional information related to design and installation of the PRT system to assure that existing reactor protection systems and engineered safety features (auto-depressurization) were not affected by the new PRT system connections or by activation of the PRT system after completion of the modification. Because we have been unable to resolve our expressed concerns and since NSP has been unable to complete, at this time, a response to our request[8] for information related to the potential for short circuits that could provide power to the safety/relief valve solenoids, thus opening all six of the valves resulting in unintentional primary system blowdown-depressurization, the PRT panel will not be connected[14]. The PRT system will, therefore, remain inactive[14] when the plant returns to power production after refueling and completion of the plant modifications to connect the new safety/relief valves (replacing the four safety valves) to the torus suppression pool. The status of the PRT "hook-up" has been reported[14] by NSP, and we concur that plant safety margins have not been reduced by the wiring between relief valve solenoids and drywell penetration. Accordingly, the PRT system is eliminated from further evaluation at this time pending resolution of the potential for power shorts that could open all six safety/relief valves.

The considerations involving changes to the Technical Specifications before normal reactor operation can be resumed (with the proposed PRT system[1] inactive) are listed below in the order of decreasing importance. Also included are two items (analysis of abnormal core transients and fuel densification considerations) which were not addressed in our Safety Evaluation dated April 8, 1974, on the subject of operation of the Monticello facility with 8x8 fuel.

1. transient analysis, safety/relief set points, steam relief flow capacity, and primary coolant system boundary stresses (proposed[13] changes 4, 5, 18, 19b, 20a, 20b).

2. control rod scram times (proposed[13] change 16).

3. fuel densification (proposed[12] changes 1, 2, 3, and 4).

4. 8x8 fuel rod array, reload 2 fuel assemblies (proposed[11] changes 1, 2, 3, 4, 5, 6, 7, and 8).

5. standby gas treatment system (proposed[13] change 24).

6. reactor vessel temperature measurements (proposed[13] change 19a).

Each of the above items is considered in the following evaluation.

EVALUATION

Safety/Relief Valves

The reactor pressure relief system limits overpressurization of the reactor coolant system to prevent failure of the reactor vessel, piping, or components due to excessive coolant pressure.

Prior to the April 1974 plant outage for core refueling, excessive stress in the reactor vessel and core coolant recirculation system due to transient pressure increases was prevented by automatic opening of as many of the four safety/relief valves and four spring-loaded safety valves as necessary whenever coolant pressure exceeded safety/relief set points (≈1080 psig and 1240 psig). High pressure reactor scram was set at 1075 psig and remains unchanged for fuel cycle 3. Each of the safety/relief valves was set to relieve steam pressure when pressure increased to a nominal value of 1080 psig. Each safety/relief valve when fully opened relieves ≈800,000 pounds of steam per hour. Prior to April 1974, if the pressure transient exceeded the relief flow capacity of these valves and reached 1240 psig, as many of the four safety valves opened as necessary to limit the peak transient pressure to acceptable levels. Each of these valves, when completely opened, could relieve 642,000 pounds of steam.

Closure of all main steam isolation valves (MSIVs) with delayed reactor scram from the high neutron flux signal and end of cycle (EOC) 2 scram reactivity assumptions was the basis for determining the pressure relief flow capacity requirements. The peak calculated pressure (at the bottom of the reactor vessel) for this assumed condition was 1308 psig or 67 psi below the maximum overpressure design limit of 1375 psig.

Whenever the safety valves opened, steam was relieved into the containment drywell causing pressurization. Since the period of core coolant overpressure is normally of short duration, the safety valves reclosed within seconds and the resultant containment pressure increases were very small and well within the containment design capability. The unanticipated release of steam into the containment drywell through premature opening of safety valves has occurred several times at Monticello and prompted technical specification changes to increase safety valve set points(15) to reduce the probability of unnecessary valve opening.

To eliminate this safety valve potential for unnecessary containment pressurization, the four spring-loaded safety valves were replaced during the April 1974 outage with four target rock safety/relief valves equivalent to the four that have been installed at Monticello since the plant startup. The safety/relief valves have higher relieving capacity than the safety valves. Also, new piping, which does not adversely affect the safety function of the safety/relief valve system, was installed to deliver steam relieved through the valves to the torus suppression pool (four new 10" lines similar to those provided for the originally installed safety/relief valves). The modifications were designed in accordance with the same code requirements to which the original plant was designed. The modifications do not involve a change in the technical specifications.

General Electric has compared the analytical results(1) for the worst abnormal transient, turbine trip without turbine bypass, and delayed scram due to high neutron flux with the calculated results from the assumed closure of all four MSIVs and delayed scram due to high neutron flux to determine the total required safety/relief valve capacity. Based on these calculations, the assumed isolation of MSIVs results in the greatest demand for steam relief capacity and continues, as before, as the basis for determining steam relief capacity requirements.

For the most conservatively limiting conditions at the EOC 3, GE has calculated(1) that the peak transient pressure at the bottom of the reactor vessel with only six safety/relief valves (set points 1080 psig) is 1285 psig. This is noticeably lower than the 1308 psig calculated value for the limiting period at the EOC 2 operation just prior to the April 1974 modifications when four safety/relief (set point 1080 psig) and four safety valves (set point 1240 psig) were required for over-pressure protection. Since the calculational methods are unchanged(1), NSP has proposed(13) that a minimum of six safety/relief valves be operative during normal reactor operation. We have discussed(2) this proposed reduction in pressure relief capacity with NSP representatives. Six safety/relief valves can relieve 71% of nuclear boiler rated steam flow(1) compared with pre-modification capability of about 54% flow through four safety/relief valves and about 38% through four safety valves for a total of about 92% and have concluded[3] that such a reduction cannot be approved before we have completed our evaluation of the calculational methods[17,2,3] used by GE. Additional information provided in NSP letter dated April 10, 1974, shows that there is sufficient margin to core coolant boundary and nuclear fuel failure thresholds for the abnormal turbine trip without bypass transient with six safety/relief valves since the maximum transient pressure during the first 4200 MWd/T of cycle 3 fuel depletion is 1192 psig (compared with 1375 psig limit) and the lowest critical heat flux ratio in the same period is 1.68 (compared with a MCHFR limit of 1.0). For the period beyond 4200 MWd/T during cycle 3 fuel depletion, the calculated margins to failure thresholds are acceptable with steady state power level restricted below 95%. We have changed the proposed technical specification with NSP telephone concurrence (April 24, 1974), to require a minimum of seven operative safety/relief valves instead of the six proposed by NSP because at the same 1240 psig pressure where steam flow capacity with four safety and four safety/relief valves was calculated to be about 92% of the 100% rated steam flow capacity, seven safety relief valves provide about 95%, a slight improvement over the core coolant pressure relief capability prior to April 1974. This change is considered to be an interim change until we have completed our review of the GE calculational methods. We have concluded, therefore, that coolant pressure relief capability is unchanged by the conversion from safety to safety/relief valves and that abnormal transient core behavior with 8x8 Reload 2 fuel assemblies in the core as well as 7x7 fuel assemblies is acceptable and consistent with earlier core performance characteristics. We also concur that the PRT system connections that have been made so far do not affect existing engineered safety features or reduce reactor safety because they have not been energized and thus serve no function.

Control Rod Scram Time

We have reviewed and previously approved[16] an increase in the required rate of insertion for the first half of control rod insertion travel following a reactor scram signal. Reactor operating experience has shown that actual measured control rod scram times are significantly less than the performance limits that had been previously established and that the average rod scram time for 90% insertion can be reduced from 5.00 to 3.50 seconds. Also, the time for the three fastest control rods of all groups of four control in two by two arrays can be reduced to "no greater than 3.80 seconds at 90% of the rod length inserted instead of 5.30 seconds". The reduced control rod scram times, based on reactor operating experience to date, are sufficiently longer than measured control rod scram times to allow for normal changes in control rod performance without reaching the technical specification limits. The original scram time limits were conservatively specified to allow for uncertainties related to control rod scram time deterioration. There are now sufficient control rod scram time measurements from operating reactors to reduce the allowance for uncertainties.

Calculated scram reactivity is based in part on the technical specification scram time limits and using the new more stringent technical specification scram times specified, the calculated scram reactivity for the last half of rod insertion is somewhat faster and the magnitude of the power mismatch and pressure increase when the normal heat sink is unavailable is reduced; i.e., the calculated consequences of abnormal transients are less severe because control rods are assumed to scram at technical specification values.

Fuel Densification - LHGR

Changes to the General Electric fuel densification model have been reviewed[19] by the Directorate of Licensing on a generic basis with the conclusion that the General Electric calculational model, as modified by the Regulatory staff, is suitably conservative for the evaluation of densification effects in BWR fuel, where possible densification effects are listed as:

1. power spikes due to axial gap conductance,
2. increase in linear heat generation rate (LHGR) due to pellet length shortening,
3. creep collapse of the cladding due to axial gap formation, and
4. changes in stored energy due to increased radial gap size.

Since the assembly average stored energy is one of the important inputs to BWR LOCA evaluations, a technical specification limit on maximum permitted assembly power is imposed. This requirement is satisfied by maximum average planar LHGR restrictions for initial, reload 1, and reload 2 type fuel assemblies which constitute the Monticello core during fuel cycle 3 operation. The calculated peak clad temperatures do not exceed 2300°F with the proposed MAPLHGR technical specifications which we have concluded are acceptable. Although the power produced by the 8x8 reload 2 fuel rods is reduced (assuming fixed fuel bundle power), on the average, to 49/63 of 78% of the fuel rod power in the 7x7 fuel assemblies, the average heat transfer surface of the 8x8 fuel assemblies is only increased by 13%. According to GE thermal-hydraulic calculations, the modest improvement in heat transfer capability is nearly offset by the reduced flow due to increased flow resistance. To maintain minimum critical heat flux ratios above 1.9, as in the past, the technical specifications must be changed to limit 8x8 fuel rod LHGR to 13.4 kW/ft where the equivalent 7x7 fuel rod LHGR is 17.5 kW/ft. We concur that these changes in technical specifications will maintain equivalent margins to the fuel design limit (MCHFR > 1) and fuel damage thresholds following design basis loss-of-coolant accidents.

8x8 Reload Fuel

Where only 7x7 fuel assembly characteristics were identified, it is necessary to change the Technical Specifications to include the 8x8 fuel assembly characteristics. The changes include calculation of reactor scram and rod block set points based on peak heat flux in the 8x8 fuel assembly rods. The 7x7 equations remain unchanged.

A new relationship of peak fuel rod heat flux versus reactor power level is included in the Technical Specifications to show the relationship of the 8x8 fuel rods as well as the 7x7 fuel rod that had been included previously. The 8x8 fuel assembly rod power peaking factor of 3.04 is now referenced in the Technical Specifications as well as the 3.08 value for 7x7 fuel assemblies. The reduced fuel pellet diameter of the 8x8 fuel assembly rods reduces the fuel time constant and necessitates appropriate changes in the Technical Specifications. We have previously reviewed[20] the performance characteristics of the 8x8 fuel assemblies, including the effects of unpowered center rods, and abnormal transients (refer to item 1 of this evaluation) and have concluded that reactor operation with 8x8 reload fuel assemblies in the Monticello core as well as 7x7 fuel assemblies is acceptable.

Standby Gas Treatment System

The proposed technical specifications, we agree, clarify intentions and eliminate inconsistencies between plant charcoal filter test capability and shop test capability with reference to aerosol particle size. The changes involve changing the dioctyl phtalate particle size from 0.3 micron to 0.7 micron and should be made. This change is acceptable based on Regulatory Guide 1.52.

Reactor Vessel Temperature Measurements

Temperatures on the outside surface of the reactor vessel are dependent on the coolant temperature, metal thickness, outside insulation, and drywell ambient temperature. Since core coolant temperature is the only important variable; i.e., during heatup and cooldown, the surface temperatures will change in a time dependent manner whenever loop temperatures change. Having determined the relationship of the vessel surface temperatures to changes in loop temperature and verified that the relationship is constant by repeated measurements during many primary coolant system heatup and cooldown operations, there is no need to continue this verification indefinitely. The data collected have been used to assure that the rate of temperature change during normal heatup or cooldown of the reactor coolant system is not excessive. Loop temperature restrictions, as specified in the Technical Specifications, are adequate to guard against damage due to rapid temperature changes. We, therefore, concur that the requirement for reactor vessel shell and flange thermocouple measurements of Technical Specification 4.6.A.1 should be deleted. This change, we have concluded, does not affect reactor safety or the health and safety of the public.

CONCLUSION

We have reviewed changes to technical specifications involving coolant pressure relief, control rods, standby gas treatment system, and reactor vessel surface temperatures, and have concluded that the changes do not present significant hazards considerations since the probability of accidents is not increased and safety margins to design limits are not decreased and the severity of forseeable consequences are not increased. We have also concluded that there is reasonable assurance that the health and safety of the public will not be endangered by the above changes, the changes to the fuel densification limits, and the use of 8x8 reload 2 fuel assemblies. Accordingly, the changes, as presented by NSP and modified by us, in the replacement pages for the Technical Specifications should be made.

/S/

James J. Shea
Operating Reactors Branch #2
Directorate of Licensing

/S/

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

Date: MAY 14 1974

REFERENCES

1. NSP submittal and errata dated January 23, 1974, and March 19, 1974, respectively - "Permanent Plant Changes to Accommodate Equilibrium Core Scram Reactivity Characteristics".

2. Minutes of NSP and AEC-L representatives meeting regarding replacement of Safety Valves and Prompt Relief Trip - by Directorate of Licensing dated March 14, 1974.

3. Directorate of Licensing letter dated March 14, 1974 - "Approval of Plant Modifications for Fuel Cycle 3".

4. NSP submittal dated March 8, 1974 - "Supplement to January 23, 1974 Report".

5. NSP request for fuel loading and testing authorization dated March 21, 1974.

6. Minutes of NSP and AEC-L meeting on March 22, 1974 - by Directorate of Licensing dated April 2, 1974.

7. NSP submittal dated November 19, 1973 - "Second Reload Submittal".

8. Directorate of Licensing request for additional information related to PRT dated April 4, 1974.

9. Safety Evaluation by the Directorate of Licensing related to Insertion of 8x8 Fuel Assemblies into Monticello Core and Testing at Reactor Power Levels Less than 1% dated March 30, 1974.

10. Safety Evaluation by the Directorate of Licensing related to 8x8 Fuel Assemblies - dated April 8, 1974.

11. NSP proposal to change Technical Specifications to allow for 8x8 reload fuel dated February 27, 1974.

12. NSP proposal to change Technical Specifications related to fuel densification - dated February 28, 1974.

13. NSP proposal to change technical specifications related to substitution of safety/relief valves for four spring-loaded safety valves and other PRT system changes - dated March 1, 1974.

14. NSP response to AEC PRT concerns - dated April 26, 1974.

15. NSP submittal dated September 13, 1973 - "Proposed Change in Safety Valve Set Point" and Directorate of Licensing Change No. 10 dated October 2, 1973.

16. Directorate of Licensing Change No. 8 - Control Rod Scram Time Change - dated July 2, 1973.

17. "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor" NEDO-10802 by GE dated February 1973.

18. NSP submittal dated February 8, 1974 - "Calculations Pertaining to the Densification Effects on 8x8 Fuel".

19. Directorate of Licensing Report - "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels" - dated December 14, 1973.

20. Directorate of Licensing - "Technical Report on the General Electric Company 8x8 Fuel Assembly" - dated February 5, 1974.

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF CHANGES TO TECHNICAL SPECIFICATIONS OF PROVISIONAL OPERATING LICENSE

The U. S. Atomic Energy Commission (the Commission) issued on February 11, 1974, and published in the Federal Register on February 13, 1974 (39 F.R. 5529), a notice of consideration of a proposed change in the Technical Specifications of Provisional Operating License No. DPR-22 issued to the Northern States Power Company to permit the use of fuel assemblies using a partial loading of 8 x 8 fuel (containing U-235 and including a fuel assembly containing segmented test rods) and to authorize changes in the limiting conditions for operations associated with fuel densification for the 8 x 8 and 7 x 7 fuels for the Monticello Nuclear Generating Plant Unit 1 (the facility).

The Minnesota Pollution Control Agency (MPCA) filed a "Request for Hearing and Petition to Intervene" dated March 15, 1974, under 10 CFR 2.714 of the Commission's Rules of Practice. Subsequently, on April 16, 1974, MPCA filed a "Withdrawal of Request for Hearing and Petition to Intervene" based upon the consolidation of these issues with the licensing proceeding involving the conversion of the provisional operating license of the Monticello facility to a full term license (see: Atomic Safety and Licensing Board's Memorandum and Order Ruling on Petition for Leave to Intervene dated April 30, 1974). Accordingly, the Commission has issued Change No. 14 to the Technical Specifications of Provisional Operating License No. DPR-22 to the Northern States Power Company (the licensee). This change, effective immediately, authorizes the items which were the subject of the February 11, 1974 notice, as referenced above.

The licensee is presently authorized to possess and operate the facility located in Wright County, Minnesota, at power levels up to 1670 MWt using a full core 7 x 7 fuel (containing U-235).

The Commission has found that the application for the above action dated November 19, 1973, as supplemented by filings dated December 14, 1973, January 15, 1974, February 8, 27 and 28, 1974, and April 1, 1974, complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations published in 10 CFR Chapter I. The Commission's Directorate of Licensing completed the major portion of its evaluation of the action and issued a Safety Evaluation on April 8, 1974, concluding that there is reasonable assurance that the health and safety of the public will not be endangered by the operation of the facility with the 8 x 8 fuel and the related changes to the Technical Specifications as authorized by Change No. 14, which is incorporated in License No. DPR-22 as Amendment No. 3 thereto.

Copies of (1) the Atomic Safety and Licensing Board's Memorandum and Order Ruling on Petition for Leave to Intervene dated April 30, 1974, (2) Amendment No. 3 with Change No. 14 to the Technical Specifications of Provisional Operating License No. DPR-22, (3) the Directorate of Licensing's Safety Evaluation dated April 8, 1974, (4) the Safety Evaluation by the Directorate of Licensing concurrently issued with Amendment No. 3 which considers abnormal core transients and the effects of fuel densification, (5) the Technical Report on the General Electric Company 8 x 8 assembly by the Directorate of Licensing dated

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February 5, 1974, and (6) the Report of the Advisory Committee on Reactor Safeguards dated February 12, 1974, on the subject of operation of boiling water reactors with 8 x 8 fuel bundles, are available for public inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C., and at the Environmental Library of Minnesota at 1222 S. E. 4th Street, Minneapolis, Minnesota 55414. Single copies of these items may be obtained upon request sent to the Deputy Director for Reactor Projects, Directorate of Licensing, U. S. Atomic Energy Commission, Washington, D. C. 20545.

Notice is being given that the Commission also has issued, as part of the above Amendment No. 3 to the license, revisions to the Technical Specifications for the facility which include changes in the specification provisions relating to (1) pressure relief, (2) control rod scram times, (3) standby gas treatment systems, and (4) reactor vessel temperature measurements.

The application, as supplemented, for these four changes complies with the standards and requirements of the Act and the Commission's rules and regulations. The Commission has found that these changes do not involve a significant hazards consideration and that the approval of these actions will not be inimical to the common defense and security or to the health and safety of the public. The Regulatory staff's review of these changes is reflected in a concurrently issued Safety Evaluation.

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For further details with respect to these four items, see (1) the applications for amendment dated January 23, 1974 and March 1, 1974, as supplemented on March 8 and 19, and April 10 and 26, 1974, (2) Amendment No. 3 to License No. DPR-22, with any attachments, and (3) the Commission's concurrently issued Safety Evaluation. All of these items also are available for inspection at the two public document rooms previously stated herein. Copies of items (2) and (3) also may be obtained upon request addressed to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Deputy Director for Reactor Projects, Directorate of Licensing - Regulation.

Dated at Bethesda, Maryland, this MAY 14 1974

FOR THE ATOMIC ENERGY COMMISSION

Original signed by
Dennis L. Ziemann

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

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Page 1 of Notice + Finding
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based upon the consolidation of these issues with the licensing proceeding involving the conversion of the Provisional Operating License ~~for the~~ of the Montrcello facility to a Full Term Licensex (See: Atomic Safety and Licensing Board's ~~Memorandum Ruling and Order of April 30, 1974~~ Memorandum and Order Ruling on Petition for Leave to Intervenex dated April 30, 1974).

Steve [illegible]

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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 3
License No. DPR-22

1. The Atomic Energy Commission (the Commission) has found that:

   A. The applications for amendment by the Northern States Power Company (the licensee) dated November 19, 1973, January 23, 1974, and March 1, 1974, as supplemented, comply 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 license, 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;

   E. The request for a hearing and petition for leave to intervene (by the Minnesota Pollution Control Agency) on the proposed action of those items relating to operation with 8 x 8 fuels and limiting conditions for operation associated with fuel densification for 8 x 8 and 7 x 7 fuels has been withdrawn and the proceeding dismissed, and Insert by Lewis

   F. Prior public notice of those items relating to pressure relief, control scram times, standby gas treatment, and reactor vessel temperature measurements is not required since they do not involve a significant hazards consideration.

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF CHANGES TO TECHNICAL SPECIFICATIONS OF PROVISIONAL OPERATING LICENSE

The U. S. Atomic Energy Commission (the Commission) issued on February 11, 1974, and published in the Federal Register on February 13, 1974 (39 F.R. 5529), a notice of consideration of a proposed change in the Technical Specifications of Provisional Operating License No. DPR-22 issued to the Northern States Power Company to permit the use of fuel assemblies using a partial loading of 8 x 8 fuel (containing U-235 and including a fuel assembly containing segmented test rods) and to authorize changes in the limiting conditions for operations associated with fuel densification for the 8 x 8 and 7 x 7 fuels for the Monticello Nuclear Generating Plant Unit 1 (the facility).

The Minnesota Pollution Control Agency (MPCA) filed a "Request for Hearing and Petition to Intervene" dated March 15, 1974, under 10 CFR 2.714 of the Commission's Rules of Practice. Subsequently, on April 16, 1974, MPCA filed a "Withdrawal of Request for Hearing and Petition to Intervene". On April 30, 1974, the Atomic Safety and Licensing Board, designated to rule on the MPCA request, accepted the MPCA withdrawal and dismissed the proceeding designated in the Commission's notice published on February 13, 1974. Accordingly, the Commission has issued Change No. 14 to the Technical Specifications of Provisional Operating License No. DPR-22

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