Docket No. 50-263

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x w/cy of GE ltrs. of 8/24 and 11/4/76.
RD.

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

MAY 24 1977

Gentlemen:

The Commission has issued the enclosed Amendment No. 27 to Provisional Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment is in response to your letter dated November 15, 1976.

In their letters dated August 24 and November 4, 1976, General Electric (GE) discussed a potential problem concerning the effects of core flow on loss of coolant accident analyses. GE recommended that a reduction in maximum average planar linear heat generation rate (MAPLHGR) limits accompany a reduction in core flow for some BWR-3 and BWR-4 facilities. In your letter dated November 15, 1976, you stated that the recommended limits were being imposed.

We have reviewed the information submitted by GE and your letter dated November 15, 1976. Your facility is a BWR-3 with a reactor vessel inner diameter of 205" and is addressed in the enclosed generic safety evaluation.

This amendment incorporates into the Monticello Technical Specifications reduced MAPLHGR limits for operation at less than full rated flow.

A copy of the related Notice of Issuance is also enclosed.

Sincerely,

Original signed by
Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures and cc:
See next page

subject to attached memo (Lane) RD 5/24

OFFICE	DOR:ORB-2	DOR:ORB-2	DOR:ORB-4	OELD	DOR:ORB-2	
SURNAME	RMDiggs	RPSnaider:esp	CNelson	B. Smith	DKDavis	
DATE	4/ /77	4/20/77	4/25/77	5/23/77	5/24/77	

NRC FORM 318 (9-76) NRCM 0240

MAY 24 1977

Enclosures:
1. Amendment No. 27 to
   License No. DPR-22
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures:
Gerald Charnoff, Esquire
Shaw, Pittman, Potts and
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Buffalo, Minnesota 55313

cc w/enclosures and cy of NSPCo
filing dtd. 11/15/76:
State Department of Health
ATTN: Secretary and Executive Officer
University Campus
Minneapolis, Minnesota 55440

OFFICE▶						
SURNAME▶						
DATE▶						

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

May 24, 1977

Docket No. 50-263

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

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We have reviewed the information submitted by GE and your letter dated November 15, 1976. Your facility is a BWR-3 with a reactor vessel inner diameter of 205" and is addressed in the enclosed generic safety evaluation.

This amendment incorporates into the Monticello Technical Specifications reduced MAPLHGR limits for operation at less than full rated flow.

A copy of the related Notice of Issuance is also enclosed.

Sincerely,

Don K Davis

Don K. Davis, Acting Chief
Operating Reactors Branch "2
Division of Operating Reactors

Enclosures and cc:
See next page

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cc w/enclosures and cy of NSPCo filing dtd. 11/15/76:
State Department of Health
ATTN: Secretary and Executive Officer
University Campus
Minneapolis, Minnesota 55440

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 27
License No. DPR-22

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

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

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

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

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

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

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

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 27, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 24, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 27

PROVISIONAL OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated license with the attached pages bearing the same numbers. The changed areas on the revised pages are reflected by a marginal line.

Remove	Insert
189B	189B
189E	189E
189F	189F

3.11 REACTOR FUEL ASSEMBLIES

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figures 3.11.1. When core flow is less than 90% of rated core flow, the APLHGR shall not exceed 95% of the limiting value shown in Figure 3.11.1. If any time during operation it is determined that the limit for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours.

4.11 REACTOR FUEL ASSEMBLIES

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$25% rated thermal power.

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, Appendix K.

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 cladding temperature by less than $\pm$ 20°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 within the 10CFR50 Appendix K limit. The limiting value for APLHGR is given by this specification.

Reference 6 demonstrates that for lower initial core flow rates the potential exists for earlier DNB during postulated LOCA's. Therefore a more restrictive limit for APLHGR is required during reduced flow conditions.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding APLHGR limits in such cases need not be reported.

B. LHGR

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation and 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.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding LHGR limits in such cases need not be reported.

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in Reference 4 assumed the steady state MCPR prior to the postulated loss-of-coolant accident to be 1.18 for all fuel types. In addition, the ECCS analysis presented in Reference 6 assumed an initial MCPR of 1.24 for reduced flow conditions. The Operating MCPR Limit of 1.38 for 8x8 fuel and 1.29 for 7x7 fuel is determined from the analysis of transients discussed in Bases Sections 2.1 and 2.3. By maintaining an operating MCPR above these limits, the Safety Limit of 1.06 (T.S.2.1.A) applicable to all fuel types is maintained in the event of the most limiting abnormal operational transient.

For operation with less than rated core flow the Operating MCPR Limit is adjusted by multiplying the above limit by K_f. Reference 5 discusses how the transient analysis done at rated conditions encompasses the reduced flow situation when the proper K_f factor is applied.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding MCPR limits in such cases need not be reported.

References

1. "Fuel Densification Effects in General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August, 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USAEC Regulatory Staff)

3. Communication: VAMoore to IS Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. "Monticello Nuclear Generating Plant Loss-Of-Coolant Accident Analysis Conformance with 10 CFR 50 Appendix K, August 1974," L O Mayer (NSP) to J F O'Leary, August 20, 1974.

5. "General Electric BWR Generic Reload Application for 8 x 8 Fuel," NEDO-20360, Revision 1, November 1974.

6. "Additional Effects of Core Flow on ECCS LOCA Analysis," A. Levine (GE) to Z. Rosztoczy (NRC), August 24, 1976.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 27 TO PROVISIONAL OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 BACKGROUND

Conservative Loss-of-Coolant-Accident (LOCA) calculations for jet-pump BWR's predict that nucleate boiling will be maintained for several seconds following a Design Basis-LOCA (DB-LOCA). This results in the early removal of significant amounts of stored energy which, if present later in the transient, when heat transfer coefficients are considerably lower, would result in a much higher calculated peak cladding temperature (PCT) (i.e. a higher delta-temperature would be needed later to conduct the heat across the higher thermal resistance resulting from the lower non-nucleate boiling heat transfer coefficient.)

However, for certain BWR's operating at less than 100% of rated core flow, LOCA calculations in the period immediately following the break initiation predict a transient core flow reduction for which nucleate boiling cannot be sustained. This early departure from nucleate boiling was not fully considered in BWR-LOCA analyses for plant operation below rated flows, and the resulting MAPLHGR limits for several plants were established in a non-conservative manner.

This SER presents results of the NRC staff's review of generic material (1,2,3,4) presented by the General Electric Co. (GE) to document effects of lower initial core flows on all operating BWR MAPLHGR limits.

2.0 EFFECTS OF LOW CORE FLOW ON NON-JET-PUMP BWR'S (BWR/1 AND BWR/2)

Unlike jet-pump BWR's, non-jet-pump (NJP) plants have the potential for experiencing a "bottom break" which could result in an essentially instantaneous flow stagnation at the highest power plane.* Consequently, analyses of NJP-BWR's use a "dryout correlation" which takes no credit for continued flow in the core after the break and therefore predicts departure from nucleate boiling in a very short time (<1.5 seconds following the break for any size break including the DE-LOCA). NJP-BWR's thus derive no significant benefit from nucleate boiling even for full flow conditions. Therefore, the effects on PCT of lower initial core flows are insignificant (<10^{o}F) for all flows and all size breaks.

We therefore agree with the GE conclusion that no additional MAPLHGR limits are needed for NJP-BWR's operating at lower core flows.

* Humboldt Bay Unit #3 is an exception to this statement, since it is a natural recirculation unit with no external recirculation loops. The largest credible break on this unit is so small that there would be no flow dependent effect on the DNB time, and consequently the "low flow" effects discussed herein are not applicable to Humboldt Bay Unit #3.

3.0 EFFECTS OF LOW CORE FLOW ON JET-PUMP BWR MAPLHGR LIMITS

3.1 General Effects of Lower Flows

Lower initial core flows cause several effects on the calculated PCT for jet-pump BWR's (JP-BWR's) following a LOCA. Some of these effects are positive and some are negative as described below.

3.1.1 Negative Effect of Lower Steady State Core Flows on MAPLHGR (tends to increase calculated PCT): Early DNB

At lower steady state flows, the post-LOCA transient core flow decrease immediately following the break (<1.0 seconds) can reach a value low enough to cause departure from nucleate boiling (DNB) at the highest power plane. Whether or not early DNB occurs for any particular plant for a particular initial core flow depends critically on:

(1) the ratio of break area to water inventory in the reactor primary system. This could be termed an "effective break size", since this ratio determines how rapidly the reactor will depressurize and, more importantly for the present subject, it determines the minimum transient flow ("flow dip") during the first $\approx$1 second following the break. The lower this "dip", the more probable is early DNB. The absolute value of this minimum transient flow is, of course, also a strong function of the flow present before the break (the initial core flow); at higher initial flows, this minimum is higher and early DNB is less likely to occur.

(2) the local power. The APRM Rod Block Line Technical Specification (RBLTS) limits maximum permissible core power as a function of core flow. Below a certain value of steady state flow (the exact value is plant specific), operation at full core power is not permitted. The RBLTS is a function of the total peaking factor in such a way that maximum local powers are also limited as a function of core flow. Therefore, at lower core flows, operation at full local power is not permitted. The lower the local power, the less probable is early DNB.

3.1.2 Positive Effects of Lower Flows on MAPLHGR (tends to decrease calculated PCT)

3.1.2.1 Earlier Reflood Due to Lower Core Power

As explained in section 3.1.1-(2), below a certain (plant specific) flow, operation at full core power is not permitted. At less than full core power, there is less steam generation due to vaporized core spray; this results in a smaller counter-current-flooding (CCFL) effect at the upper tie plate and consequently core spray water can more rapidly reach the lower plenum thereby causing earlier reflooding of the high power plane. This earlier reflooding results in a lower PCT.

3.1.2.2 Lower Local Power

As explained in section 3.1.1-(2), below a certain steady state (plant specific) flow, operation at full local power is not possible. Therefore, below this value of flow, lower local powers decrease PCT due to the lower stored heat present.

3.2 Net Effect of Lower Flow on DB-LOCA

For most plants, the positive effects described in section 3.1.2 more than compensate for the negative effect described in section 3.1.1 and no additional MAPLHGR limits are needed. However, for each class of plant it must be determined by calculation (or by comparison to similar plants) whether the combination of all of the above effects is positive (no additional restrictions) or negative (additional restrictions for low flow operation needed). Such determination for each operating class of JP-BWR (i.e. BWR/3 and BWR/4) is described in this section.

3.2.1 BWR/3 Low Flow Restrictions

3.2.1.1 Vessel I.D. >250 inches

These plants have a smaller "effective maximum break size" than the BWR/3 plants with vessel ID's <250 inches due to the larger vessel size. For steady state core flows above 61% of full rated flow these plants do not experience early DNB following a DB-LOCA. Below 61% of full rated flow, decreasing core and local power effects more than compensate for early DNB (if it occurs) and therefore such operation (reduced core flow) is less restrictive than at full steady state flow where higher powers are permissible. The staff agrees with the GE conclusion that no new restrictions are required for BWR/3 plants with vessel I.D. >250 inches when operating at reduced steady state core flows.

3.2.1.2 Vessel I.D. <250 Inches

The 224 inch I.D. BWR/3 was found to be more noticeably affected by reduced steady state core flows and early DNB. Plants of this type, which have the largest "effective break size", experience a core flow reduction or "dip" immediately following the DB-LOCA break initiation. Due to the larger "effective break size" these plants can experience early DNB even with a relatively high initial steady state core flow. If DNB occurs earlier, a loss of about 5 seconds of nucleate boiling heat transfer time could result (which if considered independently from other effects, is worth about 100^{o}F in increased PCT or 5% reduction in MAPLHGR). Since DNB can occur at a relatively high initial steady state core flow with correspondingly higher core and local power, the "positive effects" previously discussed relative to reduced local and core power at reduced core flows would result in a minimal benefit.

We agree with the GE conclusion that the DB-LOCA results and MAPLHGR limits for BWR/3's are most affected by early DNB for the 224 BWR/3. Further, we agree with the results of specific GE calculations for that size plant which demonstrate that for flows <90% of rated flow, a 5% reduction in the MAPLHGR curves derived for 100% flow will assure that the plant is operated in compliance to 10 CFR 50.46 at those lower flows.

3.2.2 BWR/4 Low Flow Restrictions

The design flow per bundle for BWR/4 plants is typically about 20% above the flow per bundle in a BWR/3. Thus, early DNB effects, if they occur in a BWR/4, cannot occur at flows above a relatively low initial core flow (compared to a BWR/3). That is, the transient flow "dip" immediately following the DB-LOCA does not reach a sufficiently low value to cause DNB unless initial steady state core flow is below 70%. Below 70% flow, maximum core and local power is reduced and the resulting positive effects discussed earlier (section 3.1.2) would partially or wholly compensate for the early DNB, making the MAPLHGR penalty less severe for BWR/4's than it would be if DNB occurred at higher initial flows (at higher powers).

3.2.2.1 BWR/4's with ID >250 Inches, Without LPCI Modifications

This size BWR/4 has a relatively low "effective break size" due to the large vessel I.D. Consequently, this class of BWR/4's does not experience early DNB following a DB-LOCA until the steady state core flows are reduced significantly. Since the core and local power levels would be reduced significantly at core flows sufficient to reduce the time to DNB, reduced flow operation would not represent a more limiting condition than normal operation.

3.2.2.2 <u>BWR/4's With ID >250 Inches, With LPCI Modifications</u>

As stated in Section 3.2.2.1, this size plant has a sufficiently large vessel inventory such that early DNB would occur only at significantly reduced steady state core flows and that the reduced core and local powers would more than compensate for the negative effects of earlier DNB.

The DB-LOCA for this type plant results in almost equivalent PCT's for the largest suction and discharge line breaks. However, since there are no net negative effects for either break due to earlier DNB at reduced flow conditions, the limiting break size and location and the present MAPLHGR limits need not be changed for low flow operations.

3.2.2.3 <u>BWR/4's with I.D. <250 With LPCI Modification</u>

All "LPCI modified" plants have two independent LPCI systems, one piped to a fixed point in the "A" recirculation loop discharge line, the other to a fixed point in the "B" recirculation loop discharge line. For the largest (suction line) break, closure of the broken recirculation loop's discharge valve isolates the LPCI injection point from the break so that the occurrence of the break does not disable the broken loop's LPCI system. Consequently, even with a single assumed failure, the largest (suction line) break analysis can assume credit for reflooding flow from at least one LPCI system. In comparison, although the discharge line break is smaller (flow must pass through the limiting flow area of either the jet pump nozzles or the recirculation pump throat to reach the break), no LPCI credit can be assumed for the discharge line break since the broken loop's system cannot be isolated from the break and the other system can be "single failed". The net result of these opposing effects (i.e. larger break with more ECCS, or smaller break with less ECCS) is that all LPCI modified plants (except certain 251 inch I.D. plants) show the smaller discharge break as PCT limiting, with the PCT about 200^{o}F higher than for the suction line break at 100% initial flow.

Low flow effects might influence the suction line PCT more than the discharge line PCT, which could cause the suction line break to become limiting at lower core flows (and therefore additional low flow limits might be required). This is not the case, however, as explained below.

The more severe flow "dip" experienced for the larger suction line break causes it to experience early DNB at a higher initial core flow (relative to the smaller discharge break) and with corresponding by higher core and local power (i.e. lesser positive effects to offset the early DNB negative effect). GE has shown that the "low flow" effects cause the suction line break PCT to approach the discharge line break PCT, but that such effects (typically loss of less than 10 seconds of nucleate boiling, worth at most a 150^{o}F PCT increase) are not sufficient to cause the suction line break to be limiting. Instead of the discharge line break PCT being 200°F above the suction line PCT (as at 100% flow), the difference decreases to about 50°F at the worst low flow condition, and the discharge break remains limiting by this amount (about 50°F).

3.2.2.4 BWR/4's With ID <250 Inch Without LPCI Modification

These plant sizes have the largest "effective break sizes" of all the BWR/4's discussed above. This is due to the small vessel water inventories and the large DB-LOCA limiting suction line break area.

GE has determined, and we agree, that the 218 inch I.D. BWR/4 without LPCI modification is the limiting case for this category of plants. Based on previous discussion, the smaller BWR/4's might be expected to be more limiting. However, these smaller plants include only: (1) the 205 inch I.D. size plant (LPCI modified and therefore already discussed above); and (2) the 183 inch I.D. non-LPCI-modified size, which has a smaller recirculation pipe diameter and does not experience early DNB at sufficiently high initial core flows (where core and local power can be higher) to cause reduced flow operation to be more limiting than the 218 inch I.D. BWR/4 plants without LPCI modification.

Typical plant specific calculations have shown that the 218 inch I.D. BWR/4 without LPCI modification (the worst "low flow" BWR/4 size) will be conservatively within the requirements of 10 CFR 50.46 if the 100% flow MAPLHGR limits are multiplied by a factor of 0.95 when operating below 70% flow. The staff agrees with that conclusion.

3.3 Low Flow Effects on Smaller Breaks (LOCA break areas >1.0ft^2)

The negative and positive effects of low core flow on the DB-LOCA discussed in sections 3.1.1 and 3.1.2 respectively also apply for break sizes other than the DB-LOCA. However, additional negative and positive effects must be considered for small breaks. These additional effects are discussed in sections 3.3.1 and 3.3.2 below, and the conclusions regarding the net sum of these effects for lower flow operation of operating JP-BWR's is given in section 3.3.3.

3.3.1 Negative Effects of Reduced Core Flow Operation for Break Sizes Smaller than the DB-LOCA but Above 1.0ft^2

3.3.1.1 Potential for Loss of Available Nucleate Boiling Time

At 100% rated flow conditions, small break analyses have indicated that flow decay is less rapid and that continued nucleate boiling is predicted for an extended period. This longer nucleate boiling period provides for the removal of additional stored energy which reduces the calculated PCT for breaks smaller than the DB-LOCA.

Any reduction of initial steady state core flow will act to reduce the time to DNB and will contribute to the "negative effects" when the benefit of extended nucleate boiling is removed.

3.3.1.2 Longer Adiabatic Heatup Time

For smaller breaks, it takes longer for reactor pressure to decay to a level which would permit the core spray and LPCI pumps to supply ECCS flow to the core. This results in a longer time between high power plane uncovery and initiation of spray cooling, i.e. in the time available for decay heat to raise the PCT. This effect is primarily dependent on break size and not on initial core flow. Only negligible differences exist (due to different average void fraction) at different initial core flows.

3.3.2 Positive Effects of Reduced Core Flow Operation for Break Sizes Smaller than DB-LOCA but Above 1.0ft^2

3.3.2.1 Flow Reduction Necessary to Experience Early DNB

As break size is reduced below the DB-LOCA, lower and lower initial core flows have to be assumed before early DNB is predicted. This is due to the fact that the flow "dip" following the break is less severe following smaller size breaks. Therefore a lower initial flow has to be assumed before the absolute value of flow at the minimum point in the "dip" is sufficiently low to cause early DNB.

At the lower flows, the RBLTS limits maximum core and local power and this lowers PCT as discussed previously.

3.3.2.2 Later High Power Plane Uncovery

As break size decreases, water level remains above the high power plane for a longer period following the break. This positive effect is not significantly changed by initial steady state core flow, showing only a negligible effect due to changes in average void content at reduced flow. This later high power plane uncovery has two positive effects:

(1) longer pool boiling period. Following DNB, the "Appendix K" model assumes pool boiling heat transfer at the clad surface until the water level drops below the high power plane. The pool boiling coefficient is much lower than the nucleate boiling coefficient present before DNB (about 50 as compared to about 10,000, respectively), however, significant cooling occurs during this pool boiling period. As break size decreases, this pool boiling period is extended relative to the DB-LOCA due to the delayed high power plane uncovery. This is one of the reasons smaller breaks are less limiting than the DB-LOCA in GE JP-BWR ECCS analyses at 100% flow.

(2) later adiabatic heatup period. After water level falls below the high power plane, zero heat transfer is assumed until the plane refloods. The heatup rate during this period is largely influenced by decay heat, which decreases as a function of time following the break. Consequently for smaller breaks when uncovery occurs later, this decay heat is lower and the cladding temperature increases at a slower rate during the adiabatic heatup period.

3.3.3 Net Effect on PCT of Break Size at Lower Flow

Previous sections have demonstrated that, for the largest break (i.e. the DB-LOCA), the steady state core flow (or flow range) giving the highest peak cladding temperature has been identified (ie, in most, but not all cases, this is 100% flow). This section will show that when break size is varied and initial core flow is held constant at any fixed value, the highest PCT will be calculated for the largest break. Demonstration of the above statement will establish the objective of this section, which is to show that the worst combination of break size and initial flow has been considered in establishing the MAPLHGR limits for all JP-BWR's. The underlined statement has been demonstrated in several ways, as discussed below.

3.3.3.1 Quantitative Break Spectrum Calculation

For the 224 inch I.D. BWR/3, GE provided analyses of a break spectrum at 70% flow. The 70% flow case in the 224 inch BWR/3 was chosen because it represents the lowest value to which flow could be reduced without local power decreasing due to the effect of the RBLTS (as previously discussed). Also, 70% flow is sufficiently low that early DNB will exist for the smaller breaks analyzed. We note that many plants do not have such a flow range: that is, if flow is reduced low enough to cause early DNB for the smaller breaks, then it has been reduced below the minimum flow at which full local power operation is possible. Such cases would not be as limiting as the example presented.

The break spectrum calculational results presented for the 224 inch BWR/3 showed the same effect on PCT at 70% flow as at 100% flow, which was a 220°F decrease in PCT for a variation in break size from 1.0 to 0.6 times the DB-LOCA. Further qualitative arguments which show that low flow effects do not cause a smaller break to become limiting at any initial core flow for any other type of BWR, are presented below.

3.3.3.2 Qualitative Break-Spectrum-at-Low-Flow Discussion

Of the two effects presented in Section 3.3.1, which tend to raise the PCT for breaks smaller than the DB-LOCA, only one (Section 3.3.1.1, "potential loss of more nucleate boiling time than DBA") is a strong function of initial core flow. The other (section 3.3.1.2, "longer adiabatic heatup time") is a function of break size and is only negligibly affected by initial core flow (as already stated in 3.3.1.2). Therefore any significant penalties for smaller breaks associated with the latter (3.3.1.2) are already taken into account in the 100% flow break spectrum, and those penalties would be essentially the same for any break spectrum at lower flow. Therefore the only penalty due to low flow operation remaining to be considered when comparing a low flow break spectrum to the 100% flow spectrum is 3.3.1.1, "potential loss of more nucleate boiling time than during the DB-LOCA".

Of the effects presented in Section 3.3.2 which tend to lower the PCT for breaks smaller than the DB-LOCA only one (Section 3.3.2.1, "flow reduction necessary to experience early DNB") is related to core flow. Any significant benefits of the other effects on the smaller break (Section 3.3.2.2, "longer pool boiling period" and "later adiabatic heatup period") have already been taken into account by the 100% flow break spectrum, and would be essentially the same when comparing any break spectrum at a lower flow to the 100% flow break spectrum.

The evaluation of the JP-BWR type plants at 100% of rated flow, using the GE-LOCA-Appendix K evaluation model, has shown that the DB-LOCA for all plants is the limiting break size (highest PCT) by a considerable "margin" over the smaller break sizes. To determine if the DB-LOCA remains the limiting break for lower steady state core flows, it is necessary to evaluate the trade-off between the single flow-dependent negative effect (Section 3.3.1.1) and the single flow-dependent positive effect (Section 3.3.2.1). All other effects, having been previously addressed, have been included in the 100% flow break spectrum and as stated above, would have the same net effect on a break spectrum at any steady state core flow.

GE argues that the potential for "loss of more nucleate boiling" (3.3.1.1) may in some cases be a slightly larger effect than the partially compensating power decreases associated with reduced initial flows (low enough to cause early DNB for the smaller break sizes), thereby raising smaller break PCT relative to the DB-LOCA . However, GE concludes that in no case would the combination of these two opposing effects be such as to eliminate the "margin" present in the break spectrum from which the MAPLHGR limits are derived (i.e. 100% flow), and although break spectra at certain fixed lower flows may show "smaller break" PCTs more closely approaching the DB-LOCA PCT, in none of these lower flow spectra will the smaller break's PCT become higher than the DB-LOCA PCT.

3.3.3.3 Conclusions Regarding Low Flow Break Spectra

Considering the quantitative example presented, the margin present between the DB-LOCA PCT and the smaller breaks' PCT, and the qualitative arguments presented, we agree with the above stated GE conclusion. We therefore also agree that the worst combination of break size and initial flow has been considered in establishing the MAPLHGR limits for all JP-BWR's, which logically follows as explained in the underlined portion of the first paragraph of Section 3.3.3.

Adding further confirmation to the above conclusion is a discussion presented by GE in a different context concerning effects on break spectra due to early boiling transition. This discussion was presented in reference 5, One Recirculation Loop Out of Service. The problem associated with operation with one loop out of service involves lack of coastdown flow from the unbroken loop and resulting early DNB. To conservatively bound these effects, GE assumed loss of nucleate boiling (DNB), for all break sizes, at 0.1 second following the break. This assumption is more conservative than the results presented in this "low flow" SER for two reasons: (1) In this SER, for a given initial core flow, the break spectrum would contain effects due to early DNB only on the larger breaks. (Recall that in the break spectrum for any given initial steady state core flow, below a certain size break, transient flow at 1.0 sec following the break will not cause early DNB). (2) Early DNB in this SER typically occurs at about 0.3 sec following the break, not the more restrictive 0.1 second assumed in reference 5.

Explaining the resulting break spectrum with this more restrictive, arbitrarily assumed early DNB, GE states:

> "To further understand the relative core heatup for a large and small break (for single-loop operation) consider the following. Immediate (0.1 second) loss of nucleate boiling is assumed independent of break size. Thus, the initial temperature response is identical for breaks of different sizes. The larger break uncovers earlier and therefore it

has a higher temperature after the time of uncovery than the small break. Very late in the transient, the later spray initiation for the case of the smaller break causes the temperature difference between the large and small breaks to be reduced. However, reflooding for the smaller break occurs at early enough times such that the larger break has the higher temperature."

3.4 Low Flow Effects on Smallest Breaks (Less than 1.0 ft^2)

For breaks smaller than 1.0 ft^2, GE states that they have performed calculations showing that nucleate boiling persists until the two phase level drops below the high power node. This is true at 100% flow, and GE states that "this assumption will be valid at all flows". We do not find it necessary to agree or disagree with the above quoted GE statement, but we concur with GE's further argument that, for these smallest breaks, early DNB will be predicted only for very low initial core flows where lower core and local powers (as previously discussed) will compensate sufficiently for the early DNB to prevent the smaller breaks from becoming limiting.

3.5 Other Break Locations

We agree with the GE statement that the core spray, feedwater, and steamline breaks "will remain essentially unchanged at low core flows because the margin to boiling transition is very great for these breaks and they are not subject to the short-duration flow 'dip' characteristic of recirculation line breaks at low flow."

3.6 <u>Early DNB at Lower Power Axial Planes Above the High Power Plane</u>

All of the previous discussion has concerned early DNB at the high power plane. However it is known that DNB occurs first near the top of the core where more voiding exists and "penetrates" downward into the core at later times. The NRC staff expressed the concern that a higher axial plane might be at only slightly lower power but might experience DNB significantly sooner, thereby having the highest PCT. GE stated(4) that DNB "jumps" from spacer to spacer, due to the flow disturbances caused by the spacer. Therefore the closest DNB could approach the high power plane without reaching the high power plane would be one spacer away, where power is typically 15% lower. Since 15% power decrease translates roughly to 300°F PCT decrease (at 20°F per 1% power), if early DNB at the higher plane causes it to be limiting, such early DNB would have to cause a PCT increase of about this magnitude. However, PCT sensitivity to changes in time of DNB at the high power plane (worse than the lower plane now being discussed) is about 100°F increase for 4 to 5 seconds' earlier DNB, 150°F for 10 sec, 175°F for 15 sec, etc. It is therefore argued that the potential losses (PCT increase) due to early DNB (which at most would be about 10 seconds earlier) cannot override the gains due to lower power. We conclude that lower power planes above the high power plane might more closely approach the high power plane's PCT at lower initial core flows, but they will not become limiting.

3.7 Single Failures

By far the most significant effect on PCT of loss of any JP-BWR ECCS equipment is the effect on reflood time. Any effects of the equipment's loss on heat transfer coefficients, etc. would be of minor, secondary importance. We therefore concur with the GE statement in reference 2:

> The single failure analysis required by Appendix K is conducted to determine that combination of ECCS pumps which results in the least favorable reflooding time. As shown in..., the reflooding time decreases slightly with decreasing core flow. The depressurization rate of the reactor is essentially unaffected by initial core flow, so ECCS injection times and flow rates are little affected. All of these effects are small compared to the large differences in reflooding time caused by various postulated single failure conditions. As a result, the single failure analysis conducted in the 100% flow analysis is valid at lower core flows.

4.0 CONCLUSIONS

We conclude that analysis results of the most limiting combination of break size, location, single failure, and core flow will be conservatively included in the MAPLHGR limits of any operating General Electric Co. BWR whose Technical Specifications are amended to include the applicable portion of the table below. Those General Electric Company BWR plants that have previously been found to be in conformance to all requirements of 10 CFR 50.46 will continue to be in such conformance including low flow effects when such applicable portion of this table is included in the plant's Technical Specifications.

Plant	Multiplier on MAPLHGR Based on 100% Core Flow	Core Flow Below Which Multiplier is to be Applied
All BWR/3's with vessel I.D. <250	0.95	90%
183 and 218 BWR/4, without LPCI modification	.95	70%
All other BWR's	None	Not applicable

Further, we recommend that the Technical Specifications Bases of Millstone Unit #1, Monticello, and Pilgrim Unit #1 be changed to reflect the fact that low flow ECCS analyses assumed an initial MCPR of 1.24. Therefore, if at any time in the future those plants wish to operate with a new operating limit MCPR (OL-MCPR) less than 1.24, a new ECCS Appendix K analysis will be required assuming the new OL-MCPR.

These changes would be accomplished for Monticello by issuance of the proposed amendment and we therefore find the amendment acceptable.

4.1 ENVIRONMENTAL CONSIDERATION

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

4.2 CONCLUSION

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

Date: May 24, 1977

REFERENCES

1. Minutes of meeting at San Jose, California, on July 27, 1976 on "ECCS Model Application to BWR/3 and BWR/4 Reactors at Less than Full Core Flow", August 16, 1976.

2. Letter from A. Levine, (GE), to Z. Rosztoczy, (NRC), "Additional Effects of Core Flow on ECCS LOCA Analysis", August 24, 1976.

3. Letter from A. Levine, (GE), to Z. Rosztoczy, (NRC), "Additional Effects of Core Flow on ECCS LOCA Analysis", November 4, 1976.

4. Personal Communication, J. Leatherman and O. Rao, (GE), to R. Woods, (NRC), (numerous telephone conversations).

5. Letter to Director, NRR, from A. J. Levine, GE, November 30, 1976, One Recirculation Loop Out of Service, with attachment proposed new Section II.A.7 for NEDO-20566.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 27 to Provisional Operating License No. DPR-22, issued to Northern States Power Company (the licensee), which revised Technical Specifications for operation of the Monticello Nuclear Generating Plant (the facility) located in Wright County, Minnesota. The amendment is effective as of its date of issuance.

The amendment modified the existing Monticello Technical Specifications to incorporate reduced Maximum Average Planar Linear Heat Generating Rate (MAPLHGR) limits at reduced core flow. These changes were in response to revised General Electric analyses which had been acknowledged by the licensee and acted upon voluntarily.

The filing by the licensee 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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the licensee's filing dated November 15, 1976, (2) General Electric Company's letters dated August 24, 1976, and November 4, 1976, (3) Amendment No. 27 to License No. DPR-22, and (4) the Commission's related Safety Evaluation. 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, Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A single copy of items (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 24th day of May, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors