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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NO. DPR-22

(CHANGE NO. 22 TO THE TECHNICAL SPECIFICATIONS)

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

Northern States Power Company (NSP) has proposed to operate the Monticello Nuclear Generating Plant with additional 8 x 8 fuel assemblies as requested in its application dated August 4, 1975, using:

(1) Modified operating limits based on an acceptable emergency core cooling system (ECCS) evaluation model that conforms with Section 50.46 of 10 CFR Part 50 as requested in NSP's application dated August 4, 1975 and supportive filings dated August 20, 1974, July 9, 1975 and September 16, 1975.

(2) Operating limits based on the General Electric Thermal Analysis Basis (GETAB) as requested in NSP's application dated March 12, 1975 and supplement dated July 10, 1975.

Since proposed changes No. 3 and 4 as described in the March 12, 1975 application are not directly related to GETAB or ECCS analysis, they will be considered at a later date.

2.0 EVALUATION

A. Emergency Core Cooling Systems

1. Conformance to all Requirements of Appendix K to 10 CFR 50

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that "...the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, §50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications as may be necessary to implement the evaluation results.

On July 9, 1975 the licensee submitted an evaluation of the ECCS performance for Monticello.[1] An amendment requesting changes to the Technical Specifications for Monticello to implement the results of the evaluation was submitted on August 4, 1975.[2] The licensee incorporated further information relating to the details of the ECCS evaluation by reference to the Quad Cities Unit No. 2 submittal[3] on ECCS evaluation as an appropriate lead plant analysis to show compliance to the 10 CFR 50.46 criteria and Appendix K to 10 CFR Part 50. The Order for Modification of License issued December 27, 1974, stated that evaluation of ECCS cooling performance may be based on the vendor's evaluation model as modified in accordance with the changes described in the staff Safety Evaluation Report of the Monticello Nuclear Generating Plant dated December 27, 1974.

The background of the staff review of the General Electric (GE) ECCS models and their application's to Monticello is described in the Staff Safety Evaluation Report (SER) for that facility dated December 27, 1974 issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October 1974 and the Supplement to the Status Report of November 1974 which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also describes the various changes required in the earlier GE evaluation model. Together the December 27, 1974 SER and the Status Report and its supplement describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Monticello evaluation which is covered by this SER properly conforms to the accepted model.

With respect to reflood and refill computations, the Monticello analysis was based on the modified version of the SAFE computer code, with explicit consideration of the staff recommended limitations, as described in the December 27, 1974 SER. The Monticello evaluation did not attempt to include any further credit for other potential changes which the December 27, 1974 SER indicated were under consideration by GE at that time.

During the course of our review, we concluded that additional individual break sizes should be analyzed to substantiate the break spectrum curves submitted in connection with the evaluation provided in August 1974. We also requested that other break locations be studied to substantiate that the limiting break location was the recirculation line.

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1). Monticello Nuclear Power Station LOCA Analyses Conformance with 10 CFR 50 Appendix K (Jet Pump Plant), July, 1975.

2). License Amendment Request Dated August 4, 1975, Monticello Nuclear Generating Plant.

3). Quad Cities Unit 2, Special Report No. 15, Supplement C, April 8, 1975, April 21, 1975 (proprietary), and July 21, 1975 (non-proprietary version of April 21, 1975 filing).

The additional analyses (performed on the lead plant, Quad Cities Unit No. 2 [3] and incorporated by reference) supported the earlier submittal which concluded that the worst break was complete severence of the recirculation line. These additional calculations provided further details with regard to the limiting location and size of break as well as the worst single failure for the Monticello design. The limiting break continues to be the complete severence of the recirculation line assuming a failure of the LPCI injection valve.

We have reviewed the evaluation of ECCS performance submitted by Northern States Power Company for Monticello and conclude that the evaluation has been performed wholly in conformance with requirements of 10 CFR 50.46 (a). Therefore, operation of the reactor would meet the requirements of 10 CFR 50.46 provided that operation is limited to the maximum average planar linear heat generation rates (MAPLHGR) of figures 3.11.1-A, 3.11.1-B, 3.11.1-C, 3.11.1-D and 3.11.1-E of the Northern States Power Company letter dated August 4, 1975 [2], and to a minimum critical power ratio (MCPR) greater than 1.18.

However, certain changes must be made to the proposed Technical Specifications to conform with the evaluation of ECCS performance. The largest recirculation break area assumed in the evaluation was 3.9 square feet. This break size is based on operation with a closed valve in the equalizer line between the two recirculation loops. Therefore, the Technical Specifications have been modified to limit reactor operation for a period not to exceed 24 hours unless the valve in the equalizer line is closed. This change was discussed with and found acceptable by the licensee.

The ECCS performance analysis assumed that reactor operation will be limited to a MCPR of 1.18. However, a more restrictive technical specification limits operation of the reactor to a MCPR of 1.33 for 7 x 7 fuel and 1.41 for 8 x 8 fuel based on consideration of a turbine trip transient with failure of bypass valves.

The Technical Specifications have been modified to require the licensee to report as a reportable occurrence, operation in excess of the limiting MAPLHGR values even if corrective action was taken upon discovery. The change was discussed with and found acceptable by the licensee.

An evaluation was not provided for ECCS performance during reactor operation with one recirculation loop out of service. Therefore, continuous operation in excess of 24 hours under such condition will not be permitted until the necessary analyses have been performed, evaluated and determined acceptable.

The steamline break accident analysis as presented by the licensee (by reference to Quad Cities Unit 2 [3]) is acceptable based on our generic review of NEDO-20360.[4]

2. Technical Specification Changes to Implement Conformance to Appendix K to 10 CFR 50

The proposed Limiting Conditions of Operation present two limitations on power distribution related to the LOCA analysis. These are the limiting assembly MAPLGHR and MCPR. The MCPR value used in the LOCA analysis was 1.18 and this value is less than the value determined from the transient analysis which has been incorporated in the proposed Technical Specifications. The bases for establishing the limiting value of MAPLHGR are indicated above in Section 2.0.A.1.

The licensee did not include the equalizer line area in the LOCA analysis, therefore, the Technical Specifications will require that the equalizer line valves remain closed at all times during reactor operation. The LOCA analysis did not address one loop operation, therefore the Technical Specifications will not allow continuous operation with one loop out of service.

The LOCA analysis assumed all Automatic Depressurization System (ADS) valves operated for small line breaks with HPCI failure. Therefore, the Technical Specifications will not permit continuous operation with any ADS valve out of service except as with other ECCS equipment one valve may be out of service for 7 days.

3. Conclusions Regarding Conformance to all Requirements of Appendix K to 10 CFR 50

On the basis of our review of the information provided by the licensee for Monticello, we conclude that the safety analyses are acceptable with respect to conformance with all requirements of paragraph 50.46 of 10 CFR Part 50 after the referenced MAPLHGR and MCPR technical specification changes are incorporated.

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4). Status Report on the Licensing Topical Report "General Electric Boiling Water Reactor Generic Reload Application for 8 x 8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, April, 1975.

B. General Electric Thermal Analysis Basis (GETAB)

1. Evaluation of GETAB-Based Technical Specifications

The GE generic 8 x 8 fuel reload topical [5] describes the thermal-hydraulic methods used to establish the thermal margins. However, based on our review of this topical we have found the GETAB application description to be incomplete. Therefore, we have evaluated the Monticello thermal margins based on the NEDO-10958 report [6] which the staff has previously found to be acceptable and plant specific input information provided by the licensee in its application dated March 12, 1975, as supplemented by NSP letters dated July 10, 1975 and July 24, 1975

The fuel cladding integrity safety limit MCPR for both the 8 x 8 and 7 x 7 fuel is 1.06. It is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are expected to avoid boiling transition. The uncertainties in the core and system operating parameters and the GEXL correlation (Table 4-1 of NEDO-20694) [7] combined with the relative bundle power distribution in the core form the basis for the GETAB statistical determination of the safety limit MCPR. The bases for these uncertainties are reported in NEDO-20340 [8] and are acceptable. The bundle power distribution used in the GETAB analysis conservatively assumes more high power bundles than would be expected during operation of the reactor.

In comparing the tabulated lists of uncertainties for Monticello with those in NEDO-10958 we have found only one difference. The Monticello standard deviation for the TIP readings uncertainty is 8.7% whereas the GETAB NEDO-10958 report shows 6.3%. The increase in uncertainty for Monticello is a consequence of the increase in uncertainty in the measurement of power in a reload core. A TIP reading uncertainty of 6.3% would be applicable if this were the initial core. In both cases the TIP reading uncertainties are based on a symmetrical planar power distribution and are acceptable.

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

6). "General Electric BWR Thermal Basis (GETAB): Data, Correlation and Design Application," NEDO-10958, 73NED9, Class I, November, 1973.

7). "General Electric BWR Reload No. 3 Licensing Submittal for Dresden Unit 3," NEDO-20694, December, 1974.

8). "Process Computer Performance Evaluation Accuracy," and Amendment 1, NEDO-20340 and NEDO-20340 1, dated June, 1974 and December, 1974.

The bypass flow has been considered in the determination of the MCPR limit. Finger springs have been attached to the lower end fittings of the reload fuel to maintain the core bypass flow within the range of the bounding analysis. In the bounding analysis, 12% bypass flow is assumed. The uncertainty of this bypass flow is factored in the total core flow uncertainty that is used in the GETAB analysis.

The operating limit MCPR is based on the most limiting transient, a turbine trip without bypass from 90% power and 100% flow conditions. The calculated decrease in MCPR during this transient is 0.27 for 7 x 7 fuel and 0.35 for 8 x 8 fuel. The resulting operating limit MCPR is 1.33 for 7 x 7 fuel and 1.41 for 8 x 8 fuel.

The required operating limit MCPR is a function of the magnitude and location of the axial and rod-to-rod power peaking. In determining the required MCPR, axial and local peaking representative of beginning-of-cycle were assumed. That is, R-factors of 1.075 for 7 x 7 fuel and 1.102 for 8 x 8 fuel and an axial peaking factor of 1.57 at a point 1/4 of the heated length below the top of the fuel were assumed. This is the most adverse set of local and axial peaking factors. During the cycle the local peaking, and therefore the R-factor, is reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% by the reduced end-of-cycle R-factor, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane.

2. Conclusions Regarding Acceptability of GETAB-Based Technical Specifications

The APRM scram and rod block setting changes suggested in Mr. Mayer's July 10, 1975 letter to D. L. Ziemann are not part of the GETAB-GEXL changes. A definitive stability analysis has not been presented for the APRM scram and rod block setting changes so these changes cannot be accepted at this time. However, the GETAB-GEXL changes are well documented and are highly desirable in view of the much improved data base for the GEXL over that for the previously used Hench-Levy MCHF correlation. The proposed technical specification changes for incorporating the GETAB-GEXL analysis are acceptable.

3.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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: OCT 30 1975

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

OCT 21 1975

Docket No. 50-263

Dennis Ziemann, Chief, Operating Reactors Branch 2, DRL

NEGATIVE DECLARATION FOR TECHNICAL SPECIFICATION CHANGES TO MONTICELLO UNIT 1

Attached is the Environmental Impact Appraisal and Negative Declaration associated with proposed changes in Technical Specifications appropriate to implementation of the ECCS Acceptance Criteria.

RB [illegible] for
Wm. H. Regan, Jr., Chief
Environmental Projects Branch 4
Division of Reactor Licensing

Enclosures:
1. Environmental Impact Appraisal
2. Negative Declaration

Issued w/Amend. 14 / Ch. 22 on 10/30/75

Date for 10/23/75 [illegible] Hood because he is acting chief today but not 10/21/75 [illegible] 10/24/75

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