APR 8 1974

Docket No. 50-263

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

Gentlemen:

A copy of the staff Safety Evaluation relating to your request dated November 19, 1973, as supplemented, to use reload 2 fuel assemblies containing 8 x 8 fuel rod arrays is enclosed for your information. The evaluation reflects the staff's review of the expected performance of the General Electric 8 x 8 fuel bundles in the Monticello Nuclear Power Station. Technical specification changes related to the reload 2 fuel assemblies will be made before electric power production is resumed when other changes resulting from NSSS modifications are to be made.

Sincerely,

/S/

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosure:
Safety Evaluation

cc w/enclosure: See next page

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cc & cy sent to NSP by Reg 4/18/74

Conv - Pete Kinsey, OGC

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SURNAME ➤	JJShea:sjh	RMDiggs	DLZiemann	VStello	JGallo	DJSkovholt
DATE ➤	4/4/74	4/4/74	4/4/74	4/9/74	4/8/74	4/8/74

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

8 x 8 FUEL ASSEMBLIES

INTRODUCTION

Northern States Power Company (NSP) has submitted[1,2] a description and safety analysis of 8 x 8 (64 rods) reload fuel assemblies and a segmented test rod (STR) assembly[3,4] which will replace 116 7 x 7 (49 rods) fuel assemblies (full core contains 484 assemblies) during the spring 1974 refueling outage. All of the remaining 44 temporary control curtains are to be removed during the refueling outage. Fuel assemblies containing gadolinia have been authorized[8] by the Directorate of Licensing for use in 20 reload 1 fuel assemblies that were inserted[6] into the Monticello core during the spring 1973 refueling outage. Therefore, the use of gadolinia in the 8 x 8 fuel assemblies is not a new feature. The outside dimensions of the fuel assembly remain unchanged.

The principal differences between the 7 x 7 and 8 x 8 fuel assemblies, in addition to the greater number of fuel rods in the 8 x 8 fuel assemblies, are:

1. The average uranium enrichment which is 2.62% is higher for the 8 x 8 fuel assemblies.

2. The 8 x 8 fuel rods are smaller in diameter than the 7 x 7, but the clad thickness is greater; i.e., 0.034 inch vs 0.032 inch clad thickness.

3. The 8 x 8 assembly contains an internal asymmetric water-filled spacer capture rod.

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4. The 8 x 8 fuel rod array has 13% more heat transfer surface than fuel assemblies previously used in Monticello.

5. The water to fuel volume ratio is higher for the 8 x 8 reload fuel assemblies than the 7 x 7 fuel assemblies.

6. The 8 x 8 fuel assembly contains only 94% of the total uranium in the original 7 x 7 fuel assemblies, but 10.5% more U-235.

NSP also submitted(3,4) proprietary information related to the design features of an 8 x 8 segmented test rod (STR) assembly for the Monticello core. The fuel assembly, according to the NSP submittal, satisfies the same design and damage criteria as the fuel assemblies that have been used in Monticello and the reload 2 fuel assemblies that are to be inserted in the Monticello core during April 1974.

EVALUATION

Since there are 63 UO_2 rods in the 8 x 8 fuel assembly compared with 49 in the 7 x 7 fuel assembly, the average fuel rod linear heat generation rate in an average 8 x 8 fuel assembly at rated core power level (1670 MWt) will be reduced from 5.8 kW/ft to 4.57 kW/ft. The combination of reduced power per rod and increased heat transfer surface will result in lower fuel rod temperatures which may result in some improvement in fuel rod integrity. Peak fuel clad temperatures following the design basis loss-of-coolant accident (LOCA) for the entire spectrum of breaks calculated in accordance with AEC Interim Acceptance Criteria (IAC) with allowance for densification are acceptably below the 2300°F limit. Metal-water reactions following a LOCA are similarly calculated to be less than 0.2%, well within the AEC-IAC limits. Although the average fuel temperatures will be lower because of more surface and more fuel rods, the capacity to store heat is reduced about 6% due to the 6% reduction in the amount of fuel. This effect, however, is partially offset by the 17% increase in the amount of zircaloy clad. The combined effect on fuel heatup during adiabatic conditions has been considered in the calculations. As noted, peak clad temperatures and metal-water reaction following the postulated design basis LOCA remain within the IAC limits with larger margins than calculated using the same methods for the 7 x 7 fuel assemblies.

With regard to the postulated design basis refueling accident, the fission product inventory of interest is about the same for the 8 x 8 fuel as the 7 x 7 fuel. The lower 8 x 8 fuel temperatures could reduce the amount of noble gas and halogen activity that collects in the fuel

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rod plenums and is available for release if the cladding is damaged, but the amount would be negligibly small. If the noble gas and halogens activity in the fuel plenums is conservatively assumed to be the same for the 8 x 8 and 7 x 7 fuel assemblies, the fission products released to the water during the design basis refueling accident are unchanged and the fission product release to the environs, therefore, remains unchanged. Since the 8 x 8 fuel rod cladding is thicker and because there are more rods to absorb the fall shock, a lower percentage of the 8 x 8 fuel assembly rods may be damaged and the release would be correspondingly lower than that assumed. We have concluded that the consequences of the postulated design basis refueling accident will not change significantly because of the change to 8 x 8 fuel assemblies.

The release of radioactive materials to the environment as a result of a postulated main steam line break outside of containment is governed by the mass flow rate of steam from both open ends of the break. The steam flow rates are limited by critical flow conditions at steam line flow restrictors. Since the source of radioactivity, steam flow rates following the assumed pipe rupture, and valve closure times are unaffected by the change to 8 x 8 fuel assemblies, we have concluded that the consequences of the postulated design basis steam line break accident are unchanged from those previously evaluated.

The postulated design basis control rod drop accident is based upon a maximum rod worth of 1.3% delta k/k causing fuel damage to the extent that (1) peak fuel enthalpy during the accident is no more than 280 cals/gm, the conservatively assumed value for prompt energy deposition into water, and (2) all fuel above 170 cals/gm causes clad failure. For these assumed accident conditions, the release of fission products into the primary coolant would be about the same. We have concluded that the release of radioactive material to the primary coolant due to a postulated rod drop accident is about the same for 8 x 8 fuel as for 7 x 7 fuel assemblies although comparison shows that the maximum rod worth is further below 1.3% delta k/k limiting condition and the consequences of an assumed rod drop accident with 8 x 8 fuel assemblies instead of 7 x 7 fuel assemblies, therefore, would be reduced.

The increased enrichment of the 8 x 8 fuel assembly is not a result of changing fuel. Such an increase would have been required if 7 x 7 fuel assemblies were used as replacement fuel. The increase in enrichment is required and had been planned to compensate for the change in core characteristics to an equilibrium core rather than a startup core; i.e., core constituted with partially depleted fuel rather than 484 new fuel assemblies and poison curtains. We have reviewed(5) the nuclear design of the 8 x 8 fuel assemblies, including the increased enrichment which is not unique for the 8 x 8 fuel assemblies, and have found the design to be consistent with previously approved fuel and, therefore acceptable.

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The increased clad thickness with smaller rod diameter and reduced power per rod will provide greater resistance to clad failure. Nevertheless, NSP has agreed(7) to participate with General Electric in a surveillance program to monitor the performance of a precharacterized 8 x 8 fuel bundle in the Monticello reactor beginning with cycle 3. We have concluded(5) that the cladding integrity of 8 x 8 fuel is acceptable and the effect of the unheated rod near the center of the bundle will not be significant.

The beneficial effects of additional heat transfer surface as well as the increased coolant flow resistance due to increased surface drag have been included in the thermal-hydraulic evaluation. We have noted(5) that in general the 8 x 8 fuel has greater thermal margins to design limits than the 7 x 7 fuel, indicating that improved heat transfer offsets the slight decrease in 8 x 8 assembly flow.

The increase in water to fuel volume poses no particular problem. Increases of this magnitude can be and have been accommodated. For example, the water to fuel ratio for reload 1 fuel inserted during the spring 1973 plant outage(1) is 2.53 in contrast to the initial core fuel ratio of 2.47 - approximately the same magnitude of the change from 2.53 to 2.60 for the 8 x 8 fuel water to fuel volume. We have concluded(5) that the nuclear design of the 8 x 8 assembly is acceptable.

The STR assembly(3,4), in addition to satisfying existing design criteria and quality assurance requirements and operating at a more restrictive linear heat generation level, will be removed from the core during each refueling outage to remove selected rods for destructive testing. Careful inspection before returning the bundle to the core will provide added assurance that the fuel performance is in accordance with design expectations.

The proposed operating characteristics for the 8 x 8 fuel assemblies; i.e.,

> *13.4 kW/ft maximum LHGR (Operating Limit), 45,000 MWd/Te maximum local exposure, and 4-6 years incore residence time,

are consistent with the design changes that have been identified and are indicative of continued improvement in quality assurance during fuel fabrication. The changes in the 8 x 8 fuel assembly design, including fuel pellet preparation, are relatively small and can be accommodated by existing calculational methods. The changes result from extensive performance analysis, inspection, and destructive examination of fuel rods that have been irradiated in nuclear power plants. We note that

*STR maximum LHGR 10.5 kW/ft

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the trend toward smaller diameter and more rods per bundle to improve fuel rod integrity and reliability is supported by the reactor operating experience at Big Rock Point. It should be noted, however, that if contrary to expectations the fuel integrity is reduced resulting in fission product release into the coolant, the existing technical specification limits on radioactive releases into the primary coolant and effluent releases to the environs will continue to determine the extent of allowable fuel degradation and provide the same level of protection to the health and safety of the public.

Coolant flow characteristics during normal and accident conditions are nearly the same since the flow cross sectional area for the 8 x 8 and 7 x 7 fuel assemblies is identical. The small decrease in flow due to the increased heat transfer surface in the 8 x 8 rods is less than 8% and is accounted for in the calculations that show an MCHFR of 2.3 for the 8 x 8 fuel assembly compared with 2.03 for the 7 x 7 fuel assembly; i.e., the 8 x 8 assembly has a larger thermal margin.

CONCLUSION

Based on our evaluation of the differences between the 8 x 8 fuel assemblies to be inserted during the April 1974 refueling outage and the 7 x 7 fuel assemblies, we have concluded that design margins for safety or fuel damage limits are larger for the 8 x 8 fuel assemblies than the 7 x 7 fuel assemblies used up to the present time. The fuel should be more resistant to failure during normal and accident conditions according to the design calculations. Inspection during each refueling outage of a precharacterized reload 2 fuel assembly and the STR assembly plus destructive tests of selected STR rods will provide timely information to confirm design performance expectations or identify deviations from predicted performance. Changes to the Technical Specifications to permit reactor operation above 1% power level with reload 2 fuel inserted will be made to reflect limits on the 8 x 8 linear heat power generation (13.4 kW/ft), maximum average planar LHGR as a function of fuel bundle exposure, and other changes related to control rod scram time (3.5 seconds instead of 5 seconds). Plant modifications that are in progress will be completed before the Monticello nuclear plant returns to power in May 1974. However, based on our evaluation of the 8 x 8 fuel assembly performance characteristics and the "Technical Report on the General Electric Company 8 x 8 Fuel Assembly" prepared by the Regulatory staff(5),

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we have concluded that the health and safety of the public will not be endangered by reactor operation with 8 x 8 fuel assemblies replacing 7 x 7 fuel assemblies in the manner described by NSP.

The staff is evaluating the acceptability of a new system concept for Monticello, the Prompt Relief Trip (PRT) that has been proposed by NSP to provide core protection. However, this system is not required to protect the reactor until the end of cycle and even then only if it is desired to extend the fuel cycle to the "all rods out" condition without programming power downward during this period prior to shutdown for refueling.

/S/

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

/S/

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

Date: APR 8 1974

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REFERENCES

1. NSP "Second Reload Submittal" dated November 19, 1973.

2. NSP submittal, "Supplemental Information to the Monticello Second Reload Submittal", dated February 8, 1974.

3. NSP submittal of NEDE-20179 - Monticello Segmented Test Rod Bundle Submittal (Proprietary Information) - dated December 14, 1973.

4. NSP submittal, "Change to NEDE-20179 Monticello Segmented Test Rod Bundle Submittal" (Proprietary Information) dated January 15, 1974.

5. AEC - Directorate of Licensing letter (February 11, 1974) to NSP regarding 8 x 8 reload fuel with the following enclosures:

   1. Federal Register Notice regarding authorization to operate Monticello with 8 x 8 reload fuel, including a segmented test rod assembly.

   2. Technical Report on the General Electric 8 x 8 Fuel Assembly by AEC Directorate of Licensing dated February 5, 1974.

6. NSP submittal, "Supplementary Information Regarding the First Monticello Reload", dated April 13, 1973.

7. NSP submittal, "Special Surveillance Program for 8 x 8 Fuel", dated February 14, 1974.

8. Directorate of Licensing authorization to operate Monticello Nuclear Power Plant with reload 1 fuel assemblies in core dated March 5, 1973.

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