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U S Nuclear Regulatory Commission
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PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket No. 50-282 License No. DPR-42

PRAIRIE ISLAND UNIT 1, PROPOSED IRRADIATION OF FUEL RODS BEYOND CURRENT LEAD ROD BURNUP LIMIT

Prairie Island Nuclear Generating Plant (PINGP) plans to irradiate a Westinghouse VANTAGE+ fuel assembly that will attain end of life rod average burnups ranging from about 52,000 to 75,000 MWD/MTU. Irradiation of these rods is intended to provide data on fuel and material performance that will support industry goals of extending the current fuel burnup limits, and will provide data to address questions related to fuel performance behavior at higher burnups. This fuel assembly has already been irradiated for two cycles in Prairie Island Unit 2 and currently has cumulative rod average burnups ranging from approximately 37,000 to 56,000 MWD/MTU. The assembly will be irradiated for one additional cycle in Prairie Island Unit 1.

As detailed in Attachment 1, the use of this fuel assembly will be fully evaluated as part of our normal reload design process, and it is expected that all design criteria will be satisfied. Lead Test Assembly (LTA) application has been endorsed by the Nuclear Regulatory Commission (NRC) in Section 5.0 of the NRC approved WCAP 12488-A. The proposed irradiation of this assembly to high burnup does not require any Technical Specifications changes.

The NRC approval of rod burnup limits affects the implementation of this proposed program. WCAP-12488-A, approved by the NRC, includes the process by which a lead rod burnup limit of 62,000 MWD/MTU may be attained for the Westinghouse fuel in the Prairie Island Plant. Since the high burnup fuel rods are planned to operate to burnup levels exceeding this limit, NRC approval is requested prior to the implementation of this program.

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The fuel assembly containing the rods that will be irradiated to high burnup is scheduled to be used in Prairie Island Unit 1 Cycle 21, which will begin operation in February 2001. In order to support the core reload design schedule for this cycle, we request NRC concurrence with this irradiation program by January 20, 2001.

In this letter we have made no new Nuclear Regulatory Commission commitments.

Please contact Jack Leveille (651-388-1121, Ext. 4142) if you have any questions related to this letter.

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Attachments: High Burnup Lead Test Assembly

High Burnup Lead Test Assembly

INTRODUCTION

PINGP plans to irradiate a Westinghouse VANTAGE+ fuel assembly containing ZIRLO fuel rods to high burnup. Irradiation of this fuel assembly will provide data on fuel and materials performance that will support industry goals of extending the current fuel burnup limits and will provide data to address Nuclear Regulatory Commission (NRC) questions related to fuel performance behavior at high burnups. The data will also help confirm the applicability of nuclear design and fuel performance models at high burnups.

The fuel assembly to be used in this program was fabricated by Westinghouse as part of the Prairie Island Unit 2 Cycle 16 reload. This was the first reload of VANTAGE+ fuel used at Prairie Island. The assembly was irradiated in Unit 2 cycles 16 and 17, then discharged with an assembly burnup of 52 GWD/MTU. The assembly, (T81), was part of a fuel inspection that was done in July 1997. A visual inspection, control rod drag testing, assembly growth, and oxide thickness measurements were taken at this time. The ZIRLO clad fuel rods have accumulated rod average burnups ranging from approximately 37 to 56 GWD/MTU. The fuel assembly will be irradiated for one additional cycle in the center of Prairie Island 1 Cycle 21. The end of cycle rod average burnups of the ZIRLO fuel rods are expected to range from 52 to 75 GWD/MTU. Irradiation of a single assembly in this manner will generate fuel performance data at high burnups with minimal impact on core operation.

The use of this fuel assembly will be fully evaluated as part of our normal reload design process, and all design criteria are expected to be satisfied. It will be confirmed that all design criteria are met prior to loading T81 into the core. In the event that T81 cannot meet the design criteria or is damaged during fuel handling another assembly has been identified as a replacement. This replacement assembly would have a negligible effect on the core loading pattern and would have a burnup that will remain below the licensing limit.

Based on our preliminary evaluation, no unreviewed safety questions will exist as a result of irradiating this assembly to high burnup in the Prairie Island 1 core. However, as the fuel rods will operate to burnup levels in excess of the lead rod burnup limit currently identified for the Prairie Island Units (References 1 and 8), NRC concurrence is requested prior to implementation of the program.

Westinghouse has performed the fuel rod design analysis for all fuel used in the Prairie Island 1 Cycle 21 reload design, including the demonstration assembly. Additional conservatism was used in the evaluation of the high burnup fuel rods since these rods will exceed the current lead rod burnup limit. The fuel rod design analysis of this assembly is documented separately, and shows that all fuel rod design criteria that are applicable for the current lead rod burnup limit of 62,000 MWD/MTU are also satisfied for the high burnup fuel rods (Reference 9).

BACKGROUND

In December 1993, the first reload of VANTAGE+ fuel was inserted into the Prairie Island Unit 2 Cycle 16. The burnup limit for this fuel is documented in Reference 1 and 8 , and currently remains applicable at both Prairie Island Units although it is not explicitly stated in the License Conditions or Technical Specifications for Prairie Island (Reference 7). The proposed irradiation of a small number of fuel rods to extended burnups at Prairie Island therefore requires NRC approval to exceed this restriction on lead rod burnup.

Pre- and Post- Irradiation Testing of Fuel Rods

The assembly, T81, was part of a fuel inspection that was done in July 1997. A visual inspection, control rod drag testing, assembly growth, and oxide thickness measurements were taken at this time. Corrosion data for the peripheral rods at approximately 52 GWD/MTU burnup showed peak oxide thickness of about 25 microns. Prairie Island is a low temperature PWR with a T_{ave} of approximately 560°F. The data was typical of ZIRLO performance at this burnup level. After the discharge of this assembly from cycle 21 a similar inspection is planned. Tentatively this would entail assembly length measurement, visual inspections for crud and the rod/nozzle gaps, and measurement of the oxide thickness on the peripheral rods and guide tubes. PINGP recognizes the importance of gathering data on fuel performance at high burnups and will be working with Westinghouse to support the proposed examinations.

SAFETY SIGNIFICANCE SUMMARY

The extended burnup of the ZIRLO fuel rods in Fuel Assembly T81 will be fully addressed as part of the Prairie Island 1 Cycle 21 Reload Safety Evaluation, using the Nuclear Analysis Department's, NRC approved, reload design methods and Westinghouse's approved fuel rod design models and methods. The fuel rods will satisfy all design criteria that are applicable for the current lead rod burnup limit. The existing analyses of record are expected to remain applicable. Likewise, there will be no impact on core operation, including setpoints. A final determination of whether an unreviewed safety question exists will be made after the cycle specific reload calculations are complete. NRC approval to exceed the lead fuel rod burnup limit imposed on PINGP is requested for this fuel assembly.

PROPOSED EXTENDED BURNUP PROGRAM

1. Description of Fuel Assembly

Fuel Assembly T81 is a twice-burned VANTAGE+,(Reference 2). The assembly has ZIRLO fuel rod cladding, guide thimbles and Zircalloy-4 mid grids. This assembly, like most of the VANTAGE+ assemblies used in Prairie Island includes 4.95 w/o U235 enrichments, gadolinia burnable absorbers, enriched annular axial blankets, a

removable top nozzle, and a debris resistant bottom nozzle. VANTAGE+ fuel at Prairie Island does not include intermediate flow mixing mid-grids. The assembly T81 contains twelve gadolinia rods enriched to 2.97 w/o U235 and containing 6 w/o gadolinia.

Currently, Fuel Assembly T81 is susceptible to top nozzle spring screw failure. Therefore, the top nozzle will be replaced with an improved design top nozzle prior to use in cycle 21.

2. Mechanical Design Evaluations

Westinghouse has performed the mechanical design assessment of Fuel Assembly T81. The history of the assembly and planned operating conditions in Prairie Island 1 Cycle 21 were considered. All current licensed fuel design criteria are expected to be satisfied, even when accounting for the end of life burnups.

2.1 Fuel Rod Design for the ZIRLO Rods

Westinghouse has assessed the fuel rod design criteria for all rods in Fuel Assembly T81, using their NRC approved models and methods. Calculations were performed as part of the normal reload design analysis to demonstrate that all fuel rod design criteria that are normally evaluated for reload fuel have been satisfied for the projected lead rod burnup levels. For the high burnup fuel rods in Assembly T81, the most limiting criteria is rod internal pressure.

A conservative corrosion model for ZIRLO was used to predict the cladding corrosion on the high burnup fuel rods. Use of this model demonstrated reduced margin to the fuel rod internal pressure limits, but the design criterion was satisfied. In addition, the upper bound steady state corrosion screening criteria was not exceeded, and thus the 10 CFR 50.46 total local oxidation limit of 17% is satisfied. Additional conservatisms are considered appropriate for assessing the fuel rod design criteria to the higher burnups that will be reached by this fuel assembly.

It should also be noted that at least one extended burnup program similar to the Prairie Island program is already in progress at another U.S. utility using Westinghouse fuel. The lead rod burnups in that program will be comparable to those in the ZIRLO rods in Assembly T81.

The performance of the high burnup fuel rods will continue to be assessed against the current fuel related Technical Specifications throughout the cycle. Specifically, the fuel will be required to meet the current reactor coolant activity limits. The impact on the current safety analyses will also be evaluated, as discussed in Section 6 below.

2.2 Fuel Assembly Design

The fuel assembly design for Assembly T81 is described in Reference 2. The measurements of T81 after two cycles of burnup to 52 GWD/MTU showed no unusual results when compared to the operating experience of other ZIRLO fuel assemblies. No unusual conditions exist that would affect the ability of the assembly to meet all mechanical design requirements, including areas such as: compatibility with all in-core fuel handling, and storage interfaces: grid impact strength; grid cell force and fretting wear resistance requirements; and fuel assembly growth allowances.

3. Thermal Hydraulic Design

Fuel assembly and core component pressure drops will not be affected by the fuel burnup of T81. The thermal hydraulic performance of Fuel Assembly T81 will therefore be performed in accordance with the Nuclear Analysis Department's normal reload design methodology (Reference 4), using NRC approved codes and methods. The fuel assembly will be required to meet the same design criteria as other fuel assemblies in the core.

4. Neutronic Performance

Consistent with Reference 3, a nuclear design evaluation will be performed for Prairie Island 1 Cycle 21 to demonstrate that the reload core will meet all applicable design criteria.

The physics parameters for the high burnup fuel rods will be determined using approved calculational methods and procedures, Present design calculations account for substantial levels of plutonium and fission products, and the current methods are expected to adequately treat the neutronics change associated with the extended burnup of the Assembly T81.

As only one assembly in the core will be operated to high burnup levels, the effect on the reload physics design for Prairie Island 1 Cycle 21 is expected to be relatively small. The fuel assembly will be located in the center of the core, and will operate at an assembly average power near 75% of the core average power throughout the cycle. The high burnup fuel rods will not be in the highest fuel rod power density locations in the core, and will not be limiting with respect to any safety analysis limit. If the effect on the design calculations is significant, it will be addressed by explicit calculations to ensure that the assembly is treated in a conservative manner.

5. Impact on Spent Fuel Pool

In general, higher burnup fuel is expected to have only a minor impact on evaluations for the Prairie Island spent fuel pool. High burnup fuel could impact both criticality calculations and calculations of the decay heat load.

Criticality calculations for the Prairie Island spent fuel pool currently take credit for the decrease in fuel reactivity with increasing burnup (References 5 and 6). The analyses

of record has credited the effect of burnup on reactivity to assembly burnups of 60,000 MWD/MTU. Credit for the decrease in reactivity for exposures beyond 60,000 MWD/MTU will not be taken for Assembly T81. The criticality analysis will therefore remain conservatively bounding for the Assembly T81.

With respect to possible impact on the spent fuel pool heat load, the major contributor to the heat load immediately after the core offload is decay heat from short-lived isotopes. These isotopes tend to reach an equilibrium condition during normal operation, and the concentrations do not change significantly with increasing burnup. Therefore the presence of one high burnup assembly will not affect the limiting case analysis currently described in the Prairie Island USAR.

A portion of the spent fuel pool heat load is due to long-term decay heat from assemblies stored in the spent fuel pool. This long-term decay heat load is caused by longer-lived actinides, which increase with burnup. Therefore, higher fuel burnups will slightly increase this contribution to the pool heat load. However, under the proposed program only one assembly will reach high burnup. The ultimate impact on this long-term decay heat contribution to the spent fuel pool heat load is therefore expected to be negligible. It is therefore concluded that the presence of this high burnup fuel assembly ,T81, will not affect the spent fuel pool heat load analysis currently described in the Prairie Island USAR.

6. Safety Evaluations

No fundamental change in the safety and accident considerations are anticipated as a result of the extended burnup of this assembly.

6.1 LOCA Analysis

Fuel temperatures and pressures used in the LOCA analysis are calculated by Westinghouse using their fuel performance code. Provided that the fuel rod design criteria are satisfied, the limiting conditions for the safety analyses occur near beginning of life, when fuel temperatures are at a maximum due to fuel densification. All fuel rod design criteria, including the criteria on rod internal pressure, will be satisfied for the extended burnup of this fuel assembly, in which case the limiting conditions for the input to the safety and LOCA analyses remain unchanged. Therefore, operation of these rods is not expected to impact the existing LOCA analysis for Prairie Island 1 Cycle 21.

6.2 Non-LOCA Safety Analyses

The potential impact of the high burnup fuel assembly on the Prairie Island non-LOCA safety analyses will be addressed as part of the Prairie Island 1 Cycle 21 reload safety analysis. As Westinghouse has shown (Reference 9) that all fuel design criteria are satisfied for this fuel assembly for the proposed operating conditions, the limiting fuel temperature inputs to the safety analyses will remain unchanged.

The melting temperature of U02 decreases with burnup which has the potential to affect safety analyses. As part of the normal reload safety analysis, it will be necessary to verify that fuel melt will not occur in the high burnup fuel rods at linear heat generation rates that may be reached during Condition II transients.

The analyses of record for Condition III and IV transients will not be affected by this physical property because the predicted temperatures in most cases remains below the fuel melting temperature.

It is therefore expected that the cycle specific reload analyses for Prairie Island 1 Cycle 21 will confirm that operation of this fuel assembly to extended burnups will not increase the probability of occurrence or consequences of any postulated accidents.

6.3 Radiological Impact

For purposes of determining the radiological impact of normal operation it is assumed that some small number of fuel rods will leak their gap inventory into the reactor coolant system. The radiological assessment of high burnup fuel cycles with lead rod burnups of up to 75 GWD/MTU was discussed in the Westinghouse topical report for the VANTAGE+ Fuel Assembly Design (Reference 2). Although the VANTAGE+ design has not been licensed to these burnup levels the information in the report supports the conclusion that the operation at high burnup does not result in large increases in liquid releases. Even if some of the rods of T81 were to fail, the technical specification limits on dose equivalent iodine-131 limit the primary coolant activity and assure that the existing analysis remains bounding.

For accidents in which the core remains intact (i.e., cladding failure but no fuel melt) the release will involve only volatile fission products. The radionuclides contributing most to these doses are short-lived, and so do not increase with burnup. Therefore, no increase in the consequences of such accidents are expected as the result of operating one fuel assembly to high burnup.

For the fuel handling accident, which is a single assembly or individual rod accident, the releases are evaluated based on the peak operating assembly. The analysis of record assumes this accident occurs 100 hrs after shutdown, when the short-lived fission products are the major contributors to any calculated doses. These short-lived isotopes tend to reach an equilibrium during operation, and do not increase with burnup. The fuel handling accident for Prairie Island follows the assumptions of NRC Regulatory Guide 1.25, dated 3/23/1972, " ASSUMPTIONS USED FOR EVALUATING THE POTENTIAL RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING AND STORAGE FACILITY FOR BOILING AND PRESSURIZED WATER REACTORS," and is based on a fission product inventory of an assembly at 1.65 times the core average power. The Fuel Assembly T81 operates at less than average power in Prairie Island 1 Cycle 21, and so will have a fission product inventory that is bounded by the analysis of record for this accident. The source term for the Fuel Handling accident assumes an assembly burnup of 75,000 MWD/MTU.

For accidents such as the steam generator tube rupture (SGTR) and the main steam line break (MSLB) no fuel failures are predicted to occur as a consequence of the accident and the calculated doses are based on the failures that exist at the time of the accident. Even if some of the rods of T81 were to fail, the technical specification limits on dose equivalent iodine-131 limit the primary coolant activity and assure that the existing analysis remains bounding.

For the locked rotor accident the amount of fuel failure is limited to 20% of all of the fuel rods as calculated in the FSAR. During the accident the higher power rods in the core experience departure from nucleate boiling (DNB) and may fail. The assembly T81 will be operating at low power and it is unlikely that it would experience DNB in a locked rotor accident. The low power also will result in a lower fission product inventory of the short-lived nuclides that are important in dose calculations. To bound this accident it will be confirmed that the 20% fuel failure limit is still met even assuming the failure of all the rods in T81.

For the rod ejection accident the amount of fuel failure is limited to 20% of all of the fuel rods as calculated in the FSAR. Fuel failure in the rod ejection accident is caused by DNB. T81 will be in a rodded location but its ejected rod worth is low, approximately 31¢, so that a rod ejection is unlikely to cause fuel failure. T81 is not located near to the other high worth rods for the rod ejection accident, is at low power, and is unlikely to go into DNB in the event that one of those rods should eject. The low power in T81 will result in a lower fission product inventory of the short-lived nuclides that are important in dose calculations. To bound this accident it will be confirmed that the 20% fuel failure limit is still met even assuming the failure of all the rods in T81.

For LOCA the dose calculations are bounded by the large break LOCA where it is conservatively assumed that the entire core fission product inventory is released. Assembly T81 is less than 1% of the core and operates at low power. The increased burnup of this assembly will have little effect on the nuclides that are important in dose calculations. The burnup of the remainder of the core will remain at levels representative of our standard operation, i.e. a batch average discharge burnup of about 52,000 MWD/MTU. The fission product inventory of the Prairie Island 1 Cycle 21 core as a whole should not differ significantly from that of our standard core. Analysis for Prairie Island assumes a core average burnup of 50,000 MWD/MTU. The environmental impacts of such severe accidents will remain applicable for high burnup irradiation of Assembly T81.

7. Preliminary Safety Assessment

The assembly to be irradiated is of the same design as fuel used in previous reloads at Prairie Island. The fuel rods in this VANTAGE+ assembly will be required to meet all design criteria for the proposed operating conditions.

Use of this assembly will not affect the set of key analysis parameters defined for the current safety analyses (Reference 4). As discussed in Section 6 above, the safety analyses of record are expected to remain applicable for the operation of this assembly in Prairie Island 1 Cycle 21. Cycle specific evaluations will verify that the assumed values for any key analysis parameters are not exceeded.

The final determination of whether an unreviewed safety question exists will be made after the cycle specific reload calculations are complete, and will be documented as part of the normal Reload Safety Evaluation.

SUMMARY

A Westinghouse VANTAGE+ fuel assembly with ZIRLO cladding that has been irradiated for two cycles in Prairie Island 2 is scheduled to be irradiated for one additional cycle in Prairie Island 1 Cycle 21. The proposed irradiation of this assembly to high burnup does not require any Technical Specifications changes. However, because the end of life burnups of this fuel assembly are expected to exceed the 62,000 MWD/MTU lead fuel rod burnup limit for VANTAGE+ fuel, NRC concurrence is required for this program to proceed.

The extended burnup of the assembly will be fully addressed as part of the Prairie Island 1 Cycle 21 Reload Safety Evaluation, using the Nuclear Analysis Department's NRC approved reload design methods and Westinghouse's approved fuel rod design models and methods. Additional conservatisms will be applied to the fuel rod design analysis since the end of life burnup of these rods will exceed the current lead rod burnup limit. All fuel rod design criteria that are applicable for the current lead rod burnup limit are satisfied for this fuel assembly.

Operation of this small number of fuel rods to high burnup in the Prairie Island 1 Cycle 21 core is not anticipated to result in the acceptable safety limits for any accident being exceeded, or in an unreviewed safety question as defined in 10 CFR 50.59. This will be confirmed as part of the cycle specific Reload Safety Evaluation.

REFERENCES

1. WCAP-12488-A, "Westinghouse Fuel Criteria Evaluation Process," October 1994.
2. WCAP 12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995.
3. NSPNAD-8101-A, Revision 2, "Prairie Island Nuclear Power Plant Qualification of Reactor Physics Methods for Application to PI Units," October 2000.
4. NSPNAD-8102-PA, Revision 7, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods For Application to PI Units," July 1999.
5. WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," Revision 1, November 1996.
6. CAA-97-042, "Northern States Power Prairie Island Units 1 and 2 Spent Fuel Rack Criticality Analysis Using Soluble Boron Credit," February 1997.
7. Technical Specifications - Prairie Island Nuclear Generating Plant. Docket 50-282, 50-306.
8. Letter from Marsha Gamberoni (U.S. Nuclear Regulatory Commission) to Roger O. Anderson (Northern States Power Company). "Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2 – Issuance of Amendment RE: Increased Fuel Enrichment Limit Changes (TAC Nos. M86727 and M86728)," September 3 1993.
9. Westinghouse, "Extended Burnup Operation Assessment for the VANTAGE+ Design in Prairie Island Nuclear Station," October 2000.