Xcel Energy

JAN 3 0 2009

L-PI-09-011
10 CFR 50.90

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

Response To Request For Additional Information Regarding License Amendment Request For Technical Specifications Changes To Allow Use Of Westinghouse 0.422-Inch OD 14x14 Vantage+ Fuel (TAC Nos. MD9142 and MD9143)

References: 1) Letter from M. Wadley (NMC) to Document Control Desk (NRC), L-PI-08-047, "License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 0.422-inch OD 14x14 VANTAGE+ Fuel," dated June 26, 2008 (ML081820137)

2) Letter from T. Wengert (NRC) to M. Wadley (NSPM), Prairie Island Nuclear Generating Plant, Units 1 and 2 Request For Additional Information Related to License Amendment Request For Technical Specifications Changes to Allow Use of Westinghouse 0.422-Inch OD 14x14 Vantage+ Fuel (TAC Nos. MD9142 and MD9143), dated December 15, 2008 (ML083300210)

By letter dated June 26, 2008 (Reference 1), Nuclear Management Company, LLC, (now Northern States Power, a Minnesota corporation (NSPM)) requested approval of amendments to the Operating Licenses and associated Technical Specifications (TS) for Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, as well as certain supporting analyses, in support of the transition from Westinghouse 0.400-inch outer diameter (OD) VANTAGE+ (hereinafter referred to as 400V+) fuel to 0.422-inch OD VANTAGE+ (hereafter referred to as 422V+) fuel.

On December 15, 2008, the NRC staff notified NSPM (Reference 2) that additional information was necessary for the staff to complete its review. The enclosed response addresses Questions 1 and 2. NSPM is continuing to evaluate the concerns put forth in Question 3. As discussed with the staff, NSPM will respond to Question 3 no later than February 20, 2009.

The supplemental information provided in this letter does not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the June 26, 2008 submittal.

In accordance with 10 CFR 50.91, NSPM is notifying the State of Minnesota of this License Amendment Request supplement by transmitting a copy of this letter to the designated State Official.

<u>Summary of Commitments</u>

NSPM shall respond to Question 3 no later than February 20, 2008. No previous commitment is being revised.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: **JAN 30 2009**

Joel P. Sorensen

Joel P. Sorensen
Director Site Operations
Prairie Island Nuclear Generating Plant
Northern States Power Company-Minnesota

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC
State of Minnesota

ENCLOSURE

Responses to Requests for Additional Information
License Amendment Request for Technical Specifications Changes to Allow Use of Westinghouse 422V+ Fuel

By letter dated June 26, 2008 (ML081820137), Nuclear Management Company, LLC, (now Northern States Power, a Minnesota corporation (NSPM)) requested approval of amendments to the Operating Licenses and associated Technical Specifications (TS) for Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2, as well as certain supporting analyses, in support of the transition from Westinghouse 0.400-inch outer diameter (OD) VANTAGE+ (hereinafter referred to as 400V+) fuel to 0.422-inch OD VANTAGE+ (hereafter referred to as 422V+) fuel. On December 15, 2008 (ML083300210), the NRC staff notified NSPM that additional information was necessary for the staff to complete its review. NRC requests for additional information (RAI) are repeated below with the NSPM response following:

1. **The TS changes requested to support the fuel upgrade include a change to TS 2.1.1.2, Reactor Core Safety Limits. The peak fuel centerline temperature for non-gadolinia bearing fuel is requested to increase. Three points of justification are provided; two are repeated in this request:**

    (i). The fuel centerline temperature melting limits are referenced in WCAP8720, Addendum 3. These melting limits are integrated into the peak fuel centerline temperature evaluation methodology used to confirm that the fuel centerline melt design criteria are met.

    (ii). The request is justified based on WCAP-12488-A, "Westinghouse Fuel Criteria Evaluation Process," October 1994.

    a. For Item (i), confirm that the fuel centerline temperature melting limits correspond acceptably to the requested TS change, that is, that the TS limit matches or is conservative with respect to the referenced fuel centerline temperature melting limit.

<u>NSPM Response</u>:

The changes to TS 2.1.1.2 involved two separate changes. One to add limits for gadolinia bearing fuel and another to modify the fuel melt limits for non-gadolinia bearing fuel (UO_2 only). The points of justification provided in the LAR were not clearly aligned with gadolinia content differences. Therefore, the following clarification is made to answer this question:

The fuel melt limit for non-gadolinia bearing fuel matches the fuel melting limit for UO_2. WCAP-8720 Addendum 3 primarily forms the licensing basis for gadolinia bearing fuel. The licensing basis for non-gadolinia bearing fuel, including the melting temperature of 5080°F -58°F per 10,000MWD/MTU, is spread throughout a number of topical reports in Westinghouse licensing history dating back to the time of the Atomic Energy

Commission (WCAP-6065). Most recently Westinghouse refers to the NRC Safety Evaluation Report for WCAP-10125 and WCAP-12488/14204-A, both of which call out the UO_2 fuel melting temperature of 5080°F -58°F per 10,000MWD/MTU.

b. Also for Item (i), confirm that the referenced centerline temperature melting limits are specifically applicable to all fuel designs planned for incorporation in the 422V+ transition cycles.

NSPM Response:

The proposed centerline temperature melting limits are the point at which either UO_2 or UO_2-Gd_2O_3 melts. This is a material property of the fuel pellets and is unrelated to the geometry of the fuel. The 422V+ pellet materials remain the same as the existing 400V+ fuel which both utilized a UO_2 and a UO_2-Gd_2O_3 mix matrix. Lastly, compliance with the fuel melt limits will be confirmed for both the 400V+ and 422V+ fuel types present in the transition cycles on a cycle specific basis.

c. Regarding Item (ii), is there a specific Fuel Criteria Evaluation Process (FCEP) notification letter referencing this change to fuel centerline melting temperature? What specific aspect of the FCEP provides justification for the requested TS change? Please explain.

NSPM Response:

There is no Fuel Criteria Evaluation Process (FCEP) notification letter associated with the fuel temperature limits for non-gadolinia bearing fuel. The fuel melting temperature is a material property defined by the AEC/NRC (WCAP-6065) and listed in WCAP-12488-A and the Westinghouse specified acceptable fuel designs limits, (SAFDL) WCAP-10125-A. The melting temperature does not vary by fuel types using UO2 fuel matrixes and is unrelated to fuel type changes as defined in the Fuel Criteria Evaluation Process discussed in WCAP-12488-A. In addition, the proposed fuel melt limits are consistent with NUREG 1431, Revision 3 used by several other facilities for application of a UO_2 fuel matrix.

2. **In a post-LOCA scenario, when analyzing boric acid precipitation, it is generally accepted as conservative to assume that no boric acid carries over into the coolant loops from the core. Please consider the opposite of this -that boric acid could carry, through entrained liquid or due to volatility in water vapor, into the reactor coolant loops and discuss whether it would be possible to precipitate boric acid in steam generator tubes and cause excessive tube plugging. Available analyses do not appear to account for this possibility. How susceptible is PINGP to such a scenario?**

NSPM Response:

NSPM understands the significance of the precipitation phenomenon posed by the Staff in that it could result from the following: (1) buildup of boron precipitate which restricts flow through steam generator U-tubes, (2) reduced steam flow from the core, (3) increased pressure in the core, and (4) core mixture level depression which could expose the core. The impact of boric acid precipitation in the steam generator has been assessed by industry and academic experts [2-1] in the areas of thermal-hydraulics, heat and mass transport processes, and chemistry. These experts concluded that this phenomenon is of medium[1] importance regarding its impact on the Phenomena Identification and Ranking Table (PIRT) Figure of Merit, "Boric Acid Concentration in the Reactor Vessel Liquid Mixing Volume." To assess the susceptibility of PINGP to this scenario, the likelihood of carrying a significant amount of boric acid to the steam generators is assessed below.

The partition coefficient (acid-to-water ratio in vapor to that in liquid) of boric acid is approximately 0.005 at atmospheric pressure [2-2] and, similar to other solutes, increases proportional to the density of the vapor phase relative to the liquid phase [2-3] at higher pressure. Therefore, in a two-phase mixture with a liquid phase boric acid concentration of 40,000 ppm, the vapor phase boric acid concentration would be 200 ppm. For this reason, the boric acid content in the vapor phase is considered insignificant with regard to precipitation in the steam generator.

The boric acid concentration of entrained liquid will be the same as that of the liquid pool or film from which it was entrained; thereby, potentially increasing the significance of boric acid precipitation due to carryover in entrained liquid. The rate of liquid entrainment is related to the gas flux passing through or over the liquid [2-4, 2-5, 2-6]. The highest rate of liquid entrainment will occur during the emergency core cooling system (ECCS) injection phase and early in the recirculation phase when decay heat, hence, gas flux is highest. However, at this time (early in the transient) the boric acid concentration is lowest. Counteracting the effect of entrainment within the upper plenum of the reactor vessel is de-entrainment on the structures present in this region. The capture (or de-entrainment) efficiency in plant designs such as PINGP is >90% using the predictive relation in Reference 2-7. Therefore, only approximately 10% of the liquid entrained from the pool in the reactor vessel has the potential to enter the reactor coolant loop piping. Assuming that the inner vessel mixture level is at the hot leg bottom elevation such that the near surface regime applies, the net entrainment rate out of the vessel has been calculated to be of the same magnitude as the mass boil-off rate due to decay heat (approximately 20 lbm/sec per loop at 20-mintues after the event assuming the decay heat rate specified in 10 CFR 50 Appendix K I.A.4 – 1.2 times the values for infinite operating time in the ANS Standard).

This is a reasonably high mass flow rate; however, compensatory effects can significantly reduce the entrained liquid flow rate out of the reactor vessel and potential for boric acid precipitation in the steam generator tubes. During the post-LOCA scenario when the steam generators are acting as a heat source, entrained liquid droplets regardless of their boric acid concentration that impinge on the steam generator tubes will rapidly evaporate and increase the loop pressure drop. The inner vessel mixture level will depress due to the increased pressure drop and the entrainment regime will transition from near surface regime to momentum controlled regime. In this regime, the entrainment fraction is approximately 150 times less than the near surface regime [2-8], e.g., approximately 0.13 lbm/s per loop at 20 minutes after the event. Similarly, if the increased loop pressure drop is due to the build-up of boric acid precipitate in the steam generator tubes, the entrainment fraction will be reduced by the same magnitude. The amount of mixture level depression needed to cause the transition from near surface to momentum controlled is not sufficient to cause a core dryout resulting in a fuel cladding heat-up, i.e., the upper plenum can remain nearly filled.

In order for the entrained liquid from the reactor vessel to reach the steam generator tubes that are approximately 8 feet above the inner vessel mixture level, the droplets must traverse the length of the hot leg horizontal piping; turn 90° toward the vertical while traversing an elbow, the steam generator inlet nozzle, and inlet plenum; and then enter the tubes without either impinging upon each other, pipe walls, other structures or de-entraining due to the several-fold area expansion and corresponding gas flux reduction encountered along the path. The entrained liquid fraction reaching the steam generator tubes can be approximated for the case where the aforementioned de-entrainment mechanisms are neglected. The steam generator tube entrance is approximately 8 ft above the hot leg bottom elevation so the entrainment regime will be the deposition controlled region of [2-4] which is a strong function of height. In this region, the entrained liquid fraction is approximately 4 to 6 times less than the momentum controlled region [2-8] or, on average, approximately 750 times less than the near surface region resulting in an entrained liquid mass flow rate of approximately 0.03 lbm/s per loop at 20 minutes after the event. Another significant compensatory effect that should be considered is reflux condensation in the steam generator when it acts in a heat sink mode once the operators initiate a cooldown of the steam generators. The boric acid precipitate that formed while the steam generator was in the heat source mode will be readily re-dissolved by the liquid film draining from the tubes once the steam generators are acting as a heat sink. Even if it assumed that the entire quantity of entrained liquid is evaporated in the steam generator tubes, excessive tube plugging is not expected to occur, especially if compensatory effects are considered.

Based upon the assessment performed, precipitation of boric acid is possible in the steam generator tubes but is limited due to compensatory effects of de-entrainment on upper plenum structures, mixture level depression (due to steam binding or increased resistance due to the precipitate) as it impacts the height needed to transport droplets to the steam generator tubes, decay heat reduction as it impacts gas flux, and reflux condensation. Hence, the steam generator precipitation phenomenon is considered to be of medium importance in the PIRT as it relates to calculations performed to determine the boric acid concentration in the reactor vessel following a loss-of-coolant accident.

Footnotes

1. The cited PIRT indicates 'low' importance; however, the importance has been elevated to medium in a more comprehensive PIRT expected to be published and distributed to the NRC during the first half of 2009.

References

2-1 Brown, W. L., et al, "Phenomena Identification and Ranking Table (PIRT) for Unbuffered Boric Acid Mixing/Transport in Reactor Vessel During Post-LOCA Conditions," WCAP-16745-NP, June 2008.

2-2 Byrnes, D. E. and Foster, W. E., "Literature Values for Selected Chemical/Physical Properties of Aqueous Boric Acid Solutions," WCAP-1570, May 1960.

2-3 Collier, J. G. and Thome, J. R., "Convective Boiling and Condensation," 3rd Edition, 1996.

2-4 Kataoka, I. and Ishii, M., "Mechanistic Modeling and Correlations for Pool Entrainment Phenomenon," NUREG/CR-3304, 1983.

2-5 Ishii, M. and Grolmes, M. A., "Inception criteria for droplet entrainment in two-phase concurrent film flow," AIChE Journal Vol. 21 No. 2, 1975.

2-6 Wallis, G. B., "One-Dimensional Two-Phase Flow," 1969.

2-7 Dallman, J. C. and Kirchner, W. L., "De-Entrainment Phenomena on Vertical Tubes in Droplet Cross Flow," NUREG/CR-1421, 1980.

2-8 Welter, K. B., et al, "APEX-AP1000 Confirmatory Testing to Support AP1000 Design Certification," NUREG-1826 (Non-Proprietary), 2005.

3. **The NRC staff reviewed documents during an October 1-2, 2008, audit at the Westinghouse Energy Center in support of the NRC staff's review of the requested fuel upgrade. During its audit, the NRC staff identified potentially non-conservative assumptions made regarding the PINGP capability for post-LOCA, long-term core cooling. Please re-evaluate the post-LOCA, long-term core cooling at PINGP, and demonstrate acceptable safety injection capability. The re-evaluation should at least consider the following:**

   a. The acceptance criteria set forth in 10 CFR 50.46(b).

   b. The 10 CFR Part 50, Appendix K, decay heat requirements.

   c. Differences in lattice pitch over generically studied plants referenced in the analysis (Westinghouse Proprietary Calculation CN-LIS-07-126, "Prairie Island Units 1 and 2 (NSP/NRP) Post-LOCA Long-Term Cooling Analysis in Support of the 422V+ Fuel Transition Program").

   d. Differences in post-LOCA decay power shape compared to the power shape evaluated in CN-LIS-07-126 (Westinghouse Proprietary) and the effect that this difference could have on core boiloff.

NSPM Response:

As discussed with the staff, NSPM is continuing to work on completion of the re-evaluation requested. NSPM will respond to this question no later than February 20, 2009.