

ENCLOSURE 1

Non-Proprietary 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 (ADAMS Accession No. 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 outside diameter (OD) VANTAGE+ (hereinafter referred to as 400V+) fuel to 0.422-inch OD VANTAGE+ (hereafter referred to as 422V+) fuel. On February 11, 2009 (ADAMS Accession No. ML090140334), the NRC staff notified NSPM that additional information was necessary for the staff to complete its review. The NRC request for additional information (RAI) is repeated below with the NSPM response following:

Containment and Ventilation Systems (SCVB) Requests for Additional Information (RAIs)

- 1. In Enclosure 1, Attachment 4, Section 5.3.3 of the June 26, 2008, license amendment request (LAR), it is stated that [PINGP] has applied [leak-before-break (LBB)] to the main [reactor coolant system (RCS)] piping. Please verify that ruptures of piping that are not subject to LBB do not result in subcompartment pressurizations that exceed pressure or structural design limits.**

NSPM Response:

The Updated Safety Analysis Report (USAR) Section 12.2.4 discusses evaluation of containment subcompartments that may be pressurized during various high energy line breaks. USAR 12.2.4.1 explains that the following subcompartments are evaluated for the effects of primary RCS loop piping and other high energy line ruptures:

1. steam generator vault compartment (12.2.4.1.1)
2. reactor cavity region (12.2.4.1.3)
3. nozzle cavity (12.2.4.1.4)
4. RCS compartment – including pressurizer (12.2.4.1.5)

These subcompartments are qualified and operable based on thermo-hydraulic operating conditions, and have been operable for all previous fuel types including OFA (Optimized Fuel Assembly) and Standard (a fuel type that is dimensionally similar to 422V+). Although application of leak-before-break (LBB) technology has essentially obviated the original subcompartment pressurization analyses related to main coolant loop ruptures (and the Unit 1 pressurizer surge line), the USAR (12.2.4.1) has retained description of these original subcompartment pressurization analyses as bounding for design of the compartment structure. Where subcompartments have been analyzed to safely withstand the differential pressure from main coolant loop breaks (31-inch inside diameter), it is reasonable to conclude that the subcompartments will then withstand the differential pressure from breaks originating from smaller breaks (e.g., 12-inch

ENCLOSURE 1

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

accumulator injection line). In general terms, the four limiting branch line breaks are relatively close in proximity to the RCS loop breaks originally postulated in design of the steam generator (SG) and reactor coolant pump (RCP) compartments.

The proposed amendment involves loading a new fuel type into the reactor and has no significant bearing on subcompartment pressurizations. The amendment involves no significant effect on the mass and energy released from any RCS or high energy line in containment. To have such an adverse effect would require a notable change in the operating pressure or temperature of a process fluid. No such changes are made in conjunction with the transition to 422V+ fuel. In summary, existing subcompartment pressurizations caused by postulated breaks in high energy lines in compartments (i.e., high energy lines other than main loop piping and Unit 1 surge line) are bounded by analyses of main coolant loop piping ruptures represented in the USAR.

2. **As described in Enclosure 1, Attachment 4, Section 504.1.2.3 of the LAR, liquid entrainment is included in the break flow for the main steamline break inside containment analysis. In the LAR, the licensee states that the entrainment characteristics for large steamline breaks are not sensitive to the steam generator design and that the NRC staff agreed to this position for a [Point Beach Nuclear Plant (Point Beach)] LAR.**

However, the NRC staff notes that this agreement was based upon different Westinghouse steam generator designs than those at PINGP.

- (a) **Please explain why the same conclusion holds for the ANP 56/19 steam generator.**

NSPM Response:

Section 2.2.1 in WCAP-8822 is titled "Effects of Steam Generator Design." The first sentence in this section is: "Although the emphasis of the program to define steam line break mass and energy releases centered on plants having steam generators with integral preheaters (Westinghouse model D units), the studies include plants having older, non-preheat model steam generators (model 51 units)." The only element discussed in this section of the WCAP is the feedwater addition that enters at different locations and causes different resistances in the SG as the depressurization occurs. Sensitivities are presented to different feedwater assumptions for the model D and model 51 SGs. It is found that more feedwater causes more entrainment, and that the shape of the transient response looks somewhat different for the two SG types because of the different location of the feedwater insertion. Nevertheless, the end conclusion is "that studies conducted using the preheat steam generator computer model would also be conservative for the non-preheat steam generator."

ENCLOSURE 1

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

The preheater / non-preheater design is not the only difference between the model D and model 51 steam generators considered in WCAP-8822. There are 12 or 16 primary separators for the model D steam generators while there are 3 primary separators for the model 51 steam generators. Note that when there are more separators, they are much smaller in diameter, such that the total area of the primary separators is similar. []^{a,c}

The two SG designs that were explicitly analyzed in WCAP-8821 and WCAP-8822 were very different. The main differences were the pre-heat vs. feeding design and differences in the number of separators. Regardless, it was concluded that []^{a,c}

Nevertheless, a comparison was made between the Framatome Model 56/19 steam generator in Prairie Island Unit 1 to the Westinghouse Model 51 steam generator that is in Prairie Island Unit 2, and for which the break quality results from WCAP-8822 have been applied. The Framatome SG was designed as a direct replacement to the model 51 SG and is therefore very similar in size and capacity. The largest difference is the moisture separator. The Framatome Model 56/19 steam generator has 16 swirl vane separators while the Westinghouse Model 51 steam generator has only 3. However, as mentioned above, the model D steam generators had either 12 or 16 swirl vane separators. Yet this difference was determined []^{a,c}

The conclusion from the review of the Framatome Model 56/19 design compared to the Westinghouse Model 51 design is that they are very similar. Differences in the separator design are not important since the two steam generator designs used in the TRANFLO calculations in WCAP-8821 and WCAP-8822 were also diverse. Yet those differences were determined to not be important relative to the liquid released from a large steamline break.

- (b) For the referenced Point Beach analyses an additional 0.1 was added to the quality. Was this done for PINGP steamline break mass and energy release analyses?**

NSPM Response:

Yes, the additional 0.1 was added to the quality.

ENCLOSURE 1

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

Containment and Ventilation Systems (SCVB) Requests for Additional Information (RAIs)

3. EMCB RAI-1:

- (a) **Section 2.4, Cladding Stress and Strain, first bullet, page 2-6 of Reference 1 mentions that the stress limit is based on the American Society of Mechanical Engineers (ASME) code. Provide the specific section, subsection, and edition of the ASME code utilized.**

NSPM Response:

ASME Boiler and Pressure Vessel Code,
Section VIII: Pressure Vessels,
Division 2: Rules for Construction of Pressure Vessels - Alternate Rules,
Appendix 4: Mandatory Design Based on Stress Analysis, (2001 version)

- (b) **Section 2.4, Cladding Fatigue, page 2-8 of Reference 1 states that the cumulative usage factor (CUF) is less than the design limit. Please provide the numerical value of the computed CUF for cladding of 422V+ fuel cladding as well as for the current 400V+ fuel in operation at PINGP.**

NSPM Response:

In order to verify that the cladding fatigue design limit is met for the Prairie Island Units 1 and 2 422V+ Reload Transition, a cumulative usage factor (CUF) is calculated []^{a,c}. This CUF must be below the design limit specified in the FSAR of the plant. Cladding fatigue is a cycle-specific calculation and is dependent on operating conditions and loading patterns. Therefore, there is no direct comparison for the CUF between the current 400V+ fuel in operation and the 422V+ Reload Transition. However, based on a review of recent reload design operating conditions and loading patterns, there is approximately []^{a,c} for the 422V+ Reload Transition.

- (c) **Section 2.4, End Plug Weld Integrity, page 2-9 of Reference 1 states that the fuel system will not be damaged due to excessive end plug weld tensile pressure differential loads. Please provide the computed values of the end plug weld tensile pressure differential loads for the 422V+ and 400V+ fuel rod end plug welds along with the acceptable or allowable limits.**

ENCLOSURE 1
Non-Proprietary Responses to Requests for Additional Information
License Amendment Request for Technical Specifications Changes to Allow Use
of Westinghouse 422V+ Fuel

NSPM Response:

Fuel rod end plug weld integrity analyses are performed on a generic basis to provide that the fuel system will not be damaged due to excessive end plug weld tensile pressure differential loads. The generic end plug weld integrity analysis consists of a checklist of the limiting values for []^{a,c}. For both the current 400V+ fuel in operation and the 422V+ Reload Transition, each of the conservative weld integrity generic criteria are met with significant margin. Therefore, there are no computed values of the end plug weld tensile pressure differential loads for the 422V+ and 400V+ fuel rod end plug welds. However, the 422V+ fuel rod design, at equivalent HZP SLB statepoint conditions, would tend to have []^{a,c}.

4. EMCB RAI-2: Section 2.5.2, Grid Load Analysis, page 2-10 of Reference 1:

- (a) Provide numerical values of the computed maximum grid impact force (combined from safe-shutdown earthquake (SSE) and loss-of-coolant accident (LOCA) analyses) and the allowable grid strength for the homogeneous and mixed cores.**

NSPM Response:

PINGP specific LOCA and seismic grid impact force analyses were performed in support of the 422V+ Licensing Submittal. The limiting results from these analyses are presented in Tables 1 and 2 on the following page.

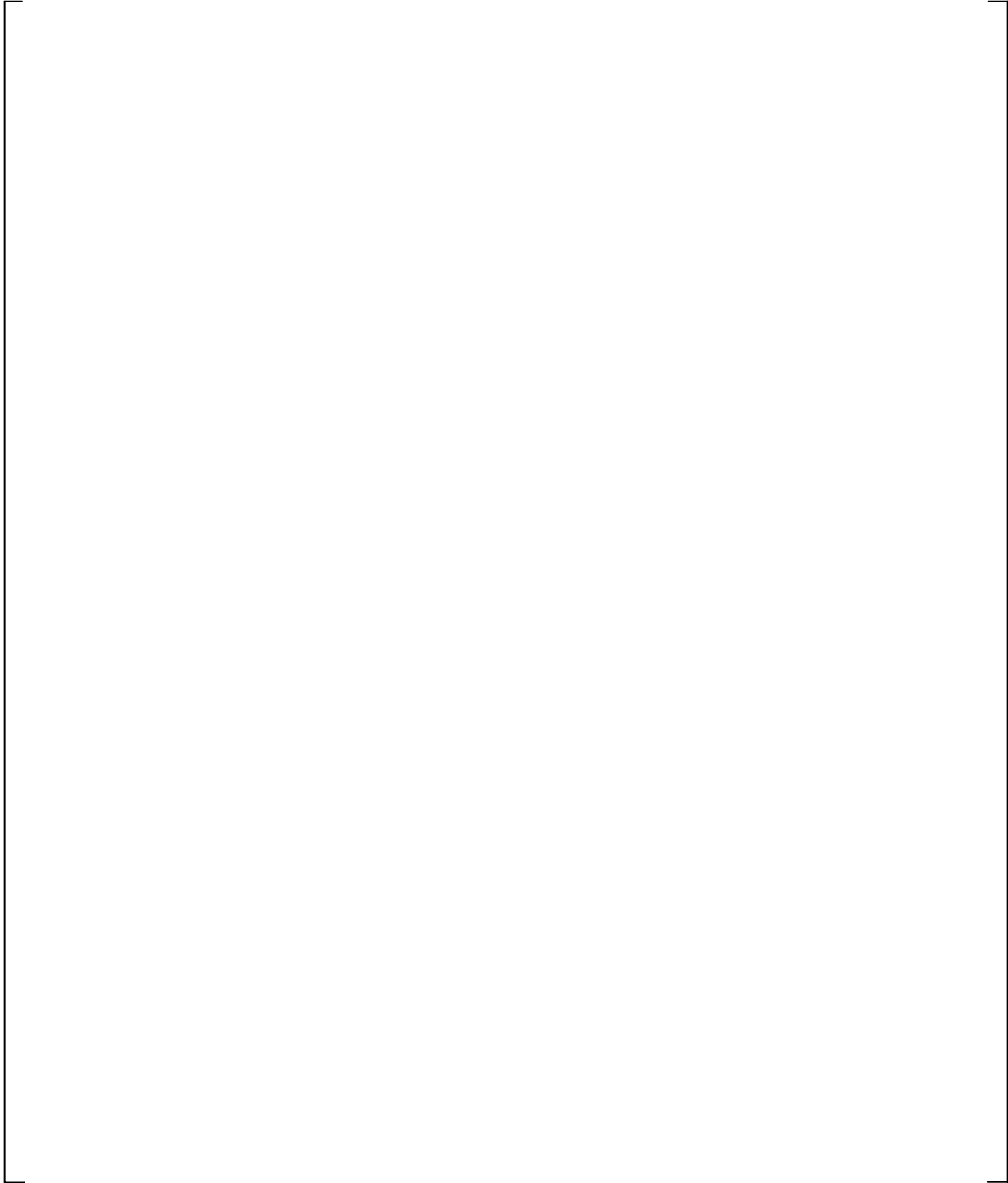
- (b) Provide a summary of fuel assembly (fuel rods and thimble tubes) stresses for operating-basis earthquake, and combined SSE seismic and LOCA loadings along with the corresponding allowable stresses for the 422V+ assembly design.**

NSPM Response:

The stresses are specific to the fuel type and not the plant. Therefore, in support of the 422V+ Licensing Submittal, an evaluation was performed based on existing conservative stress analyses to demonstrate that the stress limits were satisfied. Based on the PINGP LOCA and seismic analyses mentioned above, limiting deflections and axial loads were identified, as shown in Table 3. These were compared to generic stress analyses which assumed bounding deflections and axial loads. This comparison is presented in Table 3.

ENCLOSURE 1
Non-Proprietary Responses to Requests for Additional Information
License Amendment Request for Technical Specifications Changes to Allow Use
of Westinghouse 422V+ Fuel

a,c



ENCLOSURE 1
Non-Proprietary Responses to Requests for Additional Information
License Amendment Request for Technical Specifications Changes to Allow Use
of Westinghouse 422V+ Fuel



a,c

ENCLOSURE 1

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

5. EMCB RAI-3: Section 6.1.4, pages 6-2 to 6-5 of Reference 1 addresses mechanical system evaluations for LOCA and seismic loads using a mathematical model consisting of 3 sub-models of reactor vessel, internals such as core barrel, and support plates and fuel based on the ANSYS finite element code.

- (a) Provide a summary of the results of maximum combined LOCA and SSE loads and the corresponding allowable loads for 14X14 type guide tubes.**

NSPM Response:

The SSE and LOCA loads were combined using the square root of the sum of the squares (SRSS) method:

$$F = [F_{SSE}^2 + (F_{flow} + F_{acoustic} + F_{system})^2]^{1/2}$$

[]^{a,c}

The maximum applied SSE and LOCA load to the guide tube is calculated to be []^{a,c} pounds. This load is less than the allowable 14x14 guide tube load of []^{a,b,c} pound.

- (b) Provide a summary of the flow induced vibration levels and the acceptance limits during normal operation at the fuel upgrade analyzed reactor coolant system conditions.**

NSPM Response:

Summary of Calculated Flow Induced Vibration High-Cycle Stresses and Code Endurance Limit		
Component	Alternating Stress (psi)	ASME Code Endurance Limit for High-Cycle Fatigue (psi)
Core Barrel Upper Girth Weld at Flange	[] ^{a,c}	≥23,700
Core Barrel Lower Girth Weld	[] ^{a,c}	≥23,700
Core Barrel Outlet Nozzles	[] ^{a,c}	≥23,700
Thermal Shield Flexures	[] ^{a,c}	≥23,700
Thermal Shield Bolts	[] ^{a,c}	≥23,700
Lower Core Support Plate	[] ^{a,c}	≥23,700

ENCLOSURE 1
Non-Proprietary Responses to Requests for Additional Information
License Amendment Request for Technical Specifications Changes to Allow Use
of Westinghouse 422V+ Fuel

Summary of Flow Induced Vibration Forces Relative to Reference 2 Loop Plant				
Location	Unit 1/Unit 2 Force		Reference 2 Loop Plant Force (lb)	
	Shear (lb)	Moment (in-lb)	Shear (lb)	Moment (in-lb)
Upper Support Column Base Welds	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}
Upper Support Column Top welds	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}	[] ^{a,c}

Summary of Calculated Reactor Coolant Pump-Induced Vibration Thermal Shield Support Stresses			
Component	Alternating Stress (psi)	Allowable Stress (psi)	Factor of Safety
Top Support Bolts	[] ^{a,c}	23,700	[] ^{a,c}
Flexures	[] ^{a,c}	23,700	[] ^{a,c}
Flexure Bolts	[] ^{a,c}	23,700	[] ^{a,c}

ENCLOSURE 1

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

- (c) Provide a summary of the results (computed stresses, allowable stress limits, and fatigue usage factors) of structural evaluations of the reactor vessel internal components for the 422V+ assembly design.

NSPM Response:

Summary of Critical Reactor Internal Components Stresses and Fatigue Usage Factors				
Component	Category	Range of Max Stress	Allowable Stress	Fatigue Usage
Lower Core Plate	Pm + Pb + Q	[] ^{a,c}	48.6 ksi	[] ^{a,c}
Upper Core Plate	Pm + Pb + Q	[] ^{a,c}	48.6 ksi	[] ^{a,c}
Baffle-Former Bolts	-	-	-	U _{cum} = <1.0
Core Barrel Upper Girth Weld	Pm + Pb + Q	[] ^{a,c}	49.20 ksi	[] ^{a,c}
Core Barrel – Lower Girth Weld	Pm + Pb + Q	[] ^{a,c}	49.20 ksi	[] ^{a,c}
Core Barrel Outlet Nozzle	Pm + Pb + Q	[] ^{a,c}	34.44 ksi	[] ^{a,c}
Thermal Shield Flexures	Pm + Pb + Q	[] ^{a,c}	49.20 ksi	[] ^{a,c}
Upper Core Plate Alignment Pins	Pm + Pb + Q	[] ^{a,c}	34.44 ksi	[] ^{a,c}
Lower Radial Restraint inserts	Pm + Pb + Q	[] ^{a,c}	69.00 ksi	[] ^{a,c}
Notes:				
1. Exceeded the 3Sm limit; therefore, simplified elastic-plastic analysis was performed to calculate fatigue strength.				