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Gentlemen:

DOCKETS 50-266 AND 50-301
TECHNICAL SPECIFICATIONS CHANGE REQUEST 177
MODIFICATION TO TS 15.3.1.G.3
REACTOR COOLANT SYSTEM RAW MEASURED TOTAL FLOW RATE
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In accordance with the requirements of 10 CFR 50.4 and 50.90, Wisconsin Electric Power Company (Licensee) hereby requests amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant (PBNP) Units 1 and 2, respectively, to incorporate changes to the plant Technical Specifications. The proposed revisions will modify Technical Specification Section 15.3.1.G, "Operational Limitations," Specification 3.b, to reduce the reactor coolant system raw measured total flow rate by 5200 gallons per minute (gpm) for Unit 2. To support the reduction in reactor coolant system flow, changes are proposed to modify Specification 15.3.1.G.2, Section 15.2.3.1.B(4), "Overtemperature Delta T," and Figure 15.2.1-2, "Reactor Core Safety Limits, Point Beach Unit 2." Marked-up Technical Specifications pages, a safety evaluation, and the no significant hazards consideration are enclosed.

DESCRIPTION OF CURRENT LICENSE CONDITION

Specification 15.3.1.G, "Operational Limitations," specifies the Reactor Coolant System (RCS) operational limitations for DNB (Departure from Nucleate Boiling)-related parameters. Specification 15.3.1.G.3 presently specifies that reactor coolant system raw measured total flow rate must be ≥181,800 gpm for Unit 1, and ≥179,200 gpm for Unit 2. Specification 15.3.1.G.2 states that RCS pressurizer pressure shall be maintained ≥2205 psig during operation at 2250 psia or ≥1955 psig during operation at 2000 psia.

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Specification 15.2.1, "Safety Limit, Reactor Core," specifies the reactor core safety limits that are used to maintain the integrity of the fuel cladding. The specification states that the combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown on Figure 15.2.1-1 for Unit 1 and on Figure 15.2.1-2 for Unit 2.

Technical Specification Section 15.2.3, "Limiting Safety System Settings, Protective Instrumentation," Specification 15.2.3.1.B(4) is the overtemperature ΔT core limit protection setpoint function. This function provides setpoints that prevent exceeding the reactor core safety limits shown in Figures 15.2.1-1 and 2.

DESCRIPTION OF PROPOSED CHANGES

This Technical Specification Change Request (TSCR) proposes to revise Specification 15.3.1.G.3 as follows:

"3. Reactor Coolant System raw measured Total Flow Rate (See Basis):

a. Unit 1 ≥181,800 gpm
b. Unit 2 ≥174,000 gpm"

The associated basis is also being changed to reflect the revision to TS 15.3.1.G.3. The proposed basis revision is as follows:

"The reactor coolant system total flow rate for Unit 1 of 181,800 gpm is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (178,000 gpm). The reactor coolant system total flow rate for Unit 2 of 174,000 gpm is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (170,400 gpm). The raw measured flow is based upon the use of normalized elbow tap differential pressure which is calibrated against a precision flow calorimeter at the beginning of each cycle."

Specification 15.3.1.G.2 is being modified as follows:

"2. Reactor Coolant System (RCS) pressurizer pressure shall be maintained:

a. Unit 1: ≥2205 psig during operation at 2250 psia, or
≥1955 psig during operation at 2000 psia.
b. Unit 2: ≥1955 psig during operation at 2000 psia."

Specification 15.2.3.1.B(4)(b) is being modified as follows:

"(b) for each percent that the magnitude of $q_t - q_b$ exceeds +5 percent, the ΔT trip setpoint shall

> be automatically reduced by an equivalent of 2.0 percent of rated power for Unit 1, or by an equivalent of 3.1 percent of rated power for Unit 2."

Figure 15.2.1-2, "Reactor Core Safety Limits, Point Beach Unit 2," is being modified to support the reduction in RCS flow.

BASIS AND JUSTIFICATION

The proposed 5200 gpm reduction in RCS raw measured total flow rate for Unit 2 has been determined to be acceptable based on evaluations performed by Westinghouse and reviewed by Wisconsin Electric. The results of these evaluations are discussed in the attached safety evaluation.

It has been determined that the proposed amendments do not involve a significant hazards consideration, authorize a significant change in the types or total amounts of any effluent release, or result in any significant increase in individual or cumulative occupational exposure. Therefore, we conclude that the proposed amendments meet the requirements of 10 CFR 51.22(c)(9) and that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared.

On August 26, 1994, we submitted TSCR 175, "Modifications to Section 15.4.2, 'In-Service Inspection of Safety Class Components.'" The amendment was submitted to address steam generator tube degradation detected by new inspection techniques used for the first time at Kewaunee Nuclear Plant in the spring of 1994. The proposed changes modified Technical Specifications Section 15.4.2, "In-Service Inspection of Safety Class Components," by incorporating the use of acceptance criteria to allow sleeved tubes with certain upper sleeve joint parent tube indications to remain in service as described in Westinghouse WCAP-14157, "Technical Evaluation of Hybrid Expansion Joint (HEJ) Sleeved Tubes Containing Indications Within the Upper Joint Zone."

Based on the proposed acceptance criteria, our estimate of the total number of tubes which would require plugging and resultant RCS flow reductions indicated that adequate margin existed between the projected RCS flow rate and the current Technical Specification (TS) limit. However, analyses were begun in September to support operation with a lower TS limit for RCS flow in the event that TSCR 175 was not approved.

Negotiation between your staff and ours continued on this issue until October 4, 1994, when you informed us that TSCR 175 would not be approved. Steam generator eddy current testing was completed on October 12, 1994, and indicated that 245 tubes required plugging.

The total level of tube plugging (approximately 17%) exceeded our estimates based on past experience and potentially reduces the RCS flow rate to less than the current TS limit for PBNP Unit 2.

Our analysis of predicted RCS flow upon return to full power has been on-going since October 12, 1994, when we finalized the total number of tubes that required plugging. Based on this analysis, we have concluded that we will likely have sufficient RCS flow to meet our current TS requirements at rated power. However, our conservative predicted RCS flow is approximately 180,400 gpm, only 1200 gpm greater than the current TS limit. Additionally, with a 95% confidence level, the lower bound of predicted flow falls below the current TS limit of 179,200 gpm. Thus, there is a slight possibility that RCS flow will be less than that currently allowed by our TS. We believe that it is prudent to proceed with this Technical Specification change request and that this request is timely based on the unexpected large number of tubes that required plugging.

A reduction in the RCS raw measured total flow rate may be required to support full power operation of PBNP Unit 2 following its annual maintenance and refueling outage. Unit 2 is presently scheduled to return to full power and be on-line by November 1, 1994. We believe this submittal is timely and could not have been avoided and thus meets the criteria of 10CFR50.91 for processing as an emergency change. As such, we request this change request be processed as an emergency Technical Specification Change Request and be issued by October 28, 1994.

Please contact us if there are any questions.

Sincerely,

Roger A. Newton for

Bob Link
Vice President
Nuclear Power

KVA/jg

cc: NRC Resident Inspector
NRC Regional Administrator

TECHNICAL SPECIFICATIONS CHANGE REQUEST 177
SAFETY EVALUATION

INTRODUCTION

Wisconsin Electric Power Company (Licensee) has applied for amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant Units 1 and 2. The proposed revisions will modify Technical Specification (TS) Section 15.3.1.G, "Operational Limitations," Specification 3.b, to reduce the reactor coolant system raw measured total flow rate by 5200 gallons per minute (gpm) for Unit 2. The proposed revisions will also modify Specification 15.3.1.G.2, Section 15.2.3.1.B(4), "Overtemperature Delta T," and Figure 15.2.1-2, "Reactor Core Safety Limits, Point Beach Unit 2," to support the flow reduction.

EVALUATION

Westinghouse performed an evaluation of the affects of a 5200 gpm reduction in RCS raw measured total flow rate on the accident analyses for Point Beach Nuclear Plant, (PBNP) Unit 2. The scope of the evaluation included the Loss of Coolant Accident (LOCA), Transient(non-LOCA), Steam Generator Tube Rupture (SGTR), and Containment Analyses. Additionally, the affects of reduced RCS flow were assessed for mechanical component integrity considerations.

The 5200 gpm flow reduction corresponds to a uniform steam generator tube plugging level of approximately 24%. Based on the number of tubes plugged in each steam generator, we expect a difference in RCS flow between the two steam generators of approximately 3%. Westinghouse has considered this in their evaluations for the flow reduction. Other assumptions in the evaluations include a nominal RCS pressure of 2000 psia, a nominal full power T_{avg} of 570°F, and a steam generator pressure of 718 psia.

LOCA Analysis

The effect of a 5200 gpm reduction in RCS raw measured total flow rate on the LOCA-related analyses was evaluated using Westinghouse's NRC-approved methodologies. For a minimum measured flow (MMF) of 174,000 gpm, the corresponding thermal design flow (TDF) is 170,400 gpm. The difference between the MMF and TDF is the measurement uncertainty. The evaluation shows that, in all cases, the effect of the flow reduction would not result in exceeding any design or regulatory limits for PBNP Unit 2 at full power conditions.

Westinghouse also completed a Peak Cladding Temperature (PCT) assessment for PBNP Units 1 and 2 Large Break LOCA analysis to support an increase in the available steam generator tube plugging (SGTP) margin from 20% to 25%. The level of SGTP assumed in the PBNP Units 1 and 2 Large Break LOCA analysis is 25%. However, 5% of that margin is unavailable as a result of an issue concerning the postulated steam generator tube crush resulting from combined LOCA and seismic loads. This issue is described in Westinghouse Letter, WEP-91-171, dated June 20, 1991, "Wisconsin Electric Power Company, Point Beach Units 1 and 2, ECCS Evaluation Model Changes." Westinghouse has converted the 5% effective SGTP into a PCT increase in order to increase the SGTP margin from 20% to 25%.

Transient (non-LOCA) Analysis

The impact of the reduced flow on the non-LOCA FSAR analyses for PBNP Unit 2 was evaluated. As a result of the reduction in minimum measured flow, the core thermal safety limits become more limiting at all powers and pressures. A new core thermal safety limits plot for the PBNP Technical Specifications is required. This figure applies only to Unit 2. Additionally, an increase in the positive wing slope of the overtemperature ΔT $F(\Delta I)$ penalty is required to protect against adverse axial power shapes which could arise during Condition I and II transients. The current TS value is 2.0. The new safety analysis value for the revised core thermal safety limits is 3.1.

The reduction in RCS raw measured total flow rate is a departure from nucleate boiling (DNB) penalty. Generic DNB ratio (DNBR) margin has been allocated to maintain the current Revised Thermal Design Procedure (RTDP) DNBR limit of 1.33. The most DNB-limiting non-LOCA accidents were reanalyzed to demonstrate this limit remains satisfied for the reduction in RCS flow. Based on this reanalysis, all conclusions of PBNP FSAR Chapter 14 with respect to the DNB acceptance criterion for non-LOCA accidents remain valid for a reduction in the raw measured total flow rate to 174,000 gpm.

An evaluation of the Point Beach FSAR non-LOCA accident analyses that contain non-DNB acceptance criteria was also performed. All acceptance criteria continue to be met with lower RCS flow of 174,000 gpm.

Steam Generator Tube Rupture Analysis

An evaluation was performed to determine the effect of the reduction on RCS raw measured flow rate on the SGTR analysis. The evaluation assumed a 24% tube plugging level with a reduction in thermal design flow to 170,400 gpm, an RCS pressure of 2000 psia, a T_{avg} of 570°F, and a steam pressure of 718 psia. The

current PBNP SGTR analysis assumes an RCS pressure range of 2000-2250 psia, a T_{avg} temperature range of 573.9-575.0°F, and a steam pressure range of 740-821 psia.

It has been determined that the decrease in the break flow due to the lower initial RCS pressure of 2000 psia will more than offset the increase in break flow due to the lower T_{avg} of 570°F. The decrease in steam pressure to 718 psia increases primary to secondary break flow by approximately 1% over the current SGTR analysis. This will have a minimal effect on the offsite radiation doses (approximately 1% increase), which will still be significantly less than the permissible limits of 10CFR100. Thus, the reduction in RCS flow will not increase the consequences of a SGTR.

Containment Integrity Analysis

There is no significant impact of the reduction in RCS flow to 174,000 gpm on the LOCA Mass and Energy releases. The only major effect of changes in thermal design flow are the resulting changes in RCS initial temperatures. Thermal design flow has essentially no direct affect on the mass and energy releases, and no impact on the total energy content of the RCS. The changes in thermal design flow do not adversely affect the normal plant operating parameters, system actuations, accident mitigating capabilities or assumptions important to the short term and long term LOCA mass and energy releases, and the subcompartment and containment response to these events, or create conditions more limiting than those assumed in these analyses.

An increase in the steam generator tube plugging level is a benefit to the long term mass and energy and containment integrity calculations as the energy content of the RCS is reduced. There is no impact on the short term mass and energy and subcompartment analyses.

The current design basis LOCA short and long term mass and energy release analysis, the containment integrity analysis, and the subcompartment analysis have been evaluated relative to operation at an average RCS temperature of 570°F, raw measured RCS flow rate of 174,000 gpm. It has been concluded that the current accident analyses remain bounding.

Mechanical Component Integrity Analysis

The affect of the reduced RCS flow on design transients used for component fatigue calculations was evaluated. The evaluation focused on the number of fatigue cycles as well as the magnitude of the temperature changes. The evaluation concluded that the design transients remain appropriate as long as the plant is operated primarily in a base-load mode and operation with the reduced RCS flow does not continue past December 31, 1996.

The impact of the reduced RCS flow on the operation of the PBNP Unit 2 steam generators was also evaluated. A key parameter to be factored into the evaluation is the new steam generator pressure of 718 psia. It was concluded that operation under the new conditions is bounded by previous evaluations if operation is restricted to a nominal RCS pressure of 2000 psia.

Therefore, it has been concluded that analyses of systems and components remain valid under the new operating conditions for PBNP Unit 2 with the following limitations:

- the plant is operated primarily in a base-load mode,
- operation at a thermal design flow of 170,400 gpm does not continue past December 31, 1996, and
- the plant is operated at a nominal RCS pressure of 2000 psia.

CONCLUSION

The effect of a 5200 gpm reduction in the RCS total flow rate limit was assessed for each of the PBNP accident analyses. The acceptance criteria of all the accident analyses are still met at this lower flow rate limit. A 5200 gpm reduction in RCS total flow rate limit has been determined to be acceptable. Additionally, this reduction in the reactor coolant system raw measured total flow rate limit will not cause any safety limits to be exceeded and the margins of safety for Point Beach Nuclear Plant Unit 2 are not reduced.

TECHNICAL SPECIFICATION CHANGE REQUEST 177
NO SIGNIFICANT HAZARDS CONSIDERATION

In accordance with the requirements of 10 CFR 50.91(a), Wisconsin Electric Power Company (Licensee) has evaluated the proposed changes against the standards of 10 CFR 50.92 and has determined that the operation of Point Beach Nuclear Plant, Units 1 and 2, in accordance with the proposed amendments does not present a significant hazards consideration. The analysis of the requirements of 10 CFR 50.92 and the basis for this conclusion are as follows:

1. Operation of this facility under the proposed Technical Specifications will not create a significant increase in the probability or consequences of an accident previously evaluated. This proposed change reduces the Unit 2 Reactor Coolant System raw measured total flow rate limit by 5200 gpm. Evaluations performed by Westinghouse and Wisconsin Electric have determined that all the safety analysis requirements are still met at the reduced flow rate limit without increased consequences. A reduction of the RCS flow limit does not affect any parameters that could affect the probability of an accident. Therefore, there is no increase in the probability or consequences of an accident previously evaluated.

2. Operation of this facility under the proposed Technical Specifications change will not create the possibility of a new or different kind of accident from any accident previously evaluated. This proposed change reduces the Unit 2 Reactor Coolant System raw measured total flow rate limit by 5200 gpm. Evaluations performed by Westinghouse and Wisconsin Electric have determined that all the safety analysis requirements are still met at the reduced flow rate limit. There is no physical change to the facility, its systems, or its operation. Thus, a new or different kind of accident cannot occur.

3. Operation of this facility under the proposed Technical Specifications change will not create a significant reduction in a margin of safety. This proposed change reduces the Unit 2 Reactor Coolant System raw measured total flow rate limit by 5200 gpm. Evaluations performed by Westinghouse and Wisconsin Electric have determined that all the safety analysis requirements are still met at the reduced flow rate limit. Generic Departure from Nuclear Boiling Ratio (DNBR) margin has been allocated to maintain the current Revised Thermal Design Procedure (RTDP) DNBR limit of 1.33. The most DNB-limiting non-LOCA accidents were reanalyzed to demonstrate this limit remains satisfied for the reduction in RCS flow. The modification to the overtemperature ΔT function and core safety limits figure

for PBNP Unit 2 prevent the possibility of exceeding the core safety limits. Therefore, this reduction in RCS total flow rate limit does not reduce any existing margin of safety.