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VPNPD-95-10 CFR 50.4
NRC-95-10 CFR 50.90

October 19, 1995

Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555

Gentlemen:

DOCKETS 50-266 AND 50-301
ADDENDUM TO TECHNICAL SPECIFICATIONS CHANGE REQUEST 181
MODIFICATION TO TS 15.3.1.G.3
REACTOR COOLANT SYSTEM RAW MEASURED TOTAL FLOW RATE
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On September 13, 1995, we submitted Technical Specifications Change Request 181 to request 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 (TS). The proposed revisions modified Technical Specification Section 15.3.1.G, "Operational Limitations," Specification 3.b, to reduce the reactor coolant system (RCS) raw measured total flow rate limit by 4000 gallons per minute (gpm) for Unit 2. Our September 13, 1995, letter did not contain a safety evaluation or marked-up Technical Specifications (TS) pages as evaluations of the flow rate limit reduction were not completed.

We have since determined that an RCS flow rate limit reduction of 4500 gpm more closely correlates to a 30% steam generator tube plugging level. An evaluation of the effects of a 4500 gpm reduction of the RCS flow rate limit has been completed. Assumptions in the evaluation include a nominal average RCS temperature of 570°F and a main feedwater temperature of 374°F. Changes to TS Section 15.1, "Definitions," Figure 15.2.1-2, "Reactor Core Safety Limits, Unit 2," and the Basis for TS Section 15.3.1.G are also required to support the flow rate limit reduction.

This addendum to TSCR 181 includes our safety evaluation for the 4500 gpm RCS flow rate limit reduction, a revised no significant hazards consideration, and marked-up Technical Specifications pages.

A001

DESCRIPTION OF CURRENT LICENSE CONDITION

Technical Specification (TS) Section 15.1, "Definitions," contains definitions of frequently used terms applicable to PBNP.

TS Section 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 ≥174,000 gpm for Unit 2.

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.

DESCRIPTION OF PROPOSED CHANGES

This Technical Specification Change Request (TSCR) proposes to modify TS Section 15.1 by adding a note to the definition of rated power as follows:

> "* For Unit 2: if the Reactor Coolant System raw measured total flow rate is <174,000 gpm but ≥169,500 gpm, Unit 2 shall be limited to ≤98% rated power."

This note is also being added to Specification 15.3.1.G.3.b. The current Unit 2 RCS flow limit of 174,000 gpm, as indicated in Specification 15.3.1.G.3.b, is not being changed as this limit applies to operation at rated power. The current value for rated power (1518.5 MWt) was used in the analysis to support the RCS flow rate limit reduction. Therefore, the RCS flow limit for operation at rated power cannot change. However, as described in the attached safety evaluation, operation must be limited to less than or equal to 98% rated power if RCS raw measured total flow rate is less than 174,000 gpm but greater than 169,500 gpm.

The Basis for TS Section 15.3.1.G is being modified to read as follows:

> "The reactor coolant system total flow rate limit 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 limit for Unit 2 at rated power is 174,000 gpm. This is based on an assumed measurement

uncertainty of 2.1 percent over a thermal design flow of 170,400 gpm. However, Unit 2 is analyzed to support operation with a reactor coolant system total flow rate limit of 169,500 gpm. This is based on an assumed measurement uncertainty of 2.1 percent over a thermal design flow of 166,000 gpm. If the Unit 2 RCS raw measured total flow rate is less than 174,000 gpm but greater than or equal to 169,500 gpm, operation is limited to less than or equal to 98% rated power as described in the note to Specification 15.3.1.G.3.b. The raw measured flow is based upon the use of normalized elbow tap differential pressure which is calibrated against a precision flow calorimetric at the beginning of each cycle."

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

Based on the eddy current test results from the 1994 refueling outage and industry experience since that time, it is possible that the number of tubes that will have to be removed from service during the Unit 2 refueling outage could result in the need for a reduction of the current Technical Specification RCS flow rate limit. We will not know the exact amount of tube plugging required until the eddy current testing is completed.

We believe a 4500 gpm reduction in the RCS raw measured total flow rate limit will bound any reasonably expected increase in the level of steam generator tube plugging. The proposed 4500 gpm reduction in the RCS raw measured total flow rate limit for Unit 2 has been determined to be acceptable based on evaluations performed by Westinghouse and 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.

We submitted TSCR 181 on September 13, 1995, without a safety evaluation or marked-up TS pages in an attempt to avoid an exigent condition. However, your staff determined that the information contained in our submittal was not sufficient to publish in the Federal Register. The analysis of the RCS flow rate limit reduction has been ongoing since we submitted TSCR 181. We received the results of this analysis on October 13, 1995.

A reduction in the RCS raw measured total flow rate limit 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 8, 1995. We believe this submittal is timely and could not have been avoided and thus meets the criteria of 10CFR50.91 for processing as an exigent change. As such, we request this change request be processed as an exigent Technical Specification Change Request and be issued by November 6, 1995.

Please contact us if there are any questions.

Sincerely,

Gary M. Kriesur
for

Bob Link
Vice President
Nuclear Power

KVA/kmc

cc: NRC Resident Inspector
NRC Regional Administrator
PSCW

Subscribed and sworn before me on
this 19th day of October 1995.

Gloria J. Monsoor
Notary Public, State of Wisconsin

My commission expires 6-2-96.

TECHNICAL SPECIFICATIONS CHANGE REQUEST 181
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," to reduce the reactor coolant system raw measured total flow rate limit by 4500 gallons per minute (gpm) for Unit 2. The proposed revisions will also modify TS Section 15.1, "Definitions," and Figure 15.2.1-2, "Reactor Core Safety Limits, Point Beach Unit 2," to support the flow reduction.

EVALUATION

Westinghouse has performed an evaluation of the affects of a 4500 gpm reduction of the RCS raw measured total flow rate limit 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 fluid and mechanical component integrity considerations.

The 4500 gpm flow rate limit reduction corresponds to a uniform steam generator tube plugging (SGTP) level of approximately 30%. Assumptions in the evaluations include a nominal RCS pressure of 2000 psia, a nominal full power T_{avg} of 570°F, and a main feedwater temperature of 374°F.

LOCA Analysis

The effect of a 4500 gpm reduction in the RCS raw measured total flow rate limit on the LOCA-related analyses was evaluated using Westinghouse's NRC-approved methodologies. For a minimum measured flow (MMF) of 169,500 gpm, the corresponding thermal design flow (TDF) is 166,000 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.

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. It has been determined that the existing Overtemperature ΔT and Overpower ΔT safety analysis setpoints continue to apply to the revised core thermal safety limits.

The reduction in the RCS raw measured total flow rate limit is a departure from nucleate boiling (DNB) penalty. The current Revised Thermal Design Procedure (RTDP) DNBR limit of 1.33 remains valid for the reduced flow conditions. The most DNB-limiting non-LOCA accidents were reanalyzed to demonstrate this limit remains satisfied for the reduction in RCS flow.

For the underfrequency event, acceptable results were not obtained with an initial core power level of 100% of 1518.5 MWt. Acceptable results were obtained for an initial core power level of 98% of 1518.5 MWt. A limitation has been added to the Technical Specifications to limit operation of PBNP Unit 2 to 98% of rated power when Reactor Coolant System raw measured flow is less than 174,000 gpm but greater than 169,500 gpm.

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 limit below 174,000 gpm down to a flow rate of 169,500 gpm, assuming that the core power level is maintained at or below 98% of 1518.5 MWt.

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 a lower RCS flow rate limit of 169,500 gpm.

Steam Generator Tube Rupture Analysis

The Steam Generator Tube Rupture (SGTR) analysis was reanalyzed to ensure that the offsite radiation doses remain below the limits defined in 10CFR100. The primary thermal and hydraulic parameters which affect the calculation of the offsite radiation doses for an SGTR are the amount of radioactivity assumed to be available in the reactor coolant, the amount of reactor coolant transferred to the secondary side of the ruptured steam generator through the ruptured tube, and the amount of steam released from the ruptured steam generator to the atmosphere.

An increase in the steam generator tube plugging level will not affect the amount of radioactivity in the reactor coolant. Thus, an evaluation was performed to determine the effect of 30% SGTP on the primary to secondary break flow and the amount of steam released to the atmosphere. A 3% main steam safety valve (MSSV) tolerance with an additional 13% MSSV blowdown was assumed.

The results of the evaluation show that the offsite radiological doses for a SGTR event are 0.7 and 0.14 rem for the site boundary thyroid and whole-body gamma respectively. These results show a minimal increase over the current offsite doses presented in the PBNP Final Safety Analysis Report and remain a small fraction of 10CFR100 limits. Thus, the reduction in the RCS flow rate limit will not increase the consequences of a SGTR.

Containment Integrity Analysis

There is no significant impact of the reduction of the RCS flow rate limit from 174,000 gpm to 169,500 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. There is also no impact of changes in the feedwater temperature (i.e., to 374°F). The changes in thermal design flow and feedwater temperature 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, a thermal design flow of 166,000 gpm, a feedwater temperature of 374°F, and 30% SGTP. It has been concluded that the current accident analyses remain bounding.

Mechanical Component Integrity Analysis

The effect of the reduced RCS flow rate limit on design transients used for component fatigue calculations was evaluated. The evaluation concluded that the design transients remain appropriate under the new operating conditions for PBNP Unit 2 with the following limitations:

- the plant is operated in a base-load mode without load follow,
- operation at a thermal design flow of 166,000 gpm does not exceed two fuel cycles, and
- the plant is operated at a nominal RCS pressure of 2000 psia.

CONCLUSION

The effect of a 4500 gpm reduction of the RCS total flow rate limit on the safety analysis of record for PBNP Unit 2 has been evaluated. The scope of the evaluation included LOCA, non-LOCA, steam generator tube rupture, and containment analyses as well as mechanical component integrity analyses. The results of the evaluations and analyses support operation of PBNP Unit 2 with a 30% SGTP level under the following conditions:

- the plant is restricted to operation at less than or equal to 98% reactor power when RCS raw measured total flow is <174,000 gpm but ≥169,500 gpm,
- the plant is operated in a base load mode without load follow,
- operation under the conditions defined for 30% SGTP does not exceed two fuel cycles, and
- the plant is operated at a nominal RCS pressure of 2000 psia.

The acceptance criteria of all the accident analyses are still met at this lower flow rate limit. A 4500 gpm reduction of the 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 181
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 4500 gpm. Evaluations performed by Westinghouse and Wisconsin Electric have determined that all safety analysis and regulatory requirements are still met at the reduced flow rate limit without exceeding acceptable limits. 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 4500 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 4500 gpm. Evaluations performed by Westinghouse and Wisconsin Electric have determined that all the safety analysis and regulatory requirements are still met at the reduced flow rate limit. The current Revised Thermal Design Procedure (RTDP) DNBR limit of 1.33 remains valid for the reduced flow conditions.

The most DNB-limiting, non-LOCA accidents were reanalyzed to demonstrate this limit remains satisfied for the reduction in RCS flow. The modifications to power level 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.