Wisconsin Electric
POWER COMPANY

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VPNPD-94-083 10CFR50.4
NRC-94-059 10CFR50.90

September 2, 1994

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

Gentlemen:

DOCKETS 50-266 AND 50-301
REQUEST FOR EXIGENT PROCESSING
TECHNICAL SPECIFICATIONS CHANGE REQUEST 175
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On August 26, 1994, we submitted Technical Specifications Change Request 175, "Modifications to Section 15.4.2, 'In-Service Inspection Of Safety Class Components,'" which requested amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant, Units 1 and 2, respectively. 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." The basis for Technical Specifications Section 15.4.2 was also revised to support the above changes. Marked-up Technical Specifications pages, a preliminary draft of Westinghouse WCAP-14157, a safety evaluation, and a no significant hazards consideration were enclosed.

In order to support in-service inspection of the PBNP Unit 2 steam generator tube sleeves during the upcoming Unit 2 Fall refueling outage, we requested NRC approval of this Technical Specifications Change Request by September 24, 1994. Thus, this request would not permit the normal 30-day public comment period. Therefore, in accordance with the requirements of 10 CFR 50.91(a)(6), we request that the proposed license amendments be processed as an exigent request.

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BASIS FOR REQUEST

As stated in our August 26, 1994, submittal, Point Beach Nuclear Plant plans to inspect the Unit 2 steam generator sleeves during the upcoming fall refueling outage. Since June 1994, personnel from Wisconsin Electric Power Company (WEPCO), Wisconsin Public Service Corporation (WPS), Commonwealth Edison Company (CECO), American Electric Power Corporation (AEP), and Westinghouse Electric Corporation have been coordinating efforts to resolve NRC concerns regarding disposition of HEJ indications similar to those detected by Kewaunee Nuclear Power Plant in April 1994. Point Beach Nuclear Plant, Unit 2, is serving as lead plant for submitting the proposed acceptance criteria due to its upcoming fall 1994 refueling outage. Before a request could be submitted, however, further analysis and results of additional leak rate and structural tests performed by Westinghouse were required. This information was not available until August 25, 1994. Therefore, we believe we could not avoid the circumstances associated with this request for exigent processing.

ENCLOSURES

Enclosed with this request are two updated copies of WCAP-14157. This document, along with the previously submitted draft version, contains information which is proprietary to Westinghouse Electric Corporation. Accordingly, we request that this information be withheld from public disclosure. We will comply with the requirements of 10 CFR 2.790 to provide a non-proprietary version of this material together with an affidavit as soon as the non-proprietary version has been prepared. We will submit the required number of copies of the non-proprietary versions of the information and the required affidavit at that time. In the meantime, we are providing two copies of the proprietary version for your information and use. This updated version supersedes the draft version submitted to you on August 26, 1994. Revision 1 to WCAP-14157 will be issued when additional test results become available.

Also enclosed is a revised "No Significant Hazards Consideration" which removes reference to a bounding Final Safety Analysis Report assumption of 1 GPM primary to secondary leakage which was not applicable to Point Beach Nuclear Plant. As initially submitted, we have 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. We therefore 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.

Please contact us if you have any questions.

Sincerely,

Bob Link

Bob Link
Vice President
Nuclear Power

DAW/jg

Enclosures

cc: NRC Regional Administrator
NRC Resident Inspector
Public Service Commission of Wisconsin

Subscribed and sworn before me on
this 2nd day of September 1994.

Jacqueline Gorecki
Notary Public, State of Wisconsin

My commission expires 10-27-96.

TECHNICAL SPECIFICATIONS CHANGE REQUEST 175
REVISED "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 change will not create a significant increase in the probability or consequences of an accident previously evaluated.

The limiting EOC crack angle of 224° (determined by using the conservative structural model and no allowance for tube/tube support plate friction) would not represent a potential for failure of a tube during recovery from a MSLB. While some slippage of the joint may occur, the maximum forces developed by the joint would support the hardroll joint slipping past the degraded area. Therefore, only a limited leakage potential exists. For degradation below the hardroll region in the upper HEJ, sufficient structural and leakage integrity is provided such that a complete circumferential separation of the parent tube will not affect either structural or leakage integrity of the joint during all plant conditions.

Wisconsin Electric Power Company has performed an analysis in accordance with NUREG-0800 guidelines which shows a maximum primary to secondary leakage of 25.0 gpm in the faulted loop with reactor coolant activity of 1 μCi/g dose equivalent I-131 will not result in off-site dose exceeding a small fraction (10 percent as defined in NUREG-0800) of the 10 CFR 100 requirements. In addition, limiting the maximum primary to secondary leakage during operation to 150 gpd will help to identify tubes with rapid growth rates not bounded by the 25 percent growth allowance that result in leakage which could possibly affect tube integrity during a MSLB.

As previously stated, the postulated degraded tube would have to experience about 3 inches of axial motion prior to tube rupture-type release rates being achieved. Due to tube proximity in the U-bend region and bending restraint provided by the tube support plates, tube motion would be limited to 0.4 to a maximum of 1 inch. Therefore, even if a tube were to experience rapid crack growth not bounded by the structural model assumptions or circumferentially separate and experience slippage, the amount of slippage and subsequent leakage would be limited. In this case, 1 inch of tube motion would still provide intimate contact

between the tube and sleeve in the hardrolled region. If it is further postulated that the remaining length of tube-to-sleeve hardroll interference (about .25 to .5 inch) for a tube which has slipped 1 inch is deformed to a "bell" shape due to the pressure difference, or the tube were to experience slippage up to 2 inches, leakage would be limited by the thin gap between the sleeve hydraulically expanded region and the tube hardrolled region. The maximum leakage would be expected to be approximately 30 percent to 50 percent of the normal Unit 2 makeup capacity.

The results show that we remain within the acceptance criteria of the aforementioned FSAR Chapter 14 accident analyses. Therefore, the proposed changes will not create a significant 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.

Implementation of the proposed acceptance criteria does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism which could result in a tube rupture during a faulted event outside of the PBNP design basis. Neither a single or multiple tube rupture event would be expected in a steam generator in which the sleeved tube plugging criteria has been applied. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of this facility under the proposed Technical Specifications change will not create a significant reduction in a margin of safety.

The use of the proposed acceptance criteria is demonstrated to maintain steam generator tube integrity commensurate with the criteria of Regulatory Guide 1.121. Regulatory Guide 1.121 describes a method acceptable to the NRC staff for meeting reactor coolant system general design criteria (GDCs) by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by in-service inspection, for which tubes with unacceptable cracking should be removed from service or repaired. Upon implementation of the circumferential crack acceptance criteria for sleeved tubes, even under worst case conditions, the occurrence of circumferential cracks at the lower hardroll transition of the upper HEJ elevation is not expected to lead to a steam generator tube rupture event during normal or accident plant conditions.

Addressing the considerations in Regulatory Guide 1.83, "In-Service Inspection of Pressurized Water Reactor Steam Generator Tubes," Revision 1, implementation of the proposed acceptance criteria is supplemented by enhanced eddy current inspection using a probe capable of detecting parent tube circumferential flaws in sleeve joints.

In addition, implementation of the proposed acceptance criteria will decrease the number of tubes which must be plugged. The installation of steam generator tube plugs reduces the RCS flow margin. Thus, implementation of the proposed acceptance criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging, thereby maintaining departure from nucleate boiling (DNB) margins.