Wisconsin
Electric
POWER COMPANY

231 W. Michigan, P.O. Box 2046, Milwaukee, WI 53201-2046 (414) 221-2345

VPNPD-96-0088

10 CFR 50.4
10 CFR 50.90

October 21, 1996

Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555-0001

Gentlemen:

DOCKETS 50-266 AND 50-301
SUPPLEMENT TO TECHNICAL SPECIFICATIONS
CHANGE REQUESTS 188 AND 189
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In a conference call on October 8, 1996 you requested additional information on Technical Specifications Change Requests (TSCRs) 188 and 189. TSCRs 188 and 189 were submitted in letters dated June 4, 1996. Supplements to TSCRs have been submitted in letters dated August 5, 1996 and September 26, 1996. These requests propose amendments to the Point Beach Technical Specifications that were identified by analyses performed in support of Unit 2 operations following replacement of steam generators this fall.

We are providing additional information as attachments to this letter. We have determined that the additional information does 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. The original "No Significant Hazards" determinations for operation under the proposed Technical Specifications remain applicable.

If you require additional information, please contact us.

Sincerely,

Gary M. Krieser
for
Bob Link
Vice President
Nuclear Power

CAC/kmc

9610290214 961021
PDR ADOCK 05000266
P PDR

cc: NRC Resident Inspector
NRC Regional Administrator
PSCW

A001 1/1

Subscribed and sworn before me on
this 21st day of October, 1996.

Jacqueline Gorecki
Notary Public, State of Wisconsin

My commission expires 10-22-2000.

290051

Attachment 1

Additional information for Technical Specifications Change Requests 188 and 189:

(The numbered items are based on requests made by the NRC staff during telephone conversations on October 8).

1. Provide additional information as to why the peak pressure analysis for loss of load (FSAR Section 14.1.9) starts from the initial condition 30 psi less than the maximum nominal operating pressure of 2250 psia (i.e. 2220 psia).

Applying the pressure uncertainty in the negative direction to establish the initial pressure for this transient results in a higher peak pressure because the reactor trip on the high pressurizer pressure signal is delayed. This causes more heat to be generated after the loss of load and prior to the reactor trip, thereby resulting in a somewhat larger heatup of the reactor coolant and pressurization rate at the time of reactor trip. Thus, the resultant peak RCS pressure is slightly higher than if a higher initial pressure is assumed. This methodology is based on previous calculations performed for similar plants. A tabled titled "Loss of Load Sensitivity Analysis Summary" is provided that shows in each case analyzed starting at the lower RCS pressure results in a higher peak RCS pressure. (These sensitivity analyses were performed for a reactor power uprate feasibility study for Point Beach completed in 1995).

2. Provide additional information regarding why we analyze only hot shutdown cases for the main steam line break (MSLB) event.

The topical report on steam line break core response, WCAP-9226-P-A "Reactor Core Response to Excessive Secondary Steam Releases" which has been reviewed and approved by the NRC, concludes that it is the post-trip portion of the at-power steam line break which is of importance, since the pre-trip portion will be effectively mitigated by the reactor protection system, primarily the Overpower Delta-T trip function. The post-trip portion of the at-power steam line break is bounded by a steam line break from hot zero power (HZP) conditions where the reactor is assumed to be in a tripped condition (i.e., with all RCCAs inserted less the most reactive RCCA which is assumed to be stuck in the fully withdrawn position). WCAP-9226 Rev.1 concludes that the largest double-ended steam line rupture at end of life, hot shutdown conditions with all RCCAs inserted less the most reactive RCCA is a limiting and sufficiently conservative licensing basis analysis to demonstrate that the Westinghouse PWR is in compliance with 10 CFR 100 criteria for Condition II, III, and IV steam line break transients. These conclusions are reflected in the Point Beach FSAR. The steam line rupture analysis based on the hot shutdown initial condition is the current licensing basis for PBNP.

3. Initial break flow on page 14.2.4-1 of the steam generator tube rupture (SGTR) accident was not shown as changed in the submittal. Is this correct?

We inadvertently failed to identify the FSAR change associated with the primary leakage rates for the steam generator tube rupture accident. Section 14.2.4.2 item 1. of the FSAR will be modified as follows:

> *"Primary leakage takes place initially at a high rate (about 87 lbm/sec) but rapidly drops to a lower leakage rate (about 55 lbm/sec)."*

The integrated flow rates shown on page 14.2.4-6 of the FSAR markup are correct. Results for the first 30 minutes of the event are reported as 114,500 pounds of liquid and the amount of steam released from the ruptured steam generator is calculated to be 68,900 pounds. Integrated flow rates are used in the analysis to calculate dose. Therefore, this change in leakage rates in the FSAR markup does not change the result of the calculation.

4. Provide additional information regarding evaluation of the RCS average temperature range evaluation for the small-break loss-of-coolant-accident (SBLOCA). In particular, how was 573.9°F average temperature covered? The evaluation mainly focuses on the effects of average temperature less than 570°F.

The RCS average temperature range of 557 to 573.9°F was considered in the Point Beach replacement steam generator small break LOCA evaluation. The original small break LOCA analysis (documented in FSAR §14.3.1) was performed at a average temperature 570°F, as was noted in the evaluation. Additionally, the evaluation documents that a lower RCS average temperature would provide a benefit.

The initial plant conditions analyzed are not always the same as the initial conditions supported by the analysis for Appendix K LOCA analyses, and reporting the actual analysis average temperature can be misleading. This occurs because Appendix K LOCA analyses typically are performed using "bounding" conditions, but not all parameters can be identified as having a bounding direction.

This was recognized by the NRC in the 1976/1977 timeframe, when LOCA sensitivity to average temperature led to NRC requests for information from all licensed LOCA vendors in the US. Westinghouse provided documents to the NRC which demonstrated that the sensitivity to average temperature was plant and break-size dependent, and Westinghouse proposed a methodology and analyses which supported a very specific LOCA set of initial plant conditions as a resolution. Although the analyses are performed to support a specific set of operating conditions, the actual analysis conditions are slightly different to bound operation of the plant. The NRC verbally agreed with the Westinghouse average temperature methodology, but no official written approval was provided in 1977.

The actual analysis average temperature was 570.0°F for the Point Beach small break LOCA, but the operational average temperature supported by that analysis is 573.9°F. This 3.9°F apparent discrepancy is in fact the difference you get in average temperature when you generate LOCA specific parameters. This methodology includes allowance for a +/- 4°F uncertainty in the supported plant conditions. Therefore, the current small break LOCA analysis supports operation of the Point Beach Units at any average temperature below 573.9°F with uncertainty bands of 577.9 to 569.9°F. No peak clad temperature (PCT) penalty has been assessed based on the evaluation for the RCS average temperature range of 557 to 573.9°F because operation at lower temperatures is a PCT benefit.

5. Provide additional information to explain why harsh environment is not included in uncertainty for SBLOCA setting limits.

A normal environment, rather than a harsh environment, is used to calculate SBLOCA setting limit uncertainty for low pressurizer pressure because the protective action is completed before the harsh environment can affect the transmitter output, as explained below:

The uncertainty calculation (PBNP-IC-12 provided as an attachment to the letter dated August 5, 1996) uses normal environmental conditions in calculating the uncertainty of the low pressurizer pressure reactor trip signal for SBLOCA. Justification for not assuming a harsh environment is provided in assumptions 5.6 and 5.8 of this calculation. Normal conditions can be assumed because the low pressurizer pressure setpoint is reached early enough in the SBLOCA event so that the transmitters are not affected. The reactor trip will be complete before the harsh environment caused by a SBLOCA affects loop uncertainty.

In order to predict the transmitter environment, an assumption is made in the uncertainty calculation that the harsh environment caused by a LBLOCA bounds the harsh environment caused by a SBLOCA. LBLOCA containment conditions are used because SBLOCA conditions are not available. LBLOCA is considered to be limiting for containment conditions because the mass and energy release rate to containment is greater. A LBLOCA will pressurize containment faster and create a higher average temperatures in the containment building because of the larger break area.

The use of average containment conditions is appropriate for calculating SBLOCA trip settings because the low pressurizer pressure transmitters and cable are not located in the path of the steam jet. These transmitters and cable are located outside the shield walls for the reactor coolant system. Therefore, the environmental conditions of concern are those of the containment atmosphere and not the steam jet. An exception is a break in an instrumentation line. The jet issuing from an instrumentation line may impinge on the low pressurizer pressure transmitters. However, an instrumentation line break is so small, 3/8 inch OD, that the normal makeup flow rate is typically adequate to maintain pressurizer pressure long enough for the operator to respond without activating protective features (as described in the PBNP FSAR page 14.3.1-1).

The preceding information about break area and location supports the following conclusion: The containment environmental conditions assumed in pressurizer pressure uncertainty calculation (PBNP-IC-12) of 45 psig and 265°F are a conservative estimate of the environmental conditions, at the time of reactor trip (t=5.6 seconds), affecting the pressurizer pressure transmitters when they are required to perform their function following a SBLOCA. Environmental qualification test data for these transmitters shows that the transmitter output is not adversely affected even if these conditions exist at the transmitter for up to 60 seconds.

Transmitter performance as a function of time in a harsh environment is provided in test reports from the manufacturer. The results of transmitter testing shows that the output error will be less than 1% of calibrated span after 60 seconds of exposure to the harsh environment. The reason for the relatively small change in output after 60 seconds of exposure is that it takes time for the transmitter itself to heat up.

Test data shows that the components inside of the transmitter take more than 30 minutes to reach ambient conditions. After 60 seconds in a silicon oil bath at 333°F the components inside of the transmitter have increased ≤5°F. The largest increase measured was from 75°F to 80°F. A 5°F increase in this temperature range is expected to contribute only ±0.125% to instrument uncertainty. This uncertainty is very small compared to the temperature uncertainty allowed for normal operating conditions.

In summary, normal environmental conditions are used to calculate the uncertainty of the low pressurizer pressure reactor trip signal for SBLOCA. The reactor trip is complete before the harsh environment caused by a SBLOCA increases loop uncertainty. Test data show that exposure to ambient conditions up to 333°F have a small impact on transmitter output, less than 1% of span, for exposures of up to one minute. The ambient conditions of the test are more severe than the ambient condition expected in containment following a SBLOCA. The SBLOCA conditions are not known, but they are bounded by LBLOCA conditions due to break size and location considerations. LBLOCA conditions at the 5.6 second trip time are known (45 psig and 265°F), at the time a low pressurizer pressure reactor trip is expected and as stated previously these conditions are bounded by the environmental qualification test conditions for this instrumentation.

An additional request was made on October 11, 1996 to provide the time sequence of events and pressure response, including the points at which the reactor trip and SI would actuate for the dropped rod, steam generator tube rupture, steam line rupture and small break LOCA analyses. The following information is provided:

1. Dropped Rod

The Point Beach dropped rod analysis is based on generic statepoints. The analyses which define the generic statepoints used a nominal pressure of 2250 psia and a low pressurizer pressure (LPP) trip setpoint of 1860 psia. Since it is the deviation from the nominal conditions which is of concern for dropped rod, the "effective" trip setpoint assumed is 1860 psia for 2250-psia operation and 1610 psia for 2000-psia operation (i.e., trip setpoint 390 psi below nominal pressure). Since there were many cases run for the generic program, and it is known that the cases which trip on LPP are non-limiting, a sequence of events and pressure plots are not available.

The following information is offered regarding the treatment of the LPP trip in this analysis. The only reactor trip actuation credited in the statepoint calculations is the LPP reactor trip function. A reactor trip on LPP generally occurs in cases with a small negative or zero MTC when the dropped rod worth exceeds the control bank worth. For these cases, the reactor is effectively being shut down with only a limited return to power. Modeling the LPP trip function provides a convenient measurement of non-limiting dropped rod and control bank worth combinations with respect to the potential return to power. Without modeling the LPP setpoint, the analysis could artificially predict low RCS pressures outside the range of the DNBR correlations.

2. Steam Line Break - Core Response

The sequence of events for the Steam Line Break event will not include the requested information (time of reactor trip and SI actuation on LPP) because these functions are not actuated during the transient. Likewise, the LPP reactor trip and SI setpoints are not relevant to the SLB analysis, because they are never used. Since the functions are not actuated the setpoints are not relevant.

3. Steam Generator Tube Rupture

The time sequence of events for the steam generator tube rupture (SGTR) case which resulted in the limiting primary to secondary break flow is:

Tube rupture occurs	0.0 seconds
Reactor trip/SI	201.6 seconds
Break flow terminated	1800.0 seconds

The time sequence of events for the case which resulted in the limiting steam releases is:

Tube rupture occurs	0.0 seconds
Reactor trip/SI	62.1 seconds
Break flow terminated	1800.0 seconds

The calculation is not a transient calculation, therefore an RCS pressure as a function of time plot was not generated. The analysis assumes that the transient starts at the nominal RCS pressure and the pressure is reduced linearly to the low pressurizer pressure SI setpoint (1770 psia or 1880 psia, respectively) at the times provided above where SI and reactor trip occur simultaneously. Directly following reactor trip/SI, the pressure is assumed to stabilize at the point where break flow equals SI. This pressure is 1525 psia. This pressure is maintained until break flow termination at 30 minutes. A figure is provided that shows the pressure profiles that were used.

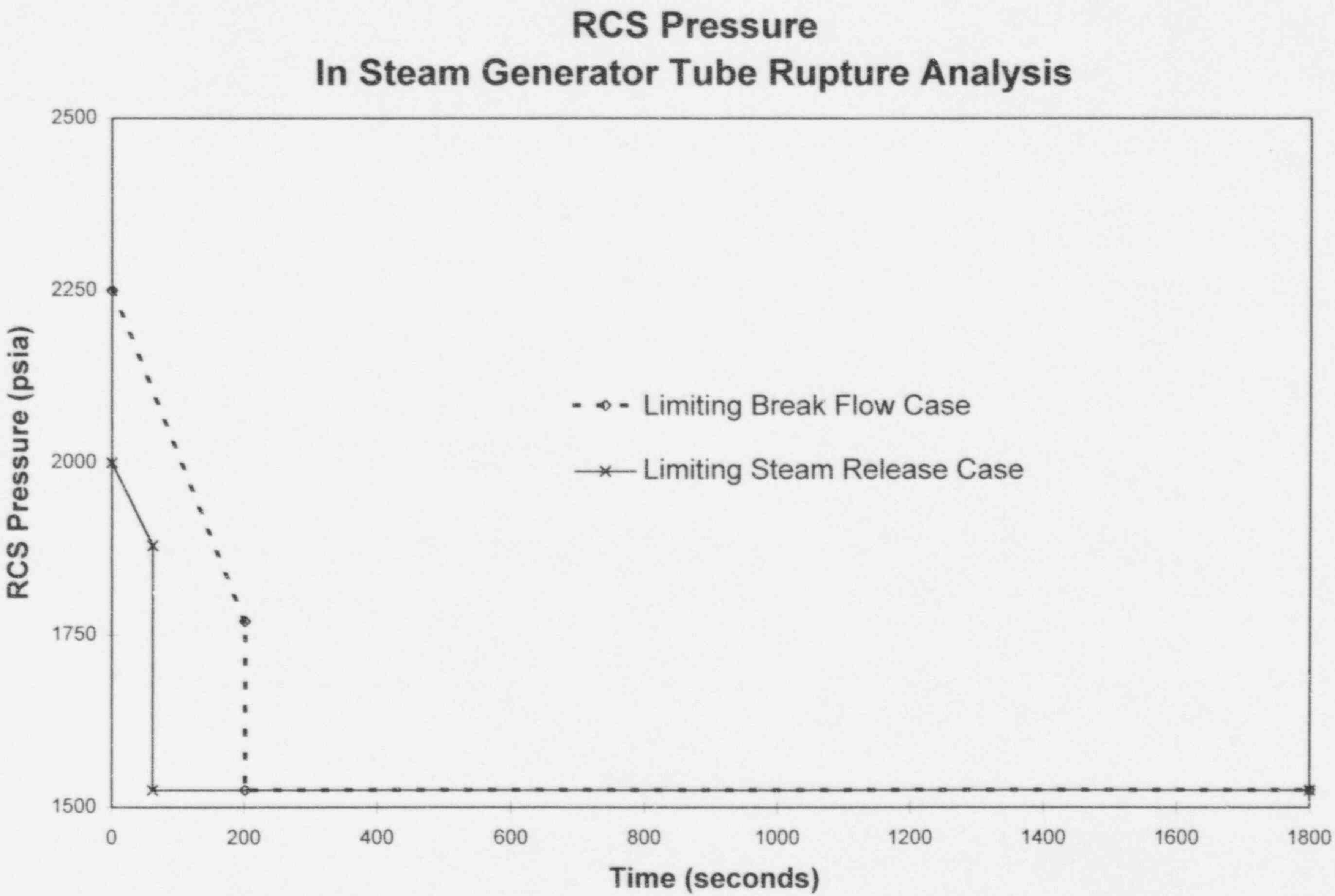
4. Small Break LOCA

The low pressurizer pressure safety injection trip used in the Small Break LOCA analysis for both 2250 psia and 2000 psia operation was 1500 psig. The reactor trip setpoint used was 1760 psig for both RCS pressures. The time sequence and pressure response from the PBNP FSAR are provided.

Loss of Load Sensitivity Analysis Summary

Run Title	NSSS Power (MWt)	Nominal Vessel T-avg (°F)	Initial Vessel T-avg (°F)	Initial Przr Pressure (psia)	Nominal Steam Temp. (°F)	Hi Pres Setpoint Reached (sec)	PSVs Open (sec)	Peak RCS Pressure (psia)	Peak SG Pressure (psia)
lol-1	1610	570.0	566.0	2219.6	508.0	6.7	10.2	2726	1184
lol-2	1610	570.0	566.0	2280.4	508.0	5.6	9.3	2717	1178
lol-3	1610	574.4	578.4	2219.6	512.8	6.5	9.9	2732	1205
lol-4	1610	574.4	578.4	2280.4	512.8	5.4	9.1	2722	1201
lol-5	1660	570.0	566.0	2219.6	506.1	6.6	10.0	2731	1184
lol-6	1660	570.0	566.0	2280.4	506.1	5.5	9.2	2721	1178
lol-7	1660	573.5	577.5	2219.6	509.9	6.4	9.7	2736	1204
lol-8	1660	573.5	577.5	2280.4	509.9	5.3	9.0	2726	1199

RCS and SG pressure limits are 2748.5 psia and 1208.5 psia, respectively. Peak RCS pressure corresponds to the RCP outlet pressure.



RCS Pressure
In Steam Generator Tube Rupture Analysis
RCS Pressure (psia)
2500
2250
2000
1750
1500
Limiting Break Flow Case
Limiting Steam Release Case
0
200
400
600
800
1000
1200
1400
1600
1800
Time (seconds)

TABLE 14.3.1-2

FOUR-INCH SMALL BREAK LOCA

TIME SEQUENCE OF EVENTS

	4 in.
Start	0.0
Reactor Trip Signal (sec.)	5.6
Safety Injection Signal (sec.)	14.5
Top of Core Uncovered (sec.)	157.1
Top of Core Covered (sec.)	181.0
Accumulator Injection Begins (sec.)	326.0
PCT Occurs (sec.)	179.0

FIGURE 14.3.1-3

DEPRESSURIZATION TRANSIENT (4 INCH)