DCSMS-016

APR 8 1983

DISTRIBUTION:
Docket File
NRC PDR
L PDR
NSIC
ORB#3 Rdg
DEisenhut
JHeltemes
OELD
ACRS-10
ELJordan
JMTaylor
PMKreutzer-3
RAClark
TColburn
BSheron
TMarsh
Gray File
BMann

Docket Nos. 50-266
and 50-301

Mr. C. W. Fay
Vice President - Nuclear Power
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

Dear Mr. Fay:

By letter dated September 17, 1982 you requested changes to Technical Specifications (TS) for Facility Operating Licenses DPR-24 and DPR-27 for the Point Beach Nuclear Plant Units 1 and 2 respectively. The proposed TS would allow operation at reduced power and reduced reactor coolant thermal design flow (TDF). The changes were requested because of the possibility that anticipated steam generator tube-plugging associated with the then forthcoming steam generator eddy-current inspection might reduce reactor coolant TDF below its 100% rated power TS limit of 178,000 gallons per minute (GPM).

Additional information was requested by the NRC staff in letters dated October 22, 1982 and November 19, 1982 and during various telephone conference calls which you responded to by letters dated October 15, November 9, and December 10, 1982.

Reactor coolant system flow measurements indicate that Point Beach Unit 1 reactor coolant TDF has remained above the TS limit of 178,000 GPM following the steam generator tube plugging conducted during the Fall 1982 refueling outage. However, even though the proposed TS changes are no longer necessary, you have requested that the NRC staff complete its review of these changes.

We have completed our review of the proposed TS changes. Based on our review we have concluded that the consequences of a design break LOCA at 84% power, 95% TDF and 18% steam generator tube plugging (SGTP) are bounded by a previously accepted analysis at 100% power and TDF and 18% SGTP. Therefore, if future calorimetric reactor coolant flow tests indicate that TDF is less than 100% but not less than 95%, SGTP is held to a maximum of 18% and the power level is administratively limited to a maximum of 84%, you need not submit a detailed supporting LOCA analysis justifying the proposed TS changes unless required by other considerations. However, if operation at conditions other than those described above is contemplated, a supporting LOCA analysis for the new operating conditions is required. Our Safety Evaluation (SE) is enclosed.

OFFICE							
SURNAME							
DATE							

As discussed with members of your staff, we are withholding issuance of the proposed TS changes supported by the enclosed SE. If these TS changes become necessary we request that you notify us in writing confirming that plant operating conditions are within the limits described above.

Sincerely,

Original signed by
Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosure:
Safety Evaluation

cc: See next page

OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL				
SURNAME	PMKreutzer	TColburn/pn	RAClark				
DATE	4/8/83	4/8/83	4/8/83				

Wisconsin Electric Power Company

cc:
Mr. Bruce Churchill, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

USNRC Resident Inspectors Office
6612 Nuclear Road
Two Rivers, Wisconsin 54241

Mr. James J. Zach, Manager
Nuclear Operations
Wisconsin Electric Power Company
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, Wisconsin 54241

Mr. Gordon Blaha
Town Chairman
Town of Two Creeks
Route 3
Two Rivers, Wisconsin 54241

Ms. Kathleen M. Falk
General Counsel
Wisconsin's Environmental Decade
114 N. Carroll Street
Madison, Wisconsin 53703

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: Regional Radiation
Representative
230 S. Dearborn Street
Chicago, Illinois 60604

Chairman
Public Service Commission of Wisconsin
Hills Farms State Office Building
Madison, Wisconsin 53702

Regional Administrator
Nuclear Regulatory Commission, Region III
Office of Executive Director for Operations
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Enclosure

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION (SE)

POINT BEACH NUCLEAR PLANT UNIT 1

OPERATION WITH REDUCED THERMAL DESIGN FLOW (TDF)

Introduction

By letter dated September 17, 1982, Wisconsin Electric Power Company (licensee) requested changes to the Technical Specifications (TS) for Point Beach Nuclear Plant Unit 1. These proposed changes would allow operation at reduced power level (91%), reduced thermal design flow (TDF) (95%) and with an increased percentage of steam generator tube plugging (SGTP) (24%). These changes were prompted by the results of the licensee's previous calorimetric flow test (178,900 GPM or 100.5% of the TS limit of 178,000 GPM at 100% rated power) and the anticipation that further SGTP might occur as a result of the then forthcoming Unit 1 steam generator eddy current inspection.

Discussion

The purpose of this SE is to present the NRC staff's evaluation of the Point Beach Unit 1 Safety Analysis for operation at reduced TDF presented in Attachment A of reference 2. This SE also presents the NRC staff's evaluation of licensee-submitted sensitivity study results related to the licensee's previously approved large break loss of coolant accident (LOCA) analysis. Reference 2 proposes changes to the TS to enable operation of Point Beach Unit 1 at 91% rated power and a minimum primary flow rate of 169,000 GPM or 95% of rated TDF. This is the predicted primary flow rate

if 24% of the steam generator tubes are plugged. Attachment A to reference 2 presents the non-LOCA accident and transient analyses for operation at reduced TDF. The following assumptions were utilized:

Maximum core thermal power	1382 Mwt (91%)
TDF	169,000 gpm (95%)
Steam Generator Plugging Level	24%
$T_{average}$	572.9°F
ΔT	55.5°F
Primary Pressure	2000 psia

This SE includes our evaluation of transients and accidents that could be significantly affected by the above operating conditions. These include loss of external load, loss of normal feedwater, locked rotor and steam line break. The following transients are not adversely and/or significantly affected by the above conditions and are therefore not further discussed: CVCS malfunction, startup of an inactive reactor coolant loop, reduction in feedwater enthalpy, excessive load increase, loss of reactor coolant flow.

Evaluation of Transient and Accidents

1. Loss of External Electrical Load

The FSAR analyses for the loss of external electrical load were performed for four cases, i.e., with automatic reactor control and credit taken for pressurizer relief and spray, at both beginning of core

life (BOL) and end of core life (EOL), and with manual reactor control, no credit for pressurizer relief valve actuation and spray, at both BOL and EOL. Initial conditions were assumed to be 102% power, 581°F T_{avg}, and 2250 psia. No credit is taken for direct reactor trip due to loss of load, and it is assumed that the reactor trips on high pressure at 8.5 seconds. For each case analyzed the DNBR increases during the transient. The most severe peak pressure is 2514 psia for the manual control case at BOL. The primary safety valves lift but no water relief occurs.

In reference 2 the licensee compares this transient at reduced TDF conditions with the FSAR analysis and indicates that the pressure rise will be slightly more rapid because of reduced TDF and extensive steam generator plugging. The time to reach the high pressure trip set point would be less than for the FSAR case and therefore the total energy input to the coolant would be less. However, this is not a good comparison since the FSAR analysis was performed at 2250 psia, while operation at reduced TDF will be at 2000 psia. Since the high pressure trip setpoint is the same for both operating pressures (i.e., 2400 psia) the time to trip may actually be longer for reduced pressure operation.

The staff questioned the licensee's assumption that reactor trip due to high pressure would be more rapid for the reduced TDF case than during conditions described in the FSAR analysis and requested additional confirmatory information justifying the analysis. In Reference (13), the licensee

indicated that for operation at reduced pressure, DNB is limiting, while peak pressure is limiting for operation at rated pressure. The reactor would trip on overtemperature delta T at reduced pressure. The consequences of this transient with regard to DNBR would be bounded by the "uncontrolled rod withdrawal at power" (URWAP) analysis. The URWAP analysis at reduced TDF, presented in Reference (2), indicates that minimum DNBR does not fall below 1.3.

We find, based on our review of previous analyses and the additional information provided by the licensee, that the consequences of loss of load transient at reduced TDF will not result in unacceptable fuel performance and that the primary system pressure will not exceed allowable values. The licensee's loss of external load analysis is, therefore, acceptable.

2. Loss of Normal Feedwater

The FSAR analysis for the loss of normal feedwater transient assumed this event to occur at 102% power, at minimum normal steam generator level, and loss of the reactor coolant pumps. The reactor trips on low-low steam generator level. One auxiliary feedwater pump starts one minute after the low-low steam generator level signal, delivering flow to one steam generator. Secondary steam relief is via the steam generator safety valve. The tube sheet of the steam generator receiving auxiliary feedwater flow is always covered. The capacity of one auxiliary feedwater pump is sufficient to prevent water relief from the primary relief and safety valve. The peak $T_{avg.}$ is 609°F at about ½ hour after transient start. The peak pressurizer liquid volume is 790 ft^3.

In reference 2 the licensee indicates that at reduced TDF the maximum pressurizer liquid volume could be 905 ft^3, which is less than the 1000 ft^3 capacity of the pressurizer and therefore no reanalysis was necessary. This is based on an assumption "that the average temperature would increase 50 percent due to flow reductions", which we interpret to mean that the primary temperature rise during this transient is 1.5 times the temperature rise at rated conditions. We conclude that this is a conservative assumption since the total primary mass reduction due to 24% steam generator tube plugging is 8%.

However, the licensee did not address in Reference 2 the effect of 24% reduction in heat transfer area on the capability for shutdown without primary water relief utilizing one steam generator and one auxiliary feedwater pump. We then requested that the licensee provide additional information regarding the effect of reduction in steam generator heat transfer area. In Reference (13), the licensee indicated that the decrease in heat transfer area is offset by the decreased decay heat since operation is at reduced power level. Therefore, the pressurizer will not be filled, the RCS pressure limit would not be reached, and the tube sheet would remain covered, with only one steam generator and one AFW pump available.

We conclude, based on our review of previous analyses and the additional information provided by the licensee, that the consequences of loss of normal feedwater transient will not result in unacceptable fuel performance and that the primary system pressure will not exceed allowable values. The licensee's analysis is, therefore, acceptable.

3. Locked Rotor

The FSAR analysis for the locked rotor accident assumes that seizure of one reactor coolant pump (RCP) shaft occurs at 102% power. Reactor trip occurs on a low flow signal. Upon reactor trip, it is assumed that the most reactive RCCA is stuck in its fully withdrawn position. The time from pump seizure to initiation of control rod motion was assumed to be 0.9 seconds. The licensee has stated that test data indicates a measured time interval of 0.45 seconds from the time the low flow trip setting is reached until the instant the rods are released. Another 0.1 second is assumed for the interval between pump seizure and reaching the low flow trip set point, for a total of 0.55 seconds. Thus 0.9 seconds is conservative (Reference 6). No credit was taken for the pressurizer relief valves, pressurizer spray and steam dump. The licensee assumed offsite power to be available and continued operation of one RCP. This is further discussed below.

The FSAR analysis showed the peak pressure to be 2778 psia. We consider this value acceptable, since it is below 120% of design pressure (service limit "C" of the ASME code), and thus meets the acceptance criteria of the June 15, 1982 revision of Standard Review Plan (SRP) Section 15.3.3-15.3.4 for peak pressure. The results of this analysis further indicate that about 22% of the fuel rods reach a DNBR less than 1.3 and about 15% of the fuel rods reach a DNBR less than 1.0. This occurs for a very short time period (about 2 seconds). Peak clad surface temperature is 1522°F. The licensee indicates that the peak clad surface temperatures are below the threshold for metal-water reaction, and therefore, the results are not unacceptable.

Reference 5 contains the licensee's analysis of this event at reduced operating pressure. While the resulting peak pressure is lower than in the FSAR analysis, the results of the DNB calculations are more severe, predicting that 63% of the fuel rods reach a DNBR of less than 1.3. The licensee indicates that this analysis was performed on a highly conservative basis, since the coolant pressure increase as a result of the transient was ignored, and rods for which the fluid conditions are beyond the range of the DNB correlation were assigned DNB ratios less than 1.3. In view of the high percentage of potentially damaged fuel as a result of this postulated accident, the staff has performed independent site boundary calculations to determine whether the radiological consequences of the postulated accident meet the guidelines of 10 CFR Part 100. The licensee is adopting standard technical specification (STS) limits for primary coolant iodine. Assuming primary coolant STS limits and 63% fuel cladding damage, the radiological consequences at the site boundary would be less than a small fraction of the 10 CFR Part 100 guidelines. The licensee's analysis did not assume loss of offsite power (LOOP) and thus the radiological consequences could conceivably be higher if LOOP occurred. Therefore, a limiting calculation was also performed assuming that all the fuel cladding is damaged. The resulting site boundary dose is still less than the 10 CFR Part 100 guidelines, and thus meets the acceptance criteria of the June 15, 1982 revision of SRP Section 15.3.3-15.3.4 for site boundary dose.

With regard to the effect of operation at reduced TDF and power, references 2 and 4 indicate that the expected fuel and clad temperatures would remain about the same as at rated conditions, since the effect of reduced flow would be offset by the lower power level. The effect of reduced flow and primary mass would not be detected by the core in the time frame of interest since the peak values are reached in considerably less than one loop transport time constant. We concur with this assumption and find that the locked rotor analysis is acceptable.

4. Steam Line Break (SLB)

The FSAR steam line break analysis was performed using 7 combinations of break sizes and initial plant conditions, including large breaks upstream and downstream of the flow limiting nozzle, one and two-loop operation, offsite power available and unavailable, and a break equivalent to steam release through one steam generator safety valve. The analyses were performed assuming end of core life, hot shutdown with the most reactive rod stuck in its fully withdrawn position, 2.77% shutdown reactivity, and one safety injection pump failing to function. The most severe case involves a break upstream of the flow limiting nozzle, two loops in operation, and loss of offsite power, and results in a peak power after return to criticality of 24%. For the break downstream of the flow limiting nozzle, peak power after return to criticality was of the order of 10%. Utilizing the MacBeth critical heat flux correlation provided acceptable DNBR values for all the transients analyzed.

For operation at reduced pressure and temperature, Reference 5 indicates that, as a result of slightly less stored energy in the coolant system, cooldown is slightly faster and the resulting thermal power is about 1% higher. Minimum DNBR is still above 1.3.

Reference 2 provides reanalyses at reduced TDF for the following cases: large SLB inside containment with and without offsite power, large SLB outside containment with and without offsite power, and a break size equivalent to one open safety valve. The assumptions for these analyses are: end of core life, the most reactive rod stuck in its fully withdrawn position, one safety injection train not functioning. While the text of reference 2 states that the initial shutdown margin is 2.77% for all cases, figures 4 through 8 indicate the initial reactivity to be 0 for the first 4 cases. As noted above, the FSAR analyses were all performed with an initial reactivity of -.0277. We consider that this may be a more conservative assumption for the SLB initial conditions. The licensee was requested to clarify these apparent discrepancies and justify the assumptions utilized or submit new analyses.

In Reference (13), the licensee provided additional information including better figures, and indicated that the assumptions utilized are consistent with Reference (14). The minimum DNBR for the postulated breaks was greater than the 1.3 limit. Reference (2) indicates that the increased level of steam generator tube plugging would, because of reduced heat transfer coefficient and flow, result in slightly lower peak power levels when compared with the FSAR analysis.

The results of the above SLB analysis indicate that the largest power excursion occurred for SLB inside containment with outside power available. DNBR remained above 1.3 for all runs. We conclude that, based on previous analyses and additional information provided by the licensee, the consequences of an SLB at reduced TDF will not result in unacceptable fuel performance. The licensee's SLB analysis is, therefore acceptable.

5. Large Break Loss of Coolant Accident (LOCA) Analysis

The licensee has indicated that the most applicable existing large-break LOCA analysis to be used for operation with reduced TDF was performed with 18% steam generator tube plugging and peaking factors (F_Q) equal to 2.32. References 10 and 11 contain such analyses for operating pressures of 2250 psia and 2000 psia, respectively. Reference 10 indicates that 100% TDF would be obtained even with 18% tube plugging. Reference 8 contains our evaluation of the LOCA analysis submitted in Reference 10. The staff concluded that a large-break LOCA during operation at Point Beach Unit 1 while at a primary pressure of 2250 psia and with up to 18% tube plugging would result in a peak clad temperature (PCT) of 2053°F and would be in conformance with 10 CFR Part 50.46 criteria. Reference 8 provides a LOCA analysis for reduced pressure operation. PCT is calculated to be 2062°F. A correction factor of 60°F is applied to these numbers to account for the effects of upper plenum injection. (Ref. 8) The criteria of 10 CFR Part 50.46 are still met.

The licensee has not performed a detailed calculation of PCT for the large break LOCA at reduced TDF and pressure operation. The licensee has, however, submitted the results of sensitivity calculations for PCT at 91% rated power, 95% TDF, 24% tube plugging and 2000 psi RCS pressure. Assuming an F_Q of 2.52, the resulting PCT, when corrected for upper plenum injection, was 2188°F. This is close to the allowable limits of 2200°F in 10 CFR Part 50.46. In Reference 12, we questioned the use of sensitivity analyses to correct for an increased F_Q and indicated concern about the small margin to PCT limit of 2200°F. We also questioned assumptions regarding linearity and superposition of sensitivity analyses and requested clarification regarding apparent inconsistencies in the analysis. In subsequent conversations with the licensee we indicated that our major concern is the utilization of F_Q of 2.52, which apparently increases PCT by 200°F over a utilization of F_Q of 2.32.

In Reference (13), the licensee indicated that 18% steam generator tube plugging is the maximum expected. Reference (13) also indicates that, while operation at reduced power and TDF would involve higher ratios of peak to average linear power, the peak kw/ft value for 91% power and 95% TDF would be bounded by the full power case. $F_{\Delta H}$ (enthalpy rise hot channel factor) is also slightly lower for the reduced TDF case. The submittal also indicates that the effect of small flow or coolant temperature changes on PCT is small when compared to the effect of the power level. For the large cold leg break LOCA, the core flow reverses direction

within 0.1 seconds of the LOCA transient, so the initial flow rate through the core is of relatively little importance. The licensee concludes that the LOCA analysis @ 100% power and TDF bounds the 91% power, 95% TDF, F_Q 2.52 case and, therefore, the latter meets 10 CFR Part 50.46 criteria.

In Reference (13), the licensee also proposes to administratively limit power to 84%. This would be equivalent to an F_Q LOCA limit of 2.32 for 91% power. Such power limitation would reduce linear kw/ft by about 7.5%. We conclude that operation at a maximum power level of 84%, a minimum TDF of 95%, and 18% SGTP would not result in values exceeding 10 CFR Part 50.46 acceptance criteria in the event of the design base LOCA.

Subsequent to the licensee's submittal of Reference 13, a calorimetric flow test was performed at Unit 1 which indicated that TDF was slightly above 100% with 14% SGTP. However, there is a concern that if additional tube plugging is required, TDF may be reduced to less than 100%.

Conclusion

Based on our above evaluation, if future calorimetrics indicate that TDF is less than 100% but not less than 95%, SGTP is held to a maximum of 18%, and the power level is administratively limited to a maximum of 84%, the licensee need not submit a detailed LOCA analysis. However, if operation at lower TDF than 95%, higher power levels than 84%, and higher SGTP than 18% is contemplated, the licensee must furnish to NRC for approval a LOCA analysis for the new operating conditions.

Principal Contributors:
T. G. Colburn
B. Mann

REFERENCES

1. Point Beach Nuclear Plant Units 1 and 2 Final Safety Analysis Report (updated)

2. WEPCo. letter of September 17, 1982, forwarding Technical Specification Change Request No. 85 for Reduced TDF

3. WEPCo. letter of October 15, 1982 forwarding additional information for Technical Change Request No. 85

4. Point Beach Nuclear Plant Unit 1 "Steam Generator Repair Report" August 1982

5. WCAP 8151, Point Beach Unit 2 Low Pressure Analysis, June 1973

6. WEPCo. letter of November 9, 1982, forwarding responses to our request for additional information regarding the locked rotor analyses.

7. NRC Confirmatory Order for Modification of Point Beach 1 License, November 30, 1979.

8. NRC Modifying Confirmatory Order of November 30, 1979, dated January 3, 1980.

9. NRC letter to WEPCo. dated April 29, 1980, forwarding Amendments No. 44 to Point Beach Unit 1 Facility Operating License.

10. WEPCo. letter of November 19, 1979, forwarding ECCS Reanalysis for 18% Steam Generator Tube Plugging Limit, Point Beach Unit 1.

11. WEPCo. letter of November 27, 1979, forwarding Low Pressure ECCS Evaluation for 18% Steam Generator Tube Plugging, Point Beach Units 1 and 2.

12. NRC letter to WEPCo. dated November 19, 1982, forwarding request for additional information regarding the LOCA analysis.

13. WEPCo. letter of December 10, 1982 forwarding additional information for Technical Specification Change Request No. 85.

14. WCAP 9226 "Reactor Core Response to Excessive Secondary Steam Releases" January 1978.