Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

June 30, 1982

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Dear Mr. Denton:

DOCKET NOS. 50-266 AND 50-301
FURTHER RESPONSE TO NUREG-0737, ITEM II.D.1
RELIEF AND SAFETY VALVE TESTING
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In accordance with the initial recommendation of NUREG-0578, Section 2.1.2, as clarified by NUREG-0737, Item II.D.1, and the Nuclear Regulatory Commission letter of September 29, 1981, each pressurized water reactor licensee is requested to submit by July 1, 1982 plant-specific evaluations, supported by test results, of the ability of existing safety and relief valves to function for expected operating and accident conditions. Our preliminary evaluation and EPRI Test Program results were submitted on April 8, 1982. The enclosure to this letter provides the current status of our continuing evaluations and analyses in response to NUREG-0737 requirements.

We have received a letter from EDS Nuclear, the contractor conducting this evaluation for the Point Beach Nuclear Plant, advising that because of a delay in the loop seal evaluation data required for our NRC submittal will be delayed by at least one and possibly two months. A copy of the EDS Nuclear June 8, 1982 letter (less enclosures) is attached. Accordingly, the final plant-specific analyses will require an evaluation period of one to two months beyond July 1, 1982 to complete the piping and support analyses. A short additional period of time will also be required to determine if modifications to existing supports must be made and to review the final EPRI Test Program data to confirm the applicability of these data to Point Beach. We, therefore, expect that results of these analyses and evaluations can be submitted by November 1, 1982. We would also expect to provide at that time details of piping or support modifications which are found to be necessary, if any, and a schedule for implementation of the modifications.

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Should you have questions concerning this submittal, please contact us.

Very truly yours,

C. W. Fay

Assistant Vice President

C. W. Fay

Enclosures

Copy to NRC Resident Inspector

WISCONSIN ELECTRIC POWER COMPANY

STATUS OF EVALUATION

POINT BEACH NUCLEAR PLANT PRESSURIZER
SAFETY AND RELIEF VALVE SYSTEM

June 1982

WISCONSIN ELECTRIC POWER COMPANY
POINT BEACH NUCLEAR PLANT
SAFETY AND RELIEF VALVE SYSTEM

1.0 INTRODUCTION

This report summarizes the status of the evaluation of the Point Beach Nuclear Plant (PBNP) pressurizer safety and relief valve system being conducted in accordance with NUREG-0578, Section 2.1.2, clarified by NUREG-0737, Item II.D.1, and by the U. S. Nuclear Regulatory Commission's letter of September 29, 1981.

2.0 SYSTEM DESCRIPTION

PBNP is a two-unit Westinghouse plant. The pressurizer safety and relief valve system for each unit includes two Crosby safety valves and two Copies-Vulcan power-operated relief valves. Water loop seals are included for each Crosby safety valve. A detailed description of the system was provided in Wisconsin Electric's preliminary evaluation of safety and relief valve test program results which was submitted on April 8, 1982.

3.0 EVALUATION STATUS

Wisconsin Electric has retained EDS Nuclear Inc. for assistance on the specific evaluations required for PBNP. These evaluations are currently in progress.

The following tasks have been completed to date:

1. The applicability of the Generic PWR Safety and Relief Valve Test Program (Test Program), implemented by the Electric Power Research Institute (EPRI), to the PBNP system has been confirmed. This includes consideration of the type of valves tested and the conditions under which they were tested.

2. The preliminary test data from this Test Program have been reviewed to evaluate the operability and functionability of the as-built valves, piping, and supports under the full range of transient conditions for which the system is required.

3. An evaluation of the feasibility of modifying the water loop seals has been completed.

4. RELAP5/MOD1 and REFORC computer models of the system have been developed for PBNP. This model is currently being utilized to determine the piping thrust loads induced by valve actuation.

5. Structural piping models of the pressurizer safety and relief valve inlet and discharge piping for each unit have been developed. These models will be used to determine the response of the piping and its supports to the valve actuation thrust loads.

4.0 EVALUATION OF SYSTEM ADEQUACY

As part of the work completed to date, applicable preliminary test data from the Test Program have been reviewed. These preliminary evaluations, which confirm the operability and functionability of the system, are summarized in this section. Final test data are scheduled to be released by EPRI on July 1, 1982. The final data will be reviewed to confirm results of preliminary evaluations.

Safety Valve Operability

PBNP has Crosby HB-BP-86, 4 K2 6 safety valves. The EPRI Test Program included Crosby 6M6 and 3K6 safety valves. The 6M6 and 3K6 valves are structurally and functionally similar to the 4 K2 6, and their test results are applicable to the PBNP valves.

The Crosby 6M6 and 3K6 safety valves operated successfully on saturated steam and saturated water, exhibiting rapid lift and stable behavior. In all cases, they functioned to relieve pressure and prevent excessive overpressurization. During testing of valves with loop seals, valve chatter did occur on the test piping configuration; however, this did not substantially affect subsequent valve performance. Also, in some tests, full lift was delayed until the loop seal had cleared the valve. Again, subsequent valve performance was not substantially affected, and the valves functioned to relieve pressure and prevent overpressurization. Thus, it is concluded on the basis of the preliminary Test Program results that the PBNP safety valve operability is confirmed.

As part of the evaluation of safety valve operability, Wisconsin Electric will investigate whether, in the final EPRI Test Program results, the ring settings on the safety valves should be altered. If it is concluded that they should be, this would be performed during the next scheduled valve maintenance following completion of the evaluation.

Power-Operated Relief Valve Operability

PBNP has four Copes-Vulcan relief valves. One of the PBNP valves is essentially identical to a valve included in the EPRI Test Program. The only difference between the three remaining PBNP relief valves and the test valve is in the valve body size (two-inch diameter compared to three-inch diameter). This difference would have no significant effect on the test data.

Extensive tests were performed on the Marshall (steam only) and Wyle (steam and water) test facilities. While the valves occasionally did not fully close, the maximum reported leakage was less than 0.8 gpm in the Marshall tests and 0.08 gpm in the Wyle tests. These test results demonstrate the acceptability of the PBNP relief valves.

Safety Valve Inlet Piping Functionability

High-frequency water hammer pressure spikes were observed during the Test Program in the test piping between the safety valves and the pressurizer. This is due to valve flutter and chatter during clearance of the loop seal water.

These pressure spikes may result in localized exceedance of the code-allowance stresses. However, the code allowances are based on quasi-statically applied pressure throughout the pipe and not on localized pressure pulses. Thus, these pressure spikes are outside the range of applicability of the PBNP design basis. Preliminary results indicate that these very high-frequency loads do not induce any significant strains in the pipe wall.

Discharge Piping Functionability

Test Program results indicate that higher-than-anticipated thrust loads may be experienced immediately downstream of the safety valves. This is due to the discharge of the subcooled water from the loop seal. Due to the relatively small volume of the PBNP water loop seals compared to the Test Program loop seals, these thrust loads are expected to be considerably less than observed in the Test Program; however, they may exceed those used in the original PBNP design. Wisconsin Electric is proceeding with thermal-hydraulic and piping analyses to determine the effect of these loads on the discharge piping and supports. The results of these analyses will be provided with the final evaluation report. Should modifications be required to the discharge piping, a schedule for completing the modifications will also be provided.

It is considered that the functionability of the piping and its supports is acceptable even if these valve thrust loads exceed the loads on which the current design is based. The reasons for this are as follows:

1. The thrust loading is of very short duration, whereas the piping is designed for quasi-statically applied loads.

2. There is considerable margin for loads in excess of design in the elastically-designed piping and supports, due to inherent ductility of the piping and supports.

3. The principal function of the discharge piping, which is not part of the pressure-retaining boundary, is to provide support for the valves. Thus, considerable plastic deformation can be experienced without any loss of functionability.

Conclusions

The review of preliminary data and evaluations to date indicate that the PBNP pressurizer and relief system will operate under expected NSSS operating and accident conditions.

eds nuclear

E D S Nuclear Inc.
2333 Waukegan Road
Bannockburn, Illinois 60015
(312) 940-2000

June 8, 1982
0870-005-014

Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, WI 53201

ATTENTION: Mr. D.L. Dill
Project Engineer

SUBJECT: Evaluation of Pressurizer Safety and Relief Valves: Project Schedule, Loop Seal Temperatures, and As-Built Drawings

REFERENCE: EDS Calculation
0870-005-013 Rev. 0

Gentlemen:

Please find enclosed the revised project schedule. Basically, this revised schedule reflects a delay of one month in outstanding target dates. This is due to the finalization of the loop seal evaluation not being possible until the end of May. The original target for this was the start of May.

As discussed with Mr. D.L. Dill, the data required for the NRC submittal will be correspondingly delayed. We recommend that this submittal be deferred at least one, and preferably two months so that it contains a substantial resolution of the issue.

It must be noted that this schedule does not include piping modifications. Any need for other than minor modifications would, of course, impact the targeted completion dates for Task E and subsequent tasks.

Also enclosed is a plot showing the steady-state loop seal water temperatures we have calculated. The temperatures are plotted from the pressurizer end of the loop seal water ($x = 0$) to the safety valve inlet flange ($x = 3.45$ feet). Curve 'c' is the existing temperature, as calculated. Curve 'a' is the probable temperature, assuming 4-inch thick insulation is added to the loop seal

piping, safety valves, and first segments (approximately one foot) of discharge piping. Curve 'b' is the most conservative (lowest) estimate of the probable temperature of the loop seal water, assuming hydrogen gathers in the first loop.

We recommend that the 4-inch insulation be added, and that the RELAP5/MOD1 analysis proceed using curve 'b'. This assumes an average water temperature of 287°F, 35°F less than the calculated, probable average temperature.

We would also like to take this opportunity to clarify the recent discussions on as-built drawings. As part of our piping analysis, we will develop (and transmit to Wisconsin Electric) piping 'math-models' for each unit. These will reflect the actual as-built geometry of the systems, per the data we have received. These drawings can, with addition of appropriate Wisconsin Electric approvals, be used as a record of the as-built configuration.

The SUPERPIPE piping models description and piping criteria definition are following shortly under separate cover.

If you have any comments or questions, please call Patrick Strange or me.

Very truly yours,

W.J. Razlaff

W.J. Razlaff
Project Manager

WJR/sas

Enclosure (2 copies)