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

October 18, 1991

Docket Nos. 50-528, 50-529
and 50-530

Mr. William F. Conway
Executive Vice President, Nuclear
Arizona Public Service Company
Post Office Box 53999
Phoenix, Arizona 85072-3999

Dear Mr. Conway:

SUBJECT: NRC BULLETIN 88-08, "THERMAL STRESSES IN PIPING CONNECTED TO REACTOR COOLANT SYSTEMS" (TAC NOS. M69664, M69665, AND M69666)

By letter dated October 3, 1988, Arizona Public Service Company responded to NRC Bulletin 88-08. Your response stated that a review was performed of piping connected to the reactor coolant system (RCS). However, your response to Action 3 of the bulletin does not provide sufficient assurance that unisolable portions of all piping connected to the RCS will not be subjected to combined cyclic and static thermal stresses and other stresses that could cause fatigue failure during the remaining life of the unit. Because the fundamental precept of the actions of the Bulletin is to prevent the initiation of cracks in piping, inservice inspection is not an acceptable technique identified in the Bulletin for preventing such cracks. A program must be implemented to provide such assurance by means of either redesign or modification, or temperature and pressure monitoring. If you plan to provide such assurance for certain lines by inspection alone, such as by an inservice inspection program, further justification must be provided. Criteria contained in the enclosure (Evaluation Criteria) should help you in preparing an acceptable response. Your response is requested within 90 days.

Although no response was required relative to Supplement 3 of the bulletin, Some licensees have addressed Supplement 3 in their response to the bulletin. Those who have not will not be required to provide a specific response to

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Supplement 3. However, you are reminded that having been informed of the phenomenon identified in that supplement, you are responsible for adequate review of both its applicability to your plant and any considered actions. The NRC staff may audit or inspect the implementation of Bulletin 88-08 and its supplements at a future date.

This request for information affects fewer than 10 respondents; therefore, OMB clearance is not required under Public Law 96-511.

Sincerely,

/S/

Catherine M. Thompson, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosure:
Evaluation Criteria

cc w/enclosure:
See next page

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Enclosure

EVALUATION CRITERIA FOR RESPONSES TO NRC BULLETIN 88-08, ACTION 3 AND SUPPLEMENT 3

1.0 OBJECTIVE

To provide continuing assurance for the life of the plant that unisolable sections of piping connected to the reactor coolant system (RCS) will not be subjected to thermal stratification and thermal cycling that could cause fatigue failure of the piping.

2.0 PURPOSE

To provide guidelines for evaluation of licensee responses, including acceptable procedures and criteria to prevent crack initiation in susceptible unisolable piping.

3.0 IDENTIFICATION OF POTENTIALLY SUSCEPTIBLE PIPING

(1) Sections of injection piping systems, regardless of pipe size, which are normally stagnant and have the following characteristics:

A. The pressure is higher than the RCS pressure during reactor power operation.

B. The piping sections contain long horizontal runs.

C. The piping systems are isolated by one or more check valves and a closed isolation valve in series.

D. For sections connected to the RCS:

a. Water injection is top or side entry.

b. The first upstream check valve is located less than 25 pipe diameters from the RCS nozzle.

Examples of such sections in PWRs are the safety injection lines and charging lines between the reactor coolant loop and the first upstream check valve, and the auxiliary pressurizer spray line between the charging line and the main pressurizer spray line.

(2) Sections of other piping systems connected to the RCS, regardless of pipe size, which are normally stagnant and have the following characteristics:

A. The downstream pressure is lower than RCS pressure during reactor power operation.

B. The piping systems are isolated by a closed isolation valve, or a check valve in series with a closed isolation valve.

C. There is a potential for external leakage from the isolation valve.

Examples of piping containing such unisolable sections in PWRs are the residual heat removal (RHR) lines. Examples of such piping for BWRs are the RHR lines and the core spray injection lines.

4.0 ACCEPTABLE ACTIONS

The following actions are considered as acceptable responses to Bulletin 88-08, Action 3 and Supplement 3, as applicable, provided that the requirements of Bulletin 88-08, Action 2 have been satisfied.

(1) Revision of system operating conditions to reduce the pressure of the water upstream of the isolation valve below the RCS pressure during power operation.

(2) Relocation of the check valves closest to the RCS to be at a distance greater than 25 pipe diameters from the nozzle.

(3) Installation of temperature monitoring instrumentation for detection of piping thermal cycling due to valve leakage.

A. Type and location of sensors.

a. Temperature sensors should preferably be resistance temperature detectors (RTDs).

b. RTDs should be located between the first elbow (elbow closest to the RCS), and the first check valve (check valve closest to the RCS).

c. For the auxiliary pressurizer spray line, RTDs should be installed near the "tee" connection to the main pressurizer spray line or on the cold portion (ambient temperature) of the line.

d. RTDs should be located within six inches of the welds.

e. At each pipe cross section, one RTD should be positioned on the top of the pipe and another RTD on the bottom of the pipe.

B. Determination of baseline temperature histories.

After RTD installation, temperature should be recorded during normal plant operation at every location over a period of 24

hours. The resulting temperature versus time records represent the baseline temperature histories at these locations. Baseline temperature histories should meet the following criteria:

a. The maximum top-to-bottom temperature difference should not exceed 50°F.

b. Top and bottom temperature time histories should be in-phase.

c. Peak-to-peak temperature fluctuations should not exceed 60°F.

C. Monitoring time intervals.

a. Monitoring should be performed at the following times:

1. At the beginning of power operation, after startup from a refueling shutdown

2. At least at six-month intervals thereafter, between refueling outages

b. During each monitoring period, temperature readings should be recorded continuously for a 24-hour period.

D. Exceedance Criteria.

Actions should be taken to modify piping sections or to correct valve leakage if the following conditions occur:

a. The maximum temperature difference between the top and the bottom of the pipe exceeds 50°F.

b. Top and bottom temperature histories are in-phase but the peak-to-peak fluctuations of the top or bottom temperatures exceed 60°F.

c. Top and bottom temperature histories are out-of-phase and the bottom peak-to-peak temperature fluctuations exceed 50°F.

d. Temperature histories do not correspond to the initially recorded baseline histories.

(4) Installation of pressure monitoring instrumentation for leakage detection in injection lines.

(Pressure monitoring is not the preferred method since pressure measurements cannot provide a measurement of thermal cycling in the unisolable pipe sections.)

A. Type and location of sensors.

a. Pressure sensors should preferably be pressure transducers.

b. Pressure transducers should be installed upstream and downstream of the first check valve.

c. For systems having a pressure higher than the RCS pressure, pressure transducers may be installed upstream and downstream of the first closed isolation valve. (The downstream section is the pipe segment between the isolation valve and the check valve.)

B. Monitoring time intervals.

a. Monitoring should be performed at the following times:

1. At the beginning of power operation, after startup from a refueling shutdown

2. At least at six-month intervals thereafter, between refueling outages

b. Pressure readings should be recorded continuously for a 24-hour period.

C. Exceedance criteria.

Actions should be taken to modify piping sections or to correct valve leakage if the following conditions occur:

a. For pressure measurements across a check valve, the downstream pressure (RCS pressure) is equal to or less than the upstream pressure at any time during power operation.

b. For pressure measurements across a closed isolation valve, the downstream pressure is equal to or greater than the upstream pressure at any time during power operation.

Docket File

October 10, 1991

Docket Nos. 50-528, 50-529,
and 50-530

Mr. William F. Conway
Executive Vice President
Arizona Public Service Company
Post Office Box 53999
Phoenix, Arizona 85072-3999

Dear Mr. Conway:

SUBJECT: RESPONSE TO NRC BULLETIN 89-01, SUPPLEMENT 2, "FAILURE OF WESTINGHOUSE STEAM GENERATOR TUBE MECHANICAL PLUGS" PALO VERDE NUCLEAR GENERATING STATION (TAC NOS. 81644, 81645, AND 81646)

On June 28, 1991, NRC Bulletin 89-01, Supplement 2 was issued to all holders of operating licenses or construction permits for pressurized water reactors. The bulletin requested that actions similar to those requested in NRC Bulletin 89-01, "Failure of Westinghouse Steam Generator Tube Mechanical Plugs," be extended to include all Westinghouse mechanical plugs fabricated from thermally treated Inconel 600. These actions were necessary to ensure that these plugs would continue to provide adequate assurance for reactor coolant system pressure boundary integrity under normal operating, transient, and postulated accident conditions.

By letter dated July 31, 1991, you responded to Bulletin 89-01, Supplement 2, by providing the information addressed in the Actions Requested portion of that document. We have reviewed your response for Palo Verde which indicates that all Westinghouse Inconel steam generator tube plugs have been replaced with Inconel 690 mechanical tube plugs during the Unit 2, Cycle 3 refueling outage, and that Units 1 and 3 do not have any Inconel 600 steam generator tube plugs installed. This acceptably resolves the issues raised in the subject bulletin. This issue is therefore closed for Palo Verde.

Sincerely,

ORIGINAL SIGNED BY

Charles M. Trammell, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
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October 10, 1991

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Mr. William F. Conway
Executive Vice President
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Post Office Box 53999
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Sincerely,

Charles M. Trammell

Charles M. Trammell, Project Manager
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UNITED STATES
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WASHINGTON, D. C. 20555

October 16, 1991

Docket Nos. 50-528, 50-529,
and 50-530

Mr. William F. Conway
Executive Vice President, Nuclear
Arizona Public Service Company
Post Office Box 53999
Phoenix, Arizona 85072-3999

Dear Mr. Conway:

SUBJECT: PRESSURIZER SURGE LINE THERMAL STRATIFICATION, BULLETIN 88-11, PALO VERDE NUCLEAR GENERATING STATION, UNIT NOS. 1, 2, AND 3 (TAC NOS. M72152, M72153, AND M72154)

The NRC staff and its consultant, Brookhaven National Laboratory, conducted an audit on May 7 and 8, 1991, of ABB/Combustion Engineering (CE) for the purpose of reviewing the status of the CE Owners Group (CEOG) program on reanalysis of the pressurizer surge line for thermal stratification effects as related to Bulletin 88-11. During the audit, representatives of CE and the CEOG gave a presentation on the progress of the CEOG program, the analytical methods used, and the results obtained, and provided responses to staff comments and questions. A copy of the audit trip report is enclosed.

As a result of this audit, the staff concluded that CE has made progress in addressing some earlier staff concerns. However, several issues remained open and needed to be resolved. On August 14, 1991, a conference call was held between Combustion Engineering, Brookhaven Laboratory, and the NRC to discuss proposed solutions to these open issues. In that discussion, CE also proposed a schedule for the completion of all action items related to Bulletin 88-11. An NRC review meeting was held on September 18 and 19, 1991, in Windsor, Connecticut. Pursuant to the resolution of the outstanding open items and the schedule adopted at that meeting, the CEOG will submit its generic bounding analysis and final report to the NRC staff by December 31, 1991. Following staff review of the CEOG final report, the NRC staff plans to issue a generic

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safety evaluation, after which you will need to submit a plant-specific applicability report for Palo Verde. You are requested to provide this report within 60 days of the issuance of our generic safety evaluation.

Sincerely,

Catherine M. Thompson

Catherine M. Thompson, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosure:
Audit Trip Report

cc w/enclosure:
See next page

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AUDIT TRIP REPORT

PURPOSE: Audit of Combustion Engineering Owner's Group (CEOG) Pressurizer Surge Line Thermal Stratification Evaluation Program to Address NRC Bulletin 88-11 Issues

LOCATION: ABB/Combustion Engineering Nuclear Power, Windsor, CT

DATES: May 7 and 8, 1991

NRC PERSONNEL: S. Hou (NRC), G. DeGrassi (BNL)

CEOG PERSONNEL: D. Sibiga, D.J. Ayres, and others (see Attachment 1)

The staff of the Mechanical Engineering Branch and its consultant from Brookhaven National Laboratory conducted an audit at ABB/Combustion Engineering. The purpose of this audit was to review the current status of the CEOG program on pressurizer surge line thermal stratification.

CE had submitted an evaluation report on this subject (Ref. 1) in June 1989. The report described the program which included data collection, defining new thermal loads, and performing an ASME Code stress and fatigue analysis and evaluation. Based on the program results, CE concluded that a 40 year fatigue life of the surge line was demonstrated. The staff reviewed the report and disagreed with this conclusion. The staff evaluation identified a number of concerns and concluded that the information provided by the CEOG report was inadequate to justify meeting all appropriate limits for the 40 year plant life. Major concerns were that stresses did not meet all ASME Code limits, striping was inadequately addressed and the bounding analysis did not represent the worse case in CE plants. The staff, however, accepted the report as sufficient basis to justify interim operation until a final analysis is completed.

CEOG has since performed additional work and submitted additional information to the staff (Ref. 2, 3, 4). During a plant specific audit at Palo Verde last February, additional questions and open items on the generic program were identified (see Attachment 2). CE representatives committed to provide responses at this audit. In addition, the staff wanted to audit sample calculations. A summary of the information presented during this audit and the staff evaluation is provided below.

I. CEOG PROGRAM STATUS

CE has redone the elastic-plastic shakedown analysis of the limiting surge line elbow. The finite element model was revised and incorporated 3-D solid elements instead of 2-D shell elements. A total of seven cycles were applied to demonstrate shakedown. Final results were not available for the audit due to computer problems. However, CE expects to complete the work and issue a draft report to the member utilities by mid-May. CE will be holding workshops for the CEOG in June to instruct utilities on how to apply the generic report in responding to the Bulletin. The final report will be submitted to NRC by the end of July. Each owner will then submit a plant specific response to Bulletin 88-11 in accordance with their own commitments.

II. CEOG RESPONSES TO NRC QUESTIONS

CE provided responses to the NRC questions listed in Attachment 2. The following is a summary of the responses and discussions.

1. Expansion Stress Intensity Evaluation

The original CE bounding analysis concluded that the surge line exceeded the 3 S_m elastic stress limit of ASME Section III Code NB-3600 equation 12. In the reevaluation, CE performed a finite element analysis of the elbow and concluded that the alternate thermal expansion stress limit of 3 S_m of NB-3222.3 is met. CE provided an explanation of how the thermal expansion stress (P_e) was determined from the finite element stress results. As indicated in Attachment 3, the finite element analysis provided axial stresses at the inside and outside surfaces of the pipe. The surface stresses were averaged through the wall thickness and the maximum and minimum values of these average stresses were used to calculate membrane and bending stresses in the pipe cross section. CE claimed that the combination of the averaged membrane and bending stresses equals the thermal expansion stress, P_e.

The audit team questioned the adequacy of using the average mid-thickness stress since it ignores the through-wall local bending stress which is significant. In addition, the stresses reported were axial stresses only and not stress intensities. CE agreed that stress intensities should be considered and claimed that the 3 S_m limit would still be met. However, since additional discussions could not resolve the question of using average instead of maximum stress, the audit team recommended that CE initiate a Code Inquiry to determine whether the ASME Code Committee concurs with the CE interpretation of P_e stress. CE explained that this had not been pursued since it would take at least a year to obtain such concurrence. Since the 3 S_m limit on P_e stress is believed to be conservative, the audit team agreed to leave this item as

a confirmatory issue pending resolution of the Code Inquiry. If the issue cannot be resolved by June 30, 1992, CEOG must take appropriate corrective actions. CE agreed to initiate a Code Inquiry on this question and the staff will review the contents of the Code Inquiry before it is submitted to the Code Committee.

2. Code Case N47 Strain Limits

In previous discussions, the staff had questioned the use of high temperature Code Case N47 strain limits in the surge line evaluation. CE explained that these limits (1% on membrane strain, 2% on bending strain and 5% on peak strain) are only being used to support the validity of the analysis. They reviewed the Code Case and judged that no other requirements of N47 are relevant to the surge line. The audit team found their response acceptable, but requests clarification of the definition of the calculated strain to be compared to these limits (i.e. is it the accumulated strain at end of life or the largest value of strain at any point in time?).

3. Thermal Striping Loads Development

CE gave a presentation on the striping analysis methodology. A copy of the presentation slides is included in Attachment 4. CE reviewed the published test data in the open literature and developed what they believe is a conservative model. The available data on striping is primarily related to striping in feedwater lines. CE developed a model with two homogenous temperature layers separated by a thin interface. The critical parameters are frequency, amplitude and heat transfer coefficient. CE's objective was to select realistic but conservative values of these parameters. Based on the data published by Hu, Fujimoto, Wolf, and Woodward, CE considered a minimum frequency of 0.25 Hz and a maximum amplitude of 40% of the total pipe top to bottom ΔT. For heat transfer coefficient, CE used a high value of 3500 Btu/hr-ft^2°F.

The audit team found the model and the selection of key parameters reasonable but raised questions regarding the application of the model. See item 4 below.

4. Thermal Striping Fatigue Analysis

CE used a 1-D finite element model to evaluate stress and fatigue due to striping. Four load cases were analyzed with frequencies of 0.25Hz and 1.0Hz and amplitudes of approximately 10% and 40% of the pipe ΔT of 320°F. For all cases except low frequency (.25Hz) and high amplitude (40%), stresses were below the endurance limit with infinite allowable cycles. The case with low frequency and high amplitude had stresses which exceeded the endurance limit but CE judged this case to be too conservative and

did not use these results. The thermal striping stresses were not combined with stratification stresses because they occur at different locations within the pipe.

The audit team questioned the basis for disregarding the low frequency/high amplitude load case and for not combining striping stress with stratification bending stress. CE agreed to provide either stronger justification for disregarding the most severe load case or incorporate it into the evaluation. CE also agreed to address the combination of stratification and striping stresses in the evaluation.

5. Design Basis Transients Development

CE discussed the development of the design basis transients in Table 3.5.1-1 of CEN 387-P which were used in the fatigue evaluation. The transients are based on the original design basis with stratification effects included. Heatup and cooldown transients are of particular interest since they contribute the most significant fatigue usage. CE assumes 500 heatup and cooldown cycles over the life of the plant with 2 cycles of 320°F ΔT stratification per heatup and cooldown. The audit team questioned the adequacy of assuming only 2 cycles of maximum stratification per heatup or cooldown. CE was asked to confirm that this assumption was reasonable and conservative based on plant monitoring data taken during heatups and cooldown. CE agreed to review the data and confirm this assumption.

6. Surge Line Slope Effects

In the review of the Palo Verde analysis, it was noted that the horizontal portion of the surge line had a slope which was not considered in the analysis. CE explained that their models assume zero slope because this is the most conservative assumption. The interface is assumed to be at the elevation of the centerline for the entire horizontal length of the line. The audit team agreed that this assumption is conservative and found the CE response acceptable.

7. Use of SUPERPIPE Program

CE had been previously asked to explain how the SUPERPIPE program had been used in the stratification analysis since the program does not have the capability to apply a transverse temperature gradient across the pipe. CE explained that SUPERPIPE was primarily used to identify the most highly stressed surge line of all CE plants for which the inelastic bounding analysis would be performed. The analyses were recently rerun with higher ΔT values of 340°F for Palo Verde, 340°F for Maine Yankee and 350°F for Arkansas. The results show that Palo Verde is the bounding CE plant.

In performing the SUPERPIPE analysis, the free end thermal expansion displacements were first calculated by hand. The pressurizer end of the pipe was fixed and the rest of the pipe was allowed to displace. The section rotations needed to calculate the displacements were determined from a two dimensional MARC heat transfer and stress analysis. Once the displacement profile was determined, the displacements and rotations at the hot leg nozzle and displacements at supports were applied in the opposite direction to the SUPERPIPE model. This provided the appropriate moments and bending stress due to stratification in the pipe. The audit team found this methodology acceptable.

8. Anchor Movements

CE had been asked to demonstrate how anchor movements were considered in the SUPERPIPE analysis. CE explained that anchor movements from the pressurizer and hot leg were applied in a separate SUPERPIPE analysis. In addition to the anchor movements, the load case also applied the appropriate average temperature along the horizontal portion of the pipe, the pressurizer temperature at the vertical pipe near the pressurizer, and the hot leg temperature at the vertical pipe near the hot leg. The moments and stresses from this load case were then superimposed with the moments and stresses for the stratification load case to give the total moments and stresses. The audit team found this procedure acceptable.

9. Nozzle Evaluation

CE provided a calculation on the ASME Code evaluation for the surge line pressurizer and hot leg nozzles and discussed the methodology and results. CE pointed out that the calculations will be redone with the higher stratification ΔT's but the methodology will be the same. The nozzle calculation used the loads from the SUPERPIPE analysis and the loads from the original nozzle evaluation. It assumed that the difference between these loads is the stratification induced load. Two cases were considered: (1) stratification stresses alone for 300 cycles, and (2) stratification plus OBE stresses for 200 cycles. By combining the stresses of these two cases with the stresses from the original nozzle stress report, a new stress range and fatigue usage factor with stratification was calculated. Nozzle stresses were calculated by formulas using bending moments and forces from the SUPERPIPE analysis. Local stratification effects were not applicable since all CE nozzles are vertical and not affected by local stratification.

The audit team questioned whether the calculation accounted for all stratification cycles. There was only one stratification cycle considered per heatup/cooldown. The revised design basis transients in Table 3.5.1-1 of CEN 387-P shows two 320°F ΔT transients for each heatup and cooldown as well as a number of

lower ΔT transients which should be considered in the fatigue evaluation. CE was asked to investigate and determine whether the appropriate number of cycles were considered in the fatigue evaluation.

10. RTD and Sample Line Connection

During the recent NRC audit of Palo Verde, a review of the original surgeline stress report indicated high fatigue usage at the RTD and sample line connections. These areas were not addressed in the CEN 387-P report. As a result, CE was asked to provide information on the fatigue usage factors in these areas. CE stated that as part of the final evaluation, they will provide each plant with appropriate stress and displacement information in these areas. CE will conduct workshops in June to explain how the results should be used in plant specific evaluations. The audit team found this response acceptable.

III. SURGE LINE FATIGUE AND SHAKEDOWN ANALYSIS

CE provided additional information on the fatigue and shakedown analyses which are being redone. Based on additional SUPERPIPE analyses, CE confirmed that Palo Verde is the bounding plant. The elbow below the pressurizer is the critical component. CE developed a MARC elastic/plastic model of the surge line similar to the one shown in Figure 3.6.2-1 of CEN 387-P. Beam elements were used to model straight pipes. Elastic/plastic 20 node solid elements with two elements through the thickness were used to model the elbows. This is a refinement over the original model which used shell elements at the elbows. The shakedown analysis used the same load history that had been previously developed for a heatup/cooldown cycle with the exception that the maximum ΔT was increased to 340°F instead of the original 320°F.

CE did not yet have the final stress results but showed some preliminary data on elbow forces, moments, and strains. The preliminary results showed evidence that after seven heatup/cooldown cycles, the cyclic response is repeatable indicating that shakedown will occur.

CE also discussed the fatigue analysis methodology. The analysis will be redone for the higher ΔT but the methodology will be the same. CE considered a total of 35 load states plus OBE and full flow transients which resulted in 595 load sets for the fatigue analysis. Alternating stress intensities were determined from the strains of the inelastic shakedown analysis. Strain levels for events which were not inelastically analyzed were predicted by interpolating the strain from the analyzed events. In order to account for uncertainties, a 10% margin was applied to all stresses. The cumulative usage factor for the bounding plant was 0.21. The highest contribution to fatigue was from a load set which ranges between a non-stratified load state and a 320°F ΔT

stratified load state. When the analysis is redone for the 340°F ΔT, the usage factor will be expected to increase.

The audit team reviewed selected fatigue usage results and questioned why OBE seismic was not combined with the 320°F ΔT stratified flow as a load state. CE stated that, by definition, an OBE does not occur during a heatup or cooldown. They performed a quick review of various FSARs and noted that Section 3.9 defines an OBE as an earthquake load which occurs while the plant is at 100% power. The audit team asked CE to further review licensing documents to better justify the basis for this assumption.

IV. PALO VERDE DIFFERENTIAL TEMPERATURE CONTROL

During the recent NRC audit of Palo Verde, a review of operating procedures indicated that there are no limits on the differential temperature between the pressurizer and the hot leg to ensure that the 320°F ΔT assumed in the analysis is not exceeded. The licensee stated that several options were being considered and committed to inform the staff of the final decision by the CE audit.

During this audit, the Palo Verde representative notified the audit team that two actions are being taken to address the issue. The licensee is having CE redo the analysis to a higher ΔT (340°F) and an administrative limit is being established to ensure that this differential temperature will not be exceeded in the future. The audit team found these actions acceptable to close out this open item from the Palo Verde audit.

V. CONCLUSIONS

The audit team found that CE had made progress in addressing the earlier staff concerns on the bounding analysis. CE confirmed that Palo Verde is the bounding plant. The shakedown analysis is being redone for a larger number of cycles. Higher stratification ΔTs are being considered in the reanalysis. The revised analysis is nearly complete and preliminary results indicate a favorable outcome. However, the audit team identified a number of additional concerns which must be resolved expeditiously and described in the final report. They include the following:

1. The staff disagreed with the CE interpretation of expansion stress (P_e) in an elbow. The CE approach determined P_e on the basis of average stresses through the wall. Surface stresses resulting from local circumferential bending of the elbow wall were neglected. The staff recommends that CE initiate an ASME Code Inquiry to obtain a Code Committee interpretation of the definition of this stress. This will be treated as a confirmatory item pending resolution of the Code Inquiry and should be closed out no later than June 30, 1992.

2. CE must clarify its definition of calculated strain for comparison with Code Case N47 strain limits. Is it the accumulated strain at end of life or maximum strain at any point in time?

3. CE must address the issue of combining thermal striping stresses with thermal stratification stresses in the fatigue evaluation.

4. CE must either provide stronger justification for disregarding the results of the low frequency (0.25 Hz)/high amplitude (40% ΔT) load case in the striping fatigue evaluation or include these results in the fatigue evaluation.

5. CE must confirm the adequacy of their revised design basis transients for fatigue evaluation (Table 3.5.1-1 of CEN 387-P) based on data collected in the monitoring programs. An item of particular concern is the assumption of only two maximum ΔT stratification cycles during each heatup and cooldown.

6. CE must confirm and justify the number of stratification cycles used in the nozzle fatigue evaluation. Only one maximum ΔT stratification cycle per heatup/cooldown was used.

7. CE must provide additional justification for excluding a combined OBE and maximum ΔT stratification as a load state in the fatigue evaluation.

VI. REFERENCES

1. CEN 387-P, "Pressurizer Surge Line Flow Stratification Evaluation," prepared for CE Owners Group, Combustion Engineering, Inc. July 1989.

2. ABB/CE letter MPS-90-903, D. Sibiga to S. Hou, "CEOG Response to NRC Request for Further Information on the Pressurizer Surge Line Fatigue Analysis," September 21, 1990.

3. ABB/CE letter MPS-90-970, D. Sibiga to S. Hou, "CEOG Response to NRC Questions on Application of Plastic Analysis," October 11, 1990.

4. ABB/CE letter MPS-91-362, D. Sibiga to S. Hou, "CEOG Response to BNL Comments/Questions on Application of Plastic Analysis to the CEOG Surge Line Evaluation," March 28, 1991.

safety evaluation, after which you will need to submit a plant-specific applicability report for Palo Verde. You are requested to provide this report within 60 days of the issuance of our generic safety evaluation.

Sincerely,

ORIGINAL SIGNED BY

Catherine M. Thompson, Project Manager
Project Directorate V
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosure:
Audit Trip Report

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