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Perry Nuclear Power Plant
Docket No. 50-440
Feedwater Nozzle Weld
Indications Identified During
Inservice Examination - Meeting Summary

Gentlemen:

By letter dated December 14, 1990 the NRC staff noted that additional information on the two feedwater nozzle indications identified at PNPP during the second refueling outage was necessary in order for the staff to complete evaluations and determine acceptability of operation of PNPP for the duration of Cycle 3. The original CEI evaluation of the indications had been submitted by letter dated November 26, 1990 (PY-CEI/NRR-1264L). By letter dated February 1, 1991 (PY-CEI/NRR-1304L), a meeting with the NRC staff was proposed to present the requested information and directly answer staff questions on this technical subject.

Therefore, on February 21, 1991, CEI and its consultants made a presentation in the NRC offices in Washington, D.C. to provide more detail and answer questions regarding CEI's evaluation of the indications.

The information presented at the meeting included an overview of the flaw evaluation methodology utilized in the original analysis, a crack growth rate discussion, PNPP water chemistry, flaw evaluation inputs, the results of a conservative analysis assuming a non-variable maximum crack growth rate, and the results of a finite element analysis assuming a variable crack growth rate.

The presentation began with a brief overview of the flaw evaluation methodology used to provide the Summary Technical Report submitted with letter PY-CEI/NRR-1264L. The basis for inputs to the ASME Boiler and Pressure Vessel Code, Section XI, simplified methodology was explained. Among these inputs were envelope nozzle design loads, weld residual stress distribution and crack growth rate (CGR). CEI explained in detail the parameters of the applicable test in EPRI Report Numbers NP-5882 M & S, dated July 1988, from which CEI extracted the CGR utilized in the original analysis. CEI acknowledged that the NRC was not familiar with that particular data, and therefore presented bounding CGR data from EPRI Report Number RP 1930-1, Amendment 22, "GE Nuclear Energy Alloy 182 SCC Test Results Final Report", dated October 1990. This more recent database contains the data points referred to in the 12/14/90 NRC letter, which implied that NRC was aware of data that could support higher crack growth rates than utilized in the original CEI analysis. However, it was pointed out that this report clearly demonstrates that the crack growth rate varies with stress intensity, and that the data base also contains lower crack growth rate values at lower stress intensities.

Data which demonstrates Perry's strict control of reactor water chemistry was presented next. This information demonstrated that the chemistry parameters associated with the EPRI RP 1930-1 data base represented a more aggressive environment regarding IGSCC than exists at Perry. Therefore, it was concluded that use of the CGR data from the report was conservative with respect to water chemistry.

The next segment of the presentation identified three unnecessarily restrictive inputs used by CEI for the flaw evaluation that was originally submitted to the NRC. One of these sources of significant margin was the number of operating hours assumed for Cycle 3; the analysis input assumed 12,000 operating hours versus the actual operating cycle length of just over 10,000 hours. Also, envelope nozzle design loads were used as input rather than piping analysis load reactions at the nozzle, which provides a margin in the calculation of the stress intensity factor. Finally, hoop stress was utilized as input instead of axial stress; membrane hoop stress is larger than membrane axial stress for these applications, but hoop stresses open longitudinal cracks whereas axial stresses open circumferentially oriented cracks, such as the feedwater nozzle indications at PNPP. Both of these two latter aspects provide substantial margin in the calculation of the ASME Code crack acceptance envelope.

Reevaluation with the more appropriate inputs as discussed above resulted in two major conclusions. First, the calculated value of K_I (the stress intensity factor) was utilized to obtain the corresponding crack growth rate value from the EPRI RP 1930-1 data base, which resulted in a maximum value of 2.9×10^{-5} in/hr (rounded up to 3×10^{-5} in/hr for analysis purposes), which is comparable to the value of 2.61×10^{-5} in/hr used in the original evaluation. Use of this 3×10^{-5} in/hr value over the entire operating cycle, even when utilized in combination with the ASME Code crack acceptance envelope developed using the original analysis input, still continued to yield acceptable results justifying operation for the full Cycle 3. Secondly, and more importantly, changes resulted in the ASME Code crack acceptance envelope which increased the overall margin between final crack depth (at the end of Cycle 3) and the ASME XI Code acceptance limit. In fact, the increase in margin is so dramatic that a CGR in excess of the maximum exhibited in the EPRI 1930-1 report (irrespective of applicable stress intensity) can be accommodated <u>without</u> exceeding Code acceptance criteria. Details of this re-evaluation are explained in a revised Summary Technical Report which is attached to this letter.

Finally, it was emphasized that the use of the more appropriate inputs as discussed above in no way violated or encroached upon the inherent ASME Code conservatisms.

As further evidence in support of CEI's flaw evaluation conclusions, representatives from SMC O'Donnell, Inc. presented the results of a sophisticated finite element analysis which fully accounted for the variable relationship between CGR and stress intensity. In addition, this analysis considered various weld residual stress distributions, applying the most conservative distribution (that in NUREG 0313/Revision 2) for the bottom-line conclusions. The results of the SMC O'Donnell analysis demonstrated a final crack depth (for Cycle 3) which was less than that predicted by either of CEI's evaluations.

CEI concluded that these evaluations of the feedwater nozzle flaws show that Perry can operate throughout Cycle 3 with more than ample safety margin, and it is not necessary to resize these indications at a mid-cycle outage.

In addition to being technically unnecessary, a planned mid-cycle inspection would result in significant radiological exposure to personnel supporting and performing the inspection. A recent analysis, performed since our November 26, 1990 letter, of the work associated with inspections of these nozzles resulted in estimates of radiological dose to workers in excess of 30 man-rem for Manual UT sizing of the flaws, and even larger doses if, in addition, Automated scans are performed, due to the setup man-hours involved. This dose-intensive work would result in an increase of 20 to 25 percent above PNPP's 1991 dose goal of 156 man-rem. Therefore, ALARA principles discourage planned inspections of these feedwater nozzles. Also, a planned mid-cycle inspection of these nozzles would result in several weeks of lost generation since PNPP is not scheduled for an outage before the third refueling outage.

However, CEI will consider performing a mid-cycle inspection during a forced outage. Several factors will contribute to the decision including the possibility for performing associated work such as Mechanical Stress Improvement (MSIP) on the nozzles, the anticipated length of the outage, and the point in time during the cycle that the outage occurs such that the inspection would provide meaningful data (the value of such an inspection in providing a confirmatory check to the crack growth rate number which will be obtained at the next refueling outage will be minimal if performed before a significant number of operating hours have been accumulated, or if performed only a short time before the planned inspection in the third refueling outage). In addition, the availability of support personnel such as Health Physics technicians and of qualified inspection personnel would be considered. The associated doses with any proposed work scope, including MSIP work, and the plants prior performance in relation to the dose goal, would also be considered. The Outage Planning Section has been tasked with development of forced outage scenarios that consider the above factors so that the appropriate decision can be made if a forced outage occurs.

Details of CEI's reevaluation are given in a revised Technical Report (attached). If there are any further questions, please feel free to call.

Sincerely,

Robert A. Stratman - for

Michael D. Lyster

MDL:BSF:njc

Attachment

cc: NRC Project Manager
NRC Resident Inspector Office
NRC Region III