CP&L

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10 CFR 50 Appendix H
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United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
REACTOR VESSEL MATERIAL SURVEILLANCE REPORT

Dear Sir or Madam:

This letter submits the Carolina Power & Light Company (CP&L) summary technical report for the evaluation of the third Harris Nuclear Plant (HNP) reactor vessel surveillance capsule (capsule 'X'). 10 CFR 50 Appendix H requires this report to be submitted within one year of surveillance capsule removal. Capsule 'X' was removed from the HNP reactor vessel on November 11, 1998.

The enclosed Framatome Technologies, Inc. (FTI) report, BAW-2355, "Analysis of Capsule X – CP&L Shearon Harris Nuclear Power Plant – Reactor Vessel Material Surveillance Program," October 1999, provides the required data. Included in the report is a revised reactor vessel surveillance capsule withdrawal schedule.

With the recent analysis, the scatter in the shift of reference temperature - nil ductility transition (ΔRT_{NDT}) for two of the six plate surveillance specimens exceed the criteria for "credibility" as defined by Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The weld surveillance data meets the credibility requirements of RG 1.99. When surveillance data is "credible" (i.e., without excessive scatter), RG 1.99 permits calculation of chemistry factors from the surveillance data, and halving of required margins applicable to reactor vessel pressure-temperature limits. Otherwise, chemistry factors are based on generic data from RG 1.99, and full margins are to be used. Consistent with the method previously approved by the NRC for another licensee (by NRC letter dated June 23, 1999 to Virginia Electric and Power Company, TAC Nos. MA0555 and MA0556), CP&L proposes to use the plate surveillance data to calculate chemistry factors, but will assume the full margin terms in developing reactor vessel pressure-temperature limits.

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Prior capsule analyses (including the initial unirradiated sample analyses) used hand-drawn curves to fit the Charpy V-Notch Impact data. For this analysis, a curve fitting program has been used to provide a more consistent curve fit. The prior data has also been reanalyzed using this program to provide consistency, resulting in some minor changes to previously reported results. Appendix D of the report identifies these changes.

Within this report, the chemistry factors used for reactor vessel materials are revised to be consistent with previous submittals made in response to Generic Letter 92-01, Revision 1, Supplement 1 (by letters dated August 17, 1995, November 16, 1995 and September 16, 1998). Based on the data from surveillance capsule 'X', the controlling material has been reestablished as plate B4197-2, consistent with the original design of the reactor vessel surveillance program.

Finally, the analysis of capsule 'X' uses a fluence methodology (BAW-2241P, Revision 1, "Fluence and Uncertainty Methodologies") developed by FTI which is consistent with Draft Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," June 1996. Revision 0 of the FTI methodology was previously approved by the NRC, although only Babcock & Wilcox (B&W) reactor vessels were addressed. Revision 1 of the FTI methodology, submitted to the NRC by FTI on April 30, 1999 to address uncertainty for non-B&W plants, is awaiting final NRC approval.

The new FTI fluence methodology results in the calculation of higher (i.e., more conservative) vessel fluence and thus greater shifts for ΔRT_{NDT}, when compared to the fluence methodology used to evaluate previous surveillance capsules. For example, previous methods used capsule flux measured-to-calculated ratio to adjust the calculated vessel flux, whereas the current FTI methodology uses this ratio only to show consistency with their uncertainty database. Further, a calculated fission energy spectrum is used, consistent with the requirements of DG-1053, instead of the assumed fission energy spectrum specified in ASTM Standard E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (IF)." Also, the ASTM standard identifies 32 effective full power years (EFPY) as the design reactor vessel end-of-life (EOL) for the design of a surveillance program. This is based on a nominal lifetime capacity factor of 80%. A higher lifetime capacity factor of 90% is anticipated for HNP. Therefore, 36 EFPY is being used as the design EOL for HNP.

The existing HNP Technical Specification (TS) pressure-temperature limits are based on the 1992 analysis of capsule 'V,' which used an earlier B&W/FTI fluence methodology. These existing pressure-temperature limits are valid through 11 EFPY, which is anticipated to be reached in summer 2000. The earlier fluence methodology does not result in pressure-temperature limits as conservative as those calculated with the new fluence methodology. The existing limits, however, are considered to provide an acceptable level of protection for the integrity of the reactor vessel.

Revised pressure-temperature curves, based on the analysis results of capsule 'X', the new FTI fluence methodology, and the various changes to assumptions and data described above, will be submitted for NRC approval in early 2000 to permit timely issuance prior to expiration of the current HNP TS curves.

Questions regarding this matter may be referred to Mr. J. H. Eads at (919) 362-2646.

Sincerely,

D. B. Alexander

D. B. Alexander
Manager, Regulatory Affairs
Harris Plant

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Enclosure

c: Mr. J. B. Brady (NRC Senior Resident Inspector, HNP)
Mr. R. J. Laufer (NRR Project Manager, HNP)
Mr. L. A. Reyes (NRC Regional Administrator, Region II)