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June 21, 1982

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Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: Arkansas Nuclear One - Units 1 & 2
Docket Nos. 50-313 and 50-368
License Nos. DPR-51 and NPF-6
Reactor Vessel Pressurized Thermal Shock

Gentlemen:

Representatives of Arkansas Power and Light Company attended the Staff/Industry meeting of June 9, 1982, where you requested comments on the staff's proposal for resolving the pressurized thermal shock (PTS) issue. At that meeting, comments were requested within two weeks. While we do not believe that two weeks is sufficient time to review the content of your proposed actions in detail due to the issues, complexities and potential impact on the industry, we do offer the following comments:

1. We do not believe basing plant operation on the RTNDT of a vessel is the correct method for ensuring vessel integrity. Rather, indexing operation to the actual fracture toughness of the vessel material should be pursued. A B&W Owners Group Program, of which Arkansas Power and Light Company is a member, is currently investigating these properties for B&W vessels. Using actual material fracture toughness properties would permit evaluation of the thermal shock problem with the latest developments in technology. This would allow flexibility for resolution of this issue.

2. We acknowledge that RTNDT could be used as a screening method for flagging plants that may have an actual PTS concern. Such plants should be allowed to perform a plant specific analysis that would more accurately characterize the true condition of the vessel material.

3. We believe that the staff's generic approach toward thermal shock is inappropriate because it does not consider basic design differences between vessel manufacturers (e.g. B&W OTSGs and vent valves) as well as differences between vessels fabricated by the same

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manufacturer (e.g. weld locations and weld composition, pump flow differences and operation procedures). These differences would have a significant bearing on the effect of transients on vessel integrity.

4. The transient selected by the staff appears unrealistic. We do not believe pressure can be maintained at 2500 psi throughout such a transient.

5. ANO-Unit 1 has not experienced any significant overcooling transients throughout its life. The probability of experiencing overcooling events similar to those that occurred at other B&W units has been further reduced by equipment and instrumentation modifications including modifications to the Non-Nuclear Instrumentation power supply system on ANO-1 and installation of reactor coolant subcooling margin displays in both units.

   The one overcooling transient that occurred in ANO-Unit 2 occurred during plant startup testing and was very moderate compared to the NRC proposed transient. Operation procedures implemented should reduce the probability of recurrence of this transient.

6. The NRC proposal appears to equate crack initiation with vessel failure. This assumption does not appear consistent with ASME Code provisions.

7. We have not had the opportunity to review the EPRI evaluation of thermally annealing a reactor vessel. This is an unproven method which may involve many engineering difficulties. Before we accept this method as a viable option of recovering material toughness, we would want to review an actual demonstration of this process. Certainly, before conducting any anneal, we would perform and submit for your review a plant specific analysis that would demonstrate whether such an action would be required.

In summary, we disagree with some basic assumptions used in your proposal. Furthermore, we do not sense the urgency for imposing inflexible regulations that do not consider all the parameters that should be factored into resolving the thermal shock issue.

Very truly yours,

John R. Marshall

John R. Marshall
Manager, Licensing

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