AUGUST 21 1981

Docket No. 50-285

Mr. W. C. Jones
Division Manager, Production
Operations
1623 Harney Street
Omaha, Nebraska 68102

Dear Mr. Jones:

SUBJECT: PRESSURIZED THERMAL SHOCK TO REACTOR PRESSURE VESSELS

We have reviewed the PWR Owners' Groups responses of May 15, 1981 and the licensees' responses of May 22, 1981 to our letter dated April 20, 1981 concerning the subject issue. The EPRI work which bears on the issue was included in the licensees' responses. On the basis of our independent review, of the plants where neutron irradiation has significantly reduced the fracture toughness of the reactor pressure vessels (RPVs), all plants could survive a severe overcooling event for at least another year of full power operation. However, we believe that additional action should be taken now to resolve the long-term problems.

This belief is based upon our analyses which indicate that reductions in fracture toughness for some RPVs are approaching levels of concern. It is also based in part on the fact that any proposed corrective action must allow adequate lead time for planning, review, approval, procurement and installation. These conclusions were recently discussed with the PWR Owners Groups on July 28-30, 1981. At those meetings, the Owners Groups reviewed the programs underway at the three PWR vendors which are designed to scope the magnitude and applicability of the generic problem and to be completed by late 1981. The three programs appeared to contain the necessary elements for resolution of the problem on a generic basis and the NRC plans to make full use of the reports due by the end of the year. While the vendors and Owners Groups are to be commended and encouraged in addressing the generic issue, there is also a need for plant-specific information for your plant.

Based on current vessel reference temperature and/or system characteristics, we have identified Ft. Calhoun, Robinson 2, San Onofre 1, Maine Yankee, Oconee 1, Turkey Point 4, Calvert Cliffs 1 and Three Mile Island 1 as plants from which we require additional information at this time.

The staff has used the time-dependent pressure and temperature data from the March 20, 1978 Rancho Seco transient as a starting point for our evaluation of this issue because: (1) it is the most severe overcooling event experienced to date in an operating plant; (2) it is a real, as opposed to a postulated, event; and (3) it was severe enough that it could challenge the RPV when combined with physically reasonable values of irradiated fracture toughness and initial crack size. In future reviews the staff plans to use the steam line break accident or other appropriate transient/accident in order to estimate minimum operational times available before plant modifications are required.

Using calculated RPV steel mechanical properties, credible initial flaw sizes, reasonable thermal-hydraulic parameters, and a simplified pressure-temperature transient similar to that observed during the Rancho Seco event. the staff has concluded that all operating plants could safely survive such an event at the present time and for at least an additional year of full power operation. However, because of the required lead times for future actions, the margins in time for long term operation are not large, and there is considerable uncertainty in the probability tnat similar or more severe transients may occur. It is clear that positive action must be initiated soon for those plants with significantly high transition temperatures. As indicated above, several such plants have been selected by the staff, based on estimates of the current reference temperature for the nil ductility transition (RT_{NDT}) of the RPVs.

The need to initiate further action at this time is emphasized by the recognition that implementation of any proposed fixes or remedial actions must allow for adequate lead time. Because long-term solutions may require a year or more, you should explore short-term approaches as well. Although clear, concise instructions should be provided to operators to reduce the likelihood of repressurization during overcooling transients, the NRC staff believes that reliance on operator actions to prevent repressurization during an overcooling transient will be very difficult to justify as an acceptable long-term solution to the problem.

In accordance with 10 CFR 50.54(f) of the Commission's regulations, you are requested to submit written statements, signed under oath or affirmation, to enable the Commission to determine whether or not your license should be modified, suspended or revoked. Specifically, you are requested to submit the following information to the NRC within 60 days from the date of this letter:

(1) Provide the RT_{NDT} values of the critical welds and plates (or forgings) in your vessel for:
   (a) initial (as-built) conditions and location (e.g., 1/4 T) and
   (b) current conditions (include fluence level) at the RPV inside carbon steel surface.

(2) At what rate is RT_{NDT} increasing for these welds and plate material?

(3) What value of RT_{NDT} for the critical welds and plate material do you consider appropriate as a limit for continued operation?

(4) What is the basis for your proposed limit?

(5) Provide a listing of operator actions which are required for your plant to prevent pressurized thermal shock and to ensure vessel integrity. Include a description of the circumstances in which these operator actions are required to be taken. Included in this summary should be the specific pressure, temperature and level values for: a) high pressure injection (HPI) termination criteria presently used at your facility, b) HPI throttling criteria and instruction presently used at your facility and c) criteria for throttling feedwater presently used at your facility. For each required operator action, give the information available to the operator and the time available for his decision and the required action. State how each required operator action is incorporated in plant operating procedures and in training and requalification training programs.

You are also requested to submit a plan for Ft. Calhoun to the NRC within 150 days of the date of this letter that will define actions and schedules for resolution of this issue and analyses supporting continued operation. We request that you include consideration and evaluation of the following possible actions:

(1) reduction of further neutron radiation damage at the beltline by replacement of outer fuel assemblies with dummy assemblies or other fuel management changes;

(2) reduction of the thermal shock severity by increasing the ECC water temperature;

(3) recovery of RPV toughness by in-place annealing (include the basis for demonstrating that your plant meets the requirements in 10 CFR 50 Appendix G IV C);

(4) design of a control system to mitigate the initial thermal shock and control repressurization.

For these, as well as for any other alternative approaches, provide implementation schedules that would assure continuance of adequate safety margins.

In the interest of efficient evaluation of your submittal, we request that you include with the above plan, a response to the enclosed request for additional information.

Due to the nature of this review, and the past review effort that has been expended, we consider the above schedules to be reasonable; however, inform us within 30 days if you anticipate conflicts with previous commitments with either submittal and a basis for any delay. We also expect participation by the appropriate PWR Owners Group and NSSS vendors in developing solutions to the problem.

Sincerely,

Original signed by

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
Request for Additional
Information

cc w/enclosure:
See next page

OFFICE	ORB#4:DL	AD-OR:DL	D:DST	D:DL			
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DATE	8/21/81	8/21/81	8/21/81	8/24/81			

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Enclosure

REQUEST FOR ADDITIONAL INFORMATION

1. Geometry

Geometrical description including design and as-built (when available) dimensions of the core, assemblies, shroud/baffle, thermal shield, downcomer, vessel, cavity, and surrounding shield and/or support structure.

2. Material Description

Region-wise material composition and material isotopic number densities (atoms/barn-cm) for the core, near-core regions and RPV, suitable for neutron transport calculations.

3. Neutron Source

Present and expected EOL:

a) Assembly-wise and core power history (EFPY).
b) Rod-wise and core power history (EFPY) for peripheral assemblies.
c) Core average axial power history distribution.

4. Vessel Fluence

a) Description of available calculations of the vessel fluence including fluence values, locations, and corresponding power histories (EFPY), including 1/4T, 1/2T and 3/4T through the RPV.
b) Description of available capsule-inferred vessel fluences including fluence values, locations, and corresponding power histories (EFPY).

5. Surveillance Capsules

a) Capsule materials, radial and axial dimensions and locations.
b) Capsule fluence measurements, together with the accumulated power history (EFPY) and a description of the lead factors used to extrapolate the measurements to the peak wall fluence location.

6. Vessel Welds

Axial and azimuthal locations of vessel weld-seams with respect to the core. Overlay of current fluence map with weld locations. Identify the critical welds, vertical and circumferential, and give the weld wire heat numbers. Give weld chemistry for the critical welds. For each weld wire heat number, report the estimated mean copper content, the range and the standard deviation, based on all the reported measurements for that weld wire heat. The welds may be surveillance weldments for your vessel or others, nozzle dropouts that contain a weld, weld metal qualification data, or archive material. In the absence of any information, assume that copper content is at its upper limit (0.35 percent when using R.G. 1.99, Rev. 1) and that the nickel content is high.

7. Systems Analysis

a) Provide a list of transients or accidents by class (for example: excessive feedwater, operating transients which result from multiple failures including control system failures and/or operator error, steam line break and small break LOCA) which could lead to inside vessel fluid temperatures of 300 F or lower. Provide any Failure Modes and Effects Analyses (FMEAs) of control systems currently available or reference any such analyses already submitted. Provide the analysis of the most limiting transient or accident with regard to vessel thermal shock considerations. Estimate the frequency of occurrence of this event and provide the basis for this estimate. Discuss the assumptions made regarding reactor operator actions.

b) Identify the computer programs used to calculate the limiting transient or accident. Indicate the degree to which the computer programs used have been verified and any other additional verification required to demonstrate that the computer program models adequately treat the identified important physical models (i.e., ECC mixing, heat transfer, and repressurization).