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Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Eisenhut:

Subject: Reactor Vessel Pressurized Thermal Shock

References: (A) NRC Memorandum from T. J. Walker to S. S. Pawlicki, April 9, 1981, "Minutes of PWR Owners' Group Meeting with NRC on March 31, 1981"

(B) C-E Letter from W. R. Corcoran to F. Schroeder, dated June 24, 1975, LD-75-431

This letter fulfills a commitment made by the C-E Owners Group in response to your request at a meeting on March 31, 1981, (Ref. A) and responds to further requests and clarifications which you provided at a meeting on April 29, 1981, concerning the issue of reactor vessel pressurized thermal shock.

This letter addresses three general aspects of the reactor vessel pressurized thermal shock issue. The letter provides a brief summary of previous analyses of the integrity of reactor vessels in C-E designed nuclear steam supply systems (NSSS) which confirm that there is sufficient safety margin to allow continued operation of these plants. The letter contains a description of further programs which are currently being reviewed by the C-E Owners Group to develop further measures to assure continued reactor vessel integrity and hence safe plant operation. We understand a general description of an industry program to assure reactor vessel integrity is being prepared by the Electric Power Research Institute and will be forwarded at a later date.

ASSURANCE OF VESSEL INTEGRITY

Prior to the March 31, 1981 meeting, the C-E Owners Group had not as a group participated in any activities involving the evaluation of reactor vessel integrity during overcooling/repressurization transients which are initiated in the secondary system. The C-E Owners Group had previously authorized Combustion Engineering (C-E) to develop a program for the purpose of

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fulfilling the requirements of NUREG-0737, Item II.K.2.13. However, this program had not yet been fully developed by C-E nor acted upon by the C-E Owners Group. The development of this program was intentionally deferred so that the manpower available could be more effectively utilized to resolve higher priorty TMI-related requirements relevant to the safe operation of our plants. Nevertheless, following the March 31 meeting, the C-E Owners Group met and authorized C-E both to research the work that had been performed to evaluate this concern in the past and to develop an integrated plan to resolve this concern. The previous work most relevant to recent NRC concerns was presented to the NRC staff at the April 29, 1981 meeting.

On June 24, 1975, C-E transmitted to the NRC in Reference (B), the results of conservative studies that specifically evaluated the effect of the main steam line break (SLB) transient on reactor vessel structural integrity. Those studies showed:

A. For new C-E vessels made of controlled residual element weld and plate material, no operator action would be required to mitigate the consequences of a steam line break over the entire 40 year lifetime of plant operation.

B. For early generation C-E vessels made of higher residual element weld and plate materials, no operator action would be required to mitigate the consequences of a steam line break transient for at least the first 20 years of plant operation.

The recent NRC concerns have been directed toward those vessels made of higher residual element weld and plate material since those are the vessels which are most affected and which have been in service for the longest period of time. The C-E vessel with the longest service life in terms of irradiation, has accumulated slightly over five effective full power years of operation. Even if one were to assume that actual fluences are twice those predicted in Safety Analysis Reports, approximately five more effective full power years of operation would have to elapse before vessel integrity could theoretically become a concern for these events.

The steam line break transient produces the largest magnitude and rate of heat removal in the C-E NSSS design. This transient would bound credible postulated control system anomalies. In addition, when this event was evaluated, the analysis contained numerous conservative assumptions described below:

1. No-load initial conditions: This assumption produces the largest potential cooldown since it involves the greatest inventory in the steam generators. In addition, these conditions also involve the lowest initial reactor vessel temperature. Together, these conditions produce the lowest potential reactor vessel temperature which would be the limiting case during repressurization.

2. No moisture carryover: Assuming no moisture carryover during the SLB transients produces the greatest total energy removal.

3. Clean Core: A clean core produces no decay heat added during the transient and a zero moderator coefficient of reactivity (i.e., no reactivity added due to moderator cooldown).

4. Maximum safety injection and charging pump flow after the Safety Injection Actuation Signal: This mode of operation refills the pressurizer most rapidly.

5. 40°F Safety Injection water: This low temperature water further cools down the reactor vessel as the reactor coolant system is refilled.

6. Failure-to-open of the pressurizer power operated relief valves: Failure of these valves produces a maximum pressurizer pressure of 2500 psia (the safety valve setpoint) instead of 2400 psia.

7. No heat addition due to Reactor Cooldown System metal cooldown.

8. No operator action to mitigate the consequences of the transient.

These assumptions were applied to a typical plant with a reactor vessel of either controlled or uncontrolled metallurgical properties. The integrity of the reactor vessel was evaluated using the conservative linear elastic fracture mechanics procedures of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, Appendix A.

Additional conservatisms used in the fracture mechanics analyses to evaluate the reactor vessel material, which were not discussed at the April 29, 1981 meeting, are listed below.

1. The temperature distribution in the reactor vessel wall was computed using a high heat transfer coefficient. The value selected was sufficient to produce the most rapid cooling which the wall can experience.

2. A spectrum of sizes of a crack hypothesized to exist in the vessel wall prior to the cooldown event was evaluated. The hypothetical initial crack size ranged from about one half inch deep to one quarter of the wall thickness. This was done to ensure that the most limiting crack size was included in the evaluation.

3. Warm prestressing, which would raise the toughness of the material at the repressurization temperature, was conservatively ignored.

4. In the case of reactor vessels with higher residual element weld and plate material, the potential for crack initiation and growth was found to exist for vessels with greater than 20 effective full power years of operation. The potential for subsequent crack arrest due to attenuation of material irradiation damage and the increase in material temperature toward the outside of the vessel wall was not assessed.

The assessment of information prepared by C-E indicates there is no immediate safety concern.

PROGRAM UNDER CONSIDERATION

The C-E Owners Group members are currently reviewing a program plan developed by Combustion Engineering to provide options for continued assurance of reactor vessel integrity under pressurized thermal shock conditions. This program plan has been developed by request of the C-E Owners Group and will be considered for authorization at a meeting of the C-E Owners Group on June 4, 1981.

The issue of reactor vessel pressurized thermal shock has four major components: (1) determination of NSSS thermal-hydraulic transients which can produce pressurized thermal shock, (2) determination of stress distributions within, and mechanical behavior of, the reactor vessel in response to such transients, (3) determination of the relevant mechanical properties of the reactor vessel, and (4) determination of the margins available to assure that reactor vessel integrity is preserved. The program plan being reviewed by the C-E Owners Group addresses each of these issues. The program under consideration will evaluate a spectrum of NSSS thermal-hydraulic transients initiated by anticipated operational occurrences (AOO), small break loss of coolant accidents (LOCA), and steam line break (SLB) accidents. The analyses will be continued beyond the initial cooldown or blowdown transient in order to evaluate the effects of injecting cold emergency core cooling (ECC) water into the reactor coolant system (RCS). The consequences of core cooling using the ECC water following a small break LOCA in the absence of all feedwater to the steam generators and the influence of mixing of the ECC water and the fluid within the RCS will also be evaluated. Designs of all participating C-E Owners Group plants will be reviewed to determine the ranges or groupings of plant parameters which must be considered.

The program includes linear elastic fracture mechanics analyses to evaluate the effect of the temperature/pressure transients including local temperature gradients produced by incomplete mixing of ECC water injection and RCS fluid. The influence of hypothetical initial cracks in the reactor vessel of various sizes and orientations will be determined. The temperature and irradiation level dependence of reactor vessel metal toughness properties will be included in the analysis. In order to determine the potential for crack initiation, crack extension, and crack arrest, the toughness properties will be evaluated as function of wall depth. Fracture mechanics analyses of both reactor vessel base material and weld material will be conducted.

The program includes an evaluation of the current irradiation condition of the reactor vessel of each participating C-E Owners Group member's plant(s). These evaluations will use as input actual plant operating history including such major parameters as core power distribution, reactor coolant temperature, and fuel management. Dosimetry data from surveillance capsules will be included where it is available and appropriate. Explanation of a comparison between the newly calculated fluence valves and values previously recorded in Safety Analysis Reports will be provided. The program includes determination of the margin of safety available to assure reactor vessel integrity. This margin will be measured by the difference between the number of equivalent full power years calculated to produce theoretical violation of the primary pressure boundary under the most limiting transient conditions and the current number of equivalent full power years of operation completed. Representative values of this margin will be determined for each participating C-E Owners Group member's plant(s). In addition, potential remedial measures will be identified including potentially beneficial operator actions. Current C-E emergency procedure guidelines will be reviewed to determine their sufficiency or identify appropriate modifications.

The program being reviewed by the C-E Owners Group will provide a generic response to NRC Action Plan Item II.K.2.13, as clarified in NUREG-0737, for C-E designed reactor vessels. The program will also provide generic response to concerns raised by the NRC at the meeting on March 31, 1981, concerning non-LOCA transients and respond to concerns raised by the NRC at the meeting on April 29, 1981, related to evaluation of irradiation conditions in reactor vessels. It is expected that the schedule for response to Action Plan Item II.K.2.13 will be met. However, the schedule for completion of response to the remaining items is still under evaluation.

The C-E Owners Group is providing the information in this letter as a participant in the general industry effort to obtain resolution of the reactor vessel pressurized thermal shock issue. However, any opinions or statements of intent contained in this letter do not necessarily represent the opinion or intention of any specific utility. Applicability of the contents of this letter to any specific licensee or license applicant docket must be provided by that licensee or license applicant. Such correspondence is the responsibility and prerogative of each individual utility.

We will be pleased to meet with you to discuss this matter in further detail if you desire.

Very truly yours,

K P Baskin

Chairman
C-E Owners Group

cc: J. Hutton
C-E Owners Group