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MEMORANDUM FOR: E. G. Case, Acting Director
Office of Nuclear Reactor Regulation

R. B. Minogue, Director
Office of Standards Development

FROM: Saul Levine, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER - #16 - WARM PRESTRESSING

Introduction and Summary

This Research Information Letter transmits verified experimental data on warm prestressing for use in the licensing evaluation of nuclear power plants.

One of the items considered in the evaluation of reactor system response to the loss-of-coolant design basis event is the effect which the relatively cold emergency core coolant has upon the hot reactor pressure vessel. The resultant temperature gradients induce thermal stresses. Under "worst case" conditions, analyses of the thermal stresses that could occur from the thermal shock, in combination with small flaws on the inside surface of the vessel, lead to the prediction that the flaws will extend. Fortunately, there exists a combination of circumstances during a loss-of-coolant accident (LOCA), whereby the crack tip region of an assumed flaw is subjected to warm prestress, a phenomenon that can preclude crack extension, when it otherwise would have been predicted. To describe the warm prestress effect, once a crack is loaded while the material is very tough, no rapid extension will occur during subsequent combined cooling and unloading. This applies even if the imposed stress intensity factor, K_I, which is a function of stress and flaw severity, exceeds the fracture toughness during the cooldown-unloading. Linear Elastic Fracture Mechanics (LEFM), without consideration of the effects of warm prestressing, predicts rapid extension of a crack when the K_I becomes equal to the fracture toughness of the material, K_{Ic}. Thus, it is necessary to have verified procedures for evaluating the initiation and propagation of flaws in reactor pressure vessel steels under realistic thermal shock conditions with inclusion of the warm prestress phenomenon.

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16-126

DEC 0 1 1977

Analytical methodology based on LEFM has been advanced in the Heavy Section Steel Technology (HSST) program at Oak Ridge National Laboratory (ORNL) to evaluate, under LOCA conditions, the factors of stress, toughness, flaw size and irradiation embrittlement of the vessel steel. Calculations were performed by the HSST staff using a reference calculational model that encompassed the worst combination of flaw sizes and vessel wall embrittlement but did not consider the benefits of warm prestressing. Four experiments were subsequently conducted to check out these LEFM analytical procedures (see Enclosures 1 and 2). The behavior of the flaws in these four experiments was accurately predicted, thus verifying the LEFM analytical methodology. Moreover, observations from these tests indicate that any crack extension resulting from a thermal shock is self-limiting to only a fraction of the wall thickness.

To improve the ORNL thermal shock analytical methodology by inclusion of a model for warm prestressing, an experimental investigation has been conducted at the Naval Research Laboratory (NRL). Experiments at NRL that simulate the behavior of the vessel material during thermal shock have demonstrated that warm prestressing can "effectively" elevate the fracture toughness of the vessel material, and therefore, limit the crack extension. While warm prestress cannot prevent the extension of shallow flaws, it can prevent deep flaws from extending further. It has been concluded that warm prestress will limit crack extension to less than one-third of the vessel wall thickness. Thus, warm prestress can form a key element upon which to base the assurance of reactor vessel integrity during a LOCA.

Results

Carefully controlled warm prestress experiments were conducted at NRL with notched bend specimens from unirradiated A533-B steel, in thicknesses of 1-1/2 and 3 inches to simulate the loading conditions of a crack in a reactor vessel wall. These experiments, which were separated into two phases, are described in this section and in the enclosures.

During Phase One of the experimental program, the phenomenon of warm prestressing was characterized in broad terms to evaluate the importance of significant variables such as specimen thickness, loading path, face grooves and failure temperature. This study employed material (HSST plate 02) whose fracture toughness had been well characterized under the HSST program. (Phase One is described in Enclosure 3). A different heat of A533-B steel was utilized in Phase Two, wherein the emphasis was to further quantify the benefits of warm prestressing under conditions of a small temperature difference (ΔT) between the failure temperature

16-127

DEC 0 1 1977

and the temperature at which the K_I was the highest. (Phase Two is described in Enclosure 4.) It is this case of a small ΔT that is most closely applicable to the postulated accident conditions being analyzed.

These research studies have shown that warm prestressing provides an effective elevation of the critical stress intensity factor, K_{Ic}, under severe accident conditions as defined by the ORNL reference calculational model. For tests run under a variety of thermal shock transient conditions, the enclosed research data demonstrate that once warm prestressing takes place, crack extension cannot occur at lower temperatures, as the vessel cools.

By applying the results of the warm prestressing experiment to the ORNL analytical studies of the reference calculational model, NRL has shown that under the most severe conditions, the crack can penetrate no more than 1/3 of the vessel wall. This means that the vessel will always be capable of retaining ECCS water, thus keeping the core cool and providing for a safe shutdown. Thus, vessel failure is not possible following warm prestressing under LOCA/ECCS conditions.

Evaluation

The warm prestress data have been reviewed by the NRC Vessel Integrity Review Group on July 20 and 21, 1977, and found to be excellent. This research was well received at the Fourth and Fifth Water Reactor Safety Research Information Meetings in September 1976 and November 1977, and the Fourth International Conference of Structural Mechanics in Reactor Technology (SMIRT) held in San Francisco, California, in August 1977. The results have been submitted for publication in a forthcoming issue of *Nuclear Engineering and Design*. Thus, RES believes there has been sufficient peer review to warrant the incorporation of these results in the licensing evaluation of nuclear power plants.

Saul Levine

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Enclosures: see next page

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1568 184

DEC 0 1 1977

Enclosures

Enclosure 1: ORNL/NUREG/TM-31, "Pressure Vessel Fracture Studies Pertaining to a PWR LOCA-ECC Thermal Shock: Experiments TSE-1 and TSE-2," R. D. Cheverton, Oak Ridge National Laboratory, September 1976

Enclosure 2: "Thermal Shock Studies Associated with Injection of Emergency Core Coolant in Pressurized Water Reactors," R. D. Cheverton, S. E. Bolt, and S. K. Iskander, Fourth International Conference on Structural Mechanics in Reactor Technology, San Francisco, California, August 15-19, 1977, Paper Number G 9/3

Enclosure 3: NRL Report 8165, "Significance of Warm Prestress to Crack Initiation During Thermal Shock," F. J. Loss, R. A. Gray, Jr., and J. R. Hawthorne, Naval Research Laboratory, Washington, D.C., in publication, also Fourth International Conference on Structural Mechanics in Reactor Technology, San Francisco, California, August 15-19, 1977, Paper No. G 9/1*

Enclosure 4: "Investigation of Warm Prestress for the Case of Small ΔT During a LOCA," F. J. Loss, R. A. Gray, Jr., and J. R. Hawthorne, Naval Research Laboratory, Washington, D.C., September 1977

17-129

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