Florida Power
CORPORATION

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

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Pressurized Thermal Shock

Dear Mr. Denton:

On June 9, 1982, at an industry/NRC meeting, you requested industry input on the regulatory position under Staff consideration for Pressurized Thermal Shock (PTS) of the reactor vessel. It is our understanding this position would utilize RT_{NDT} (Reference Temperature for Nil Ductility Transition) as an index for evaluating the PTS concern. Florida Power Corporation has generated several comments and positions concerning your request and hereby provides them for your consideration.

A significant amount of attention has been devoted to address the PTS concern over the past two years. Florida Power Corporation has actively participated in programs designed to better understand and reduce the potential risk imposed by PTS. Recently performed calculations based on Regulatory Guide 1.99 methodology indicate the current RT_{NDT} of the WF-70 weld is about 164°F. Further calculations indicate that the RT_{NDT} limit of 250°F currently expressed as a concern by the Staff will not be reached within the next seven (7) Effective Full Power Years (EFPY). This confirms there is no near-term safety risk posed by PTS at Crystal River Unit 3 (CR-3).

In evaluating PTS, the focus has primarily been on two classes of events (i.e., Small Break Loss of Coolant Accidents and overcooling events). Of the two categories, the potential risk posed by overcooling events is deemed more adverse. To more effectively reduce the probability of an overcooling event at CR-3, we have submitted a schedule for implementation of an Emergency Feedwater Initiation and Control (EFIC) System during Refuel V (see FPC letter dated June 18, 1982). EFIC will limit overcooling events such as Loss of Main Feedwater (LMFW), Main Feedline Breaks, Main Steamline Break/Auxiliary Feedline Breaks, and Steam Generator Overfill events by initiating automatic control of feedwater (i.e., independent of operator action).

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The summary of actual overcooling events presented at the June 3, 1982 ACRS meeting expresses concern over the February 26, 1980 Non-Nuclear Instrumentation/Integrated Control System (NNI/ICS) failure at CR-3. Also, a report recently received by FPC from Oak Ridge Laboratories on PTS addresses this incident as being the most severe PTS incident in U.S. experience. We have found several of Dr. Doan L. Phung's findings to be in error and take exception to others. Dr. Phung's findings indicated that both steam generators boiled dry, when actually only steam generation "A" boiled dry. In addition, reverse flow conditions existed in loop A following the boiling dry of the steam generator. The Borated Water Storage Tank (BWST) water from the high pressure injection (HPI) system in this reverse flow loop would have heated up considerably during this process, thus posing no PTS risk. Dr. Phung maintains that HPI "went full speed for 32 minutes (at about 1100 gpm) and at 250 gpm for the following 60 minutes (total injection of about 57,000 gallons)." Results from the report by B&W for this incident show HPI going full speed for 26 minutes and throttled flow for 58 minutes (total injection of about 43,000 gallons). Substantial evidence to refute Dr. Phung's assumption of completely stratified flow is also in evidence. In Electric Power Research Institute's (EPRI's) report Fluid and Thermal Mixing in a Model Cold Leg and Downcomer with Vent Valve Flow found that

> "This HPI flow undergoes a hydraulic jump at the location where the cold leg pipe bends to the horizontal direction before intersecting the downcomer. In turn, the hydraulic jump leads to high rates of mixing and entrainment of primary coolant such that fluid entering the downcomer has been warmed substantially. Additional mixing with the vent valve flow occurs in the downcomer. Thus, there are mechanisms for fluid mixing even in the absence of any forced or natural circulation of primary coolant through the cold leg."

Results based on the EPRI tests using the Froude number applicable to the event indicate a temperature between 135°-140°F where the coolant contacts the downcomer. In addition, the B&W report states natural circulation was in effect, causing further mixing. The B&W report further states brittle fracture is not a concern since the minimum inlet downcomer coolant temperature based on calculations was 250°F, and crack initiation did not take place.

Following the NNI/ICS failure at CR-3 of February 26, 1980, redundant monitoring systems were installed and operator guidelines were updated to prevent another overcooling event caused by NNI/ICS failure. The installation of EFIC would also prevent overcooling due to loss of instrumentation by control of feedwater level being independent of operator action.

In response to your request for input, Florida Power Corporation has generated the following comments:

1. During a cooldown event at CR-3, the primary system pressure does not remain constant as proposed in the design cooldown. In addition, there are systems/operating procedures that provide a unique response to an initiating event.

2. The emphasis on the change in fracture toughness should focus on the actual fracture toughness rather than on the RT_{NDT} of the material. RT_{NDT} would be helpful in pinpointing plants with potential concerns, but to redirect work that is responsibly addressing this concern would be self-defeating.

3. The Staff position on PTS should be submitted to the Committee for Review of Generic Requirements for comments before any final position is implemented.

FPC is currently participating in a B&W Owners' Group project that is performing realistic, yet conservative evaluations. As this project has shown that there is no near-term safety concerns, there is no need for hasty action that may actually decrease plant safety. Florida Power Corporation has, and will continue to work with the Staff individually or through the B&W Owners' Group, to ensure the safe and timely resolution of this concern.

Very truly yours,

David G. Mardis

David G. Mardis
Acting Manager
Nuclear Licensing

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