# SAFETY GUIDE 2

# THERMAL SHOCK TO REACTOR PRESSURE VESSELS

## A. Introduction

Proposed General Design Criterion 35 specifies design and operating conditions necessary to assure that the reactor coolant pressure boundary will behave in a nonbrittle manner. To provide protection against loss of coolant accidents, present designs provide for the injection of large quantities of cold emergency coolant into the reactor coolant system. The effect on the reactor pressure vessel of this cold water injection is of concern because the reactor vessel is subjected to greater irradiation than other components of the reactor coolant pressure boundary and, thus, has a greater potential for becoming brittle. A suitable program which may be used to implement General Design Criterion 35 to assure that the reactor pressure vessel will behave in a nonbrittle manner under loss of coolant accident conditions is described in this guide.

## B. Discussion

The injection of cold water by the emergency core cooling system into a hot reactor pressure vessel after a loss of coolant accident raises the possibility that a vessel embrittled by irradiation and having a small internal defect could fail suddenly as a result of the large thermal gradient imposed and the resulting high stresses. Analyses by the reactor vendors indicate that cold water injected into a hot reactor pressure vessel toward the end of the vessel's service life could cause incipient defects of the maximum size expected to grow; however, the maximum crack depth is predicted to be no more than 30 to 60 percent of vessel wall thickness. The vessel is not expected to fail under these conditions. The maximum crack depth expected cannot be firmly established since the vessel material fracture toughness properties assumed in the analyses have not yet been completely confirmed.

The additional data needed to resolve the uncertainties in the fracture toughness properties of reactor vessel material are expected to be provided by the Heavy Section Steel Technology (HSST) research and development program. Since reactor vessel materials are initially ductile and their fracture toughness properties are not significantly changed upon irradiation during the initial 5 years of operation, the potential for reactor pressure vessel failure as a result of cold water injection is considered to be acceptably small during this period. Sufficient data should be available from the HSST Program to permit a final judgment within this 5-year period on the acceptability of the projected behavior of vessel material throughout its service lifetime.

In the event that the results of the HSST Program or other research indicate that the potential for growth of defects in radiation embrittled reactor pressure vessel material reduces the available margin of safety against brittle fracture to an unacceptable level, an acceptable engineering solution to the problem could be applied—for example, thermal annealing of the reactor vessel material. Naval Research Laboratory data indicate that annealing of a PWR vessel at its design temperature (650°F) for a period of 168 hours should produce a recovery in fracture toughness properties and reduce the transition temperature shift due to irradiation by 30 to 50 percent (i.e., a 100°F shift in transition temperature would be reduced to 70-50° after annealing). Annealing BWR vessels at design temperatures and for equivalent time periods would, if needed, provide an equivalent degree of recovery. Based on the calculation of potential irradiation effects in presently designed PWRs and BWRs, this degree of recovery of material toughness properties combined with the potential for repeating the annealing process, if required, appears to be adequate to permit continued plant operation with the same reactor pressure vessel throughout plant lifetime.

## C. Regulatory Position

To assure that the reactor pressure vessel will behave in a nonbrittle manner under loss of coolant conditions, the following program should be followed:

1. Data collection and research work on the properties of reactor pressure ves-

sel material should be continued in order to permit verification that expected material properties assure nonbrittle behavior of the reactor vessel throughout its lifetime under postulated accident conditions. It is expected that this determination can be made within 5 years.

2. During the 5-year period necessary to develop the needed data, the potential reactor pressure vessel thermal shock problem which may result from emergency core cooling system operation need not be reviewed in individual cases unless significant changes in presently approved core or reactor pressure vessel designs are proposed.
3. Should it be concluded that the margin of safety against reactor pressure vessel brittle failure due to emergency core cooling system operation at any time during vessel life is unacceptable, an engineering solution, such as annealing, could be applied to assure adequate recovery of the fracture toughness properties of the vessel material. In the meantime, applicants should outline available engineering solutions and show that their designs do not preclude the use of such solutions.