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Radiation Weakens Pressure Vessels

The Nuclear Regulatory Commission (NRC) has become increasingly concerned about one of the most serious potential accidents in a nuclear reactor. Early in July, the agency released a list of 14 pressurized water reactors (PWR) that have become especially prone to pressure vessel rupture, an accident that one senior NRC official said would create "a 50-50 chance of an uncontrolled nuclear meltdown."

Pressure vessel rupture is such an extraordinarily disabling accident that it would probably prevent any system, includin the emergency core cooling system (ECCS from cooling the core. The Nuclear Regu latory Commission's concern is that pre surized water reactors aged three to te years have vessels that have become ur expectedly brittle from intense neutro radiation over their lives. Brittleness especially troublesome at high pressur and low temperatures, and can cause pressure vessel to crack like hot gla dunked in cold water.

A number of accidents can cause pressu vessels to rupture. The gravest danger generally caused by a small water leak the primary or secondary reactor coolir system. This results in a prompt decrea in pressure in the system, and operato turn to the emergency core cooling syste to rapidly reduce temperatures and preve formation of any steam in the system. Tl Three Mile Island accident was caus when reactor operators prematurely turn off the emergency system, mistaken guessing that the primary system had "go solid" because water was being releas from the pressure relief valve in the plan pressurizer. One official told NMA in t aftermath of the accident, "They probab did it to prevent overpressurization."

The Narrow Line Between Cooling and Cracking

Overpressurization and pressure ves rupture are precisely the NRC's curr concerns, creating a contradictory situati for PWR operators: on one hand, th must keep the ECCS on long enough to absolutely certain the primary system solid; on the other hand, they must ma sure the ECCS is not on so long that pressure vessel cools too rapidly a cracks. The penalty for failing to walk t narrow line is meltdown either way—f from preventing reactor cooling and seco from cooling it too much.

Since 1973, there have been 13 seri "overcooling" or overpressurization a dents, the most famous of which is Rancho Seco light bulb accident in Ma 1978. The accident, which tops every ot California light bulb joke, happened w repairman unscrewing a light bulb drop it behind the reactor control panel, ther shorting out the plant's integrated con system (ICS). The ICS mistakenly r catastrophic reactor conditions on n functioning gauges, and the reactor steam generator tripped, activating emergency core cooling system.

The first stage of this system, the H Pressure Injection System, cooled the actor core at a rate of 300°F per ho 200°F per hour faster than NRC reg tions permit. The pressure vessel was verely overcooled as a result.

The Department of Energy's Oak Ri National Laboratory recently repo that the probability of a pressure ve burst at Rancho Seco "would have b very high" if the reactor had been opera at full power for ten years at the time o 1978 accident. Fortunately, it had been operating at full power for four ye

Brittle Bones at Such A Tender Age!

Long periods of reactor operation result in substantial radiation exposure to the steel in a pressure vessel. A new pressure vessel can withstand temperatures of 0-40°F at high pressures without becoming so brittle that it would crack. As a reactor ages, however, the six to twelve-inch-thick steel vessels become "embrittled"—they become less resistant to fracture at progressively higher temperatures. According to the NRC, three reactors have pressure vessels that become brittle at 250-280°F. These are the San Onofre 1 reactor, Robinson 2, and Fort Calhoun; they began commercial operation in 1968, 1971, and 1973, respectively. Thirteen is a young age for brittle bones. The cost-benefit analysis prepared to justify construction of these plants assumed 30 to 40 years of trouble-free operation.

"I think we've got a year, most of the staff would probably say five," noted one senior NRC safety official. "We know [these plants] are not going to last their full design lifetimes."

All three manufacturers of pressurized water reactors admit the seriousness of the problem. In a May 1981 report to the NRC, Westinghouse states that its most vulnerable reactors can withstand a severe overcooling accident through January 1983. Combustion Engineering reports that its reactors would last at full power for five more years at twice the predicted embrittlement rates before vessel strength becomes a problem. Babcock & Wilcox reports that there is "no immediate problem" between now and 1983, "given operator action" *not to repressurize the vessel*. Combustion Engineering has found that embrittlement at the Maine Yankee power plant is occurring 2.3 times faster than they expected.

The normal operating temperature inside the pressure vessel is 600°F. At this temperature, embrittlement poses no problem. But with a rapid drop in coolant temperature, caused by a very common scram or transient, the pressure vessel's insides try to contract. The outside of the six and a half inch vessel is still very hot, and the temperature differential creates enormous tensile stresses.

It was precisely this concern, that led Britain's chief science advisor, metallurgist Sir Alan Cottrell, to recommend against the purchase of 22 Westinghouse reactors for that nation in 1974. The American Physical Society also raised this concern in their report on reactor safety in summer 1975. But APS dismissed the "British" concern, noting that "current steels for reactor pressure vessels are rather insensitive to radiation embrittlement. . .it is estimated that no substantial embrittlement will occur during the assumed 40-year life of the pressure vessel." The APS expressed graver concern for embrittlement in primary coolant system piping, a problem that has already occurred.

Some Possible Scenarios

Pressure vessels are commonly used in coal power plants, but leaks in the vessels give away the location of cracks. In the much thicker vessels, leaks do not occur until it is far too late to prevent disaster. It is not clear exactly what would happen if a pressure vessel were to crack from overcooling, but several things are possible. In a small rupture, it would be impossible to keep the reactor at a high enough pressure to prevent steam from forming inside the vessel. Forcing ECCS water into the core and "squeezing" the steam bubble—as was done at TMI—would not be possible, because higher pressures would force more and more water out of the pressure vessel. The core would then melt its way through the bottom of the pressure vessel, and start down the road to China. A large rupture would merely be worse.

All the pressurized water reactors (PWR) in the nation—44 total—are subject to vessel rupture. The NRC has found evidence of serious embrittlement in 14 reactors, including the three named above, plus Maine Yankee (200-300°F), Palisades (190-220°F), Yankee Rowe (180-210°F), Oconee 1 (160-190°F), Zion 1 (150-180°F), Arkansas 1 (150-180°F), Indian Point 2 (150-180°F), TMI 1 (140-170°F), Rancho Seco (130-160°F), Surry 1 (120-150°F), and Crystal River 3 (110-140°F). Babcock & Wilcox PWRs are particularly prone to overpressurization accidents, but all PWRs are possible candidates.

Very little can be done to forestall or avoid the problem; it is a process of aging. A number of fuel elements can be removed from the outside of the reactor core and operating temperatures can be reduced; this simply slows the rate of embrittlement and substantially reduces the output of the reactor. Thermal "annealing" is the other option; in this process the pressure vessel is heated several hundred degrees above normal operating temperatures with an exterior heat source for extended periods of time. No large vessel has ever been annealed; a small Army reactor was annealed twelve years ago, with modest, but not complete success.

It is one of the most serious problems faced by the nuclear industry, and it has no obvious technical "fix."

Sources: *Science News*, June 20, 1981; *San Diego Union*, July 9, 1981 (sent by Jim Jacobson); American Physical Society report on light water reactor safety (Panofsky, Weisskopf, Bethe, Lewis, *et al.*), summer 1975; letter from Demetrios Basdekas to Morris Udall, April 1981.