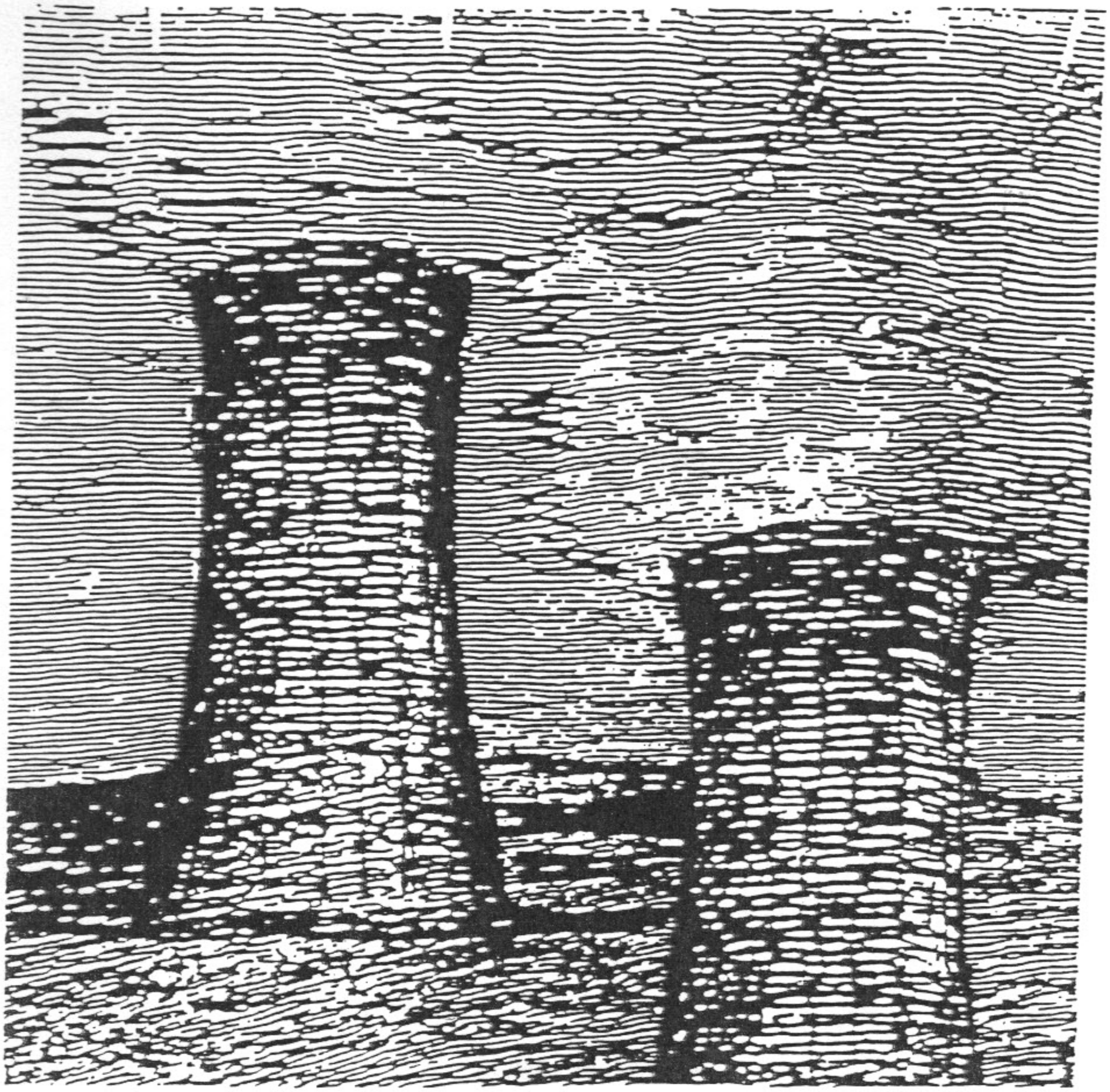
The Aging of Nuclear Power Plants:
A Citizen's Guide to Causes and Effects



Nuclear Information and Resource Service

The Aging of Nuclear Power Plants: A Citizen's Guide to Causes and Effects

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IV. Embrittlement of Reactor Pressure Vessels and Reactor Pressure Vessel Supports in Pressurized Water Reactors

Irradiation embrittlement of the reactor pressure vessels (RPVs) may be the single most important factor in determining the operating life of a PWR. The design of pressure vessels is generally the same for all PWRs. Combustion Engineering (CE) and Babcock and Wilcox (B&W) manufacture their own vessels while Westinghouse either purchases its vessels from CE, B&W, Chicago Bridge and Iron or Rotterdam Dockyard Company. Regardless of the manufacturer, PWR vessels are generally constructed from eight inch thick steel plates, formed and welded to create the vessel structure.

The major age-related mechanism associated with this component is embrittlement. Embrittlement is the loss of ductility, i.e., the ability of the pressure vessel metals to withstand stress without cracking. It is caused by neutron bombardment of the vessel metals and is contingent upon the amount of copper and nickel in the metal and the extent of neutron exposure or fluence. As the metal in the reactor pressure vessel is bombarded with radiation, high-energy atomic particles pass through the steel wall. In doing so, these atoms collide with atoms in the metal and knock them out of position. Over time this results in a loss of ductility.

In an unirradiated vessel the metal loses its ductility at about 40 degrees Fahrenheit. As the vessel becomes embrittled, the temperature at which it loses its ductility rises. This change in the mechanical properties of the metal from ductile to brittle is characterized as the "reference temperature for nil ductility transition" or RTndt. Thus as the reactor ages and the pressure vessel is exposed to more radiation, the RTndt can shift from its original 40 degrees F to as much as 280-290 degrees F or more in extreme cases.[48]

Embrittlement is of even greater concern to those plants constructed prior to 1972. According to thermal shock experts from the Electric Power Research Institute (EPRI), records show that there is copper in the walls of older vessels. Theodore Marston, who works on thermal shock for EPRI, stated that,"(w)e used a lot of auto stock (for the vessel metal), when you melt it you can't get all the wire out." The use of copper was also extensive in the welds of the vessel walls in older reactors. Copper coated wire was routinely used to weld together the large plates which make up the RPV. The NRC's director of safety technology stated that "the copper was used to prevent rust, someone probably got a $10 prize for the suggestion."[49]

The significance of reactor pressure vessel embrittlement and the concomitant shift in RTndt is the increased susceptibility to pressurized thermal shock (PTS). Pressurized thermal shock occurs when the reactor pressure vessel is severely overcooled. RPV technical specifications generally limit cool down to a rate of 100 degrees F. per hour. However, during an overcooling event the vessel may experience a drop in temperature of several hundred degrees per hour. This extreme drop in temperature of the vessel creates thermal stresses through the RPV wall. As the RPV is overcooled, there is a drop in the pressure of the primary coolant loop. This rapid decrease in the pressure of the primary coolant causes the high pressure injection pumps in the emergency core cooling system to automatically inject coolant into the primary loop. As the injection of coolant repressurizes the RPV, the vessel is subjected to pressure stresses. The stresses placed on the reactor pressure vessel by overcooling and repressurization cause pressurized thermal shock.[50]

Pressurized Thermal Shock (PTS) can be initiated by a host of mishaps including: instrumentation and control system malfunctions; small-break loss-of-coolant accidents; main steam line breaks; feed water pipe breaks; and steam generator tube ruptures. Any of these incidents can initiate a PTS event, but as long as the fracture resistance of the reactor pressure vessel remains high, i.e. the RTndt remains low, such transients are not likely to cause the RPV to fail. After the fracture resistance of the RPV is reduced through neutron bombardment, however, severe overcooling accompanied by repressurization could cause flaws in the inner surface of the RPV to propagate into a crack which breaches the vessel wall.[51]

For failure of the reactor pressure vessel to occur several factors must be present: (1) the vessel must have a flaw of sufficient size to propagate; (2) the vessel material must be susceptible to irradiation embrittlement due to copper and nickel content; (3) the vessel must be sufficiently irradiated to cause a decrease in ductility, represented by an increase in the RTndt value; (4) an event must initiate a severe overcooling transient with repressurization; and (5) the resulting crack must be of such a size and location that the RPV's ability to maintain core cooling is affected. This type of failure is beyond the design basis of PWRs: the safety systems, including the emergency core cooling system and the containment, are not designed to withstand cracks in the pressure vessel. Without the reactor pressure vessel surrounding the radioactive fuel, it would be impossible to sufficiently cool the reactor core and a meltdown would ensue.[52]

Pressurized thermal shock is a safety issue for every pressurized water reactor. PTS is of lesser concern for boiling water reactors because radiation embrittlement is not as severe a problem with BWR vessels. This is due to the greater amount of water between the reactor core and the vessel walls in BWRs. The additional water absorbs a greater amount of neutrons so that fewer bombard the walls of the RPV. The walls of a BWR vessel are also thinner than that of a PWR. Therefore, there is less of a temperature differential between the inner and outer walls of the vessel during a cooldown and thus less stress.[53]

While every PWR vessel is susceptible to pressurized thermal shock, those designed by Babcock & Wilcox (B&W) are inherently more susceptible to accidents that can initiate PTS. This is primarily due to the unique design of the B&W steam generators. B&W reactors use once through steam generators, or OTSGs (see Appendix D). OTSGs differ from other PWR steam generators in that the generator tubes are only partially covered with water and contain a smaller volume. This makes the B&W reactor much more sensitive to changes in feed water flow-changes in the flow can cause large rapid changes in the temperature of the reactor. As a consequence, incidents which interrupt feed water flow present more severe challenges to the safety systems than would be experienced in other PWRs. The result is an increased incidence of overcooling events in B&W reactors and an increased probability of pressurized thermal shock.[54]

On December 26, 1985, a severe overcooling event occurred at a B&W facility near Sacremento, California. A loss of power to the "non-safety" integrated control system at the Rancho Seco facility caused a reduction in the main feed water flow to the steam generators. Coolant level in the steam generators decreased, reactor temperature and pressure increased and the reactor scrammed. Feed water valves controlled by the integrated control system could not be operated and remained open. A rapid and severe overcooling event ensued and was exacerbated by the start up of the auxiliary feed water system which sprayed even colder water directly onto the steam generator tubes. The reactor temperature dropped 180 degrees F in 24 minutes, easily violating the technical specification limits of 100 degrees/hour. Additionally, the recommended pressure/temperature limits for pressurized thermal shock were exceeded, although the RTndt limit was not.[55]

If the overcooling event had been more severe or the reactor pressure vessel more embrittled, the RTndt limit may have been reached and the vessel could have ruptured precipitating a meltdown. Equally as disturbing as the accident itself is the fact that the failures and consequences of the event were essentially the same as those previously experienced at Rancho Seco and other plants designed by B&W. In fact, many of the safety problems experienced during the transient were identical to those supposedly resolved by the "short-term" modifications imposed on B&W plants by the NRC in the wake of the Three Mile Island accident.[56]

In May 1979, after the TMI accident, the NRC shut down every B&W

facility, including Rancho Seco. The Commission ordered that procedures and training be implemented to assure that steam generator levels could be maintained if the integrated control system failed. Approximately a month later, the NRC staff concluded that "the licensee has developed adequate procedures and operator training to control AFW (auxiliary feedwater) flow to the steam generators to specific values independent of the ICS, should a failure of the ICS occur, and therefore, is in compliance with this part of the order."[57] However, on December 26, 1985, the staff's conclusions were proven incorrect when operators at Rancho Seco were unable to control the feedwater flow to the steam generators. The NRC's reaction was to conduct a year-long review of problems that were supposedly resolved six and a half years earlier.

The Nuclear Regulatory Commission has vacillated on the issue of pressurized thermal shock for over ten years now. As early as 1977, test samples placed in B&W reactors were indicating that embrittlement was progressing at a faster rate than had been expected. RTndt limits had been originally set at 200 degrees Fahrenheit. However, as these limits were reached in the early to mid 1980s, the NRC began developing new limits within the framework of the PTS rule.

In a briefing to its Advisory Committee on Reactor Safeguards in 1982, the NRC staff considered RTndt limits of 230 and 250 degrees F for longitudinal and circumferential welds respectively. However, by 1985, the NRC sought to amend its regulations on pressurized thermal shock. The proposed amendments would establish an RTndt below which the risk from a PTS event is considered acceptable. These new reference temperatures established limits of 270 degrees F. for plate materials and axial welds and 300 degrees F. for circumferential welds.[58]

The Commission attempted to gloss over the fact that an increase in the RTndt translated into a decreased margin of safety. The NRC press release said the rule constituted "further protection from pressurized thermal shock." At least one expert was not buying the NRC's line. Demetrios Basdekas, an NRC safety engineer and long time critic of the Commission's handling of the PTS issue, opposed the new rule on the grounds that the reference temperatures were unrealistically high.

Dissatisfied with the NRC's handling of the PTS issue, Basdekas made his opinion known in a letter to the New York Times. The letter stated that while,"(t)he Nuclear Regulatory Commission is charged with ensuring that nuclear plants are operated 'with adequate protection' of the public health and safety. . . bureaucratic foot dragging and preoccupation with public relations and financial problems of the industry are contributing to a shortsighted view - that technical problems can wait or do not exist."[59]

Basdekas contended that the new PTS rule was flawed in that it failed to recognize control system failures as a possible initiator of accidents that could challenge the pressure vessel. The NRC was not only failing to acknowledge Basdekas' contentions but plant operating experience as well. On March 20, 1978, the B&W designed Rancho Seco nuclear power plant experienced a PTS event precipitated by a control system failure. While replacing a light bulb in the integrated control system, an operator dropped the bulb into the control panel shorting out the control room instrumentation which eventually led to an overcooling of the reactor accompanied by repressurization of the vessel. The event is believed to represent the most severe and prolonged overcooling event to date with a change in temperature of 300 degrees F. per hour.[60] Basdekas was able to convince the NRC that control system failures were an unresolved safety issue, but the Commission continued to ignore these failures in their calculations on pressurized thermal shock.

In response to the NRC's ambivalence, Basdekas wrote the Commissioners suggesting an independent panel review the PTS issue. The nuclear safety engineer stated that,

> . . .our understanding and treatment of both the systems/process and materials/mechanics aspects of this issue remain wanting. I also believe that the agency and the public would benefit from the opportunity of an independent panel of experts to contribute to your decision making. . . . I might not have accomplished a great deal beyond receiving punishment and intimidation, but I am satisfied that I have stayed away from what appears to be increasingly in vogue within the agency to literally give the store away.[61]

Basdekas further explained the prevailing attitude within the NRC when asked by the Chairman of the House Subcommittee on Energy and the Environment, Rep. Morris K. Udall (D-Ariz.), to comment on NRC responses to the Committee on the topic of pressurized thermal shock. Basdekas stated that:

> A satisfactory resolution, however, cannot be achieved under currently prevailing attitudes within the NRC. On one hand the NRC left it up to the utilities operating the plants chosen for evaluation to provide design and operational information on a voluntary basis, and on a schedule of their convenience, while internally establishing an arbitrary schedule for producing a "resolution" document and withdrawing previously allocated resources while engaging in a variety of prohibited personnel actions and abuse of authority to intimidate and impede if not silence those voicing concern or disagreement.[62]

The NRC adopted the PTS rule in July 1985. In less than six months from the date of its adoption, control system failure had precipitated a severe overcooling event at the Rancho Seco facility (discussed above). Yet the NRC still failed to acknowledge control system failures in their analysis of embrittlement and pressurized thermal shock.

The NRC has continued its research on the PTS issue, focusing on methods to calculate and mitigate embrittlement of reactor vessels. To cope with the most severely embrittled reactors, the NRC has allowed some plants to redesign the configuration of the fuel rods so that fewer neutrons bombard the pressure vessel wall. The NRC has also released for comment a second revision of a regulatory guide (1.99) which specifies how utilities are to calculate the extent of embrittlement and the limits for operating with embrittled pressure vessels. The revision is an improvement in that it takes into consideration the copper and nickel content of the RPV materials. However, a major source of uncertainty still exists due to the limited accuracy and the variable range of the data base (a comparison of embrittlement limits under each revision of regulatory guide 1.99 is provided for each plant in appendix E).[63]

The NRC has attempted to put the PTS issue behind it, but the problem of embrittlement has been recurring like a bad dream. New questions involve the reactor pressure vessel supports. These hold the pressure vessel in place and, depending upon the design, can be exposed to substantial amounts of radiation. There are five major types of RPV supports, four of which are used in PWRs. The major factor in determining embrittlement of the supports is their exposure to reactor core beltline neutron flux. Two types of supports are directly exposed to irradiation from this area of the reactor, the neutron shield tank supports and the column supports. These two types of supports are used in 90% of the operating PWRs in the United States.[64]

The danger of embrittlement of the structural steel supports is the possibility that the neutron bombardment has so irradiated the metal that it cracks under the stress of the combined loads it was designed to bear. In the NRC jargon this is known as catastrophic brittle failure. This type of accident is beyond the design basis for safety systems and could result in total loss of reactor cooling capability. For catastrophic brittle failure to occur three conditions must be present: (1) there must be a flaw of critical size; (2) there must be a sufficient load on the support to create critical stress at the crack tip of the flaw; and (3) the temperature must be low enough to promote a cleavage fracture at the crack.[65]

It appears that the NRC is once again attempting to finesse the issue of embrittlement. In May of 1975 it was discovered that the asymmetric loads placed on reactor pressure vessel supports because of postulated loss-of-coolant-accidents were not taken into consideration in the design of the supports for the reactors at North Anna units 1 and 2. This underestimation of the potential burden on the RPV supports, coined the "North Anna syndrome," prompted the NRC to require all PWRs to reevaluate the loads placed on the structures. It was discovered that the additional load resulting from a double ended rupture of the reactor coolant piping, also known as a guillotine break, was equal to the combined loads the structures were thought to support.[66]

The Commission responded to this issue in April of 1986 by exempting guillotine breaks from consideration. The NRC has stated that:

> . . .the dynamic effects associated with postulated pipe ruptures of primary coolant loop piping in pressurized water reactors may be excluded from design basis when analyses demonstrate the probability of rupturing such piping is extremely low under design basis conditions.[67]

The NRC bases this exemption on the "leak-before-break" theory. In essence the NRC is saying that the additional load placed on the RPV support in the event of a guillotine break need not be taken into consideration because the pipes will leak before they break. However, as previously noted, leak-before-break is neither an "established law", nor should it be the, "sole basis for continued safe operation."

Another factor contributing to the issue of embrittlement of RPV supports is the accelerated shift in RTndt of the support materials. Data from the test reactor at the Department of Energy's Oak Ridge National Laboratory (ORNL) has shown a greater than expected rate of embrittlement for steel that has been exposed to low temperature irradiation. A letter from the Advisory Committee on Reactor Safeguards (ACRS) to Victor Stello, NRC Executive Director for Operations, stated that the RTndt of steel, "irradiated slowly at 120 degrees can rise much more rapidly with exposure to fast neutrons than would be expected from the available experimental work obtained in test reactors."[68]

The ACRS requested that Stello look into the implications of the ORNL findings on embrittlement and the impact on the NRC's plans to extend reactor life past the 40 year license. Stello's response stated that,"(t)he ORNL summary coincides with our evaluation that the neutron shield tanks and support structures do not appear to pose any safety problems." However, close examination of the report reveals that the ORNL did not conclusively state that embrittlement was not a problem. In fact the report found that, "plant specific data are required for an accurate evaluation of the potential for LWR vessel support failure."[69]

The ACRS was understandably "concerned and perplexed" by Mr. Stello's response. Interpretation of the data revealed that structural steel supports are experiencing 2 to 3 times the embrittlement as might have been predicted. However, Mr. Stello failed to draw any inferences from this information. The ACRS stated that they could, "see no reason to be sanguine about the safety of operating nuclear power plants with the largest, heaviest component in the primary system supported on a structure, parts of which are fully brittle. This is unsafe by any type of analysis."[70]

The NRC's final word on embrittlement of RPV supports is still out as the staff seeks further documentation. In the meantime, support reliability has been judged to be adequate. It escapes comprehension how the supports could be found adequate without inspecting them or determining the extent of actual embrittlement.