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A BRIEF ACCOUNT OF THE EFFECT OF OVERCOOLING ACCIDENTS ON THE INTEGRITY OF PWR PRESSURE VESSELS*

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ABSTRACT

The occurrence in recent years of several pressurized water reactor (PWR) accident initiating events that could lead to severe thermal shock to the reactor pressure vessel, and the growing awarness that copper and nickel in the vessel material significantly enhance radiation damage in the vessel, have resulted in a reevaluation of pressure-vessel integrity during postulated overcooling accidents. Analyses indicate that the accidents of concern are those involving both thermal shock and pressure loadings, and that an accident similar to that at Rancho Seco in 1978 could, under some circumstances and at a time late in the normal life of the vessel, result in propagation of preexistent flaws in the vessel wall to the extent that they might completely penetrate the wall. More severe accidents have been postulated that would result in even shorter permissible lifetimes. However, the state-of-the-art fracture-mechanics analysis may contain excessive conservatism, and this possibility is being investigated. Furthermore, there are several remedial measures, such as fuel shuffling, to reduce the damage rate, and vessel annealing, to restore favorable material properties, that may be practical and used if necessary.

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INTRODUCTION

The occurrence in recent years of several pressurized water reactor (PWR) accident initiating events that could lead to severe thermal shock to the reactor pressure vessel,[1,2,3] and the growing awareness that copper and nickel in the vessel material significantly enhance radiation damage in the vessel,[4,5] have resulted in recent reevaluation of pressure-vessel integrity during postulated overcooling accidents. Analyses[6] performed following the 1978 Rancho Seco accident[1] indicated that under some circumstances transients of this type might result in failure of the vessel, if the transients occurred late in the normal life of the pressure vessel. Thus, the subject of PWR pressure vessel integrity during overcooling accidents (OCA's) deserved additional attention.

In March of 1981 the U.S. Nuclear Regulatory Commission informed the nuclear utilities of their concerns, and by March 1982 the NRC had officially declared the problem, referred to as pressurized thermal shock (PTS), an unresolved safety issue.

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PWR accidents of particular concern are those that allow cool water to come in contact with the inner surface of the pressure vessel at a time when the primary-system pressure is substantial. The rapid cooling of the inner surface results in high thermal stresses and a reduction in the fracture toughness near the inner surface. This introduces the possibility of propagation of preexistent inner-surface flaws (sharp, crack-like defects), and this possibility increases with reactor operating time because of an additional reduction in fracture toughness that is the result of neutron exposure. Because of this dependence on neutron exposure, the portion of the vessel directly across from the reactor core, where the neutron flux in the vessel wall is a maximum, is of greatest concern.

Thermal shock by itself presumably cannot drive a flaw all the way through the wall; however, if the primary-system pressure were substantial, a potential for vessel failure could exist; that is, a preexistent flaw, under proper circumstances, could penetrate the vessel wall and provide a large enough opening to prevent flooding of the reactor core.

A complete evaluation of the OCA problem in terms of its threat to pressure vessel integrity requires consideration of a number of factors, including postulated accident initiating events, reactor system and operator response to these events, specific design features of the reactor vessel and core that affect fluence-rate and coolant-temperature distributions adjacent to the inner surface of the vessel wall, sensitivity of the vessel material to radiation damage, size and orientation of preexistent flaws, and remedial measures. This paper examines primarily the fracture-mechanics-related conditions that could lead to a potential for vessel failure.

MECHANISMS AND ACCIDENTS RESULTING IN PRESSURIZED-THERMAL-SHOCK SITUATIONS

There are two basic mechanisms for creating thermal shock to the PWR pressure vessel.[7] One is the injection of cold fluid, and the other is depressurization of the primary or secondary system which reduces the saturation temperature. Cold water can be injected into the primary circuit by means of the emergency core cooling systems and into the secondary circuit by means of the feed-water systems, while depressurization can be the result of stuck-open valves, excessive power demands and pipeline failures. In either of these cases cool water eventually enters the vessel through a main coolant line, as indicated in Fig. 1 and passes down through the downcomer region, coming in contact with the inner surface of the vessel wall on the way to cooling the core.

Large decreases in pressure do not necessarily accompany a reduction in coolant temperature, and for those cases where they would, the primary system could be repressurized with the emergency core cooling system. Thus, it is possible for thermal-shock effects and high pressure to coexist.

Four classes of postulated PWR accidents that can result in pressurized-thermal-shock situations are the large-break loss-of-coolant accident (LBLOCA), the small-break loss of coolant accident (SBLOCA), a main steam-line break (MSLB) and a runaway feed-water transient (RFT).[7] The 1978 Three Mile Island accident[3] is an example of an SBLOCA (stuck-open primary-system relief valve), and the 1978 Rancho Seco accident is an example of an RFT. In the case of an LBLOCA the primary system pressure would drop very quickly to nearly one atmosphere and would remain there; thus, vessel integrity would be retained to the extent of being able to maintain coolant in the vessel.

THE TENDENCY FOR INNER-SURFACE FLAWS TO PROPAGATE DURING THERMAL-SHOCK LOADING ONLY

The tendency for inner-surface flaws to propagate as a result of thermal-shock loading is illustrated in Fig. 2, which shows the temperature, resultant thermal stress, and fracture toughness distributions through the wall of the vessel at a particular time during an LBLOCA. Also included for the same time in the transient are the stress intensity factors (K_I) for long axial flaws of different depths and the radial distribution of the fast neutron fluence. As indicated, the positive gradient in temperature and the steep attenuation of the fluence result in positive gradients in the crack initiation toughness (K_{Ic}) and the crack arrest toughness (K_{Ia}), and these positive gradients tend to limit crack propagation. However, K_I also increases with crack depth, except near the back surface, and for the particular case and time analyzed, $K_I \geq K_{Ic}$ for a broad range of crack depths. It is evident that both shallow and deep flaws can initiate, but the positive gradient in toughness provides a mechanism for crack arrest.

If the crack depths corresponding to the initiation and arrest events are plotted as a function of the times in the transient at which the events take place, a set of curves referred to as the critical-crack-depth curves is obtained that indicates the behavior of the flaw during the entire transient. A typical set of critical-crack-depth curves for a LBLOCA is shown in Fig. 3. As indicated by the dashed lines the long axial flaw would propagate in a series of initiation-arrest events and, if warm prestressing were not effective, would penetrate deep into the wall.

Warm prestressing, as referred to above, is a term used to describe a situation where K_I is decreasing with time when K_I becomes equal to K_{Ic} by virtue of a decrease in temperature. It has been postulated[8] and demonstrated experimentally[8,9] that under these conditions a flaw will not propagate; that is, a flaw will not initiate while K_I is decreasing. In Fig. 3 the WPS curve is the locus of points for $K_I = (K_I)_{max}$ ($dK_I/dt = 0$). To the left of the WPS curve $dK_I/dt > 0$ and thus crack initiation can take place, but to the right of the WPS curve $dK_I/dt < 0$, and crack initiation will not take place. For the particular case illustrated in Fig. 3, WPS limits crack propagation to $\sim$40% of the wall thickness.

Even if WPS were not effective, the flaw could not completely penetrate the wall under thermal-shock loading conditions only. This is a result of the substantial decrease in K_I as the crack tip approaches the outer surface (see Fig. 2) and has been demonstrated recently in a thermal-shock experiment.[10] However, when pressure is applied in addition to the thermal loading, the possibility of vessel failure (complete penetration of the wall) exists for some assumed conditions.

FRACTURE MECHANICS CALCULATIONAL MODEL

Linear elastic fracture mechanics (LEFM)[11] has been used thus far to analyze the behavior of a flaw during the postulated overcooling accidents. The initial flaw was assumed to be quite long on the vessel surface, to be oriented in an axial direction and to extend radially through the cladding into the base material. Fracture toughness data (K_{Ic} and

K_{Ia} vs T — RTNDT, where T is the temperature and RTNDT is the reference nil ductility temperature) were taken from ASME Section XI,[12] and the reduction in toughness due to radiation damage was taken from Reg. Guide 1.99, Rev. 1.[13] Since techniques for evaluating the behavior of a flaw when temperatures at the crack tip correspond to and exceed upper shelf (ductile) conditions, it was assumed that crack arrest would not occur if K_I were above an arbitrary upper-shelf toughness value of 220 MPa $m^{1/2}$.

This particular calculational model is believed to be conservative for several reasons: (1) long axial flaws have a greater potential than other flaws for penetrating deep; (2) the cladding may prevent short flaws from extending on the surface to become long flaws; (3) the probability of a long flaw existing as an initial flaw and of any flaw extending through the cladding presumably is very small; (4) Reg. Guide 1.99 does not account for variations in the concentration of nickel, and more recent data indicate that vessels with low concentrations of nickel will suffer less damage than presently assumed; and (5) for some of the milder postulated overcooling accidents crack arrest presumably will take place on the upper-shelf portion of the toughness curve, preventing failure of the vessel. Each of these areas is under investigation to determine the degree of conservatism involved.

LOCATION OF SENSITIVE REGIONS OF THE VESSEL WALL

The fracture toughness of the reactor vessel at any point in space and time is a function of the initiation toughness of the material and the reduction in toughness due to radiation damage. Since radiation damage to the vessel wall is a function of both fast neutron fluence and copper and nickel concentrations it is necessary to locate the areas of the vessel wall where the worst combination of initial toughness, fluence, and copper and nickel concentrations exists. The reduction in fracture toughness due to radiation damage is relatively small if the copper and/or nickel concentrations are low and is large if the concentrations of both are high.

Copper is an impurity in the vessel material, and high concentrations generally are found only in the welds that join the segments of the vessel wall. Nickel is an alloying element in both the base and weld materials, and its concentration covers a broad range among the PWR vessels and is not necessarily the same in the base and weld metals of any particular vessel. Since high concentrations of copper tend to be confined to the welds (there are exceptions), the welds take on special significance.

As shown in Fig. 4, vessels fabricated from sections of plate have both axial (longitudinal) and circumferential welds, while vessels fabricated from forging rings have only circumferential welds opposite the reactor core. In a plate-type vessel the axial welds tend to be of greater concern than the circumferential welds because, as mentioned above, axially oriented flaws have a greater potential for penetrating deep into the vessel wall during an overcooling accident. However, forged vessels, which have only circumferential welds, are not free from the adverse effects of the same accidents.

The fast neutron fluence in the vessel wall is a maximum at an elevation corresponding to about the horizontal midplane of the reactor core, and, as shown in Fig. 4, there is an azimuthal variation in fluence, resulting from the noncircular geometry of the periphery of the core. Thus, the maximum concentrations of copper and nickel, the maximum fluence and the minimum value of initial toughness do not necessarily coincide. For an accurate specific-plant analysis of the radiation-induced reduction in fracture toughness these space variations must be considered.

ANALYSIS OF THE 1978 RANCHO SECO AND SEVERAL POSTULATED OVERCOOLING ACCIDENTS

Using the fracture-mechanics model described above, pressure-vessel-integrity studies were performed for the 1978 Rancho Seco accident and several postulated accidents,[7] including an LBLOCA, an MSLB, and an RFT that consisted of a turbine trip followed by stuck-open bypass valves. Two different sets of assumptions regarding plant and operator responses to the initiating event were used for the MSLB in an effort to examine and illustrate extremes in the calculated severity of the accidents. For each of the transients considered a typically high copper concentration of 0.31% was assumed to exist in an axially oriented weld, and the initial value of RTNDT was assumed to be −7°C. For the purpose of computing a time in the life of the vessel at which a potential for vessel failure might exist a typical fluence rate of 0.05 x 10^{19} neutrons/cm^2·yr was assumed for the inner surface of the vessel at the location of the axial weld.

The LBLOCA constitutes the most severe thermal shock of any of the overcooling accidents, and for this accident the thermal shock is the result of injection of emergency core coolant. The primary-system pressure remains very low, and because of this the calculations indicate that the flaw will not penetrate the wall. Furthermore, WPS would tend to prevent crack propagation beyond midwall thickness. But even without WPS the crack would not penetrate the wall because of the lack of pressure to maintain a high stress intensity factor for very deep flaws.

The Rancho Seco accident[6] involved a loss and then a sudden reapplication of feedwater to a steam generator, resulting in rapid cooling of the primary system while the primary-system pressure was being maintained close to the normal operating level, as shown in Fig. 5. The temperature of the coolant in the cold leg dropped ∿180°C in one hour.

During the specific Rancho Seco transient, the requisite conditions for WPS existed ($dK_I/dt < 0$), and as a result the analysis discussed herein does not predict a reduction in the normal vessel design lifetime [∿32 effective full-power years (EFPY)]. However, for an otherwise similar accident the requisite conditions for WPS might not exist as a result of variations in pressure. In this case the calculations indicate a potential for vessel failure at ∿17 EFPY.

The postulated turbine-trip accident tends to represent a worst-case situation and may in fact represent an unrealistic set of circumstances insofar as system response is concerned. As indicated in Fig. 5 the transient is much more severe than that for Rancho Seco in that the primary-system temperature drops more rapidly and to a lower value, while repressurization by the high-pressure injection system provides a pressure equal to the safety-valve setting. The result of this postulated and possibly exaggerated worst-case accident is a potential for failure of the reactor vessel at about 4 EFPY, with or without WPS.

The two MSLB cases analyzed presumably represent two extremes for the postulated MSLB accident: an exaggerated worst case in the same sense that the turbine-trip case was, and a least-severe situation based on preferred and most-likely system and operator responses. For this latter case a potential for vessel failure is not predicted witnin the normal lifetime of the power plant. However, for the postulated worst-case situation the calculations indicate a potential for failure at 8 EFPY's, taking advantage of WPS, and 4 EFPY's without WPS.

REMEDIAL MEASURES

Aside from ongoing investigations that may result in the discovery and removal of excessive conservatism in the state-of-the-art fracture-mechanics model, there are several fracture-mechanics-related remedial actions that might be taken to ensure longer permissible vessel lifetimes. They include reducing the uncertainty in the estimation of the fluence in the vessel wall, redesign of the fuel loading to reduce the fluence rate in the wall, and annealing the vessel to restore the original high fracture-toughness values. There is a substantial effort under way to improve the accuracy of fluence determinations,[14] and a few PWR's in the U.S., West Germany and Finland have undergone fuel-loading changes to reduce the fluence in the vessel wall. However, the redesign of a fuel loading is a very complex task, involving a possible reduction in power to accommodate less heat transfer area and/or a less favorable power distribution, changes in control-rod worth and perhaps a shorter fuel-cycle time. Furthermore, if a vessel has alrady suffered severe radiation damage there is not much to be gained by reducing the fluence rate. For this situation, annealing the vessel may be the answer, but this too is a complex and expensive operation and has not yet been demonstrated on a full-sized PWR vessel.[15]

SUMMARY

A state-of-the-art fracture-mechanics analysis of the 1978 Rancho Seco accident, assuming a typically high concentration of copper to exist in the vessel material, indicates that because of WPS the specific transient would not result in a potential for vessel failure during the normal life of the vessel. However, the requisite conditions for WPS might not exist in an otherwise similar transient. In this case the analysis indicates that a potential for vessel failure would exist at ~17 EFPY. Depending upon the assumptions made regarding reactor system and operator response to the initiating event, potentially more severe transients can result in either no premature failure or failure at times perhaps as short as 4 EFPY.

It appears that the state-of-the-art fracture-mechanics model used for these predictions contains substantial conservatism, and programs are under way to explore this possibility and also to reduce uncertainties in fracture-mechanics-related areas, such as dosimetry, which provide important input to the pressure vessel integrity studies.

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Figure Captions

Fig. 1. Schematic Cross Section of a PWR Pressure Vessel Indicating the Flow Path of the Coolant that can Contribute to Thermal Shock of the Vessel.

Fig. 2. Mechanisms for Crack Initiation and Arrest at a Specific Time During a PWR-LBLOCA.

Fig. 3. Critical-Crack-Depth Curves for a PWR-LBLOCA, Assuming a Long Axial, High Copper and 32 EFPY.

Fig. 4. Peak Fluence, Maximum Concentrations of Cu and Ni, and Maximum $RTNDT_0$ do not Necessarily Coincide Because of Azimuthal Variations in Fluence and Material Properties.

Fig. 5. The 1978 Rancho Seco and a Postulated RFT Downcomer Coolant Temperature and Pressure Transients.

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THERMAL SHOCK IS RESULT OF ACCIDENTS THAT ALLOW COOL WATER TO CONTACT INNER SURFACE OF VESSEL

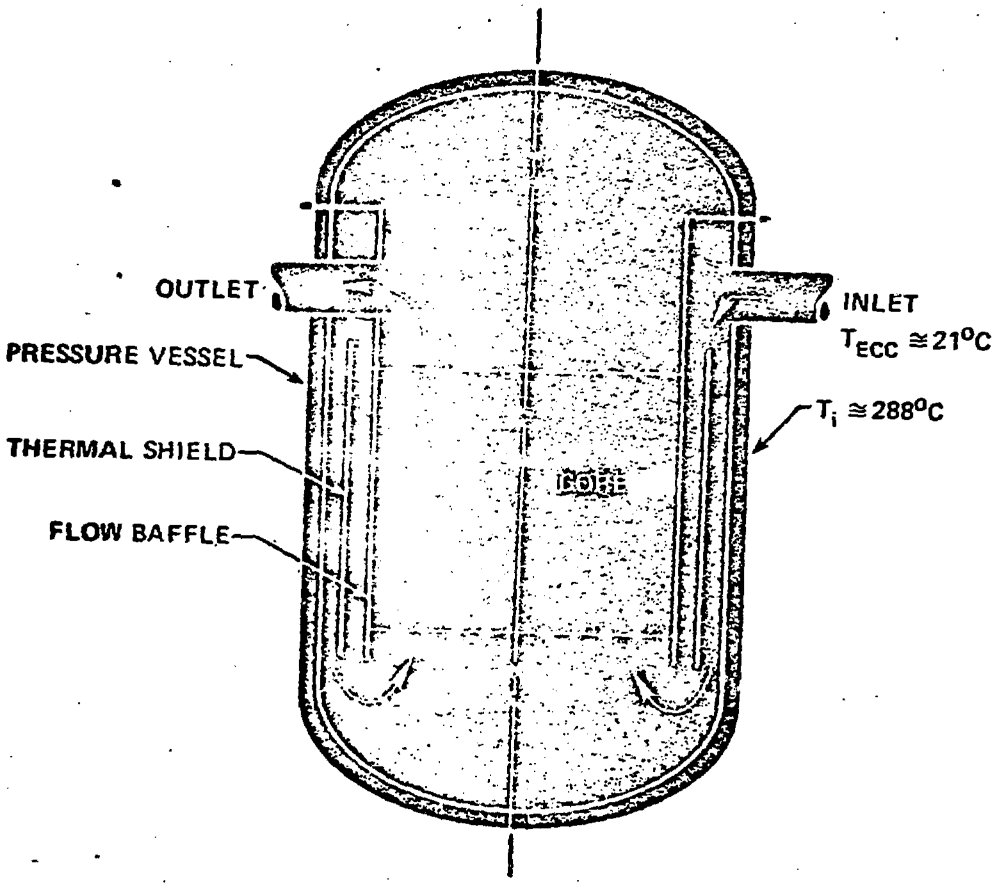

FIG. 1

ORNL-DWG 79-4353A ETD

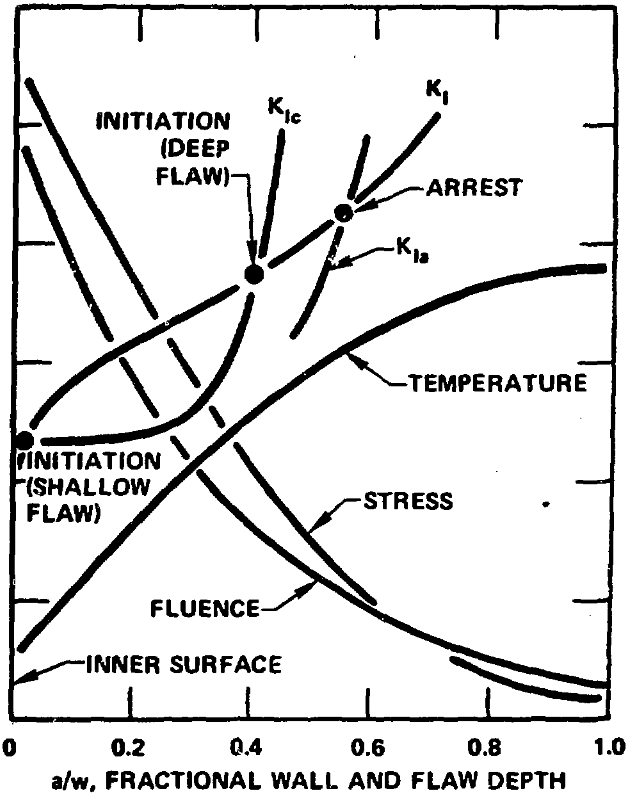

Fig. 2. Mechanisms for crack initiation and arrest at a specific time during ~~PWR-LOCA~~. PWR-LBLOCA

later than those indicated by the WPS curve will not take place because K_I will be decreasing with time. For the particular case illustrated in Fig. 1.2, WPS limits crack penetration to ~33% of the wall.

Although the conditions for WPS exist for the PWR-LOCA, they will not exist for all transients of interest, particularly those involving repressurization. Each case must be examined to determine if a maximum occurs in the K_I vs time curve at the appropriate time in the transient to limit crack propagation.

To be able to analyze numerous temperature-pressure-transient scenarios economically and expeditiously, a computer code (OCA-I) was written to perform the thermal analysis, stress analysis, and fracture-mechanics analysis for a specific set of conditions, using techniques that tend to minimize computer time and input. The present version of OCA-I calculates K_I values for a long (two-dimensional) axial flaw on the inner surface of a 172-in.-ID × 189-in.-OD (4.37-m × 4.80-m) cylinder subjected to internal pressure and cooling of the inner surface [the cylinder dimensions are typical for a 1000-MW(e) PWR system].

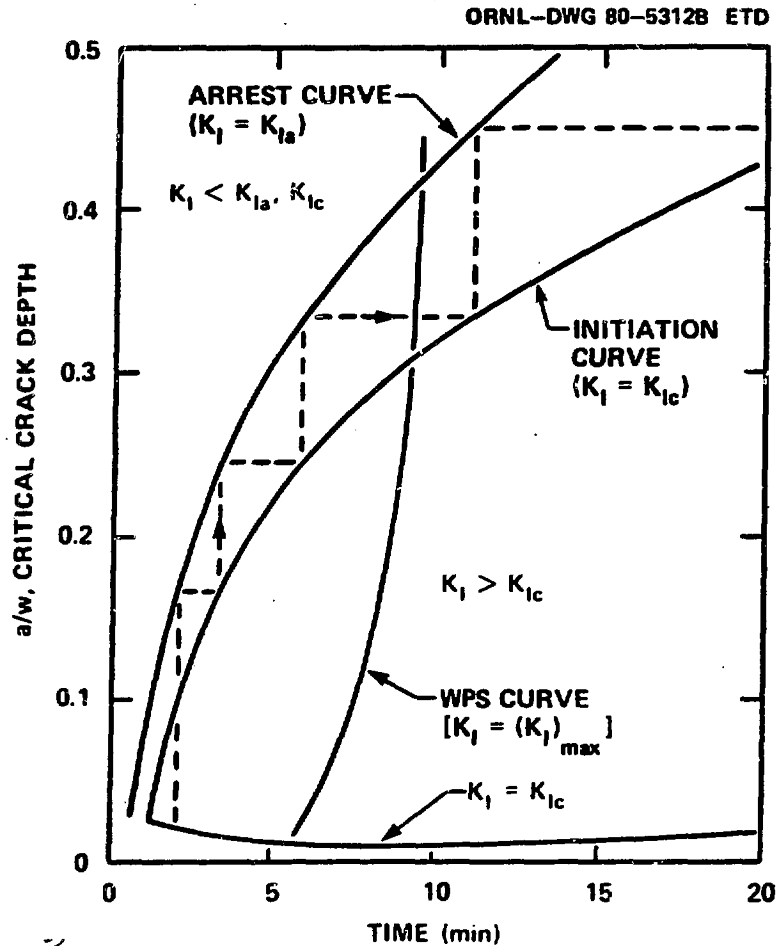

Fig. ~~1~~ 2. Critical-crack-depth curves for PWR-LOCA, assuming long axial flaw, high copper, and ~~40-year service~~.
32 EFPY

Long flaws are not likely to exist as initial flaws, but long flaws are much more amenable to accurate analysis; furthermore, during thermal-shock loading, short flaws have a tendency to grow in length to become long flaws.[4] If a short flaw does not become a long flaw, some conservatism exists in the analysis. Some conservatism is present even if the flaw is initially long, or eventually becomes long, because the two-dimensional analysis does not account for the restraint of the vessel heads or the axial gradient in material toughness, both of which tend to reduce K_I/K_{Ic}.

The OCA-I code includes all information necessary for calculating the behavior of a long axial flaw, including all material properties and the effects of fast-neutron fluence and impurities (copper and phosphorus) on radiation damage (reduction in toughness). Nearly all that is required

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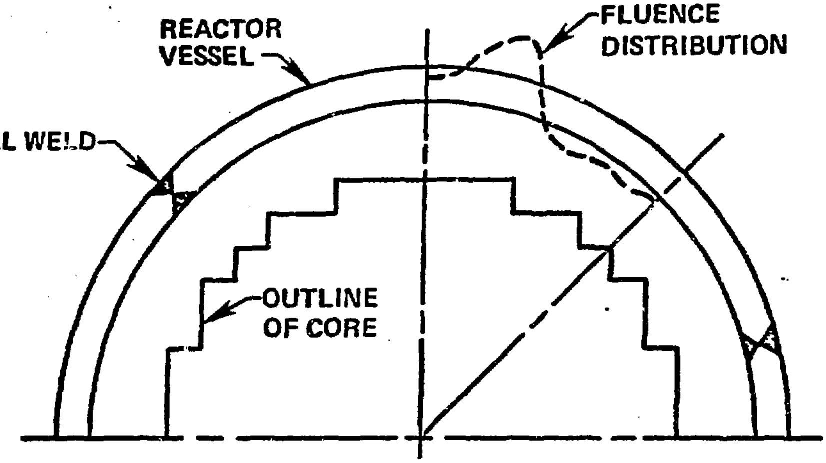
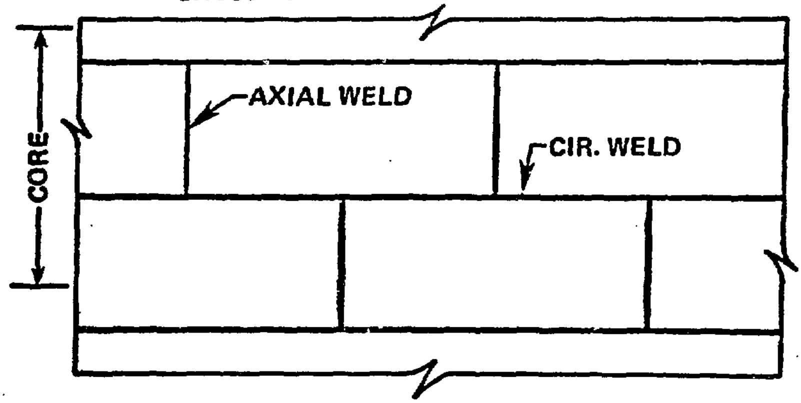
AZIMUTHAL VARIATIONS IN FLUENCE, RT_{NDT_0}, Cu AND Ni EXIST AND MUST BE CONSIDERED



CROSS SECTION OF RPV THROUGH CORE



DEVELOPED VIEW OF BELTLINE REGION OF RPV

Fig. 4

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1978 RANCHO SECO ACCIDENT NOT AS SEVERE AS SOME OTHERS ANALYZED

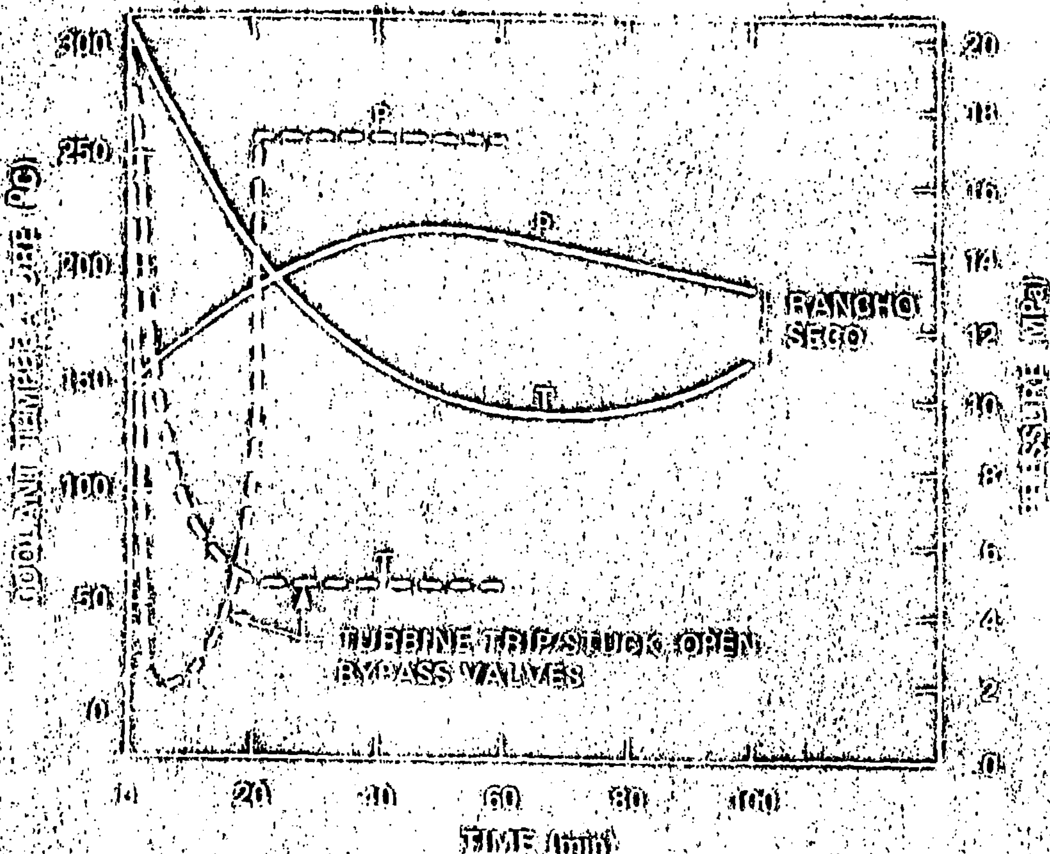

Fig 5