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August 10, 1982

Mr. El Igne
Staff Engineer
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear El:

Attached is an edited version of Bender's draft on pressurized thermal shock. Most of my comments are editorial, but there are some substantive changes proposed.

Of course, the draft needs editing badly, but that is not unusual at this stage. I have tried not to hold back too much in this respect. I have corrected spelling, improved wording (I hope), etc.

I have also made two additions. One is the addition of a couple of suggested items of supporting work on page 19. The other is an attempt to redress the absence of comment on thermal-hydraulics in the text. I have added a proposed paragraph under Safety Margins Affecting the Thermal Shock Issue. I have put this paragraph first in this section, so as to avoid breaking into the discussion of margins available in the fracture mechanics. I am confident that Catton and Theophanous will add sections on thermal-hydraulics, and in that event could easily be convinced that my contribution is less valuable and perhaps not needed.

Regards,

Herb

Herbert Kouts
Chairman

Attachment

HK:ns

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6/30/82
DRAFT 3
MB:mbw

PRESSURIZED THERMAL SHOCK (PTS) IN NUCLEAR POWER PLANT REACTOR VESSELS

Prepared for the Advisory Committee on Reactor Safeguards
by Myer Bender, ACRS Member

Since late 1965, when the ACRS reported to AEC Chairman Seaborg on the need for attention to the methodology for assuring the integrity of nuclear reactor vessels, there has been a concerted effort to improve technological understanding about matters pertaining to reactor vessel integrity. Over the past few years, a great deal of attention has been directed to the question of thermal-shock-induced damage to reactor vessels and whether some older vessels might be subject to catastrophic failure from a combination of circumstances that have either happened or can be postulated to happen in U. S. licensed nuclear power plants.

In a letter from NRC Chairman N. J. Palladino to ACRS Chairman Paul Shewmon dated March 25, 1982, the Chairman stated the following:

> "I would appreciate an ACRS critique of the staff's program on pressurized thermal shock. I am particularly interested in obtaining your views on short-term steps that should be taken by the NRC to lessen the chances of a severe problem occurring because of pressurization following thermal shock to a pressure vessel. Your critique would appear most valuable if it could be done prior to publication of the NRC plan. In that way, the staff will have the advantage of knowing your views before their plan is made final."

In response to this request, the ACRS organized a working group consisting of ACRS members, ACRS Consultants, ACRS Fellows, and Staff Members to review the thermal shock issues and assist the ACRS in formulating a response to Chairman Palladino's request. This report summarizes the information developed by the Working Group.[1]

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[1]Working Group members were: ACRS Members: M. Bender, Working Group Chairman, Paul Shewmon, Robert Axtman, Harold Etherington, David Ward; ACRS Consultants: Frank Binford, Ivan Catton, Herbert Kouts, W. R. Gall, George Irwin, T. G. Theofanous, M. S. Wechsler, Z. Zudans, E. Abbot (Assistant to Commissioner Gilinsky and former ACRS Senior Fellow), ACRS Fellow William Bock and ACRS Staff Engineer E. Igne.

RELIABILITY CONSIDERATIONS IN NUCLEAR REACTOR VESSELS

The ACRS, in its 1974 report on Reactor Vessel Integrity, concluded that for nuclear vessels built to the ASME Boiler and Pressure Vessel Code, Section III, and inspected in accord with Section XI, the probability of "disruptive failure" was less than 10^{-6} per reactor year with the probability that such failures would exceed the capability of engineered safety features being "lower." This conclusion was derived from a review of engineering and manufacturing practices being applied to the population of vessels existing and planned at that time with the prescribed operating controls required under the AEC Operating License regulations then in force.

The important factors entering this evaluation were:

(a) The materials used in vessel plates, forgings, and welding were well-known with respect to their strength and toughness properties and were being used in good practices well-defined by the ASME Code.

(b) Radiation damage effects on fracture toughness are well enough understood from experimental and theoretical investigations to enable their effects to be evaluated conservatively as they change with cumulative in-service fast neutron fluence (1 mev or greater).

(c) Inspection practices could detect significant flaws and assure control or correction before they could propogate to disruptive proportions based on what was then known about the science of fracture mechanics.

(d) Operating temperatures of reactor vessels could be maintained high enough to assure that the "upper shelf" toughness of vessel materials would be maintained under all operating conditions where fracture stress might be of concern.

(e) Recovery of fracture toughness could be attained for vessels, whose integrity might be questionable because of fast neutron radiation effects, by annealing at elevated temperatures (about 750°).

In its 1974 report, the ACRS made a number of recommendations for improvements in design, inspection and operating requirements for reactor vessels. Many of these have been implemented for newer vessels, (especially chemistry and design methods) and inspection practices have been upgraded for all vessels. Thus, the premises on which the 1974 conclusion were based still generally apply.

THE PRESSURIZED THERMAL SHOCK (PTS) ISSUE

Reactor vessels exhibit a characteristic capacity to absorb energy of deformation that increases with operating temperature. This property, called "fracture toughness" is of particular importance where stress conditions ~~exceed~~ the elastic limit of ~~the~~ material as is the case when stresses are concentrated at the tip of a crack-like flaw. When the stress ~~conditions~~ at the tip of a crack-like flaw exceed the fracture toughness capacity of the material, the crack will grow. Unless the energy released by the cracking process is absorbed in the ~~structural~~ material, the fracture can propogate through the vessel wall, disruptively and perhaps catastrophically. Reactor vessel materials have a temperature-related fracture toughness that characteristically displays a lower plateau at reduced temperature ("lower shelf") and a higher value at elevated temperature ("upper shelf"). Vessel integrity claims generally depend upon maintaining the vessel temperature at its "upper shelf" temperature even though some toughness remains at lower temperatures. Fracture would only occur under postulated loadings if flaws were located and sized such that ~~high~~ fracture stresses were induced in a crack-like flaw. Hence, if significant crack-like flaws can exist in a vessel that would lead to high localized stresses at the crack top. It is prudent to maintain vessel temperatures above the "upper shelf" temperature to assure maximum fracture resistance. This consideration alone justifies requirements for holding the vessel at "upper shelf" temperatures.

Thermal stress ~~effects~~ in the wall of a vessel are caused by thermal movement due to differences in temperature from one point in a vessel to another. Many types of temperature distributions ~~can exist in a vessel wall but~~ the ~~one~~ of particular concern for PTS is the difference in temperature from inner to outer surface caused by cooling or heating conditions ~~related~~ ~~to~~ reactor coolant temperature. This temperature ~~difference~~ causes tensile stresses at the cold surface and compressive stresses at the hot surface. A neutral thermal stress ~~point~~ thus exists in the vessel wall whose location will depend upon the shape of the temperature distribution. For rapid cooling or heating, the distribution through the wall will be non-linear. Peak stresses can be ~~controlled~~ by ~~fixing~~ the rate at which the heat is being removed or added while taking account of thermal diffusivity of the vessel material. In order to control thermal stresses, a limit is thus imposed on wall cooling and heating rates.

During startup of a plant, these thermal stresses are limited by controlling the rate of coolant temperature rise (nominally 100°F per hr) and a similar limit is assigned for cooldown. The requirement can change depending upon pressure stress conditions since pressure and temperature loadings are superimposed. It should be noted that the thermal strains change direction from heat up to cooldown. When cooling from an elevated temperature, the inner vessel surface is in tension but when the vessel is heated, the outer surface feels the tensile effects.

In recent years, several transients have been observed in which the reactor coolant was abruptly ~~cooled~~ and the low temperature coolant rapidly cooled the vessel wall. The most noteworthy* example was the Rancho Seco event in 1978 when operating circumstances caused the operator to permit cold feedwater to flow at full capacity to the steam generator of a shutdown nuclear reactor system, driving the primary coolant temperature toward feedwater temperature conditions and thus sharply cooling the vessel wall. During the transient, the system was pressurized by injection of high pressure coolant and the combined stress might have exceeded the limits at which the vessel's toughness capacity would have prevented disruptive fracture if:

(a) The vessel had accumulated sufficient fast neutron fluence to have become significantly less ductile than its initial condition.

(b) The combined stresses were high enough to cause a crack-like flaw to initiate growth.

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*The 1980 Crystal River event might have been comparable or more severe but has not been fully analyzed.

(c) A crack-like flaw of ~~sufficient~~ly large abrupt size to ~~feel a~~ cause sufficiently high localized stress concentration existed at the part of the vessel where conditions(a) and (b) were present.

(d) The Temperature of the vessel had been driven below the "upper shelf" temperature limits.

(e) The crack location and orientation were such that its behavior due to vessel deformation would lead to substantial growth in a linear elastic fracture mode from the imposed load.

The Rancho Seco vessel had not accumulated sufficient fast neutron fluence for fracture to have occurred even if all other conditions were satisfied. Even if that were not the case, the combination of probabilities to satisfy all five conditions above has led many experts to challenge the likelihood that PTS is a matter of serious safety concern if prudent operating control is exercised.

The intent of this report is to examine the various aspects of the issue that have a bearing on the probabilistic arguments in order to make a judgement as to whether the NRC Staff plans for the short- and long-term treatment of the thermal shock issue are adequate to protect the health and safety of the public.

FRACTURE MECHANICS ISSUES

Fracture mechanics, the science of materials fracture behavior, is the basis for evaluating thermal-shock-induced fracture propensity of nuclear reactor vessels. The NRC-sponsored HSST program has been exploring this technology by experimental and analytical activities since the mid-1960's with cooperation from the nuclear industry and its supplier sources. The effort has shown that vessels can fail catastrophically by brittle fracture if loads are imposed on flawed vessels under conditions of low fracture toughness (near the "lower shelf" toughness temperature). It has shown that vessels will respond in a ductile manner when similar conditions are imposed with the vessel at "upper shelf" temperatures, and will exhibit "leak before break" behavior to warn of trouble in time to take ameliorative corrective action.

The properties characterizing fracture toughness, K_{Ic} and K_{Ia} are measured for materials[2] specimens, ~~to~~ determine ~~their~~ response to impact loadings to initiate crack growth ~~or~~ crack arrest. The values are compared with a computed value of K_I to determine whether a crack-like flaw will initiate and arrest growth under specified combinations of pressure and thermally induced straining loads. For evaluation purposes, a preexisting flaw of known geometry is hypothesized to exist and its structural behavior analyzed using closed form analytical solutions or finite element computerized analysis. When analysis shows K_I will remain above the K_{Ic} value, the flaw is assumed to grow until vessel rupture occurs ~~and~~ to stop if K_I falls below K_{Ia}.

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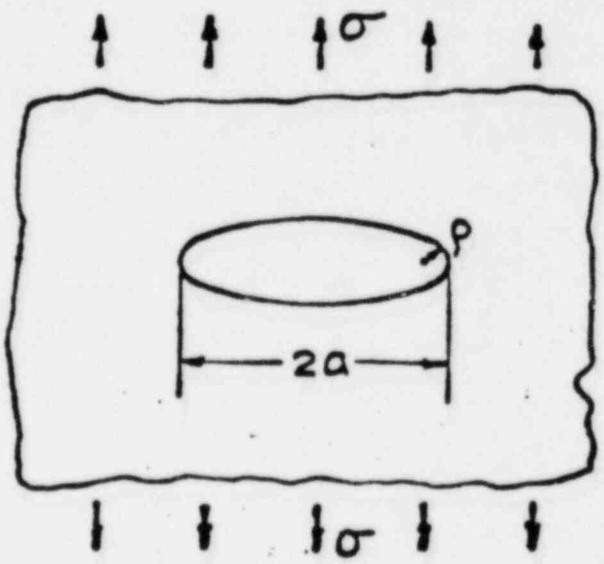
[2]A brief definiton of how K_I is computed and K_{Ic} and K_{Ia} applied can be seen by using the accompanying figure to describe a



uniaxial stress applied remotely to an elliptically shaped hole having sharp crack-like edges. The crack intensity at its edge is:

$$K_I = \sigma\sqrt{\pi a}$$

Crack extension will initiate when the stress level is such that K_I exceeds a measured value of K_I which has been observed by experiment to initiate crack extension in a specific type of material. This property is designated as K_{Ic}. The extension will cease if K_I progresses into a region where crack arrest occurs. The material can have a similarly measured value of this "arrest" property called K_{Ia}. Thus, cracking will initiate if K_I exceeds K_{Ic} and will stop its extension when K_I becomes less than K_{Ia}.

Fracture toughness, as noted previously, depends upon the service temperature of reactor vessel materials. Thus, the temperature distribution that exists in a vessel will affect its toughness. A vessel wall subjected to rapid cooling or heating will have varying toughness through the wall. When a rapidly cooled inner surface (PTS precondition) occurs, the ~~inner wall~~ toughness will increase as the temperature increases through the wall.

An extended cooling period is required for the entire wall to reach the lowest toughness value. A period of time is therefore needed to bring about temperature-related toughness reduction through the vessel wall, the length of time being a function of the cooling rate. Determining the rate at which the vessel wall will be cooled during PTS transients is therefore an important consideration. The rate can be determined analytically by conservative computational procedures or determined more precisely by a combination of experiment and analysis. Experimental work sponsored by the nuclear industry is being used to support arguments for safety adequacy with respect to some PTS transients.

Every vessel section has an initial K_{Ic} ~~value~~ that may be determined by testing representative samples of the material using fabrication techniques similar to those in the vessel. Older vessels do not always have record tests or specimens for such determinations. Archival materials have been used to fabricate welded specimens using techniques believed to be equivalent to those in the original vessel as a way of establishing a basis for judging initial toughness. There are some uncertainties associated with such post-use investigations but they are a common practice in many failure investigations. In this instance, the purpose is to recreate conditions such that the specimens have materials properties associated with chemistry and manufacturing methods equivalent to those in the existing vessels. Once the specimens are fabricated, they can then be exposed to radiation conditions to determine change in toughness with fluence (an effect assumed to be linearly proportional to time at constant fast neutron flux).

The chemistry of the vessel material is a major factor in determining the change of toughness with cumulative fluence. Copper and phospohorus are both contributors but copper is the principal variable. Earlier vessels were fabricated using welding rod coated with copper and tend to have higher residual copper content (0.1 to 0.35%) in the welds than later ones. Nickel is also thought by some experts[3] in the European community to have an effect and this must be accounted for in a conservative safety evaluation. The NRC materials engineering staff is correlating this information and developing a basis for judging initial K_{Ic} and its change with fluence for each of the vessels under investigation. Several correlations are empirically derived; the most conservative of these brackets nearly all of the measured specimens for which information is available. These empirical correlations are presently related to a reference fracture toughness (K_{Ic}) of 200 ksi $\sqrt{in}$ by specifying a reference temperature at which K_{Ic} is determined and by correlating the value with charpy impact measurements. The temperature corresponds to a charpy measured value of 50 ft lbs, ~~the~~ the corresponding reference temperature is called the RT_{NDT}. Neutron fluence causes a change from the initial RT_{NDT} value. RT_{NDT} is therefore adjusted to account for fluence-related loss of toughness. Copper and nickel content can accelerate RT_{NDT} shift with fluence and are introduced as variables by relating ~~them~~ to observed values in a number of specimens. An empirical formula developed by Guthrie correlates these effects with RT_{NDT}. In the short term, this approach. accounts for matters of concern adequately. If the physics of the materials behavior were better understood from fundamental principles, there would be a better basis for predicting longer term behavior.

An issue of importance is attenuation of the neutron fluence through the vessel wall. Measurements are now based on inside surface fluence and a factor of 10 through-the-wall attenuat~~ion effect~~ has been

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[3] It has also been suggested as a factor by French, German and British specialists but no quantitative evaluation has been offered.

cited in some instances. Change in fracture toughness because of neutron attenuation through the wall ~~might be evaluated~~ can be calculated using atomic displacement theory. A nominal adjustment to RT_{NDT} could be used to correct for this effect. Obviously, this approach needs to describe the attenuation ~~effect~~ in correct physical terms so as not to over- or under-estimate toughness ~~capacity~~ of the vessel material. The approach used could take account of an element of reserve toughness capacity that is known to exist and has not previously been included in vessel integrity assessments. Its consideration may be overly complex for the short in the evaluation activities but it should ~~tend to~~ reduce the severity of toughness loss ~~effects~~. Shift in RT_{NDT} with fluence accumulation for even conservative estimates of the worst conditions is estimated to be less than 15°F per year. There is little opportunity to reduce fluence accumulation rate in a short period (within one or two years). Other considerations dominate the risk as will be discussed later but the immediate concern may be eased if attenuation effects are recognized.

Over longer periods of time, if it is desired to assure extension of usable vessel life without other types of action, reduction ~~on~~ of the fluence accumulation rate at the earliest practical time may be of value. Techniques to reduce vessel wall neutron flux would require reconfiguring the core fueling arrangement.

CLADDING CONSIDERATIONS

Reactor vessels are clad on the inside with an austenitic stainless steel weld overlay material to provide corrosion protection. The thickness of the material is nominally 3/16" but may vary by ± 1/16" from this dimension. The cladding is equivalent in composition to A304 Stainless Steel plate but various welding materials have been used and its precise composition will vary from vessel to vessel. ~~It~~ The cladding has not previously been of interest as a factor in vessel fracture control and has been ignored in structural analysis. However, recent work has drawn attention to cracks that could develop in the cladding and extend through the vessel wall ~~material~~. The explanation for these cracks is available and is not germane to the thermal shock issue, but the potential existence of such cracks has raised questions as to whether they could propagate through the clad thickness and thereby offer a site for crack growth initiation at the inner surface.

Stainless steel has a different thermal expansion coefficient from the vessel wall material (A533 low alloy steel) and stress related effects can be of concern when the vessel changes temperature. These may be important in determining crack initiation stresses (K_I). At normal working temperatures of the vessel, the thermally induced stresses are negligible but they increase as the wall is cooled from its nominal operating temperature. If crack growth in the stainless steel overlay can be coupled to the ferritic steel vessel wall, that mechanism for fracture propagation must be evaluated. Cracks as small as 3/16" deep may have to be considered, though these may be [illegible] by any [illegible] [illegible] on the [illegible]

FLAW SIZE AND FLAW DETECTION

The ASME Code requires under Section III that flaws in vessel structures be limited in size and the basic fracture toughness limits are predicated on being able to detect such flaws. In the past, detection sensitivity for flaws of the order of 1" depth in the wall was considered to be adequate for fracture evaluation purposes. There has been some debate about the reliability of detection methods even for this size flaw, but with concerns arising for crack growth near the inner vessl surface, even smaller sized flaws may be important.

The ACRS, in its 1974 report on pressure vessel integrity, emphasized the importance of improved nondestructive inspection methods for flaw detection in vessels exposed to nuclear service conditions. Although a great deal of investigative work has been carried out in this area, there has been little improvement in detection capability.

Of current concern because of thermal shock questions is the ability to detect~~ion of~~ flaws near the surface of vessels.

Detection methods for in-service inspection use ultrasonic techniques that are generally most effective at depths around 1/4 through the thickness of the vessel and deeper. Flaws at the cladding surface are detected by visual means. Small flaws near the inner surface might be detected effectively by ultrasonics if the inner surface of the cladding had a smooth and uniform finish. Newly fabricated vessels could probably ~~be fabricated~~ in a manner that would make such an inspection improvement at the vessel beltline where neutron fluence causes such small flaws to be important but ~~it~~ is clearly not ~~practical~~ for the older vessels of concern to the thermal shock question. ~~The~~ An ~~need for~~ detection ~~of~~ small flaws needs to be established. Subsequent discussion of the crack growth analysis methods will indicate why this is the case but if the need exists, the actions required to provide suitable detection methods are not defined.

ANALYTICAL METHODOLOGY

The ASME Pressure Vessel Code for Nuclear Components, Section III, uses design practices based on plate and shell theory. Fracture mechanics techniques are used to evaluate the potential for cracks or other flaws in the vessel structure to propagate. Up to now, only linear elastic fracture mechanics (LEFM) methods have been applied, although elastic-plastic behavior is expected under many circumstances and such behavior would result in more capacity to resist flaw propagation than is presently claimed.

LEFM techniques have been demonstrated to be applicable for materials that behave in a brittle manner and many practicioners claim that the thick-walled nuclear vessels, though constructed of ductile materials, may be adequately evaluated using LEFM methods without excessive conservatism. The degree to which LEFM exaggerates the concern for PTS is not adequately understood.

A postulated condition for PTS is that a small flaw at the inner surface of the vessel propagates through the cladding and grows in length to the extent that a two dimensional fracture analysis using LEFM methods describes the condition. Some experimental evidence exists to support this position based on the behavior of precracked vessel specimens. However, little has been done to show that this type of crack condition is representative of nuclear reactor vessels and the position may be overly pessimistic. An alternate approach would be to develop elastic-plastic fracture mechanics analysis (3-D) techniques but this will require a long-term experimental and analytical program.

As presently applied, LEFM techniques can be used only by accepting a somewhat arbitrary definition of flaw geometry and flaw growth behavior that may be excessively conservative. If nuclear vessels can be shown to be adequately safe using this analytical basis, it is surely a conservative and unchallengable safety position. If there is doubt about showing safety acceptability by this route, the 3-D methods deserve attention.

THERMAL SHOCK TRANSIENT DEFINITION

Prior discussion directs attention to the basis for judging structural adequacy of nuclear reactor vessels. Understanding the structural assessment methodology is a prerequisite to assessing the importance of thermal shock transients to vessel integrity. Transients ~~are~~ of concern ~~when~~ combinations of thermal and pressure stress are sufficiently high that they can cause a flaw to propagate until vessel fracture becomes a matter of concern. ~~Most such~~ transients do not reach this level of severity. The PTS issue applies to a few specialized classes of events. The ones so far identified are:

1. Circumstances where the nuclear power plant is releasing only radionuclide decay heat, steam generator feedwater ~~is~~ being supplied at full flow to lower reactor coolant temperatures and the coolant ~~is~~ in turn cooling the vessel rapidly ~~while~~ primary system pressure ~~is~~ being maintained or increased.

2. Circumstances where, after nuclear plant shutdown, the secondary coolant system ~~has~~ been breached and steam blowdown in combination with continued auxiliary or main feedwater supply ~~is~~ cooling the primary coolant rapidly through heat transport from the steam generator.
3. Circumstances where ~~the~~ shutdown nuclear plant ~~is~~ might inadvertently feed~~ing~~ steam through the turbine bypass at full flow ~~and~~ thus drawing heat out of the secondary circuit at a rate which ~~will~~ cause temperatures in the steam generator to fall to a level where reactor coolant is being rapidly cooled.
4. Circumstances where a small break (about 2" in diameter) loss-of-coolant-accident (SBLOCA) require~~s~~ full flow of low temperature High Pressure Safety Injection (HPSI) cooling water to be fed directly to the primary coolant inlet line causing rapid cooling of the primary coolant adjacent to the reactor vessel.

In all of the above circumstances, there ~~is~~ concern for vessel fracture if the following conditions exist:

(a) Combination of accumulated fast neutron fluence and reduced coolant temperature ha~~s~~ driven the vessel wall toward "lower shelf" temperatures where fracture toughness is seriously reduced from normal conditions.

(b) A flaw of significant size exist~~s~~ in the region of reduced fracture toughness.

(c) A combination thermal and pressure stress condition ~~is~~ sufficient to cause rapid and unstable growth of the flaw until disruptive or catastropic rupture of the vessel occur~~s~~.

AVOIDING PRESSURIZED THERMAL SHOCK (PTS)

That PTS is possible is an indisputable fact. Whether it could lead to vessel disruptive or catastrophic fracture is debatable as the foregoing discussion has indicated. Nevertheless, it is not in the interest of public health and safety to accept the risk if ~~it~~ has a high frequency-of-occurrence potential. In order to reduce the occurrence probability, it is necessary to understand the contributing factors and show that are they are adequately controlled.

The Rancho Seco event was caused by a combination of failure in the steam generator feedwater system, loss of diagnostic instrumentation and operator oversight. With adequate training, the particular circumstance could be avoided and improvement in the reliability of diagnostic instrumentation would provide adequate information for effective operator action.

Secondary system blowdown followed by excessive steam generator cooling could be avoided only if feedwater supplies are adequately controlled. The feedwater supply control system, if properly designed, could prevent PTS. Operator action represents a redundant control capability that could enhance reliability if ~~he~~ has adequate diagnostic information and adequate time. The reliability of this combined capability must be shown adequate to make this combination acceptable.

Steam turbine bypass behavior requires a similar combination of control, diagnostic information and operator action, ~~time~~.

High pressure injection effects are less easily ~~corrected~~ by operator action because of contradictory need to sustain core cooling conditions, thus discouraging the operator from interrupting the high pressure injection flow.

There may be other circumstances, primarily derivatives of the above four events, that could contribute to PTS. A more complete study of these is needed. The NRC staff has initiated a study of these matters but it may be so detailed that the time to complete it will be too extended for short-term action. An investigation of a narrowly prescribed set of events based on expert opinion seems essential to short-term actions. It is not clear how this short term action is to be implemented.

The [illegible] of the thermal [illegible] must be controlled [illegible] by the [illegible] of the [illegible] wall. This in turn would depend on the thermal [illegible] in the [illegible] of the cold leg [illegible]. Calculation of thermal stress [illegible] The [illegible] of the vessel analysis is the greatest [illegible] general. This is the assumption of "thermal stripping". The assumption is generally conservative, and mixing of fluids [illegible] modified. The thermal shock

SAFETY MARGINS AFFECTING THE PTS ISSUE

To assess the safety significance of PTS, it is necessary to examine whether appropriate credit has been taken for conservative assumptions that ease concern for the safety question. The region of concern for PTS conditions is at the "beltline" of nuclear vessels, a cylindrical structure whose stress condition is easily analyzed if flaws are not present. Circumferential stresses are controlling with respect to pressure since they are double the nominal axial stresses. Thermal stresses are essentially the same over an entire sector of wall if thermal shock conditions occur. Missing / The conditions that create uncertainty are related to how those stress conditions affect the stress distribution in a crack-like flaw.

The ~~manner in which~~ the flaw geometry, shape, and orientation of ~~is assumed to~~ exist, and its geometrical change as influenced by material toughness, are major unknowns that have been evaluated on the basis of "worst case" conditions. This results in structural analysis being based on the assumption that a flaw exists whose size, though initially in the range of 1/4" deep and a few inches long, will quickly grow into a very long penetrating flaw whose behavior corresponds to LEFM principles. If the flaw does not grow to such proportions and does not penetrate the clad surface, the structural behavior can involve significant plastic deformation in the vessel wall, absorbing fracture strain energy that has been assumed to cause the crack to propagate. Most vessel cracks would behave in this manner. Only a small category of flaws near the inner wall surface fit the LEFM model conditions. The likelihood ~~that~~ of a flaw of the proportions causing concern may not be very high but because of limitations on the inspection practice, as recognized, verification of the absence of flaws is impractical.

A phenomenon known as "warm prestressing" has been identified as beneficial in limiting crack propagation. Basically, the idea is that if a crack begins to propagate through the vessel wall and the front reaches a point where the stress intensity factor (K_I) begins to ~~fall off~~ diminish with time, the crack will start to close, putting the tip of the crack into compression and thus eliminating the tensile stress ~~condition~~ that causes the crack to propagate. Some types of transient-induced cracking can be controlled by this phenomenon even if the K_I value exceeds the K_{Ic} value. The physical basis for this claim is well founded and it would be desirable to show which transients would no longer be of concern if " warm prestressing" is accepted as a controlling factor.

The fracture toughness of the vessel material, K_{Ic}, is determined from initial measurements of the toughness with appropriate accounting of copper, nickel and copper constitutive effects using specimens representative of the actual vessel. While the test specimens may not truly duplicate the actual materials, they are believed to be conservatively representative of the actual vessel. The vessel fracture toughness is being adjusted by allowing for RT_{NDT} shift with cumulative fluence at the vessel inner surface. Since there is attenuation through the wall and fracture toughness will be benefited in some measure by this effect, there is a margin of conservatism in the present evaluation basis. A precise evaluation of this effect is not possible with current information but it seems reasonable to consider how much reserve toughness capacity might be available because of attenuation effects. Atomic dislocation theory could be used to bound the conservatism available from this consideration.

Depending upon the postulated scenarios, combinations of automatic control action and operator response can affect the liklihood that reactor coolant temperatures can be driven to temperatures at which PTS is a matter of concern. If there is a high degree of redundancy in the sensor and control equipment, the liklihood of control system induced thermal shock is reduced. Failure modes and effects analysis (FEMA) of the control system can provide insight concerning this cause of PTS. Evaluation of existing designs can clarify questions concerning event liklihood. Typical systems need to be evaluated to determine what can be expected and what is normally provided to assure reliable system response. They may be adequately reliable for most installations or easily modified to provide improvements.

Operator action is a separate and possibly redundant protective capability that will reduce the liklihood of PTS. If operator action can be taken early enough to keep some or all of the affected vessel wall near upper-shelf temperatures, then inherent fracture toughness of the material will minimize the liklihood of PTS induced disruptive or catastrophic rupture. The time needed for action determines how much credit can be claimed. Potential actions are to control the source of cooling by limiting feedwater supply rates, stop the primary coolant circulators allowing the water near the vessel wall to stagnate, provide supplemental heat to the primary coolant or allow afterheat to give an equivalent result and to depressurize the system to a level where pressure loads do not impose fracture stress. The time available for these actions is a measure of the safety margins associated with operator response.

Probabilistic risk assessment (PRA) methodology has been suggested as a means of evaluating conservatism of the type listed above. While numerical analysis could give a quantitative impression of the risk, the values are not easily placed in context of acceptable risk to the health and safety of the public. It would be useful if there were a narrative discussion provided by technical experts who could relate their viewpoints to practices in comparable situations addressed in nuclear safety regulation or other public safety applications.

Ability to depressurize the primary coolant system quickly would leave only thermal stress as a fracture contributor and could eliminate the public safety concern entirely.

INDUSTRY PARTICIPATION

In response to the NRC 60-day and 150-day requirements for information placed on licensees, the NRC Staff has received from the Westinghouse, B & W, and C-E Owners Groups evaluation of the PTS problem for the nuclear vessels of concern. The information is still somewhat incomplete but it shows the following results:

1. All groups claim that there is no immediate concern for PTS as a public safety issue.
2. In all cases, Rancho-Seco-type PTS events are ruled out on the basis of reliable control system behavior or prompt operator action.
3. In some installations, operating procedures to address PTS conditions are not well defined and much is left to operator judgement.
4. The B & W group is evaluating transients caused by continuing supply of cooling water from the HPSI source and is attempting through R & D work to show that hydraulic behavior will eliminate PTS conditions at the vessel wall from this source. Warm prestress is being presented as a behavioral phenomenon that will assure that cracks do not propagate deeply into the vessel wall if K_I exceeds K_{Ic}.
5. The C-E Group is evaluating main steam line breaks as their limiting transient and are also claiming credit for warm prestressing.
6. The Westinghouse Group is using a PRA approach and developing event trees to assess the liklihood of PTS safety problems but has not identified a limiting case.

NEEDED SUPPORTING WORK

In order to establish a regulatory position on the PTS issue, a number of actions are needed. The most urgent are:

1. A complete survey of the materials used in vessels having questionable fracture toughness capacity to determine what is known about them as a basis for determining fracture toughness. This information is still incomplete.

2. A more complete definition of the transients that can lead to PTS of sufficient severity that disruptive or catastrophic rupture is a matter of concern. Examination of a selected group based on prior experience and expert judgement is needed along with an explanation of the logic so the need and basis for short-term corrective actions can be understood.
3. A determination of the control features and related operator actions on which nuclear plants depend to ameliorate concerns for PTS. Representative systems need thorough study as a basis for judging control adequacy.
4. A more complete understanding of the experimental evidence used as a basis for judging fracture potential due to PTS. Experimental work on cladding and related small flaw effects is important.
5. Identification of potential actions that might be taken to ameliorate conditions leading to concern for PTS-initiated fracture other than fluence reduction. These will show where prompt improvement might be attainable (e.g., control improvements, depressurization).

6. Research on the effect of PTS on [illegible]

7. A better understanding [illegible]

The NRC has initiated work in all of these areas but the rate at which the work will come to fruition is unclear. A schedule is being developed and will be included in the plan for resolution of the PTS issue. Realistically, however, there is a need to narrow the scope of the work program so that a basis for effective actions will be forthcoming within a one or at most two-year period.

POTENTIAL LONG TERM ACTIONS

If neutron fluence accumulation were to ~~reaches~~ reach the stage where public safety would be in jeopardy by continued operation, it would be necessary to discontinue operation. With additional investigation, it may be possible to show that no installation reaches that condition within its projected lifetime. Based on current knowledge and current operating practices, some plants

~~would~~ reach limiting circumstances within a few years. If some really are near the end of their projected life when judged by realistic circumstances and reasonable probabilities, several actions are potentially capable of extending plant life time.

Since the principal consideration is cumulative fluence, reconfiguring the core to limit fast neutron leakage to the reactor vessel is the most direct corrective measure. This has been examined for some plants and it has the potential for reducing fluence accumulation rates by as much as a factor of 4 for the highest leakage cores. ~~It is~~ of value for designs not based on low-fast neutron leakage. The margin of ΔRT_{NDT} will determine whether this ~~is~~ a useful action. Plant derating or infringement on fuel-peaking factor margins ~~may~~ be necessary.

Vessel annealing has been recognized as a potential action to recover fracture toughness by relieving atomic dislocation effects. Temperatures in excess of 700°F ~~are~~ needed and the practicalities of performing this action for the plants in question must be determined. A demonstration would be a prudent action before applying the technique to an existing plant.

Major modification of controls and instrumentation ~~may~~ reduce the liklihood of PTS ~~to acceptably low~~ levels of probabilistic circumstances. Potent~~ial enhancement~~ needs careful examination. If properly implemented, dependence on human action might be eliminated.

Alternatively, reliability of operator action might be enhanced by improved diagnostic provisions. Operators might be depended ~~upon to act more quickl~~y if they had suitable signals of PTS conditions. Contradictory requirements to maintain enhanced core cooling and still avoid PTS might be eliminated.

Techniques to isolate the vessels from PTS conditions, including improved primary system depressurization capability might be practical for some installations where PTS is a concern.

All of the above "long-term" actions can be interpreted to be a part of the NRC Staff's program. The schedule for doing the work and the resources needed have not been developed in a form suitable for

ACRS evaluation. A "short-term" effort to clarify the long-term program is needed in which contributions of the licensees, their contractors, DOE as well as NRC and its contractors are established.