Impact of Plant Operating Experience on Design Considerations for Reactor Vessel Integrity

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The recent operating experience of the Pressurized Water Reactor (PWR) Industry has focused increasing attention on the issue of reactor vessel pressurized thermal shock (PTS). Previous reactor vessel integrity concerns have led to vessel/plant system design and operating procedure modifications, and increased attention to the PTS issue is causing consideration of further improvements. Events such as the Rancho Seco excess feedwater transient (3/20/78), the Three Mile Island 2 loss of normal feedwater transient (3/28/79), and the R. E. Ginna steam generator tube rupture (1/25/82) have led to significant primary system cooldowns. Each of these cooldown transients occurred concurrently with a relatively high primary system pressure. Consideration of these and other postulated cooldown events has attracted notice to the impact of operator action and control system effects on reactor vessel PTS.

A methodology, which couples event sequence analysis with probabilistic fracture mechanics analyses, was developed to identify the events that are of primary concern for reactor vessel integrity. Operating experience is utilized to aid in defining the appropriate event sequences and event frequencies of occurrence for the evaluation.

Once the specific event sequences of concern are identified, detailed thermal-hydraulic and structural evaluations can be performed to determine the conditions required to prevent the extension of postulated flaws or ensure flaw arrest in the reactor vessel. This paper addresses key aspects of the thermal-hydraulic, stress, and fracture mechanics analysis of the reactor vessel. The effects of incomplete mixing of safety injection flow in the primary cold leg and vessel downcomer and the application of warm prestressing are emphasized. The results of these analyses are being used to define further improvements in vessel/plant system design and operating procedures.

Previous design considerations that have evolved as a result of reactor vessel integrity evaluations are discussed. These include the development of more realistic design analysis tools, specification of improved vessel material properties, and selection of plant system modifications. Improvements that are being developed or are under consideration are also discussed. These include vessel fluence reductions, additional modifications to operating procedures, increased use of probabilistic event sequence/fracture mechanics analysis methods, improvements in material fracture toughness, and reductions in the severity or frequency of occurrence of dominant reactor vessel PTS transients.
1. Introduction

Reactor vessel PTS events have been shown from PWR operating experience to be transients which result in a rapid and severe cooldown in the primary system coincident with a high or increasing primary system pressure. The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance due to neutron irradiation exists. Such an event may cause propagation of flaws postulated to exist near the inner wall surface, thereby potentially threatening the integrity of the vessel.

Reactor vessel PTS is not a new issue. Historically, vessel integrity following a PTS transient has been evaluated solely on a deterministic basis. As a result of developments in 1982 [1,2], the evaluation of reactor vessel integrity for PTS now incorporates the use of probabilistic techniques in combination with traditional deterministic methods. Figure 1 depicts the new methodology which can be used to identify modifications that may be necessary to achieve PTS safety goals for a particular plant design.

This extended paper summary amplifies the probabilistic portions of the total evaluation. Brief discussions are given on key aspects of the thermal-hydraulic and deterministic fracture mechanics analysis of dominating transient scenarios. Finally, a detailed discussion is presented on vessel/plant system modifications that have been considered in the past, and on those that are expected to be considered in the future. References [2] and [3] describe in detail many aspects of a traditional deterministic fracture mechanics analysis which are not presented here.

2. Probabilistic Evaluation

The probabilistic PTS evaluation provides an estimate of the total risk of reactor vessel failure from PTS as a function of plant life and identifies the specific transient scenarios which contribute most significantly to the total risk. The analysis is based heavily, although not exclusively, on accumulated operational experience in the nuclear industry and can be used to improve the design of existing or future plants. The analysis is performed in four principle steps which coincide with the engineering disciplines of probabilistic risk assessment, thermal-hydraulics, and fracture mechanics.

The first step is to identify all scenarios which potentially lead to pressurized thermal shock of the reactor vessel using event tree analysis. The process is shown schematically in Figure 2. The initiating event vector is selected to include direct cooldown events such as small loss of coolant accidents (LOCA's) and events which perturb the plant without themselves causing a major cooldown. Table 1 [1] lists the initiating events and the associated frequencies which are typical of U.S. PWR operating experience. The plant model is developed to identify paths by which the initiating event may develop into a PTS event. For example, the plant model will show that a reactor trip will lead to a small steam break cooldown if the steam dumps are unavailable and a secondary PORV, which is being used to remove stored energy, sticks open. The mitigation event tree is then generated. It models the influence of automatic control and protection systems and operators on each cooldown state. This tree not only considers the case where such systems and operators perform the intended actions to limit cooldown but also cases where cooldown is exacerbated by equipment malfunctions or operator error. The end state vector lists all potential PTS scenarios, grouped by common end states, along with the associated frequencies.
The next step is to associate thermal-hydraulic characteristics with each cooldown transient end state. Because there can be literally thousands of end states and since only a small number will prove to be of any practical concern, it is prudent to use simple approximations at this point. Using sensitivity studies and judgment based on experience, each end state is fit with a simple exponential curve, and an associated pressure is selected which is representative of the system pressure during the period of flaw extension. These approximations allow any cooldown transient to be characterized by three quantities; a final temperature ($T_f$) reflecting the depth of the cooldown, a time constant (a) reflecting the rate of the cooldown, and a characteristic pressure. In cases where the actual transient is erratic, approximations can be conservatively applied.

The third step is to quantify the conditional probability of reactor vessel failure given that the cooldown transient occurs. The data which is displayed in Figure 3 [2] allows this conditional probability to be ascertained graphically. The curves in Figure 3 were generated from probabilistic fracture mechanics (PFM) analyses [2] using the Monte Carlo technique. A matrix of cases for given $T_f$, $\alpha$, and inner surface reference nil-ductility transition temperature ($RT_{NDT}$) values are evaluated to obtain results for generation of the curves. (The $RT_{NDT}$ values are calculated as a function of initial $RT_{NDT}$, material residual elements, and fluence using the methodology discussed in Reference [2]). For each case, a large number of deterministic fracture mechanics analysis trials ($\approx 10^5$) are executed using random values which are selected from distributions defined for the pertinent input properties. The probability of vessel failure for each case is determined by dividing the number of failures by the number of trials. The curves in Figure 3 are plotted from the matrix of results by normalizing $T_f$ against $RT_{NDT}$.

Limitations and uncertainties exist in the use of these results since the input distributions in the PFM analyses require refinement. Several advancements which are currently being used in deterministic analyses have also not been applied in the PFM analyses. The data in Figure 3 are based upon a failure criterion of through-wall crack propagation for non-arresting flaws. It may be equally appropriate to use crack initiation or some other phenomenon as the criterion for unacceptable vessel performance depending upon the desired use of the analysis results.

The final step is to associate a probability of vessel failure with each scenario by multiplying the probability of the scenario itself by the conditional probability of vessel failure given that the scenario occurs. If the probabilities of individual scenarios are summed together, the total risk of vessel failure from PTS can be ascertained as a function of $RT_{NDT}$, and dominating transient scenarios can be identified. Figure 4 is a graph which shows total PTS risk as a function of plant $RT_{NDT}$ values based on recent generic studies. Also shown is the risk associated with each cooldown scenario that is a significant contributor to the total risk.

Once the probabilistic PTS analysis has been completed, the transient scenarios which dominate a plant's total PTS risk can be selected for more detailed analysis. Referring to Figure 4, the scenarios which exceed the PTS safety goal in the $RT_{NDT}$ range of interest are obviously selected (LOCA "B" and SSB "A"). It is also appropriate to choose scenarios which fall below the safety goal within a band which accounts for uncertainty in the probabilistic analysis (LOCA "A" and Other "A"). The very low risk scenarios (LOCA "C" is an example), which incidently include many of the "design basis" transients that have been
historically used for PTS decision making, are eliminated from further consideration.

Detailed thermal-hydraulic and deterministic fracture mechanics analyses are then performed on the small group of selected transients. The results are plotted on a graph as shown in Figure 5. The abscissa of each "X" is the lowest RT \text{NDT} at which vessel failure is predicted to occur. The ordinate is the frequency of occurrence of the scenario. A horizontal line is drawn to the right of each "X" because failure will always be predicted to occur at all RT \text{NDT} values which are higher than the RT \text{NDT} at "X". If the total risk curve (shown as a dotted line) extends into the unacceptable quadrant, some improvement is necessary to lower it to an acceptable level. In Figure 5, the SSB "A" and/or LOCA "B" PTS transient(s) must be addressed to demonstrate acceptable PTS risk.

There are three general ways of affecting the risk associated with a specific scenario. The frequency of the scenario can be reduced through the use of system modifications or man-machine interface improvements (the "X" shifts downward). For example, installation of a block valve upstream of the secondary power operated relief valves might reduce the frequency of non-isolatable small steam breaks. The severity of the transient, if it occurs, can be reduced through the use of system modifications or man-machine interface improvements (the "X" shifts to the right). Heating of emergency core cooling water is an example of such a modification. Finally, methods such as flux reductions may be employed to reduce the vessel RT \text{NDT} at the end of plant life (the vertical boundary of the unacceptable quadrant shifts to the left).

Potential improvements which may be used if it should become necessary to reduce the risk due to PTS include: heating of emergency core cooling water, auxiliary feedwater flow limiting devices, PTS control and protection systems, core modifications, vessel annealing, man-machine interface improvements, and specific equipment upgrades. A cost-benefit analysis can be performed on individual improvements, or combinations thereof, which are found to reduce PTS risk a sufficient amount. In this way, the most cost effective method of achieving PTS goals can be selected.

3. Thermal-Hydraulic and Deterministic Fracture Evaluation of Dominating Events

Probabilistic evaluations of PTS events using the above methodology show that the probability of vessel failure is often dominated by LOCA's. Furthermore, it is the small break LOCA which is large enough to cause a loss of natural circulation in the primary coolant system, that dominates the PTS risk from LOCA's. Loss of primary flow causes degraded mixing of the cold safety injection flow with the hot primary fluid.

The thermal hydraulic codes which model the system small break LOCA transient behavior for core peak clad temperature analyses are not designed to model local mixing effects, i.e., thermal stratification and secondary flows. These local effects are important in determining the local temperature response of the vessel wall for PTS analyses. Three dimensional analytical codes are currently being developed to model these local effects [4]. However, the large computational times associated with such codes at present allow them to be used only on a very limited basis.

A simple single node model (mixing cup) can be used to approximate vessel downcomer temperature if it is applied to a volume of the primary system where strong secondary circulation is known to cause nearly uniform mixing of the fluid. Experiments to date indicate that treating a portion of the downcomer, cold leg, and loop seal as a mixing cup yields results that agree well with measured data. Figure 6 shows the temperature history
which is predicted by a mixing cup model versus experimental data [5]. An extension of the mixing cup model is shown in Figure 7 for a small LOCA. Temperature histories are given showing the effect of thick metal heat addition to the model. The simplicity of the mixing cup model permits the simulation of the large number of scenarios which involve degraded mixing.

The basic deterministic method used for evaluating the effect of dominating transient events, such as a small LOCA transient, on reactor vessel integrity is linear elastic fracture mechanics (LEFM) [3]. The deterministic fracture mechanics analysis also includes the benefits of warm prestressing, when applicable [1,3].

Warm prestressing is an empirically observed phenomenon that results in an apparent increase in fracture toughness of the vessel material. The phenomenon is induced when the material is "prestressed" to a high flaw tip stress intensity while the material is at a high temperature and, therefore, has a correspondingly high fracture toughness. As the thermal portion of the PTS transient continues, the calculated material fracture toughness is reduced due to steadily decreasing material temperature. At some point in the transient, even though flaw initiation might be calculated by LEFM methods, consideration of warm prestressing would demonstrate non-initiation of an assumed flaw or reduce the arrest depth of a propagating flaw. The phenomenon is acknowledged by the technical community as a real phenomenon that is applicable in many PTS transient scenarios and is conservatively applied in PTS evaluations.

4. Previous and Future Considerations

Consideration of operating history, postulated events, measured fluence and irradiation damage has resulted in prior design modifications aimed at improved plant safety and reliability as discussed in the following paragraphs.

As information from operating plants became more available, it was determined that the residual element, copper, played a significant role in the susceptibility of the vessel material to irradiation damage. As a result of this finding, most recent vessel designs specifically limited the residual amounts of copper in the reactor vessel material and its welds.

In order to further minimize the adverse impact of fluence on weld material in the reactor vessels, recently designed reactor vessels oriented the key vessel welds into low fluence regions.

In order to minimize the adverse effect of system repressurization in combination with a postulated severe cooldown, limited head safety injection pumps were incorporated in some plant designs. This minimized the maximum pressure to which the system could be repressurized following an abnormal transient event.

In addition to actual design specification and system hardware changes, more realistic design analysis tools were developed to allow more accurate prediction of potential flaw initiation and propagation following postulated transient scenarios. These tools allowed useful design changes to be specified and implemented while not requiring unnecessary changes.

Significant progress has been made in developing and utilizing further new methods to evaluate the impact of PTS transients on reactor vessel integrity. The motivation for developing these additional new methods has been the need to realistically determine the significance of severe cooldown events that have occurred to date relative to vessel
integrity, and to evaluate the improvements or design changes that may be necessary for demonstrating continued safe operation.

The basic methods for performing a more realistic PTS evaluation have been described in this paper. However, the methods described do not include other methods currently under development that could further reduce uncertainties. These additional methods include: a more realistic determination of stagnation conditions, development of efficient multi-node mixing models, a more accurate treatment of three dimensional thermal and stress effects, a better quantified treatment of warm prestressing and clad effects, an improved method for predicting irradiation damage, and more accurate probabilistic predictions of flaw propagation. Furthermore, an assessment of total vessel integrity safety risk with a comparison of that risk to a safety goal can be established for determining acceptability when that safety goal is defined by the regulatory agency. In the future, plant specific evaluations will include an increasing number of these other methods which will result in more realistic evaluations and will identify only those improvements or design changes that are actually necessary to maintain the plant safety risk within the defined safety goal.

5. Conclusions

The original PWR designs considered irradiation damage of the vessel material and the impact of postulated severe cooldown transients. However, the safety and reliability of PWR's has been continuously improved by utilizing operating plant experience in successive design changes. Operating plants will continue to provide additional information on the severity and frequency of significant cooldown events and the effects of fluence and material chemistry on material irradiation damage. This additional information will be factored into increasingly sophisticated and more realistic design analysis tools to develop additional improvements in plant safety and reliability.

6. Acknowledgements

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References

Table 1. INITIATING EVENT FREQUENCIES [1]

<table>
<thead>
<tr>
<th>Event</th>
<th>Frequency (/R-Year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Loss of Main Feedwater (LOFW)</td>
<td>3.41</td>
</tr>
<tr>
<td>2. Closure of one Main Steam Isolation Valve (MSIV)</td>
<td>6.00 x 10^-1</td>
</tr>
<tr>
<td>3. Loss of Primary Flow (LOPF)</td>
<td>3.21 x 10^-1</td>
</tr>
<tr>
<td>4. Core Power Increase (POWIN)</td>
<td>4.77 x 10^-2</td>
</tr>
<tr>
<td>5. Turbine Trip (TT)</td>
<td>4.00</td>
</tr>
<tr>
<td>6. Spurious Safety Injection Activation (SSI)</td>
<td>1.59 x 10^-1</td>
</tr>
<tr>
<td>7. Reactor Trip (RT)</td>
<td>4.11</td>
</tr>
<tr>
<td>8. Turbine Trip due to Losses of Offsite Power (TT/LOOP)</td>
<td>1.01 x 10^-3</td>
</tr>
<tr>
<td>9. Steam Generator Tube Rupture (SGTR)</td>
<td>3.92 x 10^-2</td>
</tr>
<tr>
<td>10. Small LOCA, &lt; 1.5 in. diameter (LOCA-1)</td>
<td>9.07 x 10^-3</td>
</tr>
<tr>
<td>11. Small LOCA, &gt; 1.5 in. diameter (LOCA-2)</td>
<td>6.11 x 10^-4</td>
</tr>
<tr>
<td>12. Large LOCA, &gt; 6 in. diameter (LOCA-3)</td>
<td>3.88 x 10^-4</td>
</tr>
<tr>
<td>13. Excessive Main Feedwater (EX FM)</td>
<td>2.50 x 10^-1</td>
</tr>
<tr>
<td>14. Steamline Rupture Inside Containment (STM BRK IN)</td>
<td>3.88 x 10^-4</td>
</tr>
<tr>
<td>15. Steamline Rupture Outside Containment (STM BRK OUT)</td>
<td>3.87 x 10^-2</td>
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Figure 1. Evaluation of Reactor Vessel Integrity for Pressurized Thermal Shock
Figure 2. Overview of the Assembly Process, Showing Relationship of Pinch Points, Frequency Vectors, Event Trees, and Transition Matrices

Figure 3. Conditional Failure Probability as a Function of Cooldown Rate and Final Temperature Minus RT_{NDT} [2]

Figure 4. Plant Specific Frequency of Vessel Failure
Figure 5. Plant Specific Frequency of Vessel Failure Using Deterministic Fracture Mechanics (DFM) Analyses

Figure 6. Comparison of the Simple Control Volume Mixing Model with Creare 1/5 Scale Test No. 73 - Fluid Thermocouple No. 24 Located on the Vessel Wall at the Core Midplane [5]

Figure 7. Small LOCA Temperature Transient History with Thick Metal Heat Addition