NUCLEAR PRESSURE BOUNDARY MATERIALS RESEARCH, PROBLEMS AND PROPOSED SOLUTIONS

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SUMMARY

Nuclear generating units are supplying an ever increasing percentage of the world's electricity. The world's installed nuclear capacity in 1977 is 80.6 GWe; by 1980 the number will be 239 GWe; and by 1990 433 GWe. Despite the size and complexity of a modern nuclear power plant and the harsh material operating environment, these plants are extremely reliable engineering systems. For 1975, the average availability factor for all United States nuclear plants was 72.2% with an average capacity factor of 66.5%. During the same period, the average availability for fossil plants over 600 MWe was 73.1%. The capacity factor during this same period was 54.5%. It is interesting to note that nuclear power, despite a slightly lower availability factor (a difference of 0.9%), showed a 12% higher capacity factor. The reason for this difference is that utilities prefer to base load their nuclear plants to take advantage of the lower cost of electricity production, using more expensive fossil units for meeting peaking loads.

If we assume a 7 mil/kWh difference in coal and nuclear generation rates, and further assume that all electricity from nuclear outage is replaced by coal generation capacity, each percent of lost nuclear capacity world wide represents an increase in cost of electricity of $32.9 million per year to the world's consumers. If each lost percent in nuclear capacity were replaced by oil, which is the more probable case, the loss to the world economy would be $138.6 million per year.

Forced outage of nuclear units in the U.S. due to material related equipment malfunction or failure represented 35.8% of the total outage time through January 1, 1976. If we assume that U.S. figures are representative of all nuclear plants, the cost of material related failure in increased electricity cost alone will be between $327 million and $1.4 billion in 1977. From these figures it is clear that there is a large economic incentive to solve our material related problems and increase the availability and capacity of our nuclear units.

The complete test discusses several important materials problems facing LWR operation and some of the research programs that have been formulated at EPRI to address these problems. The problems include steam generator corrosion damage, BWR pipe cracking, BWR nozzle corner cracking, and radiation induced embrittlement. LWR materials problems are complex and will not easily yield to solution. However, the economic rewards for the solutions are great.
INTRODUCTION

Nuclear generating units are supplying an ever increasing percentage of the world's electricity. The world's installed nuclear capacity in 1977 is 80.6 GWe; by 1980 the number will be 239 GWe; and by 1990 433 GWe. Despite the size and complexity of a modern nuclear power plant and the harsh material operating environment, these plants are extremely reliable engineering systems. For 1975, the average availability factor for all United States nuclear plants was 72.2% with an average capacity factor of 66.5%. During the same period, the average availability for fossil plants over 600 MWe was 73.1%. The capacity factor during this same period was 54.5% (Table 1). It is interesting to note that nuclear power, despite a slightly lower availability factor (a difference of .9%), showed a 12% higher capacity factor. The reason for this difference is that utilities prefer to base load their nuclear plants to take advantage of the lower cost of electricity production, using more expensive fossil units for meeting peaking loads.

If we assume a 7 mil/kWh difference in coal and nuclear generation rates, and further assume that all electricity from nuclear outage is replaced by coal generation capacity, each percent of lost nuclear capacity world wide represents an increase in cost of electricity of $32.9 million per year to the world's consumers. If each lost percent in nuclear capacity were replaced by oil, which is the more probable case, the loss to the world economy would be $138.6 million per year.

Forced outage of nuclear units in the U.S. due to material related equipment malfunction or failure represented 35.8% (3) (see Table 2) of the total outage time through January 1, 1976. If we assume that U.S. figures are representative of all nuclear plants, the cost of material related failure in increased electricity cost alone will be between $327 million and $1.4 billion in 1977. From these figures it is clear that there is a large economic incentive to solve our material related problems and increase the availability and capacity of our nuclear units.

The following compact text discusses several important materials problems facing LWR operation and some of the research programs that have been formulated at EPRI to address these problems. The problems include BWR nozzle corner cracking, and radiation induced embrittlement. The complete text will also include a discussion of steam generator corrosion and BWR pipe cracking. LWR materials problems are complex and will not easily yield to solution. However, the economic rewards for the solutions are great.

**BWR NOZZLE CORNER CRACKING**

The feedwater nozzle of a boiling water reactor forms an integral part of the reactor pressure vessel, as shown in Figure 1. This nozzle constitutes the opening through which makeup water is added to the pressure vessel. It is located below the water level and is approximately 10" in diameter. The nozzle is fitted with a thermal sleeve sparger, as
### Table 1
CAPACITY AND AVAILABILITY FACTORS 1975

<table>
<thead>
<tr>
<th></th>
<th>NUCLEAR (all)</th>
<th>FOSSILS (over 600 MWe)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Availability</td>
<td>72.2%</td>
<td>73.1%</td>
</tr>
<tr>
<td>Capacity</td>
<td>66.5</td>
<td>54.5</td>
</tr>
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</table>

### Table 2
REPRESENTATIVE AVERAGE OUTAGE DURATION AND PARTIAL POWER REDUCTION IN NUCLEAR UNITS THROUGH JANUARY 1, 1976

<table>
<thead>
<tr>
<th>ITEM</th>
<th>DURATION (hours)</th>
<th>% TOTAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>FORCED OUTAGE (equipment malfunction)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Turbine/Generator</td>
<td>140</td>
<td>5.1</td>
</tr>
<tr>
<td>Condenser</td>
<td>105</td>
<td>3.8</td>
</tr>
<tr>
<td>Steam Generator</td>
<td>120</td>
<td>4.3</td>
</tr>
<tr>
<td>Pumps</td>
<td>75</td>
<td>2.7</td>
</tr>
<tr>
<td>Valves</td>
<td>122</td>
<td>4.4</td>
</tr>
<tr>
<td>Vessel &amp; Core</td>
<td>60</td>
<td>2.2</td>
</tr>
<tr>
<td>Plant Electrical Distribution</td>
<td>90</td>
<td>3.2</td>
</tr>
<tr>
<td>All Other</td>
<td>281</td>
<td>10.1</td>
</tr>
<tr>
<td><strong>Subtotal</strong></td>
<td><strong>993</strong></td>
<td><strong>35.8</strong></td>
</tr>
<tr>
<td>SCHEDULED OUTAGE</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maintenance</td>
<td>288</td>
<td>10.4</td>
</tr>
<tr>
<td>Refueling</td>
<td>1350</td>
<td>48.7</td>
</tr>
<tr>
<td>Training &amp; Administration</td>
<td>30</td>
<td>1.1</td>
</tr>
<tr>
<td><strong>Subtotal</strong></td>
<td><strong>1688</strong></td>
<td><strong>60.2</strong></td>
</tr>
<tr>
<td>REGULATORY</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>112</td>
<td>4.0</td>
</tr>
<tr>
<td><strong>TOTAL</strong></td>
<td><strong>2773</strong></td>
<td><strong>100.0</strong></td>
</tr>
</tbody>
</table>
shown in Figures 2 and 3, that distributes the water uniformly into the reactor pressure. Various designs have been applied to this subassembly such that the thermal sleeve is fitted into the nozzle by means of a slip-fit design, an interference fit, a threaded joint, or a welded configuration. Most of the earlier units employed the slip-fit design. In addition, earlier units were clad on the inside diameter of the vessel and nozzle with stainless steel weld material.

Radial cracking has been detected on the inside surfaces of the BWR feedwater nozzle. Cracking has been identified by dye penetrant examination primarily on the ID blend radius, but in some cases, cracks extend into the bore of the nozzle itself. In a few cases, the cracking was sufficiently deep to penetrate the cladding and extend into the base material of the nozzle. To date, the maximum depth of cracking including the cladding has been approximately 3/4 of an inch. Of the 13 vessels examined to date, 10 have confirmed cracks in the feedwater nozzle. Of the 3 vessels for which no cracking was observed, one had been operated for less than one year and the other two were inspected without removing the sparger. However, the inspection by dye penetrant on these latter two vessels probably was not effective because reliable nondestructive inspection by dye penetrant techniques requires that the sparger be removed from the nozzle. Because this is a time-consuming and thus costly operation, it would be highly desirable to be able to inspect the nozzle from the vessel exterior. Many techniques employing ultrasonic procedures have been attempted and recently encouraging result are being obtained. Multiaxial and multizone inspections seem to be the best. These techniques are effective only if the crack has penetrated the cladding. Detectability limits down to 100-150 mls have been claimed. The successful implementation of ultrasonic inspections will constitute a significant improvement for inspecting feedwater nozzles.

A detailed investigation of the cracking problem has led to several recommended improvements. Initially, a detailed analysis of the cracking patterns was performed. Because of the extensive nature of the cracking and of the orientations of the cracks, it appeared that some type of thermal fatigue was involved. Consequently, measurements were made of the water temperatures in the annulus area surrounding the thermal sleeve and adjacent to the nozzle cladding interface. It was observed that two types of thermal loading were present. Firstly, a high frequency, low amplitude thermal cycling of between .25-1 Hz in frequency was present. Additionally, transient temperatures (\(\Delta T \sim 350^\circ F\)) were experienced as a result of plant start up. Furthermore, a pressure stress was present which had the effect of providing a mean stress upon which to superimpose these thermal stresses. It was determined that these loadings could be broken down into a high-cycle fatigue initiation phase and a low-cycle fatigue propagation phase. By studying the plant cyclic data observed in a variety of plant operating conditions, and then applying these data to life predictive techniques assuming a high-cycle fatigue initiation step and a low-cycle fatigue propagation step, the cracking observations can be explained.
Loose fit sparger problems

Figure 2

Typical "slip fit/interference fit" (feedwater nozzle cross section)

Figure 3
Several methods of eliminating nozzle cracking are presently being investigated. The amplitude of the high frequency ripple component can be reduced by a redesign of the sparger thermal sleeve configuration such that the annulus volume is increased, thereby minimizing the influence of leakage around the thermal sleeve. This redesign would not eliminate the thermal ripple but would tend to lower its amplitude. A second method of improvement would be to eliminate or minimize leakage into the annulus area between the cladding and the thermal sleeve. This could be accommodated by using a welded joint or a piston ring construction. The use of shrouds around the sparger inlet has also been suggested in an effort to limit the mixing problem. Furthermore, it may be possible to minimize the impact of slugging the system with cold feedwater with a system redesign using auxiliary feedwater heaters or a redesigned feedwater flow control valve. Another proposal that was shown by analysis to be of value is to remove the stainless steel cladding due to the high thermal expansion characteristics of the stainless steel weld material. All of these proposals have merit and should prove beneficial. Currently, two additional areas of investigation are being pursued. A three-dimensional elastic-plastic refinement of the combined pressure thermal loading stress analysis is being performed. Also, repair techniques including techniques of in situ removal of the weld clad overlay and remote automatic weld repair techniques utilizing procedures involving no postweld heat treatment are being examined.

**RADIATION-INDUCED EMBRITTLEMENT**

Continued assurance of safety and integrity of the primary system pressure boundary requires that the applicable materials properties be known, and the change in these properties under irradiation be accounted for through the life of the plant. One of the postulated failure modes of the primary system pressure boundary, when the metal temperatures are low during reactor startup, is brittle fracture. In a reactor pressure vessel (RPV), heavy section thicknesses, geometrical discontinuities (e.g., flanges) and reinforced penetrations cause a high degree of elastic constraint. This elastic constraint reduces the apparent ductility of the material and increases the propensity to brittle fracture at temperatures less than 120°F above the brittle-ductile transition temperature. In addition, the RPV material surrounding the core is exposed to fast neutron bombardment that increases the temperature range for brittle behavior and reduces the toughness of the material. Radiation-induced embrittlement is an extensively studied phenomenon, which impacts most significantly on older reactor vessel materials. The construction materials of the older plants have the highest levels of residual element content (including copper and phosphorous) and, as a result, have a greater radiation sensitivity and greater subsequent embrittlement. This embrittlement is manifested in a loss of ductility, a reduced resistance to fracture, and an increase in the brittle-ductile transition temperature. These manifestations may be significant to the assurance of meeting technical specification limits for some nuclear plants.
Certain observations have been made with regard to the radiation embrittlement of RPV materials.

1. The pressurized water reactor (PWR) vessels receive approximately ten times the fluence of a similar size boiling water reactor (BWR) vessel. BWR's are thus generally considered immune to any significant neutron embrittlement.

2. The accumulated embrittlement is strongly a function of residual element contents, particularly Cu and P levels.

3. The concern over neutron embrittlement is generally focused on RPV materials with low initial fracture toughness.

4. The submerged arc weldments are more sensitive to neutron exposure than the corresponding base material (plate or forging).

The potential for embrittlement in RPV materials was recognized from the outset of power reactor technology. As a result, material property test specimens were placed in capsules within the reactors to monitor any degradation caused by the reactors' operating environment. Unfortunately, the space for these capsules is limited; therefore, only a few of the materials could be monitored. The following observations can be made with regard to the overall power reactor surveillance program:

1. Because the role of Cu and P in radiation damage has only recently been defined, many surveillance programs do not include the most critical belt line material which controls plant technical specification. Prior to 1970, the role of residual elements was ignored in selection of surveillance materials.

2. In certain reactor systems, the surveillance capsule supports have been damaged due to flow-induced vibration. The capsules were subsequently removed, leaving the reactors without active surveillance programs. It has been suggested to NRC that an integrated surveillance program in which specimens from reactors without active surveillance programs be put in other power reactors for irradiation by an acceptable substitute for presently required practices.

3. The simplified dosimetry of commercial surveillance programs permits defining the fluence with a level of uncertainty of ±40%.

4. The neutron flux and spectrum at the surveillance capsule position are different from those at the vessel wall and at the one-quarter thickness position. The uncertainties introduced by extrapolating the embrittlement measured by the specimens to vessel walls are significant.
5. The specimens incorporated into the reactor surveillance program do not directly measure the property desired, that is, fracture toughness. Tenuous empirical correlations are necessary to predict fracture toughness from Charpy impact energy.

The Regulatory Position on Radiation Embrittlement

The prevention of brittle fracture of pressure boundary of nuclear systems is assured by compliance with the toughness requirements set forth in 10 CFR Part 50, Appendices G and H. In light of some of the aforementioned observations on embrittlement and surveillance, the Nuclear Regulatory Commission issued Regulatory Guide 1.99. This Regulatory Guide sets recommended limits on the increase in the brittle to ductile transition due to neutron exposure. In addition, transition temperature shift and upper shelf Charpy energy decrease prediction procedures are presented to account for levels of residual Cu and P content. The minimum upper shelf energy during the life of the plant is 50 ft-lb in order to measure the temperature shift at this level and to prevent low energy ductile tearing. If the upper shelf level limited is breached, the following actions may be required for continued licensability.

1. Extensive testing and analysis of existing surveillance specimens, where available.

2. Generation of irradiated fracture toughness data on relevant steels.

3. Extensive safety analysis of reactor vessel utilizing state-of-the-art elastic-plastic analyses.

4. Extended outages to comply with nondestructive examination requirements of 10 CFR Part 50.

5. In extreme cases, an in situ anneal or replacement of the reactor vessel may be mandated, with consequent prolonged outages.

It should be noted that such consequences can also result from inadequate data, even though the vessel may in reality be within safe limits.

Improvements and Possible Remedies

Although the embrittlement problem seems bleak at first inspection, improvements for newer generation plants have been developed and remedies for existing plants are being developed. Since the detrimental effect of Cu and P was discovered in the late 1960's and early 1970's, restrictions on the maximum content of these elements in the core belttline region imposed by the vessel fabricators have essentially eliminated any significant embrittlement problem for newer generation reactor vessels. In addition, higher toughness materials are presently required by the Nuclear Regulatory Commission for insertion in higher fluence.
regions. Therefore the principal concerns regarding embrittlement focus on operating PWR systems. Possible remedies for the existing PWR systems are listed below.

1. Develop a comprehensive irradiated materials properties data base to reduce the necessary conservatism imposed by the paucity of base data for Reg. Guide 1.99.

2. Define the fracture toughness requirements for beltl ine material to provide adequate safety margin and to evaluate current impact energy requirements.

3. Develop improved analytical treatments to reduce the uncertainty associated with extrapolating fluence measurements at an accelerated or normal capsule position to or within the pressure vessel wall.

Recent Highlights

The extent of the neutron embrittlement problem has only been appreciated for the past eighteen months. Since that time the following tasks highlight ongoing research in this area:

1. The Heavy Section Steel Technology (HSST) Program has been expanded to measure the fracture toughness of irradiated low upper shelf, high residual content weld metal on large (4 inch thick) specimens and to develop a methodology to measure fracture toughness with small specimens.

2. The Reg. Guide 1.99 data base is being reanalyzed with recently developed statistical analysis procedures to define better the effect of neutron damage on pressure vessel materials.

3. A radiation damage program is being funded by the Electric Power Research Institute (EPRI) on a series of previously characterized pressure vessel materials. The program is being developed statistically to define the change in fracture toughness and charpy values with fluence.

4. An analytical program to investigate the Charpy impact specimen in order to extract quantitative fracture toughness from the most common surveillance specimen has been initiated. This work is being funded by EPRI.

5. An EPRI research program has been formulated to evaluate thermal annealing radiation damage of sensitive pressure vessel materials. The program will establish the feasibility of such an anneal as well as provide the optimal parameters and procedures.
SUMMARY

This paper has reviewed some of the key materials-related problems that compromise the availability of nuclear power plants and the various research projects that address these issues. Given the nature of the problems under review, it is anticipated that much research remains to be done in providing fixes that will ultimately lead to successful remedies to most of these problems. Note that for the problems reviewed, the actual or projected impact on plant safety is minimal. Remedies to the problems would impact favorably on plant availability, thereby making nuclear power plants even more attractive than they have been to date. The successful and safe operation of commercial light water reactors for over 300 reactor years provides strong evidence that the metals profession has provided an adequate description of reactor materials behavior, thereby enabling reactor designers and vendors to provide the utilities, and ultimately the public, with a safe, moderately priced and reliable energy source.

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REFERENCES


