A SURVEY OF WORLDWIDE EXPERIENCE WITH THE CRACKING SUSCEPTIBILITY OF ALLOY 600 AND ASSOCIATED WELDS

by

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1. INTRODUCTION

This report was developed to address one of the recommendations put forth by the Davis-Besse Lessons Learned Task Force (LLTF). Specifically, in the words found in the LLTF Final Report (Section 3.1.1.2):

> *The NRC should assemble foreign and domestic information concerning Alloy 600 (and other nickel based alloys) nozzle cracking and boric acid corrosion from technical studies, previous related generic communications, industry guidance, and operational events. Following an analysis of nickel based alloy nozzle susceptibility to stress corrosion cracking (SCC), including other susceptible components, and boric acid corrosion of carbon steel, the NRC should propose a course of action and an implementation schedule to address the results.*

This recommendation was broken into several subtasks to address it more reasonably. One such task is- the development of this report, which is limited to a survey of nickel-base alloy cracking. Other tasks include an analysis of the control rod drive mechanism (CRDM) cracking susceptibility (time-at-temperature) model – a task that was already in the NRC's Office of Nuclear Regulatory Research (RES) OpPlan (as a result of User Need Request NRR-2002-018), which was completed in early 2003 (Ref. 1), and a forthcoming report on corrosion of pressure boundary materials in boric acid solutions (due in October, 2004).

The impetus for this report was the March 2002 discovery of the CRDM flaws, leaks and pressure boundary corrosion at the Davis-Besse plant. But, US and international reactor operators continue to discover, primarily through observation of leakage products, cracks in primary boundary piping and penetrations fabricated from wrought or welded nickel-base alloys. Events of recent note are the cracks in a medium diameter pressurizer nozzle at Tsuruga 2 and the re-cracking of the repaired instrument nozzle in the head that was just removed from Oconee 1 (Licensee Event Report (LER) 40192, described further on page 14). In the US, significant findings in components other than CRDMs have been at V. C. Summer (hot leg "A" cracks, predominately axial, but with circumferential components that, with considerable additional growth, could have resulted ultimately in a piping break); South Texas Project 1 (bottom-mounted instrumentation nozzle cracks) and North Anna 2 (well-developed, outside diameter, circumferential cracks in a penetration not exhibiting leakage products). Figure 1 is a metallograph of a flaw in Nozzle 3 – the one with the major leak - from the Davis-Besse plant. Table 1 is a compilation of vessel head penetration (VHP) cracking incidents in the US over the last three years.

2. ORGANIZATION OF THIS REPORT

There are several, comprehensive publications detailing workshop or conference proceedings, literature surveys, and other reports detailing incidents of cracking at plants. The most important of these reports, covering the period from the initial observations of alloy 600 cracking and leakage through about mid-2002, are summarized in Section 4. This report, however, does not assume the objective of compilation of all the documentation of all the worldwide incidence of primary water stress corrosion cracking (PWSCC) - the References take care of that. We have provided citations and brief summaries of these important References. Additionally, we have attempted (in the Appendix) to update the picture by pulling together useful references that describe the cracking in nickel-base alloys that has taken place from mid-2002 to the present time (March, 2004). Thus, the authors have attempted to present an overview of the history of nickel-base alloy cracking in power reactors, leading to a description of the research being funded currently (a) by NRC's Office of Nuclear Regulatory Research, (b) by the domestic industry (EPRI and the MRP (Electric Power Research Institute and Materials Reliability Program , resp.)), and (c) in foreign countries (in Sections 5, 6 & 7). Lastly, Section 9 is the authors' perspective on future developments in the occurrence or mitigation of PWSCC. The last section of this document presents the authors' views on the research needed, and the issues that the regulatory community will face in the near term. The idea is to present the 15,240 meter (i.e., the "50,000 foot")[1] view of where this agency and its licensees have been, and where we may have to go in the future.

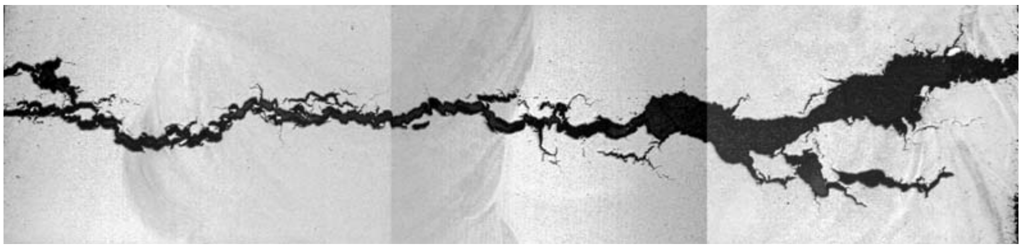

Figure 1. A metallograph showing the profile of a crack in the attachment weld of Nozzle #3 in the (now discarded) Davis-Besse reactor head (Ref. 2).

Table 1. List of VHP cracking incidents in US in last three years.

Plant	Nature of Cracking, Head Replacement Plans
Oconee 1	Leaks or cracks detected in 3 CRDMs + 5 instrument nozzles in 2000, (Head replaced in 2003)
Oconee 2	19 CRDMs repaired, 1 circ crack above the J-weld (Head replacement underway – March '04)
Oconee 3	14 CRDMs repaired, 4 with circ. cracks above J-weld, (Head replacement in '03)
Ark. Nuclear 1	8 CRDMs repaired (Head replacement in '05)
Surry 1	6 CRDMs repaired (Head replaced in '03)
North Anna 2	14 CRDMs repaired, 6 with circ cracks (Head replaced in '02)
Davis-Besse	5 CRDMs with cracks, 2 nozzles with wastage (Head replaced in '03)
Three Mile Is. 1	6 CRDMs and 8 instrument nozzles plugged (Head replaced in '03)
Crystal River 3	1 CRDM with circ. crack (Head replaced in '03)
Ginna	Head replaced ('03)
Millstone 2	3 control element drive mechanisms (CEDMs) repaired, (Head replacement in ('05)
South Texas 1	2 lower head nozzles repaired
Heads to be replaced at: Kewaunee ('05), Robinson 2 ('05), Pt. Beach 1 ('05), Farley 1 ('04) Pt. Beach 2 ('05), Farley 2 ('05), Prairie Island 2 ('05), St. Lucie 1 ('05), St. Lucie 2 ('06), Turkey Pt. 3 ('04), Turkey Pt. 4 ('05), North Anna 1 ('03), Surry 2 ('03)	

[1] Dimensions in this report are given in a metric equivalent, followed by the English measurement in parentheses. In all cases, the measurements were made, or provided, in English units, and converted to their metric equivalents.

During the course of this literature review, it became obvious quickly that many previously published literature surveys, plant condition surveys, workshops and conferences had centered on the subject of Alloy 600 and compatible weld-metal cracking in primary-side, reactor coolant environments (PWSCC – primary water stress corrosion cracking). These documents, and the references in them, provide a broad view of the inspection, stress calculations, cracking and repair of cracked nickel-base alloy components. Therefore, the scope of this particular document was revised, as compared to the LLTF recommendation. This document attempts to summarize the past experience in a way that inferences can be made regarding future findings in alloy 600 components.

3. HISTORICAL BACKGROUND

Nearly all reviews of PWSCC in Alloy 600 begin with the mention that this phenomenon was observed first (Ref. 3) in 1959, by Henri Coriou (in the laboratory now known as Commissiarat à l'Energie Atomique (CEA)-Saclay), in pure, deoxygenated (~ 1MΩ, $\leq$3 ppb O_2) water at 350°C (662°F). Tests were also conducted in water containing 1 ppt (part per thousand) chloride as sodium chloride. Cracking at 0.5% permanent strain was exhibited by Alloy 600 after nine months in the pure water, and after six months in Cl-contaminated water. By the early 1970's, the basic dependencies of Alloy 600 PWSCC on carbide precipitation, temperature and levels of minor contamination were well known. Most of the early work was propelled by cracking in steam generator tubing. Of particular importance is the work by Copson and Dean (Ref. 4), who studied effects of contaminants, especially lead, on initiation times and rates of crack extension.

Initially, the work of Coriou was challenged by other experimentalists (largely American) who had had no success in cracking Alloy 600. These detractors suggested that inadvertent, and immeasurably-low levels of contamination may have been responsible for the cracking. Repeated testing, and subsequent documents authored by Coriou (Ref. 5) assert that the CEA autoclave environments were as pure and carefully maintained as was possible. Coriou attributed his success in obtaining cracking results to the relatively high levels of strain in the test coupons, as well as a willingness (and ability) to allow tests to run for many consecutive months. Ultimately, many others were able to obtain similar indications of cracking, and the results of Coriou, together with other tests from the 1960s time frame, led to recognition of the effects of strain, temperature, microstructure, yield strength and certain environmental parameters, including oxygen level, and contaminants such as lead and chlorine. Not every one of the conclusions resulting from this work remains completely intact over the years, but a remarkably clear picture of the important parameters was developed by the CEA group. By the early 1980s, cracking of steam generator tubing, first on the secondary side, later from the primary side, provided notice that Alloy 600 would crack in field applications, given a high enough stress, and a substantial incubation time.

The tests of Coriou, and his contemporaries, are best characterized as crack initiation tests. For the most part, the test samples were small flat beams or rings, strained by clamping mechanisms, or tie rods, and contained in autoclaves without any additional instrumentation to monitor crack extension, strain relaxation, or other means of monitoring crack nucleation, or degradation. By the early 1990's, testing with fracture mechanics specimens, usually of the compact design, together with very accurate and stable crack extension measurement systems, had been established in a number of laboratories worldwide, and well-characterized crack growth rate data became available (Ref. 6).

4. LITERATURE SURVEYS AND CONFERENCES ON CRACKING IN ALLOY 600 COMPONENTS

Cracking in primary side water of thicker sections of Alloy 600 was heralded domestically by the discovery of leaks in pressurizer instrument nozzles (1986 at San Onofre Unit 3) and heater sleeves (in 1987 at Arkansas Nuclear One). In 1989, 20 leaking heater sleeves were found at Calvert Cliffs Unit 2 (8 additional had non-leaking, axial crack indications). A 1990 literature survey documents the history of this and similar cracking discovered in both domestic and French plants (Ref. 7). At this stage of understanding, incidents of field cracking were strongly correlated with temperature – pressurizer penetrations are generally exposed to a higher temperature than are CRDM penetrations and other components. Figure 2 is a timeline of the significant findings in reactor plants, along with the conference and reporting milestones of significance.

The 1990 report produced the correct conclusion that susceptibility of alloy 600 was related to its thermomechanical processing. Specifically, a low temperature final anneal, producing intragranular carbides and a yield strength at the high end of the specification, coupled with a fabrication process, such as cold work or welding, that imposed a high residual stress, led to a high susceptibility to crack initiation and growth.

The 1991 Workshop (Ref. 8) brought together about 40 engineers, from Japan (1), France (2), and the US (balance), and included over 30 presentations, characterized by an active question and answer session focused on the specifics of plant experience and specific incidents. In addition to these sessions on plant experience, other sessions covered inspections, repair strategies, testing and analysis, modeling and outage planning. One presentation [J. F. Hall, ASEA-Brown Bovari-Combustion Engineering (ABB-CE at that time)) described test results to evaluate the corrosion of low-alloy steel in concentrated solutions of boric acid. This research program by the Combustion Engineering Owners' Group (CEOG) was stimulated by NRC's Generic Letter 88-05 on that topic. This 1991 workshop included one of the first presentations describing the stress distribution in the vicinity of a vessel penetration with a J-groove weld attachment on the coolant side. The importance of temperature, and material microstructure were confirmed and described further. Other firsts include the identification of dissolved hydrogen level as an important factor; the influence of Li/B and pH was cited as uncertain.

Nearly ninety engineers representing seven countries attended the 1992 Workshop – a reflection of the growing concern worldwide. Over 50 presentations (Ref. 9) provided a much greater level of detail than had previously been described. Due to the discovery of CRDM cracking at Bugey 3, the focus of the 1992 conference shifted from pressurizer penetrations to reactor vessel penetrations. The French reported that about 3% of the nozzles inspected had flaw indications. Stress analysis and characterization of materials, including weld metals, was much more extensive than in the 1991 workshop.

A presentation at the 1992 workshop described the destructive examination and the findings of eleven axial cracks, including one leaker, on the ID of one nozzle at Bugey 3. The leaking crack in Nozzle #54 - one of the outermost locations - was about 50 mm long on the ID, and about 2 mm long on the OD. At that time it was unequivocally stated that no circumferential cracks had been observed. A second throughwall crack was found in the same CRDM, but on the uphill side, and below the weld. Axial cracks were found in one additional nozzle at Bugey 3; the remaining 63 nozzles were determined to be not cracked. At the sister plant, Bugey 4, cracking with an axial orientation, but no leaking was found in eight nozzles.

Examinations of the cracked Bugey-3 CRDM housing continued for a long time after, and were summarized in Ref. 9. There was one, small (3 mm long x 2.25 mm deep), indication on the outside of the nozzle, above the weld, at an angle of 30° off the vertical axis. It was initially postulated that this crack was a branch off the leaking axial crack. However, subsequent presentations by involved French engineers (in Reference 10) suggest that this may have been a crack that nucleated independently of the main, axial throughwall crack. There was also a 110° x 3.5 mm deep circumferentially-oriented flaw in the weld, on the uphill side, that emanated from the “triple point” (the point at which the low-alloy base metal, the J-groove weld and the CRDM tube all come together), and propagated into the buttering and the J-groove weld.

The 1994 workshop on PWSCC of Alloy 600 (Ref. 10) was attended by more than 110 representatives, and expanded on the same topics as were incorporated in the 1992 workshop. In the two-year interval between the 1992 and 1994 workshops, CRDM inspections in several nations (i.e., vessel heads fabricated by several vendors) found mostly axial cracks. Some plants with up to 20 years of operation did not exhibit flaw indications in the CRDMs, but one plant indicated that a pressurizer steam-space instrument alloy 690 nozzle, welded with alloy 182 (as compared with the more common alloy 152) produced cracking in the weld. The proceedings report that vessel head replacement began in 1993/1994. Six vessel heads were replaced on French plants during those two years, and 29 more vessel heads were placed on order. Plans were announced for replacement of “two vessel heads in Spain, one vessel head in Sweden and three vessel heads in Japan”. It was reported that computational stress analyses had shown that (a) the development of welding-induced residual stress was dependent on the fabricator, that stresses were not always higher for the downhill nozzles (those located toward the periphery of the head), and (c) that butt weld stresses varied substantially around the circumference, and depended on the specific welding procedure. It was also reported that the PWSCC susceptibility of weld metal was a function of chromium content, and that welding alloys with greater than 30% Cr were resistant to PWSCC.

A description of the head penetration inspection experience at D. C. Cook 2 plant was presented at a follow-on meeting after the conclusion of this workshop. In 1994, three axial flaws were detected in an instrument penetration (#75) after 9.88 EFPYs at 604°F (equivalent to ~13.9 EDY). The flaws were axial in orientation, with lengths of 45, 16 & 9 mm, and a maximum depth of 6.8 mm (compared with a wall thickness of 0.625 in. (16 mm). All flaws were below the weld, although the largest of the flaws was very close to the bottom of the weld. A fracture mechanics evaluation indicated that growth of these flaws would not exceed 75% of the wall thickness during the next operating cycle. They were left in place, and repaired two cycles later, in 1996.

A 1995 conference (Ref. 11) was held under the auspices of the International Atomic Energy Agency, but was essentially organized by NRC staff. Thirteen countries were represented, and seventeen papers presented, although less than 40 engineers were able to attend. This conference was one of the first opportunities for the NRC to present publicly the position of the agency *vis a vis* the safety significance of the CRDM cracking issue. J. Strosnider took the opportunity to point out that leakage and ensuing deposits of boric acid on the head could lead to high corrosion rates. He also suggested that the unique stress distribution developed by the

(Text continued on page 7)

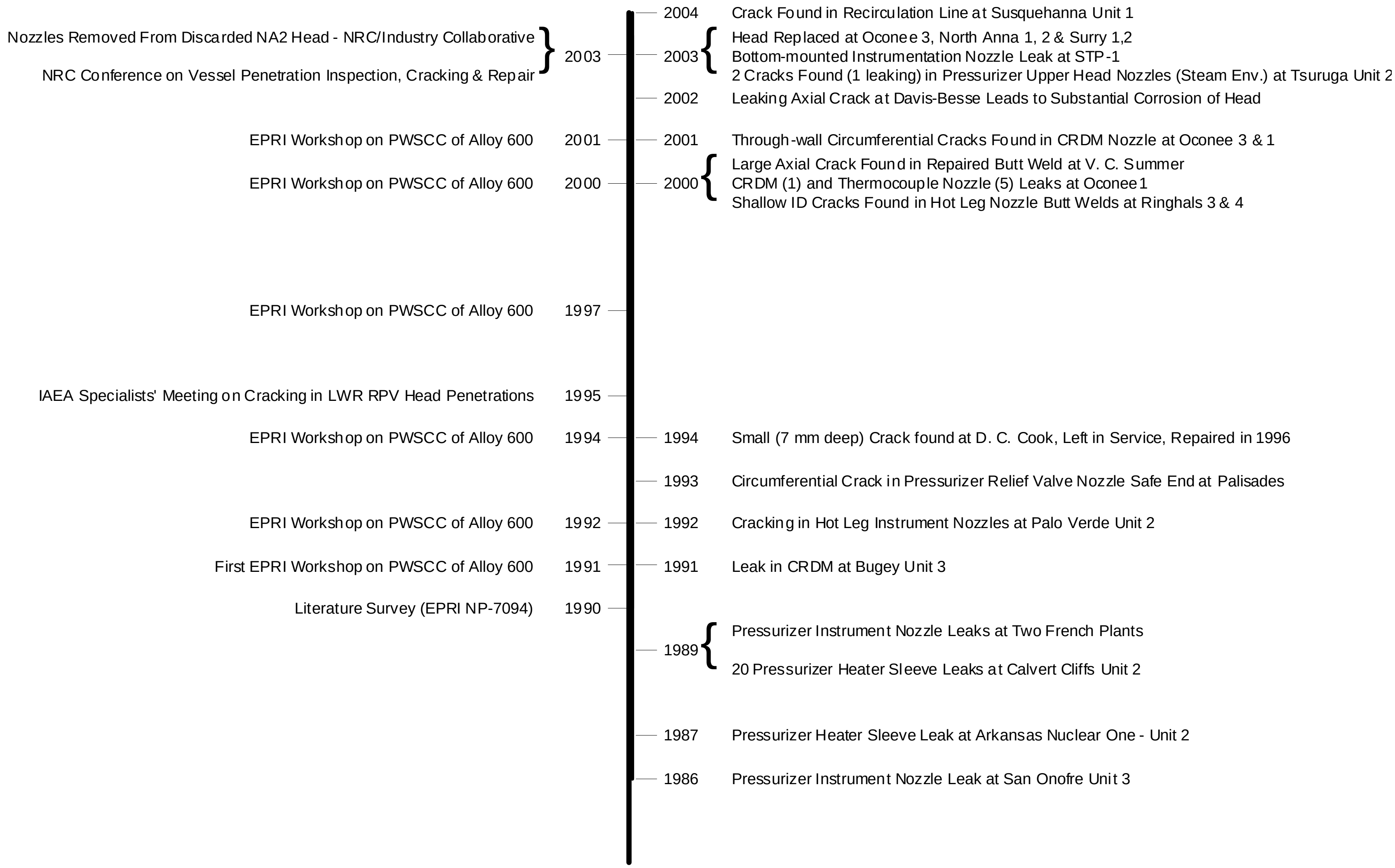

Figure 2. Timeline of findings of significant Alloy 600 cracking in NSSS plants, and important documentation, workshops and conferences held to discuss this issue.

attachment weld could lead to the possibility of circumferential cracking. This conference was also the forum for the IAEA to describe the development of an "International Database on Aging Management". Participants from Spain, France, Korea, Japan and the USA gave summaries of the vessel head cracking situation, or the vessel head replacement program in each of their countries. The results of 3-D modeling of the residual stress distribution for a J-weld attachment were the subject of two presentations.

Slightly over 100 engineers attended the 1997 workshop (Ref. 13), which was held in the shadow of the imminent issue of Generic Letter 97-01 requiring licensees to describe their plans for vessel head inspection for CRDM cracking. There had been no significant changes in the rate of incidence of CRDM cracking (in other parts of the world), and one occurrence in the US (D. C. Cook, which was repaired in 1996), between the 1994 and 1997 conferences. The French plants had entered a replacement program by this time, and cracking of pressurizer penetrations in domestic plants continued to be an economic, but not a safety problem. Nonetheless, the NRC was about to require (in Generic Letter 97-01) that plants develop a long-term plan for inspection and monitoring for PWSCC of vessel head penetrations.

Three years elapsed before the next industry-sponsored workshop, in 2000 (Ref. 14). The opening presentation, by Warren Bamford, pointed out that reactor pressure vessel (RPV) head nozzle cracking, or flaw indications had been found in eleven countries. Cracking in French plants had been found in about 6.5% of the nozzles inspected. Eight inspections of bottom-mounted instrumentation tubes in four countries had turned up no evidence of cracking at the time of the conference. Cracking in several, repaired pressurizer steam space nozzles, and laboratory testing of alloy 600 and several variations of alloy 81 (viz. alloy 182, EN 82H) showed that crack propagation rates in the weld metal were higher than in base metal.[2] Some research on crack sizing, and on mitigation of crack nucleation (by water peening, shot peening, nickel plating and others) was presented at this workshop.

The 2001 workshop (Ref. 16) was held following the discovery of several instances of leaks in domestic reactors. Leakage from CRDMs had been found at Oconee 1 (Fall 2000 outage), followed by Oconee 3 (early 2001 maintenance outage), and Oconee 2 (Spring 2001 outage). A leaking crack was also found at Arkansas Nuclear Unit 1 in February 2001. The hot leg "A" crack had been discovered at V. C. Summer in October 2000. Detailed presentations by all the affected licensees described their inspection findings and the repair procedures in which they engaged. The sense of urgency created by these findings provoked discussions on whether leakage would always be present (the industry position was affirmative), and whether cracks would become a safety concern before they would be discovered by inspection. The importance of maintaining a clean head was particularly stressed in a presentation on the Oconee findings. J. Strosnider gave the regulatory perspective on CRDM penetration cracking, stressing that this issue was attracting senior management attention at the NRC, and that inspection based on leakage detection alone may not constitute compliance with the regulations.

Recently (June 2003), EPRI has distributed MRP-87 (Ref. 17), a report that indexes and describes cracking of alloy 600 components other than CRDM penetrations, and other than steam generator tubing. The scope of this report encompasses pressurizer nozzle and heater

[2] By mid-2003, a sufficient tabulation of crack growth rates for both alloy 600 and alloy 182 allowed an evaluation showing that SCC growth rates in alloy 182 weld metal were a factor of 2.7 faster than rates in alloy 600 in well-controlled laboratory tests. A carefully constructed non-linear regression fit to this data, and elimination of the (artificial) "threshold" of 9 MPa√m that was used for the alloy 600 curve, produces a curve that is virtually indistinguishable (for K>20 Mpa√m) from the 5X "Scott" curve for alloy 182 that was proposed in MRP-21 (Ref. 14)

sleeves, small diameter instrument nozzles, coolant loop butt welds and similar applications of nickel-base alloy tubing and attachment welds. This report covers all domestic incidents through June 2002. The summary of the report alludes to a forthcoming parallel report that summarizes CRDM cracking, but as of December 2003, that report had not been issued. The stated basis for the MRP-87 report was to "compile the information necessary to understand the conditions under which PWSCC occurs . . ."

MRP-87 presents a comprehensive survey of licensees and vendors, and a tabulation of locations in each type of PWR plant in which alloy 600 or alloy 182 may be found exposed to the coolant environment. The document lists all of the Alloy 600 and associated weld cracking incidents for domestic plants beginning with the instrument nozzle and heater sleeve cracks found in the 1986-1987 time frame. Appendices in this report list each observed incidence of cracking, giving a description of the flaw indication, an inventory of the design details and materials involved, a description of the type of inspection that was used to discover and size the indication, and the repair procedure, or other response to the disposition of the flaw.

After a two-year hiatus from conferences and workshops, the NRC, with the assistance of Argonne National Laboratory, held the Vessel Penetration Inspection, Crack Growth and Repair Conference on September 29 - October 2, 2003. This conference brought together participants with interests spanning all four of the areas described above – inspection, crack growth, stress analysis and mitigation - and offered an opportunity for all participants to bring themselves up-to-date on the status of NDE, crack growth, stress analysis and mitigation programs worldwide.

The four-day conference was concluded on October 2, after hosting 220 registrants, and featuring 46 presentations by regulatory agency staff, industry representatives, and researchers worldwide. The conference was divided into five sessions, highlighting (a) Inspection, (b) Continued Operating Experience, (c) Stress Analysis and Structural Integrity, (d) Crack Growth of Nickel-Base Alloys, and (e) Mitigation of Cracking and Foreign Experience. The conference participation of the US industry was particularly encouraging, encompassing nearly half (22) of the presentations. Many of the papers focused on recent developments in the areas of inspection, understanding of the mechanisms of crack growth, and the mitigation of crack growth. The timeliness of the presentations was exemplified by three contributions by the staff at South Texas Project, describing elements of the discovery and repair of their bottom-mounted instrumentation (BMI) nozzles, and a (literally) up-to-the-minute presentation describing the findings about the pressurizer nozzle leaks discovered three weeks earlier at the Tsuruga plant.

The presentations in the lead-off technical session on non-destructive inspection technology noted that pure water stress corrosion cracking (PWSCC) of nickel-base alloys is a complex problem, and that few of the key factors are fully understood. The session on continued plant operation afforded both a look back, at the South Texas Project finding, which is also a look ahead, since many in the reactor safety assurance business feel that this may be one of the next, important generic issues. Other presentations offered approaches to repair and inspection that anticipated potential problems that plants may encounter. The session on structural analysis and fracture mechanics contained four presentations describing various approaches to finite element analyses of reactor components, as well as other presentations on specific studies of cracking phenomenology. The largest of the sessions included eleven contributions describing the results of crack growth rate studies and assessments of the micro-mechanisms of stress corrosion cracking. In particular the dependencies of crack growth rates on microstructural characteristics, and on the electrochemical state, including film development, of the material-environment couple were described. The final session contained presentations that described the foreign experience with nickel-base alloy cracking, and the regulatory approaches used in France, Sweden, Germany and Japan. The session concluded with four papers describing approaches to mitigation of cracking.

The conference presentations and discussions provided a comprehensive basis upon which the NRC may be able to build regulatory actions in the near term, as well as prepare itself for issues in the future, such as those that may pertain to replacement components. Of particular note, this conference featured corresponding presentations by NRC- and industry-funded contractors in the area of inspection, and crack growth rate applications. The opportunity to have these pairs (or more) of presentations juxtaposed allows a more clear view of the similarities and differences of the respective approaches. The proceedings of this conference will be distributed in early January, 2004, and will contain the handouts, full, written contributions from each author, and transcriptions of the session summaries. The next conference in this same topical area will be hosted by EPRI, and has a tentative schedule of spring, 2005.

5. RECENT CRACKING AND LEAKAGE EVENTS

The NRC Conference held from September 29 – October 2, 2003 included presentations that described several recent alloy 600 cracking events, the Appendix of this document contains an up-to-date summary of findings gleaned from LERs submitted by US licensees. The LER citations have been included to give a more complete picture, and to expedite traceability in the future. One theme that emerges from this compilation is that in PWR plants, cracking has progressed from pressurizer penetrations (beginning in the mid-1980s), to CRDM's (in the 90s), hot legs (late 90s), to BMIs (2003). Generally speaking, this progression follows a pattern of successive occurrence components operating at progressively lower temperatures. In BWRs, findings of cracking and leakage are just beginning to emerge. Most of the cracks or leaks found to date have been in recirculation lines, in which thermal fatigue may be a component of the driving force, and in core shroud supports. It is too early to tell whether a pattern exists that suggests locations of future cracking in BWRs.

6. CURRENT RESEARCH SPONSORED BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH

The authors believe that it is important to develop a perception of the trends of the nickel-base alloy cracking issue, and equally important to present an overview of the research being conducted to address these problems. This section presents an overview of the programs funded by (a) the NRC's Office of Nuclear Regulatory Research, and (b) the Electric Power Research Institute, and (c) a synopsis of research being conducted in several foreign countries.

The RES programs covering the scope of flaw evaluation for light water reactor applications are characterized by studies of (a) non-destructive inspection technologies, (b) materials properties and crack growth rates, and (c) stress and structural integrity analysis. These three diverse technologies have to be tied together in order to provide an accurate understanding of the crack nucleation and growth phenomenon, and from that, to predict appropriate intervals for inspection, or in some cases, computation of times to leakage (i.e., for a crack to go throughwall, be it in a steam generator tube, or a vessel penetration). Figure 3 is a schematic showing how these disciplines feed into the overall RES materials degradation program, which encompasses alsoa wide range of degradation issues that apply to light water reactors, including vessel internals, and steam generators.

These specific studies are part of a suite of RES programs that were developed over the years in response to regulatory needs. These studies were further shaped and evolved to complement the results that emanated from domestic and international programs (See Sections 7 & 8) funded by a variety of sources, including EPRI and the US industry, the nuclear Navy, the owners groups, and a long list of foreign vendors and foreign government regulatory agencies.

Reactor structural designs present significant challenges for inspection, be it visual, acoustic (ultrasonic), or electromagnetic (usually eddy current). The very thick sections and enormous area of the pressure boundary, the wide variety of materials, and the omnipresent radiation fields make the conduct and interpretation of inspections time-consuming and expensive. Many of the materials are welded joints, or cast products, in both of which the metallurgical grain size is of the same order of magnitude as the acoustic wavelengths used in ultrasonic examinations. This causes a kind of acoustic resonance, producing echos containing a large component of uninterpretable noise that can mask the acoustic returns from a flaw of important size. RES is funding research into several, promising, sophisticated non-destructive techniques, such as the use of phased arrays of transducers, and the synthetic aperture, focusing technique (SAFT) both of which have the capability of eliminating the noise using computational techniques. An example of a SAFT processed image from a coarse-grained nickel-base weld is shown in Figure 4. Development of these advanced techniques is also underway for reactor vessel structures, reactor internals, and steam generator tubing.

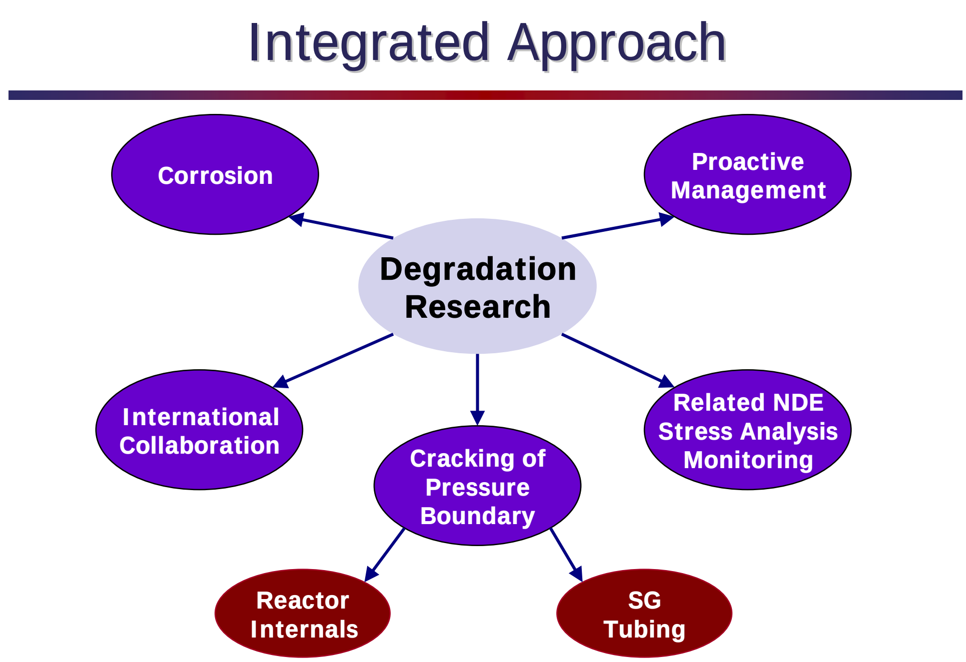

Figure 3. Schematic of the linkage among the various disciplines comprising RES programs in light-water reactor materials degradation.

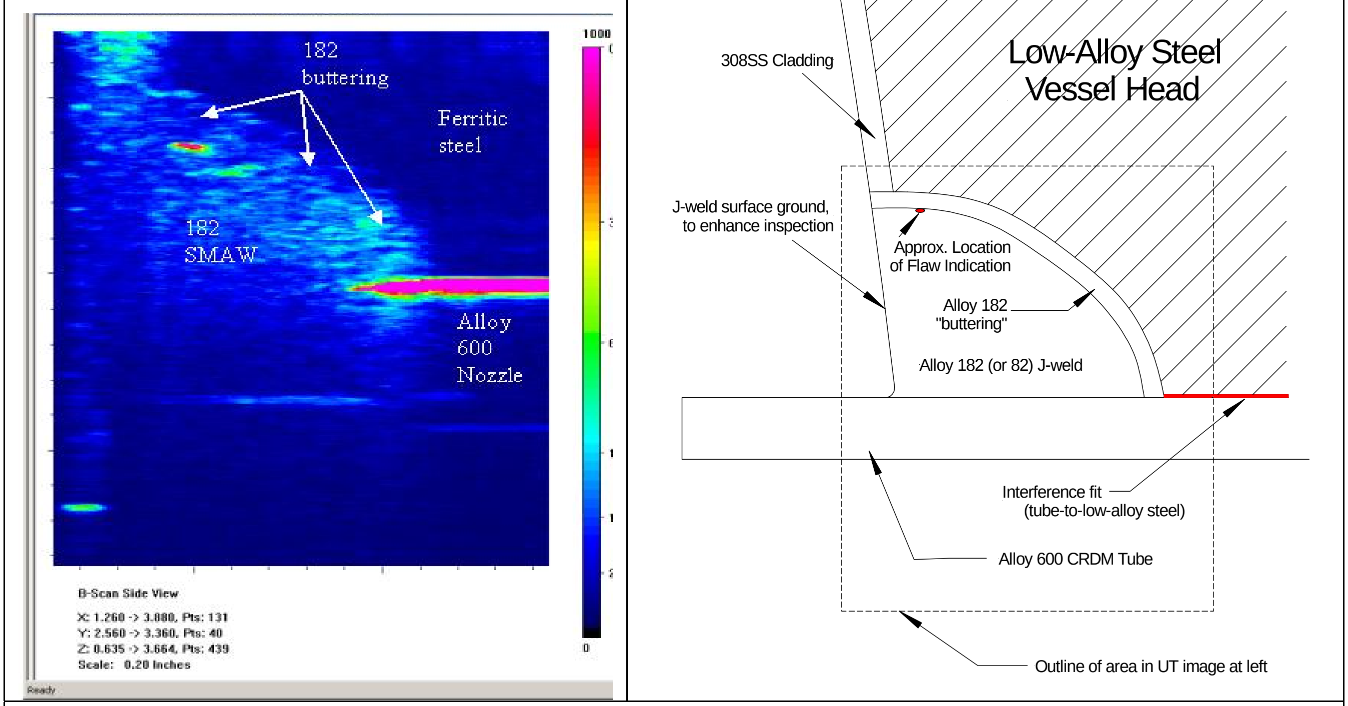

Figure 4. SAFT processed ultrasonic transducer image of the "buttering", J-weld, and CRDM tube. The ultrasonic image on the left was taken with a 10 MHz, spherically focused, longitudinal wave transducer. The guide at the right shows the important material components that can be imaged with this technique. The red "splotch" in the upper, left center is a manufacturing flaw in the weld, probably about 1 – 2 mm in size.

As detailed in Section 2 – Historical Background, cracking of Alloy 600 in high-temperature, simulated reactor coolant was noted by Henri Coriou in 1959 and published two years later. Experience with Alloy 600 in steam generator tubing in the 1970s and 1980s proved that cracking was dependent on temperature and microstructure. Crack nucleation and crack growth in Alloy 600 has become quite well understood at this time, with a substantial data base compiled in the last twenty or so years. The NRC's Office of Nuclear Regulatory Research (RES) has been a significant participant in this effort, having funded much research at Brookhaven Nat. Laboratory and Argonne Nat. Laboratory (ANL), in the areas of stress-corrosion crack (SCC) nucleation and growth, and non-destructive inspection of steam generator tubing, and more recently, at ANL, studying SCC growth rates in thicker sections of Alloy 600 and Alloy 182.

It is important to understand the crack-tip mechanisms that result in (a) intergranular SCC, (b) irradiation-assisted SCC, (c) corrosion fatigue crack growth, and the many subforms of these major categories of crack growth. RES has funded laboratory, autoclave tests of irradiated and unirradiated pressure boundary, reactor internal, vessel penetration and steam generator tubing materials for nearly thirty years (since 1975), as well as the adjunct piping integrity studies during much of the late-1970s and 1980s. Currently, RES is funding crack growth rate research at ANL, and also through a co-funded international program, the Cooperative IASCC Research (CIR) group.

Crack growth rate testing in simulated reactor coolant environments is a very challenging, and exceedingly time-consuming procedure. Measurement of a single crack growth rate data point in the 10^{-11} to 10^{-9} m/s range requires several weeks of uninterrupted high-temperature, high-pressure exposure in simulated PWR or BWR coolant. During this time, load on the specimen must be held constant, and the environmental chemistry must be carefully controlled. This environmental control must be coupled with extremely precise crack extension measurement instrumentation, capable of measuring, precisely, 0.02 mm of crack extension, without drift or any systematic error. Also, such tests must begin with a conditioning (sometimes called “coaxing”) phase, usually of about three weeks, in order to achieve equilibrium of the electrochemical environment, and the establish the desired mechanism of crack extension.

The third component of flaw evaluation in reactor components is stress analysis. Fortunately for this particular discipline, the development of high-speed and relatively inexpensive computers has made it possible to obtain results of finite element analyses that were unthinkable only twenty years ago. RES-funded research aimed at computation of the residual stress fields in J-welds for CRDM attachment has produced algorithms that simulate the welding process bead-by-bead, the cooling and contraction of those beads, including the continuous sampling of temperature-dependent stress-strain curves in order to calculate properly the strain state after that bead has cooled to room temperature (See Figure 5). The finite element analysis programs calculate stresses and strains, resolved into the three principal directions, and show that the highest strains are (a) at the root of the weld, or “triple-point”, and (b) near the location of the final pass. The way in which such a pattern builds up, during the course of the welding process, is shown in Figure 6.

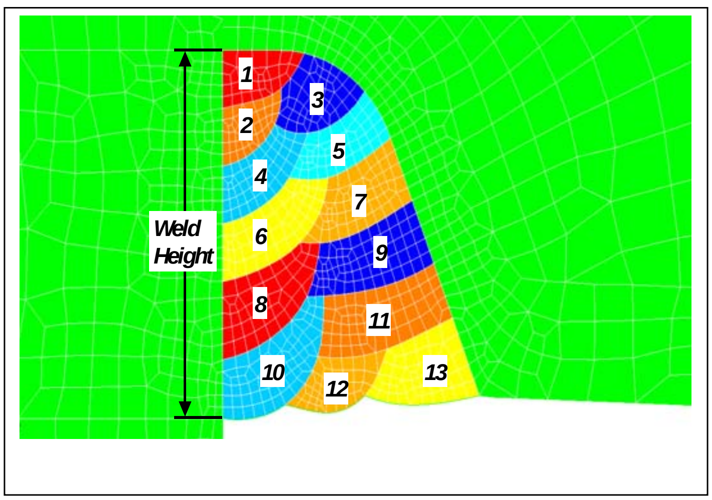

Figure 5 This figure shows the finite element grid, and the bead-by-bead pattern of weld deposition that is modeled in order to calculate the residual stresses that are developed around vessel penetration attachments.

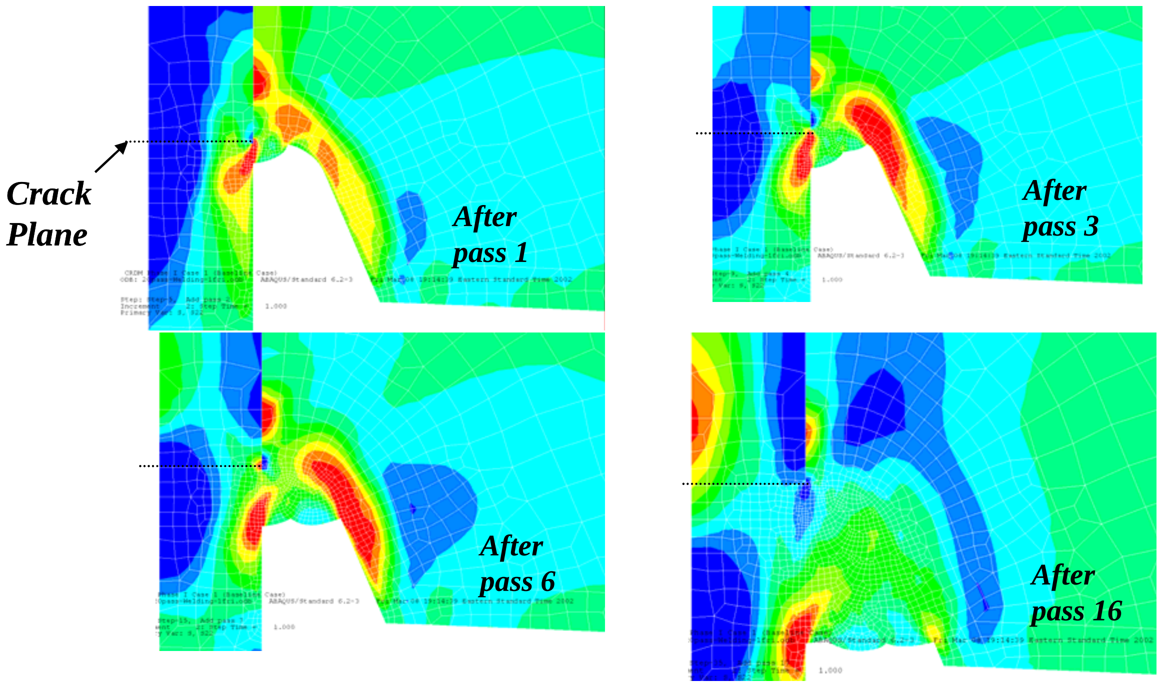

Figure 6. This figure shows the development of the axial residual stress as the beads are laid down during the assembly of a penetration attachment. It is the axial stresses that would produce circumferential cracks, which, if located above the J-weld, could possibly terminate in ejection of the CRDM housing. The regions in red indicate a tensile stress that is in excess of that generally found in Alloy 600 CRDM tubing.

Under the present terms of the inspection orders and bulletins, most licensees with plants deemed as moderately or highly susceptible to cracking have made the decision to replace their vessel heads. Therefore, flaw evaluations for the long term (for vessel head penetrations) are a somewhat academic exercise at this time. However, flaw evaluations for other components (safe-ends, small bore nozzles and piping, as examples) is a reasonable possibility. Evaluation of flaws in replacement heads, usually fabricated with nozzles and welds of the far more crack-resistant materials Alloy 690 (for the CRDMs) and Alloy 152 (for the attachment welds) is an even more likely possibility. To date, Alloy 690 has exhibited a very encouraging resistance to crack nucleation, and if problems develop in the replacement heads, they are more likely to be found in the J-weld than in the parent metal. Even so, the crack growth rates in Alloy 152 are expected to be much less than those currently established for Alloy 182 (which are about 4.2X those for Alloy 600). Thus, the incentive to develop methods of inspecting the welds, and dispositioning the potential for crack growth in welds, including the stress analysis for Alloy 152 welding techniques, establishes a very, very important goal for RES.

7. RESEARCH BEING CURRENTLY FUNDED BY US INDUSTRY

Most of the research into cracking of nickel-base alloys is being coordinated and funded through the EPRI MRP (Materials Reliability Program). The structure of the MRP activities related to the alloy 600 issue was guided through 2002 by three working groups: (a) Butt Weld Working Group; (b) RPV Head Working Group, and (c) Inspection Working Group. In 2003, a fourth entity, the Mitigation Working Group was added. The activities of these groups were summarized by Hickling (Ref. 18). Through 2002, the Butt Weld Working Group focused on stress analysis of butt welds in PWRs, and a methodology for deferral of augmented inspections of butt welds. This group also supported the activities of an expert review panel of alloy 600/182/82 crack growth rates. The principal product of this group was the MRP-55 document on crack growth rates in alloy 600 (Ref. 6). The RPV Head Working Group concerned itself with probabilistic fracture mechanics treatments of CRDM circumferential cracks, and risk assessment of the alloy 600 cracking issues. The Inspection Working Group addressed improvement of non-destructive inspections of butt welds, including development of a data base of inspection results. This included a Performance Demonstration Initiative (PDI) for the UT and ET of butt welds. This group will soon issue MRP-110 (Ref. 19), describing the technical basis for an upper head penetration inspection plan for PWRs. The industry group expects to publish the inspection plan *per se* in early summer 2004.

In response to the rapidly growing regulatory, industry and public concern about cracking in nickel-base alloys, the activities of these three groups were widened in 2003. To the core activities mentioned above, the Butt Weld Working Group added tasks associated with the safety assessment of dissimilar metal butt welds, and consideration of issues with smaller lines. The RPV Head Working Group took on three, new and significant efforts: (a) development of a Failure Modes and Effects (FME) flow chart, an extensive boric acid corrosion testing program, and salvaging (and ultimately the inspection of) six nozzles from the discarded North Anna 2 Head (See Section 8 for more information). The activities of the Inspection Working Group were expanded to include inspection demonstrations of thermal imaging technologies, and the qualification of new vendors for both UT and ET of vessel penetrations. Two new efforts were initiated by the IWG: (a) development of procedures for inspection of bottom-mounted instrumentation penetrations, and (b) development of a plan for the inspection of replaced heads with alloy 690 penetrations. In 2003, the Mitigation Working Group was formed, and took on the evaluation of several types of mitigation: (a) mechanical methods (Mechanical Stress Improvement Process (MSIP), and water peening), (b) reduction of crack growth rates (largely through zinc additions, or upper head temperature reduction), (c) chemical methods (the most efficient of which is (improved) hydrogen control), and “novel” methods, which is largely unspecified at the moment. The Alloy 600 Industry Working Group also coordinates other efforts which are not specifically aligned with the working groups responsibilities described above. These include oxide film analysis using Raman spectroscopy, development of predictive cracking models, and consideration of the basis for revision of the EPRI water chemistry guidelines.

Some additional work is being undertaken by vendors and owners’ groups, notably some alloy 600, 690 and associated weld materials crack growth rate testing underway in the Westinghouse Science and Technology Center.

Mitigation Efforts Undertaken by Licensees

The last aspect of flaw evaluation that should be mentioned in this context is the mitigation of flaw growth. There are several approaches to slowing the nucleation and growth of cracks, including temperature reduction, chemical additions (usually zinc), and careful control of the electrochemical potential (usually by maintaining hydrazine additions at the high side of the water chemistry guideline). Vessel upper head temperature reduction (UHTR), managed by diverting some of the cold leg return flow into the upper dome, has been applied to a number of US as well as non-domestic vessels. Several licensees are adding zinc to the primary water, in quantities in the 10 to 20 parts per billion range, which is allowed within the EPRI water chemistry guidelines, and most plant technical specifications.

8. FOREIGN EXPERIENCE, RESEARCH, AND ADDITIONAL INVESTIGATIONS

Japan

In Japan, vessel heads with alloy 690 penetrations have replaced the original heads at 12 of the 23 PWRs. T-cold conversion (i.e., UHTR) has been implemented at the remaining 11 plants. Six of these 11 have undergone eddy current inspections within the last ten years, and no flaw indications were found. Eddy current inspection and water-jet peening of the BMI nozzles in seven plants has been completed at the present time. This technology is being developed for CRDM penetrations. Water-jet peening technology is also being developed for safe-end nozzles. Two, large nickel-base alloy SCC programs are underway. The Electric Joint Research Project includes tasks to evaluate SCC and SSRT on Alloys A600, Alloy 132, 82, TT690, Alloys 152 & 52. The National Nickel-Based Alloy Material Project includes tasks to evaluate SCC on Alloys MA600, Alloy 132, 82, TT690, Alloys 152 & 52. Both of these programs are long-term, with ending dates in the 2005/2006 time frame.

France

In France, the discovery of CRDM cracks in 1991 led to analysis of the effects of temperature and water chemistry, the effects of stress, and the factors behind the susceptibility of nickel-base alloys to SCC. However, the early decision to replace heads allowed French to focus on repair strategies for the relatively short term, and inspection programs for the heads that had not been replaced. Development of a blade probe (by 1994) allowed inspection of CRDMs without removal of the thermal sleeve. On the heads with the highest susceptibility (in terms of yield strength of the alloy 600, and the vessel head operating temperature), N^{13} leak detection systems were installed. This consisted of a bell-like structure that was placed directly over a group of suspect CRDMs, to funnel any emerging N^{13} into the detector. The use of these was discontinued after a short time due to analysis of the issue, showing that failure of a CRDM (i.e., ejection) was a slowly-developing process, and also because the sensors were creating a significant number of false alarms.

In 1992, the French developed a methodology to compute the minimum time for crack initiation in alloy 600. This index is relative to a “base” case of a “sensitive” heat of alloy 600 that would produce indications of cracking in laboratory experiments at 10,000 hours. Three factors are combined in the index, i_m, i_θ and i_σ, ($0.0 \leq i \leq 1.0$) representing factors describing material susceptibility, temperature and stress, respectively. The factors, *i*, are calculated for a given CRDM, to have a maximum value based on the best knowledge of susceptibility, temperature and stress, and a predicted time to failure is computed using:

$$t_f = 10{,}000/(i_m \cdot i_\theta \cdot i_\sigma,) \qquad (1)$$

The parameters in this formula, including the factor of 10,000, were designed to predict the minimum amount of time required to develop a crack 2 mm in depth, or 5% of the wall thickness, whichever is less. Head temperatures are assumed to be 290°C for the T_{cold} heads, and 300°C for the T_{hot} heads. The French have adopted the position that there is no difference in the crack growth rates between the two dome temperatures. Therefore, a mean curve appropriate to 290°C is used, resulting in the following expressions for the mean and maximum growth rates:

$$da/dt = 0.3 \times (K - 9)^{0.52} \text{ (da/dt in mm/hour, K in MPa}\sqrt{\text{m}}\text{)} \qquad (2)$$

and

$$da/dt = 0.3 \times (K - 9)^{0.10} \text{ (da/dt in mm/hour, K in MPa}\sqrt{\text{m}}\text{).} \qquad (3)$$

It is the maximum growth rate that is used in calculations related to inspection intervals. Thus, the inspection policy is crack growth rate dependent: every three years for cracks less than 3 mm deep, every two years for cracks between 3 and 5 mm deep, and every year for cracks above 5 mm. In all cases, the head would be replaced before reaching a safety criterion of 4 mm of remaining ligament for the 900 MWe reactors. Other penetrations are also inspected, on a sampling basis, and include the steam generator partition plate stub welds, BMIs and earlier nozzle repairs. Twenty-six steam generator divider plates, determined to fall into a high susceptibility category - "precursors" is the French terminology, will be inspected during the period 2001 – 2008. Nine additional, randomly chosen SG divider plates will be inspected.

The ten-year inspections have been completed at several (<5) of the French plants with replaced heads, and no indications of cracking were discovered. Forty-two heads (out of 54) heads have been replaced, and two more replacements are scheduled for 2004.

Germany

Most plants in Germany have a Siemens design, featuring threaded connections, with a seal weld, and a Nb-stabilized stainless steel clad/carbon steel substrate penetration tube. These penetrations have not exhibited cracking problems. Only one plant, Obrigheim, has welded head penetrations, and these are smaller (85 mm OD, 10 mm wall thickness vs. ~100 mm OD, 15 mm wall thickness) than for a typical U. S. PWR. The entire head of this plant was strain-relieved (10 hours at 600°C) prior to startup, and no cracking of the alloy 600 has been observed up to this time. Additionally, the calculated head temperature is 295°C, or about 20°C less than the more common T_{hot} condition. Framatome-ANP (Erlangen) is participating in the ICG-EAC Round Robin (described below).

Other Foreign Countries

Sweden, Finland, Korea, Belgium, Canada, Switzerland and Spain are also conducting research and testing of nickel-base alloys. Most of these are participating through the ICG-EAC coordinated efforts.

International Cooperative Group – Environmentally-Assisted Cracking (ICG-EAC)

The ICG-EAC is in the process of conducting round robin testing of 30% cold-worked, alloy 600 and as-welded (i.e., not strain-relieved) alloy 182. The alloy 600 samples were distributed in 2002, and by the time of this writing, most testing has been completed. Reporting of these results is expected at the April, 2004 annual meeting of this group. Most laboratories are testing at a fixed value of 20 MPa√m, but a few are testing over a wider range of stress intensity factor. The testing of alloy 182 will be conducted in the 2004-2005 time frame; samples are not yet available (12/2003). Compilation of this data will provide essential information on the uniformity of test results from the disparate laboratories, as well as providing additional data for the data base used to perform flaw growth evaluations.

9. COLLABORATIVE EFFORTS BETWEEN THE NRC AND INDUSTRY

Examination of the Cavities and Materials from the Discarded Davis-Besse Head

The extensive corrosion of the low-alloy steel vessel head found at Davis-Besse in March, 2002 created extensive interest in discovering the set of conditions that led to the wastage. The initial failure analysis of that corrosion was completed in June, 2003 (Ref. 2). Figure 1, and Figure 7, a photograph of the extensive corrosion loss experienced on the head at the Davis-Besse plant, were taken from that report. RES has initiated several programs addressing this issue. Pieces of the head were removed, and have been shipped to Pacific Northwest National Laboratory (PNNL), where they will be decontaminated, and the smaller corrosion cavity surrounding Nozzle #2 (the major corrosion was at Nozzle #3) will be examined in an attempt to discover the conditions of nascent head corrosion near a (relatively) younger CRDM leak. Other pieces of metal were removed from Nozzle #3, and the metal surrounding Nozzle #11, and sent to ANL, where crack growth rate tests are being conducted currently on the Alloy 600 and Alloy 182.

Material from Nozzle #3 containing one of the SCC crack tips has been delivered to PNNL, where the Dept. of Energy is funding a program from their BES allocation to characterize the morphology and material chemistry at the crack tip, using microanalytical microscopy techniques. At ANL, an extensive program of corrosion testing of several pressure boundary materials in concentrated solutions of boric acid is underway. The output of this program will be data describing the corrosion rates, and electrochemical properties of boric acid solutions over a range of temperatures, concentrations and aeration levels. The industry is just embarking on a similar set of tests; in addition to corrosion rate testing, the industry program also includes assessment of flow rate effects. The fourth phase of the industry program, to be initiated following conclusion of more basic studies, will be the testing of several mockups providing leak rates similar to those believed to have transpired at Davis-Besse.

Examination of Nozzles Removed from North Anna 2 Head

The extensive cracking of Alloy 182 J-welds and CRDMs at the North Anna 2 plant, including the observation of cracked CRDMs that did <u>not</u> exhibit visible evidence of leakage, has provided the impetus for a large, collaborative program involving the NRC and EPRI/MRP. Before the head was disposed in Utah, six nozzles, with a substantial amount of the surrounding low-alloy steel and cladding, were cut from the head (at industry expense), and shipped to PNNL. The nozzles are being decontaminated (using RES funding), and will undergo non-destructive examination by at least three industry vendors, some using novel techniques that are still under development. In a follow-on program being drafted by RES, additional NDE techniques will be assessed and applied to these nozzles, and some of the flaw indications will be sequentially sectioned to determine if the probability of detection and crack sizing was satisfactorily accurate.

Figure 7. A photograph of the 18-inch diameter section of the discarded Davis-Besse head, containing the corrosion cavity created by leakage from the included nozzle (#3).

10. SUMMARY

The experiences described in this report, and the recent testimony of several experts suggests that alloy 600 and its associated welds (alloy 182 & alloy 82) are susceptible to crack nucleation and growth in a wide range of applications. Cracking has occurred at fairly low temperatures when the stress is high (South Texas Project, Unit 1: bottom mounted instrumentation penetrations, T ~ 290°C, asymmetric weld with fabrication flaws), and at high temperatures, when the stress is (presumably) low, and the material quality is high (Davis-Besse: T~318°C, top center head location, median yield, good microstructure). The prognosis for the wrought material (alloy 600) is not good. The prognosis for the weld is little better. Field reporting of crack growth rates for welds is not generally reported, since cracks are discovered by either dye penetrant application and visual inspection or eddy current testing, and are not generally sized for depth. Laboratory results suggest even higher crack growth rates for alloys 182, 82 and 132. Although the accepted disposition curve for alloy 600 (in MRP-55, Ref. 6) exhibits a threshold (at K_{th} = 9.0 MPa√m, or 8.2 ksi√in.), the evolving consensus of experts is that the threshold is even lower, and may not exist at all. The forthcoming companion document to MRP-55, providing crack growth rate information for alloys 182, 132 and 82, will provide a disposition curve that will not exhibit a threshold (K_{th} = 0.0 MPa√m). Recent data suggests that crack growth rates in the heat-affected zone of alloy 600 may be up to a factor of 30 faster than those in the mill-annealed, parent product (Ref. 20). The increase is attributable to both the increased strain in that region, caused by the thermal cycling associated with the weld deposition, and to the (at least partial) solutioning of the grain boundary carbides during the welding process.

While the probability of cracking in replacement heads will be much decreased compared with original vessel heads, the risk is not likely to be reduced to zero. Preliminary data indicate that resistance of alloy 690 to cracking will be much improved over alloy 600. However, preliminary and currently unpublished laboratory results indicate that alloy 152, while more resistant to cracking than alloy 182, will crack under some reactor conditions. Furthermore, alloy 152 – the weld associated with alloy 690 penetrations – has a known problem with low-temperature degradation of fracture toughness. At low temperatures (~55°C, or 130°F), and in coolant environments containing significant levels of dissolved hydrogen (generally in the 50 – 150 cc/kg range), the elastic-plastic fracture toughness, and tearing toughness of weld metals with the "52" chemistry appears to decrease drastically (References 21, 22). Whether this problem translates to an increase in subcritical crack growth rates, or crack nucleation rates, remains to be demonstrated in laboratory tests. Since the issue appears to be related to the electrochemical condition of the alloy surface with respect to the Ni/NiO phase transition (which in turn is a function of temperature and dissolved hydrogen content), it seems likely that crack growth rates will also be similarly affected. The data on HAZs of alloy 600 supports this contention.

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APPENDIX A: RECENT CRACKING AND LEAKAGE EVENTS

Although the NRC Vessel Head Penetration Conference included presentations that described several recent alloy 600 cracking events. Tables 1a through 1d contain summaries of findings, gleaned from sources such as LERs and event notification reports submitted by US licensees. In this regard, only publicly available information is presented in this table. The findings are categorized by the component which failed and then listed chronologically. This should provide an overall understanding of how Alloy 600 cracking events evolved over time, and the references are intended to provide traceability. Following the table below is a more detailed account of the individual incidents.

Table 1. Pressurized Water Reactor Plants

Table 1a. Reactor Vessel Head Nozzles

Plant Type	Plant	Date	Component	Reference
B&W	Oconee 1	November 2000	CRDM Weld and thermocouple nozzles	LER 269-2000-006 Event Notification Report 37567
	Oconee 3	February 2001	CRDM and CRDM Weld	LER 287-2001-001 Event Notification Report 37760
	ANO 1	March 2001	CRDM	LER 313-2001-002 Event Notification Report 37864
	Oconee 2	April 2001	CRDM and CRDM Weld	LER 270-2001-002 Event Notification Report 37950
	Crystal River 3	October 2001	CRDM	LER 302-2001-004 Event Notification Report 38365
	TMI 1	October 2001	CRDM	LER 289-2001-002 Event Notification Report 38416
	Oconee 3	December 2001	CRDM and CRDM Weld	LER 287-2001-003 Event Notification Report 38493
	Davis-Besse	February 2002	CRDM	LER 346-2002-002 Event Notification Report 38732
	Oconee 1	April 2002	CRDM and CRDM Weld	LER 269-2002-003 Event Notification Report 38821
	ANO 1	October 2002	CRDM and CRDM Weld	LER 313-2002-003 Event Notification Report 39254
	Oconee 2	October 2002	CRDM	LER 270-2002-002 Event Notification Report 39288
	Oconee 3	April 2003	CRDM	LER 287-2003-001 Event Notification Report 39821
	Oconee 1	September 2003	CRDM and thermocouple penetration	LER 269-2003-002 Event Notification Report 40192
W 4-Loop	D.C. Cook 2	1994	CRDM	30 day outage response to Bulletin 2001-01
	D.C. Cook 2	May 2003	Vessel head penetration	Event Notification Report 39855
W 3-Loop	Surry 1	Oct.-Nov. 2001	CRDM Weld	LER 280-2001-003 Event Notification Report 38435 30 day outage response to Bulletin 2001-01
	North Anna 2	November 2001	CRDM Weld	LER 339-2001-003 Event Notification Report 38498

Reactor Vessel Head Nozzles (continued)

Plant Type	Plant	Date	Component	Reference
	North Anna 2	September 2002	CRDM and CRDM Weld	LER 339-2002-001 Event Notification Report 39191
	North Anna 1	February 2003	CRDM	LER 338-2003-001 Event Notification Report 39635
W 2- Loop				
CE	Millstone 2	March 2002	CEDM	Response to Bulletin 2002-02
	St. Lucie 2	April 2003	CEDM and CEDM weld	LER 389-2003-002 Event Notification Report 39812
CE Standard				

Indicates there have been no reports for this type of plant

Table 1b. Reactor Vessel Bottom Mounted Instrument Nozzles

Plant Type	Plant	Date	Component	Reference
B&W				
W 4-Loop	STP 1	April 2003	Bottom mounted instrumentation	LER 498-2003-003 Event Notification Report 39754 ML032120244 ML032120667
W 3-Loop				
W 2-Loop				
CE				
CE Standard				

Table1c. Pressurizer Nozzles

Plant Type	Plant	Date	Component	Reference
B&W	ANO 1	December 1990	Pressurizer instrumentation nozzle	LER 313-1990-021
	Crystal River 3	October 2003	Pressurizer level nozzle	LER 302-2003-003 Event Notification Report 40222
	TMI 1	November 2003	Pressurizer heater sleeve	LER 289-2003-003 Event Notification Report 40296
W 4-Loop				
W 3-Loop				
W 2-Loop				
CE	San Onofre 3	February 1986	Pressurizer instrument nozzle	LER 362-1986-003
	St. Lucie 2	1987	Pressurizer instrument nozzles	Noted in LER 389-1993-004
	ANO 2	April 1987	Pressurizer heater sleeve	LER 368-1987-003

Table 1c. (continued) Pressurizer Nozzles

Plant Type	Plant	Date	Component	Reference
	Calvert Cliffs 2	May 1989	Pressurizer heater sleeves and level nozzle	LER 318-1989-007
	San Onofre 3	February 1992	pressurizer vapor space level instrument nozzles	LER 361-1992-004
	San Onofre 2	March 1992	pressurizer vapor space level instrument nozzles	LER 361-1992-004
	St. Lucie 2	March 1993	Pressurizer instrument nozzles	LER 389-1993-004
	Palisades 1	October 1993	Pressurizer temperature nozzle	LER 255-1993-011
	Calvert Cliffs 1	February 1994	Pressurizer heater sleeve	LER 317-1994-003
	St. Lucie 2	March 1994	Pressurizer instrument nozzles	LER 389-1994-002
	San Onofre 3	July 1995	Pressurizer level instrumentation nozzle	LER 362-1995-001
	San Onofre 2	March 1997	Pressurizer temperature nozzle	LER 361-1997-004
	Calvert Cliffs 1	1998	Pressurizer heater Sleeve	MRP-27
	Calvert Cliffs 2	July 1998	Pressurizer level tap	LER 318-1998-005
	Waterford 3	February 1999	Pressurizer instrument nozzles	LER 382-1999-002 Event Notification Report 35407
	San Onofre 3	1999	Pressurizer heater sleeve	MRP-27
	ANO 2	July 2000	Pressurizer heater sleeve	LER 368-2000-001 Event Notification Report 37199
	Ft. Calhoun	October 2000	Pressurizer Temperature Nozzle	ML003781299 ML023610110
	Waterford 3	October 2000	Pressurizer heater sleeve	LER 382-2000-011 Event Notification Reports 37422 and 37434
	Millstone 2	February 2002	Pressurizer heater sleeves	LER 336-2002-001
	ANO 2	April 2002	Pressurizer heater sleeve	LER 368-2002-001 Event Notification Report 38855, 38888
	Millstone 2	October 2003	Pressurizer heater sleeves	LER 336-2003-004
	Waterford 3	October 2003	Pressurizer instrument nozzles	Event Notification Reports 40278 and 40277
CE Standard	Palo Verde 1	January 1992	Pressurizer steam space nozzle	LER 528-1992-001

Table 1c. (continued) Pressurizer Nozzles

Plant Type	Plant	Date	Component	Reference
	Palo Verde 2	October 2000	Pressurizer heater sleeve	LER 529-2000-004 Event Notification Report 37411
	Palo Verde 3	September 2001	pressurizer heater sleeve	Event Notification Report 38332
	Palo Verde 3	March 2003	pressurizer heater sleeve	LER 530-2003-002 Event Notification Report 39714
	Palo Verde 3	February 2004	Pressurize heater sleeve	Event Notification Report 40556
Foreign	Tsuruga 2	September 2003	Pressurizer piping nozzle stub weld	http://www2.jnes.go.jp/atom-db/en/index.html

Table 1d. Reactor Coolant System Nozzles and Other Components

Plant Type	Plant	Date	Component	Reference
B&W	ANO 1	February 2000	RCS hot Leg instrumentation	LER 313-2000-003 Event Notification Report 36697
W 4-Loop				
W 3-Loop	Summer 1	October 2000	Hot leg nozzle	LER 395-2000-008 Event Notification Report 37423 http://www.nrc.gov/reactors/operating/ops-experience/alloy600/vcsummer.html
W 2-Loop				
CE	Palisades 1	September 1993	Relief valve	LER 255-1993-009
	San Onofre 3	July 1995	RCS hot leg instrument nozzles	LER 362-1995-001
	St. Lucie 2	October 1995	RCS hot leg instrument nozzle	LER 389-1995-004
	San Onofre 3	April 1997	RCS instrumentation nozzle	LER 362-1997-001
	San Onofre 3	July 1997	RCS instrumentation nozzle	LER 362-1997-002
	San Onofre 2	January 1998	RCS nozzles	LER 361-1998-002
	Waterford 3	February 1999	RCS hot leg instrument nozzle	LER 382-1999-002 Event Notification Report 35407
	ANO 2	July 2000	RCS hot leg instrument nozzle	LER 368-2000-001 Event Notification Report 37199
	Waterford 3	October 2000	RCS hot leg mechanical nozzle seal assembly clamps	LER 382-2000-011 Event Notification Reports 37422 and 37434
	St. Lucie 1	April 2001	RCS hot leg instrument nozzle	LER 335-2001-003 Event Notification Report 37919
	Waterford 3	October 2003	RCS hot leg instrument nozzle	Event Notification Reports 40278 and 40277
CE Standard	Palo Verde 1	October 1999	RCS hot leg valves	LER 528-1999-006 Event Notification Report 36256
	Palo Verde 1	March 2001	RCS hot leg thermowell	LER 528-2001-001 Event Notification Report 37878
	Palo Verde 3	September 2001	RCS hot leg temperature nozzle	Event Notification Report 38332

Table 1d. (Continued) Reactor Coolant System Nozzles and Other Components

Plant Type	Plant	Date	Component	Reference
	Palo Verde 3	March 2003	RCS hot leg instrument nozzle	LER 530-2003-002 Event Notification Report 39714

Table 2. Boiling Water Reactor Plants - All Components

Plant Type	Plant	Date	Component	Reference
GE Type 4	Vermont Yankee	April 1986	Core spray nozzle weld	LER 271-1986-005
	Hope Creek 1	September 1997	Core spray nozzle weld	LER 354-1997-023
	Duane Arnold 1	November 1999	Recirculation riser welds	LER 331-1999-006 Event Notification Report 36416 and 36402
	Susquehanna 1	March 2004	Recirculation weld	Event Notification Report 40605
GE Type 3	Pilgrim 1	October 2003	Reactor vessel nozzle to Cap Weld	LER 293-2003-006

Reactor Vessel Head Nozzles

B&W Plants

Oconee Unit 1 (Event Notification Report 37567 + LER 269-2000-006)

On November 25, 2000, during a visual inspection of the top surface of the reactor pressure vessel head, small amounts of boric acid deposited on the vessel head surface was discovered. These deposits appeared to be located at the base of 5 unused thermocouples and the #21 control rod drive mechanism nozzle weld at points where they all penetrate the RPV head surface. On December 4, an eddy current test was performed on the inside surface of the 8 thermocouple nozzles and revealed axial crack-like indications on the ID of the nozzles in the vicinity of the partial penetration weld. Dye penetrant testing on CRDM #21 identified two very small pin-hole indications running at a slightly skewed angle across the fillet weld. The eight thermocouple nozzles were removed and had Inconel 690 plugs welded in place. The indications in the CRDM fillet weld were ground out and a final weld repair was performed.

Oconee Unit 3 (Event Notification Report 37760 + LER 287-2001-001)

During a visual inspection of the reactor vessel control rod drive mechanism nozzles on February 18, 2001, small amounts of boron residue surrounding the base of several control rod drive mechanism head penetrations was discovered. The boric acid deposits were identified around CRDM #'s 3, 7, 11, 23, 28, 34, 50, 56, and 63. Subsequent surface dye penetrant test inspections of the nine nozzles weld area and outside diameter identified several deep axial cracks that initiated near the toe of the fillet weld and had propagated radially into the nozzle materials as well as axially along the outer diameter surface. Ultrasonic testing confirmed the existence of deep cracks in all nine leaking CRDM nozzles. Of these 47 original crack indications, 19 were outer diameter initiated flaws that were not through wall. There were 16 flaws that were outer diameter initiated through wall cracks. There were 9 circumferential flaws, with one being inner diameter initiated and the rest being outer diameter initiated. Two of the outer diameter circumferential flaws were above the J-groove weld. Finally, there were also three inner diameter initiated non-through wall cracks.

Arkansas Nuclear Unit 1 (Event Notification Report 37864 + LER 313-2001-002)

On March 18, 2001, during a routine visual inspection of the reactor vessel head area boric acid crystals were discovered. On March 24, 2001, eddy current testing and ultrasonic testing revealed a reactor coolant solution pressure boundary leak. The leak was identified in the wall of CRDM # 56. The UT data indicates that the crack is on the downhill side of the nozzle and it extends approximately 0.8 inches below the weld and extends upward to approximately 1.0 inches above the weld. The depth of the crack was approximately 0.2 inches. The axial crack in the CRDM was removed and an embedded flaw repair technique was utilized.

Oconee Unit 2 (Event Notification Report 37950+ LER 270-2001-002)

A preliminary reactor head visual inspection on April 28, 2001, revealed small amounts of boron residue surrounding control rod drive mechanism head penetrations 4, 16, 18, and 30. Subsequent surface dye penetrant test inspections of the weld area and nozzle outside diameter identified several axial cracks on four CRDM nozzles that initiated near the toe of the fillet and propagated radially into the nozzle materials as well as axially along the outer diameter surface. Eddy current tests revealed two shallow axial flaws on CRDM nozzle 16 and craze cracking on all four CRDM nozzles inner diameter surface. The ultrasonic testing confirmed the existence of some cracks axial with one short outer diameter initiated circumferential crack on CRDM 18. The circumferential flaw was outer diameter initiated and extended 11% through wall and was 1.26 inches in length.

Crystal River Unit 3 (Event Notification Report 38365+ LER 302-2001-004)

On October 1, 2001, during a visual inspection of the reactor vessel control rod drive mechanism nozzles, boric acid buildup was discovered around nozzle 32. Ultrasonic testing revealed a through wall indication. Ultrasonic testing performed on nozzle 32 revealed two through wall axial cracks. These cracks extended from the bottom of the nozzle to above the J-groove weld. These two axial cracks then joined a circumferential crack above the J-groove weld. This circumferential flaw was about 90 degrees and was 50% through wall. There was another circumferential flaw below the J-groove weld which extended 30 degrees and was approximately 75% through wall. Nozzle 32 was repaired using the ambient temperature bead repair technique.

TMI Unit 1 (Event Notification Report 38416 + LER 289-2001-002)

On October 11 and 12, 2001, during a visual inspection of the reactor vessel control rod drive mechanism nozzles, boric acid buildup was discovered around 8 different thermocouple nozzles. Liquid penetrant test and ultrasonic testing identified through-wall indications on three CRDM nozzles. These engineering evaluations concluded that the indications at CRDMs 44, 35, and 37 indicate a reactor coolant system pressure boundary leak. During later examinations of the reactor vessel head, CRDMs 29 and 64 were shown to contain RCS pressure boundary leak.

The CRDM nozzles were repaired by initially rolling the nozzle above the J-groove weld, and then machining the lower portion of the CRDM nozzle including portions of the J-groove weld. A new pressure boundary weld was formed between the CRDM nozzle and the RPV head low alloy steel at the location about the previous J-groove weld and below the rolled nozzle area. A surface remediation inducing compressive stresses was performed after the repair. The thermocouples were repaired by cutting them approximately 1 inch from the outside surface of the RPV head. The remaining nozzle portion inside the RPV head was machined out of the head. These six thermocouple nozzles were plugged by installing an Inconel 690 plug in the RPV head bore. A nozzle weld dam was then inserted into the cavity. A weld pad build up of Inconel 152 was welded over the nozzle plug.

Oconee Unit 3 (Event Notification Report 38493 + LER 287-2001-003)

During a visual inspection of the reactor vessel control rod drive mechanism nozzles on December 12, 2001, leakage indications were discovered. These indications were discovered by minor boric acid buildup around 4 CRDM nozzles. Nondestructive examination of the suspected nozzles revealed that seven of the sixty-nine total nozzle required repair. Five of these seven nozzles had a leak pathway to the top the reactor vessel head. Some of the indications were in the nozzles themselves, while other indications extended slightly into the weld. Most of the cracking was axial in nature, however, there was one circumferential flaw found in nozzle #2 above the J-groove weld.

Davis-Besse Unit 1 (Event Notification Report 38732 updated 3/6/02 +
LER 346-200-2002)

On February 26, 2002 during a refueling outage, visual inspections were conducted. The inspections were inconclusive due to previous known boric acid deposits. On February 27, 2002, ultrasonic testing data identified axial through weld indications on nozzle #3 CRDM. Engineering evaluations of this data confirmed reactor coolant system pressure boundary leakage exits. UT data for CRDM nozzles 1 and 2 exhibited axial indications that represent boundary leakage. Additionally, a circumferential indication on this CRDM nozzle was 34 degrees in length and was 50% through wall. In the process of machining Nozzle #3, unexpected movement of the nozzle had occurred. Due to this movement and further investigation, a cavity in the RPV head was discovered. The width of this cavity measured approximately 4-5 inches at its largest point. The cavity formation is due to boric acid corrosion over a long period of time. The initial cracking of the CRDM lead to leakage of reactor solution into the annulus. The combinations of PWSCC along with boric acid corrosion were main contributors to the cause of this event.

Oconee Unit 1 (Event Notification Report 38821 + LER 269-200-2003)

On April 1, 2002, a qualified visual inspection of Unit 1 reactor vessel head was conducted and found two penetrations with very slight amounts of boron accumulations were detected. Ultrasonic testing was then performed on the penetrations and revealed partial through-wall outside diameter cracks for nozzles 1, 7, and 8. There were five flaws and a potential leak path identified in nozzle 7. There was one axial flaw but no leak path identified in nozzle 8. Nozzle 1 showed three minor indications in a region of rough weld contour. Liquid dye-penetrant was then used to examine these three nozzles. PT revealed two axial flaws in the original weld on nozzle 7. PT was also used on nozzle 1 and it did not find any recordable or rejectable PT indications. Nozzles 7 and 8 were repaired by removing part of the nozzles below the reactor vessel head (RVH) and a length of 5 inches into the RVH. A new pressure boundary weld was installed within the bore, inspected, and surface conditioned with a water jet peening process.

Arkansas Nuclear Unit 1 (Event Notification Report 39254 + LER 313-2002-003)

On October 7, 2002, a routine visual inspection was performed on the reactor vessel head area. Small boric acid crystal nodules were found around the area of control rod drive mechanism nozzle #56. On the downhill side of the nozzle, boric acid residue was located, extending 180 degrees around the nozzle annulus area, with a small boric acid nodule at the most downhill point. Non-destructive examination (NDE) found indications of cracking in the nozzle, which was the cause of the boric acid residue. NDE of all the nozzles revealed indications of non-through wall cracks in six other nozzles, and a likely porosity weld defect in another. The leaking nozzle #56 had been repaired during the previous outage. The new crack indications were located just outside the previous weld repair zone. The previous repair technique was the embedded flaw repair. It is believed that the same nozzle failed because the previous repair did not isolate the 182 weld, which is a susceptible material to PWSCC in the PWR environment. The current repair technique consisted of removing the portion of the nozzle that extends below the surface of the reactor vessel head. A new half nozzle was installed using alloy 52 weld material.

Oconee Unit 2 (Event Notification Report 39288 + LER 270-2002-002)

During a visual inspection on October 15, 2002, evidence of through wall leakage was discovered on seven CRDM penetrations. These penetrations were nozzles 8, 9, 19, 24, 31, 42, and 67. None of these nozzles had been previously repaired. Additional nozzle head penetrations were masked by boric acid deposits suspected of being from separate sources of leakage. NDE was used to characterize the cracking. No circumferential cracks were reported and 10 CRDMs (11, 15, 19, 21, 24, 31, 33, 36, 38, and 42) with axial cracks were found. The repair technique included removing part of the previous nozzle and welding a new half nozzle into the reactor head. The new nozzle was treated with water jet peening afterwards.

Oconee Unit 3 (Event Notification Report 39821 + LER 287-2003-001)

Unit 3 entered its scheduled end-of-cycle 20 refueling outage on April 20, 2003. During a visual inspection of the reactor vessel head on May 2, 2003, evidence of possible through wall leakage was observed on two CRDM penetrations. The locations of these penetrations are nozzle 4 and 7. CRDM #4 contained a very thin white coating while nozzle 7 appeared to have a small accumulation of boron on the head adjacent to the annulus region. Approximately 6 to 8 additional nozzles-to-head penetrations were masked by deposits from a component cooling system leak above the RV head and were unable to be inspected. Prior refueling outage RVH inspection videotapes showed that the CRDM #7 deposits were not associated with a new leak but rather were remnants from a prior outage leak and repair where the boron residue had not been removed. The CRDM #4 boron deposits are fresher and similar to previous RVH leaks. The apparent root cause of the nozzle leak is PWSCC. The head was replaced during this refueling outage.

Oconee Unit 1 (Event Notification Report 40192 + LER 269-2003-002)

During a scheduled bare metal visual inspection on September 23, 2003, possible evidence of a through wall leak on two control rod drive mechanisms (CRDM) (nozzle 6 and 16) and one thermocouple penetration (nozzle 7) was observed. The thermocouple had been repaired (plugged) in December 2000. Reactor coolant leakage prior to the unit shutdown was varying between 0.15-0.24 gallons per minute. There were no plans to perform additional inspections or repairs since the head is to be replaced during the same refueling outage which these indications were observed.

Westinghouse 4-Loop Plants

<u>Cook Unit 2</u> Alloy 600- Bulletin 2001-01 Plant Specific Information (30 day response)

During the cycle 10 refueling outage in 1994, eddy current testing examination was performed on 71 of the 78 vessel head penetrations. The testing showed crack indications in penetration number 75. Three indications were found with lengths of 9mm, 16mm, and 45mm. These indications were axial in orientation and were closely spaced. The 3 indications were located near the 160-degree location on the high side. The 45mm crack was located near the J-groove weld, but was mostly below the weld.

<u>Cook Unit 2</u> (Event Notification Report 39855 updated on 6/11/03 (retracted))

Craze cracking indications were found on a reactor pressure vessel head penetration May 17, 2003. Shallow indications were found on the inside diameter of penetration #74 during the reactor head inspection. These indications are closely spaced 3/8 inch below the J-groove weld. Initial calculations showed a crack depth of 0.117 inches. There was no through-wall leakage detected. These same cracking indications were found during the 2002 refueling cycle and have not shown any significant growth. This report was retracted because it was determined that the craze cracking indications in penetration #74 of the Unit 2 Reactor Pressure Vessel Head do not represent a seriously degraded principal safety barrier of the nuclear power plant.

Westinghouse 3-Loop Plants

<u>Surry Unit 1</u> (Event Notification Report 38435 + LER 280-2001-003 + 30 day outage response to Bulletin 2001-01)

On October 28, 2001, through-wall indications of the J-groove weld were identified on CRDM penetrations 27 and 40. On November 2, 2001 indications of flaws in the penetration welds on around penetrations 65, 47, 69, and 18 were also uncovered. The repair of these nozzles utilized the temperbead repair procedure.

<u>North Anna Unit 2</u> (Event Notification Report 38498 + LER 339-2001-003)

On November 13, 2001, a through wall leak on penetration #63 was observed. This event was treated as a through wall leak based on the qualified visual inspection results and liquid penetrant examination. A portion of the weld around penetration 63 was excavated to a depth of approximately 1" of weld metal. The liquid penetrant exam of this excavation showed 12 indications located in the outside edge of the weld almost the full length of the excavation, and which turns into the weld at the uphill and downhill ends of the excavation. Six of the recorded cracks were transverse to the weld while the other six were parallel. Eddy current testing found a crack 31mm in length in the area of the attachment weld. Ultrasonic testing of the same crack found it to be less than 1mm in depth and 14mm in length. Penetration 51 also had boric acid residue near it on the reactor vessel head. A liquid penetrant test found 12 indications at the toe of the weld. Five of these indications were parallel while the rest were transverse. Eddy current testing of the nozzle's weld found 6 axial indications. These axial cracks were less than 2mm in depth and ranged from 6 to 24mm in length. Similarly, penetration 62 was also investigated. This nozzle had eight indications at the toe of the weld. Two of the cracks were parallel and six were transverse. Eddy current testing revealed two axial indications. The dimensions were 74mm and 42mm in length, while being less than 2mm and less than 1mm in depth respectively. The repair used for the nozzles was temperature temperbead procedure.

North Anna Unit 2 (Event Notification Report #39191 + LER 339-2002-001)

Boric acid residue was discovered during a bare head inspection of the reactor vessel head on September 14, 2002. It appears that head penetrations #21 and #31 had exhibited some amount of leakage due to boric acid residue on the reactor vessel head. Four additional penetrations are suspected of leaking, and several penetrations are masked with boric acid residue. Fifty-nine J-groove welds have been inspected using eddy current (ET). Fifty-seven of the fifty-nine penetrations inspected were identified with crack like indications. The ET identified at least one indication of a 6 mm crack in about 83% of the J-groove welds. During the previous year's outage, no boric acid residue was discovered. The six nozzles (N2-51, 53, 55, 57, 62, and 63) which could not be inspected with ET had their welds inspected using liquid penetrant tests (PT). Three of these penetrations (N2-51, 62, and 63) had been previously repaired with weld overlay of the J-groove. Each of the six penetrations that were inspected with PT had evidence of rejectable indications.

Eddy current examinations of the J-groove welds showed indications of axial and circumferential cracking with respect to the welding direction. The range in length was from 0.12 inches to 7.0 inches. Some longer flaws were recorded, but these are actually a series of small flaws with very short distances between. Eddy current testing of the inside diameter surface showed twenty of thirty-five penetration tubes had axial indications. These indications were believed to be less than 0.12 inches deep. Four penetrations (#21, #31, #51, and #63) showed evidence of a leak path in the shrink fit area between the vessel head and the tube. Penetrations #51 and 63 had been identified as leaking in the fall 2001. Repairs of these penetrations had been improperly applied because the weld overlay repair did not extend out far enough to cover the previous NDE indications. The six penetration welds inspected with PT had greater than 1/16 inch linear flaw indications.

It was decided to replace the reactor vessel head instead of making multiple repairs.

North Anna Unit 1 (Event Notification Report 39635 + LER 338-2003-001)

On February 22, 2003, Unit 1 was ramped offline for a scheduled refueling outage. During this outage, visual inspection was performed on the reactor vessel head. On March 4, 2003 an apparent reactor vessel head through-wall leak was noted on CRDM #50. The inspection was to follow up with the inspection results performed during the previous outage in 2001. Boric acid residue was found approximately one-half an inch in diameter on the lower side of the penetration-to-head transition. There were no signs of wastage on the reactor vessel head. The Unit 1 reactor head was replaced during the 2003 refueling outage.

CE Plants

Millstone 2 (Response to Bulletin 2002-02)

On March 13, 2002, Dominion Nuclear Connecticut, Inc. (DNC) informed the NRC that analysis of ultrasonic inspection results from the previous three weeks had determined that flaw indications were in three nozzles as shown in Table 3. The portions of the nozzles containing these flaws were bored out, and a "half-nozzle" repairs were completed. In this method, the flawed nozzles were bored out from the underside (wetted side) of the head, to a location approximately half-way through the head, leaving the old nozzle with a single-V, weld-prepped surface. A new section of alloy 690 nozzle tube, similarly prepped, was inserted, and welded into place. This repair procedure moves the pressure boundary from the inside surface to a point about midway through the thickness of the head.

Table 3. Flaw indications from Millstone 2 plant in March, 2002.

Nozzle Number	**21**	**34**	**50**
Number of Axial Indications	5	4	0
Number of Circumferential Indications	1	2	2

St. Lucie Unit 2 (Event Notification Report 39812 updated 5/6/03 + LER 389-2003-002)

On April 30, 2003, during a refueling outage, a defect in CEDM penetration #72 was found. St. Lucie Unit2 had approximately 14.0 effective degradation years at the start of the 2003 refueling outage, therefore, this plant has a high susceptibility in accordance with Order EA-03-009. Visual inspection of the reactor pressure vessel was clean, with no evidence of leakage from the 102 RPVH penetrations or wastage on the RPVH surface. The UT inspection identified an axial crack in the CEDM penetration 72. The defect outer diameter connected and extends into the nozzle and penetrates into the J-groove weld between the nozzle and reactor vessel head. The defect is an axial flaw, 0.28 inches deep and 0.96 inches long on the downhill side of the penetration. On May 2, 2003, a second defect in the CEDM penetration #18 was identified. The defect is also outer diameter connected and described as axial. It extends into the nozzle and through the J-groove weld between the nozzle and reactor vessel head. This second defect measures 0.26 inches deep and 2.98 inches long. It is also located on the downhill side of the penetration. Neither flaw extended through the wall of the nozzle. Neither nozzle had any evidence of leakage from the annulus between the nozzle and the reactor pressure vessel head associated with the indications.

Reactor Vessel Bottom Mounted Instrument Nozzles

Westinghouse 4-Loop System

<u>South Texas Project Unit 1</u> (Event Notification Report 39754 updated 5/22/03 + LER 498-2003-003)

During a bare metal inspection performed on the vessel bottom head on April 12, 2003 two potential leaks were identified. This leak appeared to be associated with bottom mounted instrumentation (BMI) penetrations 1 and 46. The BMI penetrations are inspected every outage and no residuals had been discovered during the previous outage on November 20, 2002. There was a small amount of residue surrounding the outer circumference of the BMI penetrations where the nozzles meet the bottom of the reactor vessel. There did not appear to be any wastage. Approximately, 150 mg and 3 mg of residue were collected from penetrations number 1 and 46 respectively. The initial indication of boron and lithium in these samples suggested that they were RCS residue. A lithium isotopic analysis was conducted and the results confirmed that the precipitate was indeed RCS. An approximation of age was developed by conducting a cesium isotope study on the samples. The results of this test suggest that the solution is roughly 3-5 years of age. This in turn suggests a small leak rate due to the time to push the leakage through the annulus. UT testing of 57 nozzles and visual inspection of all 58 BMI nozzles was completed on May 23, 2003. Nozzles 1 and 46 contained a total of 5 cracks. No cracks were identified in any of the other BMI penetrations. Nozzle 1 contained 3 cracks which were all axial. Only one of these cracks provides a leak path from either the outside of the nozzle above the J-groove weld or inside the nozzle to the annulus. Nozzle number 46 contains 2 axial cracks. Only one of the cracks in nozzle 46 provides a leak path from the outside of the nozzle above the J-groove weld to the annulus. None of the cracks in nozzle 46 extend to the inside of the nozzle. This is supported by the eddy current tests which reveal that only nozzle 1 has a crack on the inside wall.

Pressurizer Nozzles

B&W Plants

<u>Arkansas Nuclear Unit 1</u> (LER 313-1990-021)

On December 22, 1990, a potential reactor coolant system leak in the area of a pressurizer upper level instrumentation nozzle was identified. During a follow-up inspection it was verified that a very small leak at the nozzle existed. Nondestructive testing was conducted which confirmed the existence of a small axial crack in the nozzle inner surface which breached the outside diameter of the nozzle at the toe of the nozzle to vessel weld. A temporary repair was completed which initially deposited a weld pad on the shell OD around the nozzle penetration. The next step was to prepare a partial penetration weld in the pad and a new nozzle was installed into the penetration from the shell OD. This left a small gap between the original nozzle and the new nozzle.

Crystal River Unit 3 (Event Notification Report 40222 + LER 302-2003-003)

On October 4, 2003, during a routine visual inspection of the upper level instrument tap nozzles, very small reactor coolant leaks were found on nozzles RC-1-LT1, RC-1-LT2, and RC-1-LT3. The leakage evidence for RC-1-LT1 and RC-1-LT3 consisted of stains and boric acid residue. The evidence on RC-1-LT2 consisted only of stains on the pressurizer carbon steel shell. There was no evidence of leakage on any of the similar pressurizer nozzles. The last unidentified leak rate completed prior to plant shutdown was 0.15 gpm. The three pressurizer upper level instrument tap nozzles were repaired using a half-nozzle technique. This technique replaces half of the alloy 600 nozzle with alloy 690 using the similar metal weld (alloy 52/152). This technique also moves the pressure boundary from the internal weld to an external location.

TMI Unit 1 (Event Notification Report 40296 + LER 289-2003-003)

On November 4, 2003, an inspection of the pressurizer heater bundle identified a primary leak at the lower pressurizer heater bundle diaphragm plate. Boric acid residue was found between the diaphragm plate and the cover plate. Initially the leak was thought to be coming from a seal weld. Nondestructive evaluation (NDE) determined that the leak path was through the edge of the pressurizer heater bundle diaphragm plate. There were six indications in the NDE. Four of these indications were surface flaws not associated with a through-wall crack. The heater bundle was initially repaired by depositing a seal weld over the areas of the pressurizer heater bundle diaphragm plate. A leak was revealed in later testing at normal operation pressure and temperature. Due to this leak, the lower pressurizer heater bundle assembly including the pressurizer heater bundle cover plate was replaced with a new heater bundle assembly. The new diaphragm plate was constructed out of type 304 austenitic stainless steel. The diaphragm plate was welded to the pressurizer using Inconel 52 weld materials.

CE Plants

San Onofre Unit 3 (LER 362-1986-003)

On February 27, 1986, a pressure boundary leak was observed in a ¾ inch diameter pressurizer level instrument nozzle. Dye penetrant testing was utilized and discovered a crack extending from the end of the nozzle inside the pressurizer, 5/8 of an inch outward through the RCS pressure boundary.

St. Lucie Unit 2 (Cited in LER 389-1993-004)

In 1987, during the replacement of four pressurizer steam space instrument nozzles, it was determined that two nozzles had cracks but there was no evidence of leakage.

Arkansas Nuclear Unit 2 (LER 368-1987-003)

On April 24, 1987, an unusual event was declared and a reactor shutdown was commenced due to a suspected reactor coolant system pressure boundary leak of approximately 60 drops per minute from the area of the pressurizer vessel lower head. It was determined that the leakage source was the heater sleeve for the X1 pressurizer heater. Due to this leakage there was some corrosion damage to the carbon steel pressurizer shell. The dimensions of this damage in the carbon steel were one-half inches in diameter and three-quarters of an inch deep.

While trying to remove the heater element from the X1 heater sleeve it was discovered that the heater sheath had ruptured. Similarly, another heater, T4, was found to have the same sheath damage. The final determination was that the X1 heater sheath failed which resulted in the damage to the X1 heater sleeve. The damage to the internal sleeve led to the damage of the pressure cladding weld. The initial trigger event was due to fabrication residual stresses in these Watlow heaters. All heater manufactured by Watlow, except for X1 and T4, were removed. There were six available spare heaters that were installed. The other 15 empty heater sleeves were fitted with dummy Inconel heater plugs welded to the sleeves. The heater sleeves X1 and T4 were cut off approximately three-eighths inch below the internal welded area. The rest of the sleeves were drilled out. Plugs were then inserted into the X1 and T4 holes and welded to the outside of the vessel utilizing a temper bead welding repair process.

Calvert Cliffs Unit 1 and 2 (LER 318-1989-007)

On May 5, 1989, an in-service inspection of the Unit 2 pressurizer discovered evidence of reactor coolant leakage from 28 of the 120 pressurizer vessel heater penetrations and one upper level nozzle. No evidence of leakage was found on the Unit 1 pressurizer heater penetrations or pressure/level penetrations. Additional inspections using dye penetrant and eddy current tests of 28 Unit 2 and 12 Unit 1 heater sleeves were conducted. Three sleeves from Unit 2 were destructively examined. All cracks were axial and determined to have minimal safety significance. Reaming and repair operations associated with fabrication of the Unit 2 pressurizer appear to have contributed to the cause. The Unit 2 pressurizer was repaired by replacing 119 heater sleeves with dual Alloy 690 heater sleeves. One heater sleeve was sleeved and plugged with Alloy 690. All four upper level nozzles were replaced with Alloy 690.

San Onofre Unit 3 (LER 361-1992-004)

On February 18, 1992, during refueling outage cycle 6, a dye penetrant examination of the pressurizer vapor space level instrument nozzles revealed the presence of a through-wall crack. The examination was due to boric acid crystals being found near the nozzle previously. The leaking nozzle was replaced with a new nozzle made of Inconel 690. Liquid penetration testing was conducted on the three remaining vapor space nozzles. Much smaller indications on two of the three nozzles were revealed in these tests. These three nozzles were also replaced with nozzles fabricated with Inconel 690. The water space instrument nozzles were also visually checked and no signs of leakage were apparent.

San Onofre Unit 2 (LER 361-1992-004)

On March 14, 1992, an inspection on Unit 2 pressurizer vapor space level instrument nozzles was conducted. Boric acid crystals were found at two of the nozzles. An interim repair of the Unit 2 nozzles with alloy 690 was implemented prior to startup. Inspection of the remaining water and vapor space nozzles showed no signs of leakage.

St. Lucie Unit 2 (LER 389-1993-004)

On March 2, 1993, water was discovered dripping onto the floor in containment near the pressurizer. Visual inspection revealed that four upper instrument nozzles were leaking at the entry fitting to the pressurizer. Liquid penetrant and eddy current test revealed axial cracking in the four steam-space nozzles extending into the surrounding weld area. The leaking pressurizer steam space nozzles were removed and replaced with nozzles made of Inconel 690.

Palisades Unit 1 (LER 255-1993-011)

On October 9, 1993, inspection of the pressurizer upper temperature nozzle penetration TE-0101 was found to be leaking. Subsequent inspection of the lower temperature nozzle penetration TE-0102 was also found to be leaking. The cause of the leaks is due to cracking in the Inconel 600 nozzle material. The two nozzles were repaired by installing a weld pad on the outside pressurizer shell.

Calvert Cliffs Unit 1 (LER 317-1994-003)

On February 16, 1994, boron deposits were located on the pressurizer heater sleeve B-3 after removing insulation. On February 23, 1994, after removing more insulation, boron deposits were also discovered on pressurizer heater sleeve FF-1. Boroscopic and eddy current tests revealed a circumferential bulge approximately 0.5 inches long and 0.019 inches high (diametrical) in the area of the boric acid leaks. The leakage area showed evidence of surface metal smearing and cold work. Sleeve FF-1 had been reworked due to the presence of a stuck reamer. Heater sleeve B-3 and FF-1 were plugged with an alloy 690 plug welded to the outer diameter of the pressurizer lower head.

St. Lucie Unit 2 (LER 389-1994-002)

On March 16, 1994, boric acid was observed on the exterior of the pressurizer steam space C instrument nozzle during an inspection. Dye penetrant was utilized and identified indications at the A, B, and C steam space instrument nozzle welds. The D instrument nozzle weld was acceptable. The unacceptable cracks were in the “J” weld between the Inconel 690 nozzle (replaced in 1993) and the clad on the inside of the pressurizer.

San Onofre Unit 3 (LER 362-1995-001)

On July 22, 1995, during inspection of the Inconel 600 and 690 (see March 1992 event for Inconel 690 installation) instrument nozzles one pressurizer level instrumentation nozzle was found with a small amount of boric acid crystals and oxidation present. Dye penetrant testing indicated crack initiation in the heat affected zone of the weld butter. The alloy 690 pressurizer nozzle piece interior did not have indications of PWSCC. All the vapor space instrument nozzles were planned to be replaced with Inconel 690 using Inconel 52 weld filler metal.

San Onofre Unit 2 (LER 361-1997-004)

On March 3, 1997, steam was observed emanating from the pressurizer Unit 2. It was concluded that the leak was caused by primary water stress corrosion cracking of Inconel 600 type materials of the pressurizer liquid temperature thermowell nozzle. The crack was oriented parallel to the long axis of the nozzle. The nozzle was removed and replaced with Inconel 690.

Calvert Cliffs Unit 1 (MRP-27)

Unit 1 heater sleeves were nickel plated in 1994. One nickel plated heated sleeve (B-1) was found leaking during the 1998 refueling outage. Ultrasonic testing revealed a short axial indication.

Calvert Cliffs Unit 2 (LER 318-1998-005)

On July 25, 1998, a steam leak was discovered at an upper level instrument nozzle on the pressurizer. A dye penetrant test of the nozzle proved that the Inconel 690 nozzle did not contain leak pathway. Ultrasonic examination of the vessel shell was performed to look for defects in the shell material. No defects were found. Because no leak path was found in the nozzle or shell material, it was postulated that the crack was in the Inconel Alloy 600-type weld filler material of the nozzle. The leaking nozzle was cut and the outer portion of the nozzle was removed. A weld pad of Inconel 690 was installed around the penetration and a new nozzle manufactured of Inconel 690 was inserted. This nozzle was welded to the pad with Inconel 690 filler material.

Waterford Unit 3 (Event Notification Report 35407 + LER 382-1999-002)

On February 25, 1999, during a routine visual inspection evidence of reactor coolant system leakage was found on two alloy 600 instrument nozzles located on the top head of the pressurizer. The leakage was in the annulus area where the nozzle penetrates the pressurizer head. The nozzles are welded on the inner diameter of the pressurizer and are joined to instrument valves RC-310 and RC-311. The two leaking nozzles located on the pressurizer have been repaired using welded nozzle replacements.

San Onofre Unit 3 (ERPI MRP-27)

In 1999 a cracked heater sleeve was identified by eddy current testing. The heater had failed, swelled, and stuck within the sleeve. The flaw was approximately 40% through wall on the inner diameter of the sleeve near the attachment weld.

Arkansas Nuclear Unit 2 (Event Notification Report 37199 + LER 368-2000-001)

Twelve pressurizer heater sleeves were found to be leaking. On July 30, 2000, boron residue was discovered on the reactor coolant system pressurizer heater power cables. The boron came from leaks in heaters B2 and D2. B2 is on backup heater bank #4 and D2 is on backup heater bank #6. After removing insulation from the pressurizer, the licensee discovered 10 additional pressurizer heater sleeves that had previous leakage. Eddy current testing on two of the heater sleeves indicated that there was a single, through-wall, axial crack in both sleeves below the J-groove weld. These cracks initiated from the inside surface of the sleeves. Ultrasonic testing showed no cracks in the shell base metal. The pressurizers were repaired with ASME Code-qualified process.

Ft. Calhoun (ML003781229 + ML023610110)

On October 22, 2000, during a walk down inspection, leakage from the lower pressurizer liquid space temperature nozzle TE-108 was detected. A weld technique was used to repair the nozzle.

Waterford Unit 3 (Event Notification Report 37442, 37434 + LER 382-2000-011)

On October 17, 2000, during a bare metal inspection of the pressurizer heater sleeve number F-4, a small amount of boric acid residue was discovered. This pressurizer was repaired by plugging the penetration.

Millstone Unit 2 (LER 336-2002-001)

On February 19, 2002, pressurizer heater penetrations and pressurizer instrument nozzle penetration were examined with visual inspection. Two heater sleeves showed indications of minor leakage due to the boron precipitates discovered on the outside of the penetrations. The cause of this event is through wall cracks in the two pressurizer heater sleeves. The leaking heater sleeves were repaired using mechanical nozzle seal assembly clamps.

Arkansas Nuclear Unit 2 (Event Notification Report 38855, 38888+ LER 368-2002-001)

Six heater sleeves were found to be leaking in the pressurizer. Five of the leaking heater sleeves were discovered on April 15, 2002 while the other was found on April 30, 2002. On April 15, boron deposits were discovered on five pressurizer heater sleeve penetrations. On April 30, boron residue was observed around the sixth pressurizer heater sleeve. Since similar events had occurred in the July 2000 outage, no NDE was conducted. The leaking pressurizer heater sleeves were repaired using the Mechanical Nozzle Assembly technique.

Millstone Unit 2 (LER 336-2003-004)

During the October 2003 outage, two leaking pressurizer heater penetrations were identified. These two pressurizer heaters along with the two degraded pressurizer heaters found in the previous outage were planned to be removed during the current outage. Ultrasonic testing determined that the flaws were axial in nature. The leaking heater penetrations were repaired using the Mechanical Nozzle Seal Assembly technique.

Waterford Unit 3 (Event Notification Report 40278 + 40277)

Evidence of a reactor coolant leakage was detected on two Inconel instrument nozzles located on the top head of the pressurizer on October 25, 2003. One of these nozzles is the pressure transmitter that taps off of reactor coolant systems hot leg #2. The leakage was located in the annulus area where the nozzle penetrates the pressurizer head.

CE Standard Plants

Palo Verde Unit 1 (LER 528-1992-001)

On January 2, 1992, a pressure boundary leak was discovered in the pressurizer steam space nozzle. A pad weld was put in place in order to stop the reactor coolant leakage. PWSCC is believed to be the probably cause of the leakage.

Palo Verde Unit 2 (Event Notification Report 37411 + LER 529-2000-004)

On October 4, 2000, during an in-service inspection, a reactor coolant system pressure boundary leakage was discovered. The leakage was discovered at pressurizer heater nozzle sleeve A06. The leakage was detected in the form of small deposit of boron accumulation on the sleeve. Eddy current testing indicated linear axial cracking. The degraded heater sleeve was repaired by initially cutting off the heater sleeve close to the pressurizer bottom head. The degraded sleeve is then counter bored and a reinforcing pad and plug are welded to seal the sleeve location. The repairs were made using Inconel 690 material. Another pressurizer was also repaired during the same refueling outage. This heater had failed and swelled in 1991.

There was also a linear, axial indication in this sleeve. The sleeve was repaired in the same fashion as discussed before.

Palo Verde Unit 3 (Event Notification Report 38332)

On September 20, 2001, evidence of reactor coolant leakage was discovered. The leak was from a pressurizer heater sleeve nozzle. The pressurizer heater sleeve leakage is located at pressurizer heater B17. The leakage was identified by the discovery of boron deposits accumulated around the circumference of the pressurizer. There was no evidence of leakage during the last refueling outage. The Mechanical Nozzle seal assembly technique was used for repairing the nozzle.

Palo Verde Unit 3 (Event Notification Report 39714 + LER 530-2003-002)

On March 29, 2003, engineering personnel performing preplanned visual examinations of reactor coolant system piping discovered boric acid on the reactor coolant system (RCS) hot leg instrument nozzle and a pressurizer heater sleeve. There was boric acid residue discovered was on the backup pressurizer heater sleeve A01. Eddy current testing on the heater sleeve suggests that the cracking is axial in nature. The heater sleeve was repaired using Mechanical Nozzle Seal Assembly Technique.

Palo Verde Unit 3 (Event Notification Report 40556)

On February 29, 2004, engineering personnel were performing a visual examination of the reactor coolant system piping and discovered boric acid residue on the A03 pressurizer heated sleeve. The visual observation was characterized as a small white buildup of boron residue around the heater sleeve as the sleeve enters the pressurizer bottom head. There didn’t seem to be any residue running down the outside of the sleeve, and there were no signs of dripping, spraying, puddles of liquid or liquid running down the nozzle or pressurizer. The residue appeared to be dry. The heater was repaired using the Mechanical Nozzle Seal Assembly Technique.

Foreign Plants

The following summary of the alloy 600 cracking incident at Tsuruga, in Japan, is included because it closely resembles many cracking incidents in US plants.

Tsuruga Unit 2

Cracking was discovered September 5, 2003, on the pressurizer relief piping nozzle stub and safety nozzle during periodical inspection September 2003 at Tsuruga Power Station Unit 2. Boric acid precipitation was found on the pressurizer relief piping nozzle after the heat insulator was removed. Once the boric acid was removed from the surface of the piping, the area was examined using the SUMP procedure. This is a procedure where a sample of the cracked area is removed and then analyzed. The results of this evaluation revealed a minor crack on the surface of the weld portion of the piping nozzle stub. On the same relief nozzle, cracking was found in an area where boric acid precipitates were not located. This second crack indication was also found in the weld region. Interestingly enough, both cracks developed in areas where a large portion of the weld consisted of touch up weld. In another instance, cracking was also located on safety nozzle A.

Reactor Coolant System Nozzles and Other Alloy 600 Cracking

B&W Plants

Arkansas Nuclear Unit 1 (Event Notification Report 36697 + LER 313-2000-003)

On February 15, 2000, a flawed weld was identified on an instrument connection to the reactor coolant system loop 'A' hot leg piping. Once the insulation had been removed, leakage was discovered on five other nozzles. Further investigation using NDE revealed that leakage was occurring through flaws in the partial penetration weld. Both axial and circumferential flaws were found. There was also a subsurface flaw found in a seventh nozzle. Six of the seven level tap nozzles and welds were replaced with Inconel 690. The seventh nozzle weld was repaired using a weld pad buildup and fillet weld.

Westinghouse 3-Loop Plants

Summer Unit 1 (Event Notification Report 37423 + LER 395-2000-008 + Final Westinghouse investigation report WCAP-15616, Revision 0, "Metallurgical Investigation of Cracking in the Reactor Vessel Alpha Loop Hot Leg Nozzle-to-pipe Weld at the V. C. Summer Nuclear Generating Station," January 2001.

On October 7, 2000, 100 pounds of boric acid was identified in the 'A' hot leg area of the reactor vessel. The potential leak area was identified on the first weld off the reactor vessel at the nozzle to pipe connection of the 'A' loop hot leg. Dye penetrant tests of the weld have confirmed a 4" long hairline crack in the weld between the hot leg piping and the reactor vessel nozzle. A visual inspection revealed boron on the weld between the hot leg piping and the vessel nozzle. Dye penetrant test identified a 4 inch circumferential indication in the weld. This indication was due to cutting/boric acid corrosion at the nozzle butter to nozzle interface. The inside ultrasonic, eddy current, and visual inspection identified the flaw as axial oriented and less than 3 inches in length. The flawed weld was removed and a new weld made of alloy 52/152 material was utilized.

CE Plants

Palisades Unit 1 (LER 255-1993-009)

On September 16, 1993, plant personnel identified a leak in the power operated relief valve line near the nozzle connection to the pressurizer. The crack initiated in the heat affected zone of the power operated relief valve Inconel 600 safe end. NDE and visual inspection found a circumferential crack approximately 3 inches in length (about 30% circumference).

San Onofre Unit 3 (LER 362-1995-001)

On July 27, 1995, radio-chemistry evaluation confirmed that reactor coolant solution (RCS) weepage had occurred on two RCS hot leg instrument nozzles. The accessible exterior of the two RCS hot leg nozzles were replaced with new 690 nozzles. The access to the interior of the RCS hot leg piping prevents welding from the inside of the RCS. Therefore the old nozzles were cut off half way through the RCS hot leg materials and the new nozzles were welded to the exterior of the RCS pipe.

St. Lucie Unit 2 (LER 389-1995-004)

On October 10, 1995, during a routine reactor coolant system visual leak check an apparent boric acid buildup was discovered on the 'B' side RCS hot leg instrument nozzle. Further investigation confirmed that pressure boundary leakage had previously occurred. The defective instrument nozzle and other instrument nozzles of the same heat were replaced with Inconel 690.

San Onofre Unit 3 (LER 362-1997-001)

On April 11, 1997, during a routine inspection, four hot leg RCS nozzles were found to have leaks and a fifth was suspected of leaking. It was suspected that the leakage was due to cracks through the nozzle in the heat affected zone of the partial penetration weld on each of the instrument nozzles. The outer half of the Inconel 600 nozzle was replaced with Inconel 690.

San Onofre Unit 3 (LER 362-1997-002)

On July 3, 1997, RCS nozzles were inspected and one hot leg spare RTD thermowell nozzle had an increased amount of white residue. An isotopic analysis determined the residue was boric acid from the RCS. Primary water stress corrosion cracking was believed to be the root cause of the leaks reported. It was believed that the leakage came from a crack in the heat affected zone of the partial penetration weld on each of the instrument nozzles. The outer half of the nozzle was replaced with Inconel 690 material using a half-nozzle repair technique.

San Onofre Unit 2 (LER 361-1998-002)

On January 26, 1998, plant personnel visually inspected all reactor Coolant System nozzles in the hot and cold legs, the pressurizer, and the steam generator channel heads. Seven nozzles were identified for repairs. The leakage from these nozzles was not measurable and the evidence of leakage could not be detected until the RCS insulation was removed. It was believed that the leakage from the nozzles came from cracks in the heat affected zone of the partial penetration weld of the instrument nozzles. Similar cracks have been caused by primary water stress corrosion cracking of Inconel 600 materials. Three of the nozzles were replaced with alloy 690 using a half-nozzle replacement technique. The other four were repaired using a Mechanical Nozzle Seal Assembly technique.

Waterford Unit 3 (Event Notification Report 35407 + LER 382-1999-002)

On February 28, 1999 evidence of boric acid leakage was found on three nozzles. One was on the RCS hot leg #1 RTD nozzle, a second was on the RCS hot leg #1 sampling line, and a third was on the RCS hot leg #2 differential pressure instrument nozzle. The three hot leg nozzles were repaired using the Mechanical Nozzle Seal Assembly technique.

Waterford Unit 3 (Event Notification Report 37442, 37434 + LER 382-2000-011)

On October 19, 2000, during a bare metal inspection, boric acid was found on two of the three mechanical nozzle seal assembly (MNSA) clamps that had been installed on hot leg nozzles during refueling outage #9. These clamps had been installed as temporary repairs till a permanent repair could be made during refuel 10. The MNSA clamp leakage could have been caused by the flange not being flat against the pipe. The leak could have also arisen from a brief leakage while the clamps seated. All three MNSA clamps were removed and permanent weld repairs were made on the leaking RCS hot leg nozzles.

St. Lucie Unit 1 (Event Notification Report 37919 + LER 335-2001-003)

On April 14, 2001, leakage was discovered on a pipe to nozzle connection on line I-3/4-RC-126. This line has been determined to be the “B” RCS hot leg instrument nozzle connection for differential pressure (D/P) transmitter PDT-1121D. The nozzle was replaced using a half-nozzle design.

Arkansas Nuclear Unit 2 (Event Notification Report 37199 + LER 368-2000-001)

On July 30, 2003, one RCS hot leg resistance temperature detector (RTD) nozzle was found to be leaking. Ultrasonic testing of the RCS hot leg base metal adjacent to the RTD nozzle showed that there were no cracks in the hot leg pipe around the RTD nozzle. The RCS RTD nozzle was repaired with an ASME code-qualified process.

Waterford Unit 3 (Event Notification Report 40278 + 40277)

Evidence of a reactor coolant leakage was detected on two Inconel instrument nozzles located on the top head of the pressurizer on October 25, 2003. One of these nozzles is the pressure transmitter that taps off of reactor coolant systems hot leg #2. The leakage was located in the annulus area where the nozzle penetrates the pressurizer head. Similarly, boric acid leakage was found on one hot leg 1 Inconel 600 instrument nozzle on October 27, 2003. There is potential leakage found for a steam generator instrument nozzle and also the pressurizer side shell nozzle.

CE Standard Plants

Palo Verde Unit 1 (Event Notification Report 36256 + LER 528-1999-006)

On October 2, 1999, evidence of a reactor coolant system pressure boundary leakage was discovered. The leakage was discovered at two Inconel 600 nozzles, one in each of the RCS hot legs. One was at the nozzle upstream of valve RCV0285 in the line to a steam generator #2 differential pressure instrument. The other was at the nozzle upstream of valve RCV-277 in the line to a steam generator #1 differential pressure instrument. The leakage was discovered in the form on small deposits of boron accumulated around the circumference of the nozzles. Isotopic analysis of the boron accumulation detected only long-lived radionuclides, indicating that it has taken over three years for the reactor coolant to migrate through the nozzle weld and wall thickness. The repair will include cutting of the old nozzle and welding on a new 690 nozzle to the outside diameter of the hot leg pipe.

Palo Verde Unit 1 (Event Notification Report 37878 + LER 528-2001-001)

On March 31, 2001, during a visual inspection of the reactor coolant system piping boric acid residue was discovered on the Inconel 600 RCS hot leg thermowell 1JRCETW0121HB. The visual indications were characterized as white streaks fanning out from the hot leg and continuing up the taper of the thermowell with some buildup on the top of the tapered portion. The repair consisted of cutting off the old Inconel 600 nozzle and welding an Inconel 690 plug.

Palo Verde Unit 3 (Event Notification Report 38332)

On September 20, 2001, evidence of reactor coolant leakage was discovered. The leakage was discovered in a RCS hot leg temperature nozzle. The RCS hot leg nozzle was located in the RTD nozzle for and in-service temperature detector (loop #1, equipment ID: 3JRCETW112HD). The leakage was identified by the discovery of boron deposits accumulated around the circumference of the hot leg nozzle.

Palo Verde Unit 3 (Event Notification Report 39714 + LER 530-2003-002)

On March 29, 2003 engineering personnel performing preplanned visual examinations of reactor coolant system piping discovered boric acid on the reactor coolant system (RCS) hot leg instrument nozzle. There was boric acid precipitation was found around the instrument nozzle that penetrates Loop 1 hot leg. The leaking heater sleeve was replaced with an Inconel 690 nozzle.

Boiling Water Reactor Plants - All Components

GE Type 4 Plants

Vermont Yankee (LER 331-1999-006)

On April 26, 1986, Ultrasonic examination of the N5A and N5B core spray safe-end to nozzle welds indicated IGSCC. The IGSCC cracks were located in the nozzle weld butter material, Inconel 182. The indications were predominately axial in orientation with seven indications in each weld. The repair technique will utilize a weld overlay procedure.

Hope Creek 1 (LER 354-1997-023)

On September 19, 1997, a leak was discovered on core spray nozzle safe-end weld N5BSE associated with the "A" Core Spray subsystem. The N5BSE weld was nondestructively tested in the previous refueling outage. This NDE test had been improperly evaluated and the crack had been unrecorded. The cause of the through wall leakage has been attributed to IGSCC in the Alloy 182 weld metal. A weld overlay technique was used to repair the leaking core spray nozzle safe-end weld.

Duane Arnold 1 (LER 331-1999-006)

On November 5, 1999, two indication of IGSCC were identified in weld RRB-F002. One indication was approximately 44% through-wall and the other was approximately 65% through-wall. The inspection was expanded and a 65% through-wall crack was found in weld RRD-F002. These two F002 welds were repaired by completing weld overlays using Alloy 52. The cause of the cracking was IGSCC in the 182 weld metal. A reference LER which may have had similar cracking is 331-1985-010.

Susquehanna 1 (Event Notification Report 40605)

On March 23, 2004, during a routine inspection an indication was discovered on the N1B penetration. This reactor vessel penetration is associated with the reactor recirculation B loop. The crack had been detected during previous outages, however, it was not designated as a crack. The crack is circumferential and is approximately 50% through wall. The length of the crack is roughly 2.2 inches (approximately 7% of the diameter). The plant is planning to use a weld overlay technique to repair the flaw.

-GE Type 3 Plants

Pilgrim 1 (LER 293-2003-006)

On October 1, 2003, a reactor coolant leakage was detected in reactor vessel nozzle to cap weld. The crack was contained within the 182 weld metal. After nozzle was initially welded to the cap there were defects detected and the weld was repaired. The current leakage is believed to be due to a crack left in the weld materials during the previous repair procedure. The repair procedure utilized a weld overlay technique with Inconel 52.