a rapid change in leakage, some of the predominant isotopes that provide the indication include long lived Cesium137 and Cobalt 60. These isotopes have a half life in the range of 5 to 30 years. Based on this, even with a constant RCS leak rate and coolant activity, they could tend to constantly accumulate in containment over the course of a fuel cycle, giving a continuously increasing detector response that could be difficult to distinguish from subtle changes in leakage. The output would also be expected to fluctuate with filter changes. Therefore, the particulate detector was not further considered for possible long term detection of CRDM nozzle leakage.

Iodine Monitor (Channel 3)

After passing through the particulate filter, the sample is drawn through an iodine collector. The iodine monitor is a 3 inch 4 pi lead shield containing the iodine collection cartridge and a gamma scintillation detector. The iodine collector efficiency is greater than 95 percent. The output from the detector is fed to a microprocessor. The microprocessor looks at two windows from this detector. The upper window is a background (5 percent above the iodine peak) and the lower window is centered on the iodine peak. The upper value is subtracted from the lower value giving a true iodine reading with output converted to μCi/cc. The iodine detector is capable of detecting iodine radioactivity on concentrations as low as 7×10^{-7} μCi/cc of containment air. The predominant Iodine isotopes released from the reactor coolant are Iodine 131 and 133 with half lives of 8 days and 21 hours respectively. For a constant RCS leakage rate and coolant activity, these isotopes will reach a stable equilibrium value in containment and would thus theoretically provide a direct and valid indication of a slowly evolving RCS leakage trend. These isotopes have the added advantage of being actively trended in the reactor coolant for tracking of fuel defects, so that changes in coolant activity could theoretically be accounted for in so that output could be used to determine RCS leakage rates.

Output data from the detectors was manually recorded on a monthly basis from late 1992 through the present. However, due to high cycle 13 coolant activity and known increases in RCS leakage, the detectors frequently saturated during the fall of 2001. This resulted in a loss of alarm function for the remaining channels. Therefore, the carbon filters were removed from the detectors in November 2001, effectively removing the Iodine Channels from service. Data prior to this time is presented on Figure 41. Although the output indicates a clearly increasing trend, the output readings from this monitor suffer from a significant amount of scatter. The cause of the scatter is not definitively known, however, it might be related to readings being taken near the time of change-out of the carbon elements (response not at equilibrium) or it might be related to actual changes in CTMT atmosphere conditions (e.g. scrubbing of the iodine by condensate on the containment air coolers, or retention by condensate in the sample lines.) Further, a large portion of the trend is undoubtedly due to increasing RCS activity due to fuel defects. An attempt was made to separate the effects of the coolant activity by taking a ratio of detector output with coolant activity. This also resulted in an increasing trend, but it suffered doubly from the scatter in both the monitor data and RCS activity data. In the end, although increased leakage was clearly detectable, there is no means to distinguish CRDM nozzle leakage from any other RCS leakage, and so this indication was not particularly valuable.

Noble Gas Monitor (Channel 1)

After passing through the particulate and Iodine monitors, the gas sample is finally drawn through the noble gas monitor. The monitor housing is a 4 inch 4 pi lead container. The detector is a beta detector with an internal light emitting diode (LED) check source. The output of the detector is fed to a microprocessor where the counts per minute are converted to μCi/cc. Some of the predominant isotopes that remain to be counted by this detector are Xenon 133 and 135, with half lives of 9.2 hours to 5.2 days. For a constant RCS leakage rate and coolant activity, these isotopes will also reach a stable equilibrium value in containment and would thus theoretically provide a direct and valid indication of a slowly evolving RCS leakage trend. Trend data from these monitors is presented in Figure 42.

The output from these detectors is much more consistent than the output of the iodine channels. These detectors are sensitive and reflective of RCS leakage trends and changes in RCS activity. Detector output is particularly sensitive when RCS activity is high, as during cycle 13. Under this condition, noble gas activity could provide indication of very small RCS leakage prior to positive identification through the RCS inventory balance. To determine a leak rate, a representative combination isotopes in the RCS would need to be found to achieve an appropriate scaling factor to screen out the effects of RCS activity. Assuming this was accomplished, other RCS leakages could still mask the relatively small leakage expected from a CRDM nozzle leak. Therefore, these detectors are also of limited value for diagnosis of CRDM nozzle leakage.

RE4597 BA and RE4597AA Filter Changes

RE4597AA and RE4597BA have been the subject of numerous Condition Reports due to moisture in the lines, and clogging of the filter elements with boric acid. Moisture in the lines has been associated with restarting from outages and is attributable to high CTMT humidity. The humidity arises from the initial humidity when CTMT is closed and long term accumulation from a variety of primary and secondary leak sources. Temperature changes along the sample piping as the sample is continuously drawn from CTMT to the monitors can cause condensation of water in the piping before the monitors and can interfere with monitor operation. Due to the variety of sources of moisture, and the fact that this condition has occurred for a prolonged period of time (reference PCAQR 92-0346), it is not particularly associated with nozzle leakage.

Relatively large RCS leakage sources have the demonstrated potential to produce an aerosol mist due to flashing and evaporization of the jet of liquid as it exits from the leak. An example of this type of leakage occurred when the "head to hot leg vent" line developed a flange leak in 1992. When the leakage source contains borated water, the boric acid is dispersed with the aerosol as a fine particulate material. This material remains suspended in the CTMT atmosphere for a prolonged period, before eventually settling out on CTMT surfaces and appearing as a fine powder. When ingested by the CTMT radiation monitors, the boric acid will prematurely clog the monitor filters and require frequent filter changes. This condition occurred during the RPV head to hot leg vent line flange leak, but returned to normal following that repair.

Normally, the change frequency for the RE4597AA and RE4597BA filters is approximately 30 days, and is dictated by schedule rather than low flow. However, in March of 1999 fouling of the monitor filters began to recur (CR1999-0372, CR1999-0861, CR1999-0882). Initially, this was attributed to the disabling of the pressurizer code safety valve rupture discs in late 1998 (discussed in the CAC section). It was noted that the service life of the filters had decreased, particularly for RE4597BA. However, by May 19, 1999, the boric acid deposits on the filters had developed a "yellow" or "brown" appearance. Under CR1999-1300, sample filters were sent to Southwest Research Institute (SRI) for analysis. The SRI report (Project 18-2321-190) indicated that the samples contained predominantly ferric oxide from corrosion of iron components. An adjunct report from Sargent and Lundy (Project 10294-033) indicated that the fineness of the particles suggested that it was attributable to a steam leak. From May of 1999 until April 2001, filter changes on RE4597BA were required on an irregular 1 to 3 week interval. It was noted in CR01-1110 that the filter life had reduced to around 3 days. The sample point was changed to the alternate location near the personnel lock, and service life improved slightly.

However, by November of 2001, filter replacements were again required approximately every other day. On November 2, a blank (no carbon) cartridge was installed in the iodine channels of both monitors to eliminate a frequent alarm condition. Throughout the period of 1999 through 2001, RE4597AA exhibited similar, but slightly less severe symptoms.

The reactor service structure, which encloses the CRDMs, CRDM flanges, and CRDM nozzles, is ventilated by one of two fans that take suction on the area immediately surrounding the CRDMs and CRDM flanges. It takes an indirect suction on the area surrounding the CRDM nozzles, drawing through the mirror insulation that separates the CRDM nozzles from the CRDMs and CRDM flanges. The fans discharge on the 603 elevation, in the North-East quadrant of the reactor building. Airborne flange leakage or nozzle leakage would be exhausted by the ventilation fans to this area. The fan exhaust is closer to the normal suction of RE4597BA than to the normal suction of RE4597AA. This would tend to explain why boric acid fouling was more severe for RE4597BA, and why the symptoms were reduced when switching to the alternate sample location, which is diametrically opposite the CRDM ventilation fans.

Accumulation of boric acid on the radiation monitor filters was recognized to be symptomatic of an RCS leak as soon as it occurred. Significant efforts were made, especially during the cycle 12 mid-cycle outage in 1999 and 12RFO in 2000 to locate the source of leakage. During that outage, the only significant leakage potentially capable of producing the amount of boric acid necessary to exhibit the necessary symptoms was found to be leaking CRDM flanges, particularly at the nozzle 31 location. However, the presence of iron oxide in the boric acid on the filter elements was not explained.

In August 1999, four high efficiency particulate filters were placed in CTMT near the elevator on the 603' elevation. These 500 cfm filtration units were intended to help clean up the CTMT atmosphere on the theory that airborne material was left over from the cycle 12 mid-cycle outage. The filters were removed in October 1999. The filters had no notable effect on the CTMT atmosphere.

Based on the observations that there was a high boric acid accumulation near the CRDM exhaust fans and no leaking CRDM flanges found in 13 RFO, it can now be inferred that the boric acid found in the RE4597 filters (and in the CACs) originated at the CRDM nozzles and was dispersed by the CRDM exhaust fan.

3.3.6 *Containment Recirculation Fan/Fan Failures*

The Containment Recirculation System (CRS) is composed of two non-safety related fans and associated ductwork. The CRS circulates the air in the Containment Dome during all plant modes of operation to eliminate the temperature stratification. The CRS is normally operated continuously. However, in February of 1999, CRS fan 1 failed and remained out of service. In March of 2001, CRS fan 2 failed and remained out of service. The failure mode involved failure of the motor bearings, and significant destructive rubbing of the fan blades on the housings.

Failure of fan 1 significantly preceded discovery of brown deposits on RE4597 filters in May of 1999. Failure of fan 2 did not occur until well after iron appeared on the RE4597 filters. Iron continued to appear on the filters well after both fans were out of service. The out of service dates also do not coincide with other events. The particulate iron that would be expected from the fan blades is not similar in particle size that was found on the RE4597 filters. Therefore, the failure of the CRS fans does not appear to be the cause of the iron deposits on the RE4597 filters.

3.4 Programs Important to Preventing Problems

This section of the data analysis provides a discussion of programs the root cause team viewed as important to preventing this type problem. Industry programs are intended to provide advance warning and to recommend approaches to avoiding significant problems. The Boric Acid Corrosion Control and the Inservice Inspection (ISI) programs are intended to provide a level of defense by ensuring the integrity of the RCS and supporting systems used to mitigate plant transients. The review included interviews with the program owners. Both programs were reviewed as part of the root cause investigation.

3.4.1 *B&W Owners Group and Industry CRDM Nozzle Related Initiatives*

In November 1990, the B&WOG Materials Committee issued Report 51-1201160-00, Alloy 600 SCC Susceptibility: Scoping Study of Components at Crystal River 3 (reference 8.2.10). Very little attention had been given to inspection for PWSCC in Alloy 600 applications other than that associated with the steam generator tubing. As a result of the reported instances of PWSCC in the pressurizer heater sleeves and instrument nozzles in several domestic and foreign PWRs, the NRC felt that it may be prudent for licensees of all PWRs to review their Alloy 600 applications in the primary coolant pressure boundary, and, when necessary, implement an augmented inspection program (reference IN 90-10). The Materials Committee initiated a scoping study to investigate potential problems associated with PWSCC of Alloy 600 material used in B&W designed RCS components. The report summarized the completed research regarding Alloy 600 components used at a target B&WOG plant Crystal River 3. Based on this information, a susceptibility rating was given, along with recommendations for ensuring RCS integrity through inspections of appropriate components. The applications of Alloy 600 at other B&W operating plants were identified and the applicability of the target plant evaluation to these other operating plants was confirmed. This summary was to be used by the B&WOG Materials Committee in assessing the potential for future PWSCC occurrences with Alloy 600 components at B&W operating plants. The report notes that it is expected that the locations having the highest temperatures in the RCS would be the most susceptible to PWSCC. The RPV upper head is identified as one area where attention should be given. The report recommends the control rod housing bodies be inspected, if possible, at an opportune time. The report includes a table of Alloy 600 locations at Davis-Besse, which includes the 69 CRDM nozzles. The report also includes a summary of PWSCC occurrences of in-service RCS Alloy 600 components.

In December 1990, EPRI issued EPRI NP-7094, Literature Survey of Cracking of Alloy 600 Penetrations (EPRI Project 2006-18) (reference 8.5.10) to document the problem of stress corrosion cracking of Alloy 600 penetrations in PWR pressurizers and to identify corrective actions that utilities can take to address this issue. The document lists the CRDM nozzles as an Alloy 600 component.

In October 1991, the first EPRI workshop on PWSCC of non-steam generator Alloy 600 materials in PWR plants was held, with representatives from the U.S. and French nuclear facilities, all U.S. Owners Groups (Westinghouse, Combustion Engineering, and B&W), EPRI, the U.S. Navy, and various vendors/consultants. This workshop provided extensive coverage of PWSCC in pressurizer instrument nozzles, pressurizer heater sleeves, steam generator drain lines, and hot leg instrument nozzles. The B&WOG provided an update on B&W activities, including the Materials Committee scoping study of Crystal River 3 and the areas of concern, including the Control Rod Housing Bodies. Later, it was learned that during a 10-year hydrostatic test in September 1991, the French Bugey 3 plant discovered a leak in a CRDM nozzle, via a through-wall crack. The crack was caused by PWSCC in an area of high residual stresses caused by the J-groove weld joining the nozzle to the RPV head. Additional cracks were subsequently found in other plants in France, Sweden, and Belgium.

On May 12, 1992, the B&WOG Materials Committee met with the NRC staff and provided a presentation on "Work on PWSCC of Alloy 600 Nozzles and Components" which included information on the Bugey 3 CRD nozzle leakage. NRC concurred with the B&WOG that, based on the available information on the French CRDM nozzle inspection, there is no immediate safety concern due to the fact that the identified cracks are axial in nature. The NRC suggested another meeting during 1st quarter 1993 to cover the following on the CRDM nozzle cracking vis-a-vis B&WOG plants:

1. 50.59 Safety Evaluation to provide sufficient assurance that the issue is not a safety concern
2. CRDM nozzle inspection strategy/criteria
3. Evaluation of leak detection/monitoring system.

On 8/10/92 – 8/11/92, there was an EPRI Alloy 600 Coordinating Group Meeting Concerning CRDM Nozzle Cracking attended by representatives from each of the NSS vendors, several utilities, and Dominion Engineering. Work on CRDM nozzle cracking in the Owners Groups was presented and discussed. One item discussed was that no one was expected to inspect CRDM nozzles during the 1992 fall outage schedule unless required by the NRC. The NRC position was expected to be finalized at a Westinghouse Owners Group (WOG) meeting on 8/18/92.

On August 18, 1992, the NRC met with members of WOG to discuss the safety significance of CRDM penetration cracking and update the status of WOG's Alloy 600 program. The meeting was attended by representatives from each of the Nuclear Steam Supply (NSS) vendors, each of the owners groups, several utilities, and consultants. The NRC provided an overview of Alloy 600 PWSCC and their view on CRDM nozzle inspections. The staff viewed the CRDM nozzle cracking as a minimal safety impact, but that prudence suggested an orderly inspection program. The NRC was concerned that the potential for cracking exists in a large number of nozzles and that there is concern with boric acid corrosion of the RPV head. The staff presentation slides indicated the following inspection, evaluation, and repair guidance: (1) For PWR plants refueling before spring 1993, visual inspection during leakage test, with special attention to CRDM penetrations at periphery locations and visual inspections (VT-2 quality) remote or direct to inspect the inside surface of the spare CRDM penetrations; (2) For PWR plants refueling after Spring 1993, PT and eddy current (EC) inspections of the inside surface of all spare CRDM penetrations; (3) EC inspection of CRD sleeved penetrations if cracks are found; (4) Provide flaw acceptance criteria; and (5) Develop corrective actions for CRDM penetrations. Recent work on CRDM nozzle cracking in the WOG was then presented and discussed. It was stated that inspection of CRDM nozzles during the 1992 fall outage schedule was not planned by any of the owners groups unless ongoing safety evaluations indicate that there is a safety concern. The NRC appeared to agree with this, but wanted to review the WOG safety evaluation (scheduled for completion 10/31/92) and requested another meeting with the WOG in November. The NRC also stated that they would entertain a submittal without an NDE (ECT or UT) inspection plan but the basis for this decision must be very convincing. Coordination of the activities of the Owners Groups on Alloy 600 CRDM penetration cracking was planned to be done by Nuclear Utility Management and Resource Council (NUMARC). The NRC staff believed the reported cracking in CRDM penetrations was not an immediate safety issue requiring regulatory action. There was time for a thorough, disciplined analysis of the safety significance, the approach to RPV head inspection, criteria for taking repair actions, and possible regulatory guidance.

On October 2, 1992, the B&WOG issued a proprietary Alloy 600 PWSCC Time-To-Failure Models report (reference 8.2.11), presenting a PWSCC susceptibility ranking model and six susceptibility models that had been proposed within the nuclear industry to model time-to-failure of Alloy 600 components as a result of PWSCC. The PWSCC susceptibility ranking model for Alloy 600 RCS components was based on carbon content of the material, anneal temperature and duration, operating temperature, and operating and residual stresses. A ranking of 4, 4-5, or 5 indicates a high (50%) probability of failure within 20 years; a ranking of 3 or 3-4 indicates a medium (50%) probability of failure within 40 years; and a ranking of 2-3 or below indicates a low probability of failure within 40 years. All failures at the time had been ranked between 4 and 5 with this ranking model. The report provided the susceptibility ranking of the Alloy 600 components on B&W designed plants. The Davis-Besse CRDM nozzles were of four different heat numbers: heat M3935 was ranked as 2-3; heat NX5940 was ranked 3-4; heat C2649 was ranked 5; and heat M4437 was ranked 4-5. Based on one of the models, the time-to-failure calculation for the worst case (heat C2649) predicted 123 EFPY for 50% of the population to initiate cracks. The report concluded that, although none of the models addressed in this document accurately predicts any of the existing industry failures of Alloy 600 components, it contained a good base of ideas to improve the time-to-failure model.

In December 1992, the second EPRI workshop on PWSCC of Alloy 600 in PWRs was held, with representatives from U.S., French, Swedish, and Japanese nuclear facilities, all U.S. Owners Groups, the U.S. Navy, and various vendors. Workshop sessions focused on concerns about PWSCC of alloy 600 penetrations in the RPV head (CRDM nozzles) in several plants, including the Bugey 3 plant in France. A stress analysis summary concluded the stresses are highest in the outermost nozzles for Westinghouse plants, while the stresses are essentially the same for central and outer row nozzles for B&W plants. Another report indicated field experience to date shows that cracks have occurred predominantly in peripheral row nozzles, consistent with the results of finite element stress analyses.

Later that month, B&W issued a proprietary CRDM Nozzle Characterization report (reference 8.2.12), regarding PWSCC of CRDM nozzles. The fabrication and manufacturing processes for B&W-design CRDM nozzles and French-design CRDM nozzles were discussed. A comparison of this information was made, and the similarities and differences were noted. It was determined that B&W-design nozzles are not significantly different than the French-design nozzles, and, thus, are not immune to PWSCC. In the report, Davis-Besse is noted as having all 24 of its peripheral nozzles rated as "very high susceptibility" for PWSCC, as are 40 of its 45 non-peripheral nozzles. This report differs from the previous report (10/2/92) in that heat NX5940 was now ranked as 5 (instead of the previous 3-4). The report also lists the heat number for each CRDM nozzles and notes that nozzles 1-5 are all of heat number M3935, the lowest susceptibility ranking (2-3) for Davis-Besse nozzles.

An Ad Hoc Advisory Committee (AHAC) headed by NUMARC with members from all three Owners Groups and EPRI was formed to formulate the CRDM nozzle inspection criteria and coordinate the relevant industry activities. On March 3, 1993, the AHAC met with the NRC and discussed the WOG Safety Evaluation. The B&WOG committed to perform an evaluation of the safety significance of potential nozzle cracking.

On May 26, 1993, the B&WOG issued BAW-10190P, Safety Evaluation For B&W Design Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking (reference 8.2.7) summarizing the stress analysis, crack growth analysis, leakage assessment, and wastage assessment for flaws initiating on the inner surface of the B&W designed CRDM nozzles. The overall conclusion reached in this evaluation was that the potential for cracking in the CRDM nozzles does not present a near-term safety concern. Crack growth analysis predicted that once a crack initiates, it will take a minimum of six years for the flow to propagate through-wall. If a crack propagates through-wall above the nozzle-to-head weld, leakage was anticipated and a large amount of boric acid deposition was expected. Once boric acid deposition occurs from leakage, wastage of the RPV head can initiate. It was predicted that wastage of the RPV head can continue for six years before ASME code limits are exceeded. The B&WOG utilities developed plans to visually inspect the CRDM nozzle area to determine if through-wall cracking had occurred and if boric acid deposition was occurring as result of a through-wall crack. The report identifies that at each of the B&WOG utilities' plants, a walkdown inspection of the RPV head was implemented as part of the response to NRC Generic Letter 88-05. Enhanced visual inspection of the CRDM nozzle areas was also incorporated. If any leaks or boric acid crystal deposits are located during the inspection of the RPV head area, an evaluation of the source of the leak and the extent of any wastage was required to be completed. A conservative wastage volume of 1.07 cubic inches per year was believed to be possible from a leaking CRDM nozzle. The postulated corrosion wastage within and in the vicinity of the RPV head penetration from a leaking CRDM nozzle would not affect safe operation of the plant for at least six years. The boric acid deposition was expected to be detectable by the current GL 88-05 inspections. Since inspections of the RPV head area (for leakage and boric acid deposits) are performed during each outage, it was thought to be unlikely that a leak would go undetected for a period of six years. The evaluation concludes excessive wastage of the RPV head will not occur before leakage is detected either by visual observations in accordance with utility responses to GL 88-05 or the plant leakage detection system. The B&WOG also stated it was evaluating the potential for crack initiation and propagation on the nozzle outer surface, although preliminary evaluation of the through-wall stress distribution indicates that, even if a circumferential crack initiates on the outer surface, the crack will be self-relieving and will not cause separation of the nozzle. The B&WOG was continuing its involvement in the NUMARC-sponsored AHAC for PWSCC of CRDM nozzles, including the industry-sponsored crack growth testing of CRDM penetration materials. Duke Power Company scheduled an inspection of one B&W designed reactor in the fourth quarter of 1994.

On November 19, 1993, the NRC issued its Safety Evaluation for Potential Reactor Vessel Head Adaptor Tube Cracking to NUMARC (reference 8.3.20). The staff concluded that there was no immediate safety concern for cracking of the CRDM penetrations. The bases for this conclusion (reference 8.3.2) were that if PWSCC occurred at RPV head closure penetrations: the cracks would be predominately axial in orientation, the cracks would result in detectable leakage before catastrophic failure, and the leakage would be detected during visual examinations performed as part of surveillance walkdown inspections before significant damage to the RPV head would occur. This finding was predicated on the performance of the visual inspection activities requested in Generic Letter (GL) 88-05. Also, special nondestructive examinations were scheduled to commence in the spring of 1994 to confirm the safety analyses for each PWR owners group.

On December 14, 1993, the B&WOG Materials Committee issued BAW-10190P Addendum 1, External Circumferential Crack Growth Analysis for B&W Design Reactor Vessel Head CRDM Nozzles (reference 8.2.8) providing an evaluation of external circumferential crack growth, gross leak-before-break mechanism, and the stress affects of CRDM nozzle straightening. The report concludes that there was no possibility for an external circumferential flaw indication to grow circumferentially to the point of becoming a safety concern. The overall conclusions presented in

B&W-10190P remained unchanged with this addendum. It was concluded the GL 88-05 walkdown visual inspections of the RPV head areas provide adequate leak detection capability.

In March 13, 1994, the RCS System Engineer initiated PCAQR 94-0295 regarding a commitment in the commitment management system that was closed (complete) and not converted to an ongoing commitment. The commitment required a visual inspection of the RVP head every refueling to determine the potential for CRDM nozzle cracking in support of a B&W safety evaluation to the NRC. The PCAQR evaluation identifies the inspection is covered in the existing program as outlined in NG-EN-00324, Boric Acid Corrosion Control. The commitment was closed based on the NG-EN-00324 inspections and the fact that the NRC saw enhanced inspections as being "prudent" but not necessary were to be put in the next outage contract.

In November 1994, the 1994 EPRI Workshop on PWSCC of Alloy 600 in PWRs was held. The workshop summarized the field experience associated with PWSCC of Alloy 600 CRDM nozzles, reviewed the current status of inspection, repair, and remedial methods as well as strategic planning models, and discussed stress analysis results as well as PWSCC initiation and growth in Alloy 600. The workshop was attended by domestic and overseas utilities, PWR vendors, research laboratories, and consulting organizations. Three U.S. plants had inspected CRDM nozzles; no cracks were found in one plant and only minor cracking was observed on one nozzle in each of the other two plants. Results of inspections in France, Sweden, Spain, Belgium, Japan, and Brazil revealed a trend toward earlier axial cracking in plants with forged nozzles as opposed to those made from rolled bars or extrusions. It was also thought that other factors such as surface finishing could play a role (see reference 8.5.4).

On April 7, 1997, Davis-Besse received GL 97-01 Degradation of CRDM/CEDM Nozzle and other Vessel Closure Head Penetrations (reference 8.3.2). The letter requested plants describe their program for ensuring the timely inspection of PWR CRDM and other RPV head penetrations (VHP). In July1997, the B&WOG Materials Committee issued BAW-2301, B&WOG Integrated Response to Generic Letter 97-01: "Degradation of Control Rod Drive Mechanism Nozzle and other Vessel Closure Head Penetrations" (reference 8.2.1). On July 28, 1997, Davis-Besse responded to the GL 97-01 endorsing BAW-2301. The BAW topical report provides the justification and schedule for an integrated VHP inspection program.

The BAW-2301 introduction reiterates conclusions discussed in references 2.7 and 3.20. The introduction furthermore states PWSCC for CRDM nozzles and other VHPs will not become a long-term safety issue provided the enhanced boric acid visual inspections, performed in accordance with GL 88-05, are continued. An axial crack would lead to a leak on one or more nozzles and result in a significant deposition of boron crystals. It is very unlikely that this type of accumulation would continue undetected with regular walkdown inspections of the RPV head area. If the crystals remain hidden by the RPV insulation, the insulation would begin to bulge as a result of this accumulation of crystals. This deposition would easily be detected prior to significant damage to the RPV head. Therefore, the RPV head's structural integrity would not be jeopardized, thereby eliminating any safety concerns with PWSCC of these nozzles. In order to assure the assumptions of the original safety evaluation remain valid, an integrated inspection program had been developed to address this issue for the B&WOG plants.

The BAW-2301 report presents the integrated B&WOG inspection program. Oconee 2 and Crystal River 3 are identified as two of the B&WOG plants most susceptible to PWSCC, as currently ranked. These two plants either have or will perform inspections of the RPV head nozzles from beneath the RPV head. Oconee 1 and 3, Davis-Besse, ANO 1, and TMI 1 do not have CRDM nozzle inspections planned in the near term (1998-2000).

In May 1998, the Davis-Besse Materials Committee representative initiated a procedure change request to NG-EN-00324, Boric Acid Corrosion Control. The change requested the B&WOG Materials Committee Report 51-1229638 Boric Acid Corrosion Data Summary and Evaluation be add to a note that identifies material that contain helpful reference material for determining boric acid corrosion rates. The information was incorporated into the procedure as requested in April 1999.

On April 30, 2001, the NRC issued Information Notice (IN) 2001-05 to alert plants to the recent detection of through-wall circumferential cracks in two CRDMs nozzles and weldments at Oconee 3. On May 2, 2001, CR 01-1191 initiated identifying the need for a project plan with team members developed to prepare Davis-Besse for a cracked CRDM J-groove weld. The CR identifies all three units at Oconee and one unit at ANO have inspected for and found cracked J-groove welds around their CRDM nozzles.

On August 3, 2001, NRC issues NRC Bulletin 2001-01 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles. The discussion section identifies recent identification of circumferential cracking in CRDM nozzles and axial cracking in the J-groove weld has resulted in the NRC staff reassessing its conclusion in GL 97-01 that cracking of VHP nozzles is not an immediate safety concern. Circumferential cracking in CRDM nozzles were identified at Oconee 2 and 3, and axial cracking in the J-groove weld in CRDM nozzles were identified at Oconee 1 and ANO 1 (i.e., B&W plants). The findings at Oconee 2 and 3 highlight the possible existence of a more aggressive environment in the CRDM housing annulus following through-wall leakage; potentially highly concentrated borated primary water could become oxygenated in this annulus and possibly cause increased propensity for the initiation of cracking and higher crack growth rates. Regulatory Affairs initiated CR 01-2012 in response to the bulletin.

Between September 4 and November 30, 2001, Davis-Besse met with and docketed responses to the NRC regarding NRC Bulletin 2001-01. In discussion held with the NRC on November 28, 2001, Davis-Besse committed to a 100% qualified visual inspection, non-destructive examination (NDE) of 100% of the CRDM nozzles and characterization of flaws through destructive examination should cracks be detected. Several other commitments were also made at that time including moving forward the start of the scheduled refueling outage from April 1 to no later than February 16, 2001.

3.4.2 *Davis-Besse Boric Acid Corrosion Control Program*

As discussed above, and as part of the root cause evaluation of the Davis-Besse RPV head degradation, the Boric Acid Corrosion Control program was reviewed. The intent of this review was to compare various aspects of the program to GL 88-05 and how the program is currently being implemented as it relates to the RPV head.

Generic Letter 88-05 was issued March 17, 1988 to address the effects of boric acid leaks on carbon steel components. All license holders for PWRs were required to address the issues identified in the generic letter. GL 88-05 identifies four areas that must be addressed in the plant specific boric acid program. The areas include:

- determination of principal leak locations where leaks may occur that are smaller than technical specifications allow
- procedures for locating small leaks
- methods of conducting examinations and performing engineering evaluations to address the impact of the leak
- corrective actions to prevent recurrence of this type of corrosion

The Boric Acid Corrosion Control program procedure (NG-EN-00324) was reviewed against these same points.

Principal Leak Locations

The procedure identifies the following areas as principal locations for possible leaks:

- Steam Generator and Pressurizer manways and handholes
- Seal Welds
- Thermowells
- Reactor Coolant Pump and other pump seals and casing flanges
- Control Rod Drive Flanges
- Piping Flanges and Bolted Connects
- Valve Bonnets and Packing Glands
- Reactor Vessel Head O-rings

The procedure does not address industry known leakage areas such as the CRDM nozzle issue or potential leakage areas such as the lower RPV head area. Currently the procedure is limited to inspections of systems and components inside Containment.

As an example, PCAQR 94-0295 discussed the need for enhanced inspections of the RPV head in addressing the CRDM nozzle cracking issue. The later text in PCAQR 94-0295 states that B&W amended its safety evaluation following feed back from the NRC that stated enhanced inspections were not required. B&W's amended safety evaluation took credit for the GL 88-05 inspections. Discussion with Framatome indicate the safety evaluation dated May 1993 was never changed to eliminate the need for enhanced inspections. See PCAQR 94-0295 discussion in the condition report section for additional details concerning this specific issue.

Procedures for Locating Small Leaks

Step 2.1.1 of the Boric Acid Corrosion Control procedure identifies a number of station procedures that support inspection and identification of leakage. Several of the procedures were reviewed to generally determine the kinds of inspections that are required. DB-OP-06900 Plant Heatup requires an inspection at operating temperature and pressure. DB-OP-01200 Reactor Coolant System Leakage Management provides guidance and trigger points with regard to Technical Specification leakage values. The DB-OP-01200 includes trigger points for "buildup of boric acid on equipment requires frequent containment entries to clean and/or inspect". Both procedures prompt action to identify and characterize leakage however, the values of unidentified RCS leakage causing this type of a condition are relatively small.

DB-PF-03065 Pressure and Augmented Leakage Test performs a leakage inspection at temperature and pressure in support of the ISI program. This procedure will be addressed in the ISI section.

Conducting Examinations and Engineering Evaluations

The procedure outlines various activities that "should" be performed to ensure the component is serviceable and meets code requirements. These activities include assessing for corrosion, wastage, and performing engineering evaluations. The procedure provides reasonable guidance in this area; however, there are many places in the procedure that use the word "should" instead of "shall." The use of "should" allows a choice to be made in an area that involves technical insight. The use of "should" may be used if the technical staff involved with the decision making is highly experienced or reviewed by a highly experienced peer or supervisor.

The RCS System Engineer was interviewed concerning activities related to the spring 2000 refueling outage (12RFO). The Boric Acid Corrosion Control Inspection Checklist (BACCIC) provides a method to both characterize and disposition a Boric Acid leak. The RCS System Engineer was questioned concerning the dispositioning of the BACCIC that was issued to document the Boric Acid on the RPV head. The BACCIC closure process is not well defined. The RCS System Engineer could not recall how the BACCIC was closed out in 12RFO. Additionally, there was no evaluation to address the Boric Acid remaining on the RPV head during cycle 13. See CR 2000-0782 in the Condition Report section for additional details concerning this specific issue.

Corrective Actions to Prevent Recurrence

This portion of the procedure describes various types of modifications to prevent leaks and/or mitigate the outcome of the leak. The section does not discuss, however, reviewing the maintenance history of the leaking component, reviewing maintenance procedures and work practices, reviewing industry documents such as "EPRI Good Bolting Practices," and reviewing industry issues (Operating Experience) for possible improvements to recurring issues.

The program owner was interviewed on the subject of the Boric Acid Corrosion Control program. The program owner characterized his role in the Boric Acid Corrosion Control program as a caretaker. The program owner coordinates the walk down of Containment and then provides the System Engineer with a copy of the Boric Acid Corrosion Control Inspection Checklist resolution. The program owner also is responsible for maintaining and updating the administrative procedure that controls the program.

Step 6.7.4 of the Boric Acid Corrosion Control procedure (NG-EN-00324) identifies increased responsibilities for the program owner during outages. The procedure describes the program owner functions as: coordinates decontamination and insulation removal for detailed inspection of components, develop plans to resolve leaks (identify and prioritize), coordinate repairs with the pressure test engineer, and provide a status of the repair activities to Outage Management.

The Boric Acid Corrosion Control program does not require the retention of any Boric Acid Corrosion Control Inspection Checklist. The BACCIC contains both the initial assessment of the leak (including corrosion) and the results of any subsequent evaluations. The BACCIC contains signatures for the resolution of the leak, but does not require review or supervisory acceptance. Neither the Boric Acid Corrosion Control program owner or the RCS System Engineer could produce copies of the completed BACCIC sheets from the 12RFO, therefore an evaluation of the effectiveness of the actual reviews could not be performed.

3.4.3 *Davis-Besse Inservice Inspection Program*

The focus in the ISI program is related to the pressure test performed at the end of a refueling outage (Mode 3 walk down) and performing certain cold (Mode 5) inspections.

In support of this review, the following documents were reviewed:

- DB-PF-03065 Pressure Test dated 5/20/98 (11RFO)
- WO 99-000320-000 (RX VESSEL) Reactor Vessel Bolting VT-2 Examination at the start of 12 RFO
- DB-PF-03065 Pressure Test dated 5/13/00 (12RFO)
- DB-PF-03010 RCS Leak and Hrydrostatic Test dated 6/2/00 (12RFO)
- Inservice Test Plan (IST Plan) Volume II Second Ten Year Interval Pressure Test Program dated 10/27/99

DB-PF-03065 Pressure Test dated 5/20/98 (11RFO) was reviewed for program compliance and understanding of the objects being inspected. The person involved with the 11RFO Pressure Test that was performed at the conclusion of the outage was interviewed. The person was level 3 certified in NDE. The person described his entering the Reactor Cavity and walking around the RPV head looking for evidence of leakage from the CRDM nozzles. There was no requirement for a hold time at pressure. The test was performed per the requirements of the ISI program and the person demonstrated cognizance of the task.

Work order (WO) 99-000320-000 (RX VESSEL) for 12RFO included performing a VT-2 examination of the installed RPV studs, nuts, and washers. The VT-2 examination could not be performed because of the presence of Boric Acid on the RPV head flange. Condition Report 2000-0781 was written to document the condition of the RPV fasteners. See the Condition Report section for the additional details concerning this specific issue. Final inspection of the fasteners occurred after removal for refueling activities; however, no evidence was presented to document a follow up examination of the RPV head following boric acid removal.

DB-PF-03065 Pressure Test dated 5/13/00 and DB-PF-03010 RCS Leak and Hrydrostatic Test dated 6/2/00 (both from 12RFO) were reviewed. The two inspection reports include CRDM nozzle inspections with the plant at operating temperature and pressure (Mode 3). The examination report dated 5/13/00 indicates that the CRDM nozzles were included in the examination by inspection on top of the service structure looking downward; however, the CRDM nozzle to CRDM flange weld view is obstructed by the CRDM mechanisms and the CRDM flange. It is not clear what is being inspected by this line item. The 6/2/00 dated examination report identifies the CRDM nozzle to RPV head welds were inspected for leakage looking for indications of leakage under the RPV. The report identifies that the inspection for leakage below vessel meets code requirements. No leakage was identified during these inspections.

The ISI Pressure Test Engineer was asked how post boric acid removal inspections are handled specifically related to the RPV head. The person processing the BACCIC would contact the ISI Pressure Test Engineer to have the affected area inspected. If he is not contacted, there is no follow up. The RPV head has not been specifically included in the Boric Acid Corrosion Control program. In addition, the ISI program inspection techniques did not identify CRDM nozzle weld leakage when leakage is now believed to be present.

3.4.4 *Evaluation of Condition Report Responses*

The Root Cause team as part of the investigation, reviewed the completion of a number of Condition Reports. The results of this review are discussed with each Condition Report and then at the end of this section.

PCAQR 94-0295 - Addresses the need to convert commitment A16892 to an Ongoing Commitment. The PCAQR was closed by Regulatory Affairs after they determined that B&W had changed the safety evaluation related to the CRDM nozzle issue to eliminate the need for enhanced inspections. The statement goes on to take credit for the GL 88-05 program to address leaking CRDM nozzles. Discussions with Framatome (purchased B&W) indicate the safety evaluation dated May 1993 was never changed to eliminate the need for enhanced inspections.

PCAQR 96-0551 - Addresses boric acid in several areas on the RPV head and that all of the steps for the Boric Acid Corrosion Control program procedure in effect at that time (NG-EN-00324 rev 1) may not have been followed. The text within the document response takes credit for the GL 88-05 walk downs (GL 88-05 walk downs related to the RPV head at DB only address CRDM flange leakage). The response to the PCAQR endorses implementing MOD 94-0025 (inspection holes for Service Structure). The modification is outstanding.

CR 1999-1300 - Addressed the iron oxide deposits on RE4597 AA/BA. The response discusses the results of the radiation monitor action plan. The lab analysis report evaluating the iron oxide states that the iron oxide was not from magna flux powder but appears to be from an iron based component in containment. There does not appear to have been an aggressive effort to locate the actual source of iron oxide after the original proposed source was disproven. The source of the iron oxide was not conclusively determined under this CR.

CR 2000-0782 - (Categorized as "Routine") This CR was written to address the buildup of boric acid on the RPV head. The CR describes the areas affected by the boric acid. A BACCIC sheet 1 was attached to the CR. The BACCIC characterized the leak as "heavy", red/brown deposits, new leakage not seen during 11RFO, and recommended a detailed inspection. The CR response does not address the concerns discussed in the CR text or the BACCIC. The responder to the CR believed the leakage was from CRDM flange leakage. It discusses whether there is a need for issuing OE. There is no inspection or evaluation for Boric Acid corrosion and no discussion about other possible sources of boric acid.

CR 2000-1037 - (Categorized as "Routine") This CR was written to document the boric acid on the RPV head and on top of the mirror insulation. Operations Review block contained the following note "This CR should be sent to SYME for resolution. This CR will address the effects of the boron on the head. CR 2000-0782 will address the hardware issue of leaking flanges." The response to the CR states that "Accumulated boron deposits between the RPV head and the thermal insulation was removed during cleaning process performed under WO 00-001846-000. No boric acid induced damage to the head surface was noted during the subsequent inspection." The boric acid being left on the RPV head for cycle 13 was not discussed in the CR and not evaluated.

Condition Reports 2000-0782 and 1037 were characterized as "Routine". Both Condition Reports discuss accumulated boric acid on the RPV head. The Condition Reports were not elevated to the appropriate significance level. In addition reference section 4.1, Davis-Besse Experience Review regarding a discussion of the RC-2, Pressurizer Spray Valve incident in 1998.

3.5 Related Issues

After issuance of this Report, a new team was convened to investigate the management and human performance aspects of the damage to the RPV head. That team's investigation results are reported in the Root Cause Analysis Report entitled "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head" dated 8/13/02. That Report replaces Section 3.5 of Revision 0 of this report, with the exception of the subsections presented below.

3.5.1 *RPV Head Inspections*

Prior to 13RFO, head inspections were not a scheduled activity. The Framatome inspection contract only generated a video tape for retention and use by First Energy. The procedure for conducting the analysis by Davis-Besse personnel was not followed. The lack of analysis was a missed barrier to identification of the leak evaluating the structural integrity of the RPV head.

3.5.2 *Restart Readiness*

There is no standard structure to the Restart Readiness Review done at plant startup. The Restart Readiness Review for 12RFO did not identify the fact that boric acid remained on the head of the RPV. Topics were selected for review that senior management considered significant to the restart of the plant and since as previously discussed, management involvement and awareness of the significance of the boron left on the head was inadequate, it was not a requested topic for review. A structured Restart Readiness Review program that included a review of boric acid issues might have increased management awareness of the issues associated with boric acid residue left on the RPV. Information supplied for the Reactor Coolant System in regards to Restart Readiness Review did not discuss the presence of boric acid deposits that remained on the RPV head following cleaning.

3.6 CAUSAL FACTORS/CONCLUSIONS

The Events and Causal Factors Chart (Figure 27) identifies the following undesired events that if prevented would not have resulted in the degradation of RPV head base metal.

- CRDM nozzle crack initiated
- CRDM nozzle crack propagation to through wall leak
- Plant not identifying the through wall crack/leak during outages
- Plant returned to power with boron on the RPV head after outages
- Plant not identifying degradation of RPV head base metal during 12RFO

The following are the conclusions from the causal factors review and cause determination as identified during the root cause team's data analysis: The physical factors that caused corrosion of the RPV head in the regions of nozzles 2 and 3 are the CRDM nozzle leakage associated with through-wall cracking, followed by boric acid corrosion of the RPV low-alloy steel. In order to be defined as a ROOT CAUSE, the identified cause must be something that can be validated. Since it is unlikely that sufficient physical evidence is still retrievable to provide this validation, this ROOT CAUSE must be categorized as a PROBABLE CAUSE. Although it is unlikely that the physical evidence will be retrieved to prove what caused the crack(s), the report provides details why PWSCC is concluded to be the damage mechanism.

Since PWSCC of CRDM nozzles is a known degradation mechanism of Alloy 600 materials, and similar corrosion as experienced near nozzle 3 has not been reported from this cause at other nuclear plants, this PROBABLE CAUSE does not provide the explanation for the extent of damage that occurred in the evolution of this condition.

Corrosion damage of the severity experienced at nozzle 3 could only have occurred with an adequate supply of the corrosive element, in combination with environmental conditions conducive to high corrosion rates. The major question surrounding the boric acid's contribution to the extent of damage, and its rate of progression, is when, and from what source boric acid accumulated on the RPV head. In determining this, the team considered the fact that dried boric acid crystals at normal RPV head temperatures do not result in any significant attack of low-alloy steel surfaces. However, there are examples of 'wet' boric acid leaks causing damage to a RPV head.

Other plants are known to have experienced accumulation of boric acid on RPV heads, due primarily to CRDM flange leaks, or conoseal leaks, without damage similar to that of Davis-Besse nozzle 3. What made Davis-Besse's situation different were the lengths of the cracks (and associated leaks) and the length of time the leaks went undetected. Ultimately, since the leakage appears to have continued for at least 3 to 4 years, boric acid would have accumulated sufficiently during this period to have provided the necessary environment to begin significant RPV head corrosion. The pre-existence of substantial accumulation of boric acid from other sources, like flange leaks, may have accelerated the corrosion and increased its severity. The defense against damage from leaking boric acid is provided by the station's boric acid corrosion control program. For this condition, an additional ROOT CAUSE was the Less Than Adequate Program/Process, which allowed accumulation of boric acid to remain on the RPV head, and thereby allowed the nozzle leaks to go undetected and uncorrected, in time to prevent damage to the head. (Note – The Root Cause Analysis Report on the "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head", dated 8/13/02, concluded that failure to comply with specific requirements of the Boric Acid Corrosion Control Program was a ROOT CAUSE.)

The design of the RPV head/service structure makes access to the top of the RPV head difficult for cleaning and inspection. In the original design, only approximately 2 inches of clearance existed between the top of the RPV head and the bottom surface of the permanently installed reflective insulation. It also provided very limited access for maintenance, consisting only of small drainage openings near the bottom of the RPV head, along the periphery, referred to as mouseholes. Deferral of the modification to the service structure for improved access when the modification was first considered resulted in the continued limited ability to prevent significant boric acid accumulations and allow for better visual determination of leakage sources. Since the severity of the damage that occurred to the RPV head is judged to have required years to develop after the initiation of a CRDM nozzle leak, the deferral is considered a CONTRIBUTING CAUSE to the condition. This is also supported by the less than fully successful attempts to clean and inspect the head using alternate methods from the mouseholes, in refueling outages prior to 13RFO.

Environmental factors, such as temperature conditions and radiation dose, also impeded efforts to inspect and clean the RPV head, in that they affected the methods to be used, and the amount of time allocated to perform the tasks.

Boric acid that accumulated on the top of the RPV head over a period of years inhibited the station's ability to confirm visually that neither nozzle leakage nor RPV corrosion was occurring. Evidence now available shows that leakage from the nozzles began 2 to 4 operating cycles ago. Acceptance of the condition of boric acid accumulation on the RPV head was a CAUSAL FACTOR. The investigation concluded that some of the early boric acid accumulation was likely due to CRDM flange leakage, rather than nozzle leakage, but the effect of its accumulation on the RPV head would have been the same regardless of its origin. The main effect was to inhibit inspection of the top of the RPV head and associated nozzles. While this preexisting boric acid may have accelerated the initial corrosion, this effect is considered minor since borated water leaking from the cracks in the CRDM nozzle would have soon produced its own deposits.

Historically, there have been problems with CRDM flange leakage both at Davis-Besse and in the industry. This appears to have obscured the recognition that boric acid accumulation on the RPV head might also be due to nozzle leakage.

Davis-Besse's boric acid corrosion control program specifically includes the CRDM flanges as an area of concern for the RPV. Potential leakage from CRDM nozzles was not a specific consideration of the program.

The potential for significant corrosion of the RPV head as a result of accumulating boric acid and local leakage was not recognized as a safety significant issue by the staff and management of the plant. The lack of understanding of this was a CAUSAL FACTOR, and ultimately resulted in its own root cause investigation.

Containment building related conditions like iron oxide, boric acid and moisture found in radiation monitor filters, boric acid accumulations on the air coolers and boric acid accumulations on the RPV flange were all recognized, but no collective significance was recognized. However, it is not clear if these could have led to the discovery of the problem on the RPV head in time to prevent significant damage.

All three CRDM nozzles that were found to have leaks were located in the center top region of the RPV head. The team was not able to determine how important this location would be to the potential for development of corrosion as a result of an unattended leak, compared to that of a leak that might exist on the steeper sloped regions of the RPV head. It is probable that the close proximity of the RPV head to the overhead insulation layer allowed for boric acid to concentrate and remain in this region. This in turn could have provided a means for accelerated corrosion rates earlier in the process, in that large accumulations of boric acid may have been available to mix with a continuous moisture supply, once it developed from below.

The Industry continues to study why the corrosion at Davis-Besse was more severe than at other B&W design plants such as Oconee 1-3, ANO 1, TMI 1 and Crystal River 3. The team has identified two possible reasons for this:

- First, the cracks in nozzles 2 and 3 at Davis-Besse extend farther above the top of the J-groove weld (1.1" – 1.2") than cracks measured at other B&W design plants (<1.0"). Analyses in Section 5 demonstrate that the leak rate is sensitive to the length that the crack extends above the J-groove weld. However, the analyses also show that changes in support provided by the low-alloy steel RPV head material can affect the crack opening displacement and area.
- Second, presence of pre-existing boric acid deposits on top of the RPV head may have increased the initial corrosion rates at the exit of the annulus. This theory is supported by test data, which shows that placing insulation around a bolted flange tends to capture the escaping steam and increase the corrosion rate on the heaviest corroded stud, and increase the corrosion rate at other studs around the flange.

In any event, the large-scale corrosion occurred as a result of not detecting and arresting the leakage until advanced symptoms had occurred.

4.0 Experience Review

An experience review was performed and the Davis-Besse and nuclear industry searches identified the following related issues.

4.1 Davis-Besse Experience

In 1998, two body-to-bonnet flange nuts on RC-2, Pressurizer Spray Valve, were identified as missing. The CR 1998-0020 root cause analysis report identifies the nuts were missing as a result of boric acid corrosion. Boric acid corrosion resulted due to a packing leak and the nuts being carbon steel versus stainless steel. The root and contributing causes are similar to the conditions described in this root cause report. The investigation into the " Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head", dated 8/13/02, included examination of why corrective actions from that event were not effective in successfully detecting the conditions on the RPV head.

4.2 Nuclear Industry Experience

Reference discussions provided throughout the Data Analysis section and Table 7 Nuclear Industry Experience Review Results for a summary of Davis-Besse response to NRC and Institute Of Nuclear Operations (INPO) related documents.

4.3 Conclusions

Previous Davis-Besse and nuclear industry experience were not effectively used to prevent the current condition and therefore is considered a casual factor. (Note - The Root Cause Analysis Report on the "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head", dated 8/13/02, concluded that this was a ROOT CAUSE.

5.0 Root Cause Determination

This summary presents the collective judgment of the Root Cause Investigative Team based on the data and evidence that has been characterized at this time in the investigation (current to 08/05/02. The data that supports these causes is summarized in Section 3.6. Additional Root Causes are identified in the Root Cause Analysis Report on the "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head," dated 8/13/02.

5.1 Probable/Root Causes

1. **Probable Cause – Less than Adequate Material Selection. PWSCC cracking in the CRDM nozzle interface at the J-groove weld due to material susceptibility in the presence of a suitable environment resulted in:**
   - CRDM nozzle crack initiated
   - CRDM nozzle crack propagation to through wall leak
   - Boric acid corrosion of the low-alloy steel RPV head material
2. **Root Cause - Less than Adequate Boric Acid Corrosion Control and ISI programs and program implementation regarding the RPV head resulted in:**
   - Plant not identifying the through wall crack/leak during outages
   - Plant returned to power with boron on the RPV head after outages
   - Plant not identifying degradation of RPV head base metal during 12RFO
   - Boric acid corrosion of the low-alloy steel RPV head material

5.2 Contributing Causes

1. **Less than Adequate Environmental Conditions. Cramped conditions due to the design and high radiation at the RPV head** (that remained uncorrected through deferral of proposed modifications) **resulted in:**
   - Plant not identifying the through wall crack/leak during outages
   - Plant returned to power with boron on the RPV head after outages
   - Plant not identifying degradation of RPV head base metal during 12RFO
   - Boric acid corrosion of the low-alloy steel RPV head material
2. **Less than Adequate Maintenance and Testing. Corrective Maintenance did not promptly correct the problem with equipment condition (CRDM flange leakage,** especially at nozzle 31 due to its close proximity to nozzle 3). **This resulted in:**
   - Plant not identifying the through wall crack/leak during outages
   - Plant not identifying degradation of RPV head base metal during 12RFO
   - Boric acid corrosion of the low-alloy steel RPV head material

6.0 Extent of Condition

6.1 Degradation Mechanism Issues

There are two specific degradation mechanisms observed on the RPV head that will be addressed in this extent of condition evaluation. The mechanisms are PWSCC and boric acid corrosion. The 69 CRDM RPV head penetrations will be evaluated by the RPV head repair team and Engineering ensuring all necessary inspections and examinations are performed on the CRDM nozzles to address extent of condition.

The extent of condition within the containment will be evaluated via walkdowns of structures, systems and components (SCC) within containment. In defining the scope of the walkdowns, three separate criteria were developed to ensure that a bounding evaluation is performed. These three separate criteria are:

(1) Sources: As used in this evaluation, sources are components containing borated water that are considered likely leak locations. The sources are further divided into three groups: Valves, Threaded/Bolted joints (e.g. thermowells, manways, handholes, reactor coolant pumps), and Alloy 600 components/welds. The Alloy 600 components/welds are susceptible to PWSCC. The intent is to (1) verify there is no additional RCS pressure boundary leakage at Davis-Besse (from Alloy 600 components/welds) and (2) verify that evidence of RCS leakage from any source is properly evaluated (including the potential impact on susceptible materials of the RCS pressure boundary).

(2) Targets: As used in this evaluation, targets are components within the RCS pressure boundary that utilize materials susceptible to boric acid corrosion (carbon and low-alloy steels) as part of the pressure boundary. The targets include the following RCS components: RPV, steam generators, pressurizer, RCPs and individual piping sections. The intent is to verify that boric acid corrosion has not degraded the RCS pressure boundary. Additionally, although technically not within the RCS pressure boundary, the core flood tanks will be evaluated as targets. It should be noted that certain valves within the RCS pressure boundary may contain susceptible materials but for convenience the valves are listed as sources.

(3) Safety-related (non RCS pressure boundary) SSCs: This criteria refers to safety related SSCs that utilize materials susceptible to boric acid corrosion but are not part of the RCS pressure boundary. The intent is to verify that boric acid corrosion has not adversely impacted the function of safety related SSCs.

Methodology:

(1) Plant Engineering will develop a list of inspection points to address the sources and targets. A table of valves and threaded/bolted connections previously developed for the boric acid corrosion control program mode 5 walkdowns will be used to identify these sources (most of these walkdowns are complete at this date). A list of Alloy 600 components/welds within the RCS pressure boundary has been provided by Design Basis Engineering. A series of inspection points will be needed to adequately address each target. Each target will receive a visual inspection of the external surfaces of installed insulation for evidence of leakage (boric acid residue or bulging of the insulation).

Additionally, each connection point between a target and non-susceptible piping will be inspected to verify that no boric acid has migrated undetected under the insulation to reach a susceptible component. This will require removal of insulation to permit a visual inspection. These inspections (external inspection of the insulation and visual inspection of connection points) will provide adequate assurance that there is no undetected degradation of the RCS pressure boundary. It should be noted that many of the "connection points" are Alloy 600 components/welds that also require inspection as potential sources. These inspections will be performed by VT-2 qualified personnel. Representative photographs will be made to document the "as found" condition of each inspection point.

(2) The use of visual inspection of Alloy 600 components/welds to detect evidence of throughwall PWSCC requires adequate access to perform a visual inspection. Additionally the design of component/weld must provide assurance that leakage will be detectable at the surface. This may require additional evaluation of certain nozzles (such as incore nozzles) to verify that a visual inspection is adequate.

(3) In any case where evidence of boric acid deposits exists, the source of the deposits and the leak path must be traced to ensure that there is no wastage of the RCS pressure boundary. It is known that there are boric acid deposits on the insulation on the bottom of the RPV. There are boric acid deposits on the seam between pieces of insulation suggesting that the boric acid came from inside the insulation. It is therefore necessary to perform an inspection under the insulation to determine whether or not there is wastage on the RPV and to determine the source of the boric acid. Due to the difficulty of this task and ALARA considerations, a specific plan is being developed to perform this inspection.

(4) The third category, safety-related (non RCS pressure boundary) SSCs, will be addressed by general area walkdowns of the containment building. These walkdowns will be primarily conducted by Design Engineering Mechanical/Structural (DEMS) and Design Engineering Electrical/Controls (DEEC). The DEMS personnel will focus on safety related SSCs such as structural steel, concrete, pipe supports, control rod guide tube supports, susceptible non RCS piping and coatings. DEECS will focus on cabling, conduit, junction boxes, etc. Plant Engineering will perform inspections of ventilation systems within containment (such as CACs and ductwork). Photographs will be made to document any boric acid deposits/corrosion discovered during these walkdowns.

(5) It is expected that (after proper documentation) existing boric acid deposits will be cleaned up. This will prevent future degradation of susceptible materials due to re-wetting of dry boric acid deposits. It will additionally ensure a proper baseline condition for future inspections.

(6) It is also expected that any SSC that has experienced degradation due to boric acid corrosion will be evaluated then reworked or preserved as needed to ensure high standards of material condition and housekeeping.

Note: The investigation into the management and human performance aspects of the head damage determined that the potential exists for additional extent of condition considerations for other systems and programs. For details, refer to Section 7 of the report entitled "Failure to Identify Significant Degradation of the Reactor Pressure Vessel Head", dated 8/13/02.

7.0 Recommended Corrective Actions

7.1 Probable/Root Causes Corrective Actions

Less than Adequate Material Selection. PWSCC cracking in the CRDM nozzle interface at the J-groove weld due to material susceptibility in the presence of a suitable environment.

1. (Remedial) Develop a plan to monitor for CRDM nozzle leakage. The plan must include steps to repair once leakage is detected. **(Plant Engineering)**
2. (Remedial) Review Davis-Besse results for CRDM nozzle crack initiation/propagation against the susceptibility model. **(Design Basis Engineering, Completion prior to restart)**
3. (Remedial) Replace the RPV Closure Head
4. (Preventive) Obtain and install a new RPV Closure Head that does not use alloy 600 for CRDM nozzles.

Less than Adequate Boric Acid Corrosion Control and ISI programs and program implementation regarding the RPV head.

5. (Remedial) An extent of condition review for boric acid damage will be performed to ensure that there are no latent unidentified issues related to boric acid corrosion. The results will be reviewed by the senior management team prior to startup. **(Plant Engineering, Completion prior to restart)**
6. (Preventive) Perform Self-Assessments of the boric acid corrosion control and ISI programs. **(Plant Engineering Completion prior to restart)** The purpose of these Self-Assessments is to evaluate the deficiencies documented in this report. Items to be considered should include:

   <u>Boric Acid Corrosion Control Program</u>

   - Incorporating as areas for inspection, industry issues such as CRDM nozzle leakage
   - Incorporating into the inspection plan systems that carry borated water and provide mitigating type functions that help to preserve the Reactor Coolant Pressure Boundary during plant transients and/or accidents
   - Incorporate Boric Acid Corrosion Control Inspection Checklist document retention requirements (retention should be at least several fuel cycles)
   - Incorporating a signature block for the Boric Acid Corrosion Control Program Owner to document his review and concurrence with the disposition activities
   - Review the use of "should" versus "shall" throughout the procedure.
   - Incorporating requirement that boric acid "shall" be removed from affected areas and the affected area inspected to identify any signs of potential corrosion.
   - Incorporating a signature block for the System Engineers supervisor to document his review and concurrence with the disposition activities
   - Review station commitments to determine if other areas or equipment must be included in the Boric Acid Corrosion Control Program
   - Establish a hard link between the Boric Acid Corrosion Control Program and the ISI Program that requires both groups to approve the close out of a Boric Acid Corrosion Control Inspection Checklist.

ISI Program

- Improve the text descriptions of the areas to be inspected, include sketches of the area and provide a pre-job brief prior to inspecting for bolted connections and Mode 3 leakage during plant heat up
- Eliminate the conflicting text descriptions that are contained in some of the inspection plans
- Evaluate the techniques employed for monitoring CRDM nozzle welds for leakage.
- Reinforce the obligation the ISI program has to protect and preserve the RCS pressure boundary including addressing Boric Acid deposits on the RCS pressure boundary when that specific area was not included in the original inspection plan
- Establish a hard link between the ISI Program and the Boric Acid Corrosion Control Program that requires both groups to approve the close out of a Boric Acid Corrosion Control Inspection Checklist

7.2 Contributing Causes Corrective Actions

Less than Adequate Environmental Conditions. Cramped conditions due to the design and high radiation at the RPV head (that remained uncorrected through deferral of proposed modifications).

1. Provide improved access for inspection and cleaning of the RPV head. **(Design Basis Engineering; Completion prior to restart)**

Less than Adequate Maintenance and Testing. Corrective Maintenance did not promptly correct the problem with equipment condition (CRDM flange leakage, especially at nozzle 31 due to its close proximity to nozzle 3).

2. No corrective action is required. No CRDM flange leakage was noted during 13RFO. This contributing cause has been resolved. The monitoring for leakage will continue through Preventive Maintenance number 1629 and assessment for corrosion through the Boric Acid Corrosion Control Program.

7.3 Additional Actions

1. It is recommended that a historical Alloy 600 review be conducted. The review should include documents associated with the CRDM nozzles and summarize the results in a FENOC-level program document. Potential items for consideration include:
   - The 1994 EPRI Workshop report
   - The EPRI TR-103696 report referenced in the 1994 Workshop Report.
   - EPRI NP-6719-M-SD (Feb 8-10, 1989)
   - March 5, 1996 NEI white paper entitled Alloy 600 RPV Head Penetration PWSCC
   - 1997 EPRI Workshop on PWSCC of Alloy 600 in PWRs Parts 1 & 2 (TR-109138-P2).
   - EPRI Workshop on PWSCC Alloy 600 in PWRs, 2/14-16/2000, St. Pete Beach.
   - EPRI MRP Alloy 600 Industry Workshop, 6/13-6/14/2001, Atlanta, Report 1006278.
   - B&WOG Materials Committee Report 51-1229638
   - Automated Ultrasonic Inside Surface Examinations of Reactor Coolant System Alloy 82/182 Nozzle Welds Performed in Spring 2001: PWR Materials Reliability Project – Alloy 600 Issue Task Group, 82/182 Weld Integrity Inspection Committee, EPRI Report 1006225

Note – Revision 0 of this report had recommended that the results of this review be included in the report. It will be more useful as a separate document to aid in managing issues with alloy 600 components.

2. Revision 0 had recommended that a more extensive CRDM flange leak and repair historical review be conducted and summarized in the Root Cause Analysis Report. With the decision to replace the RPV head, this additional historical review provides no benefit to DBNPS. The report already contains sufficient information to understand the role of flange leaks in the issue.

3. Revision 0 recommended that the analysis of samples collected from the RPV head wastage root cause investigation be collected and tracked. The following samples had been collected:
   - Four samples of rusty boric acid from initial head investigation following insulation removal
   - Nozzle 3
   - Four samples of deposits including corrosion products from Nozzle Two Removal
   - Nozzle 2

   Additionally, the wastage area adjacent to nozzle three was removed from the head and is to be investigated destructively. The sample results are included in Section 3.1.5, Section 3.2.4, Morphology, Stage 3. References 8.2.14 and 8.2.15 are the analysis reports. The destructive testing will be tracked under corrective action CA 02-00891-114.

4. Extensive effort is currently in progress by the MRP to develop a model for how small leaks from PWSCC cracks progress to modest amounts of corrosion such as seen at nozzle 2 and much larger amounts of corrosion as seen at nozzle 3. While the corrosion is obviously due to the boric acid, the exact stages of progression are being assessed. Mechanisms being evaluated include boric acid corrosion, crevice corrosion, impingement, flow accelerated corrosion, low oxygen corrosion, steam cutting, molten boric acid corrosion, etc. This work includes finite element thermal-hydraulic modeling to determine the effect of steam leakage on locally suppressing the metal temperature in the annulus. Revision 0 recognized the possibility that this work could conceivably have affected conclusions in this report. DBNPS has remained in contact with the EPRI work, and has reviewed the draft results, which continue to support all conclusions made in this report. Sections 3.2.1 and 3.2.4 have been updated providing additional clarity on the subject of morphology.

5. Revision 0 recommended a review of the stresses of the CRDM nozzles at both operating conditions and cold conditions. This was to determine based upon the stress review if extended time periods at mode 5 conditions increase the likelihood of PWSCC crack initiation. The results of this review have been included in the revision to section 3.2.1.

8.0 References

8.1 Davis-Besse References

1. Davis-Besse 13RFO CRDM Nozzle Examination Report, Revision 1, Framatome ANP UT Report, March 11, 2002.
2. Potential Condition Adverse to Quality Reports

   90-0120 Boron Leakage and CRDM Stator Cooling

   90-0221 CRDM Flange F-2 Slight Erosion of Outer Gasket Groove

   91-0353 Boron on Reactor Vessel Head from Leaking CRDM Flanges

   92-0072 CAC Cooler Degraded Below Acceptable Performance

   92-0248 Boron Found in Filter RE4597AA

   93-0098 Reactor Head Vent Flange Leakage

   93-0132 Reactor Coolant Found Leaking from CRD Flanges

   93-0175 Service Water Piping to CAC's Have Accumulated Boric Acid

   94-0295 TERMS A16892 Requires Visual Exam of Reactor Vessel Head each Outage

   94-0912 Documents CRDM Leakage

   94-0974 Documents Scratches and Gouges on Seating Surface Location G-5

   94-0975 Document ½ Moon Gouge CRDM Flange M-3

   94-1338 Westinghouse CRDM part 21

   96-0551 Video of CRDM Flanges Shows Evidence of Leakage

   96-0650 VT-2 Exam of RCP Stud Shows Evidence of Boric Acid Leakage

   96-1018 Info Notice 96-032 Received Concerning Augmented Inspection of Rx Vessel

   1998-0649 Inspection Results of Reactor Vessel Head

   1998-0650 Video Inspection Results CRDM Nozzle/Head Interface

   1998-0824 CAC's 2 and 3 Have Accumulated Boric Acid

   1998-1164 Water Collecting in Sample Line for RE4597AA

   1998-1885 RC-2 Carbon Steel Nuts

   1998-1895 Containment Normal Sump Leakage > 1GPM

   1998-1980 Containment Cooler Plenum Pressure Decreasing
3. Condition Reports

   1998-0020 Multiple Problems with RC-2

   1999-0372 Containment Rad RE4597AA/AB High

1999-0510 RE4597AA OOS Low Flow

1999-0845 Boric Acid Clumps Room 181

1999-0861 RE4597AA Sample Line Full of Water

1999-0928 Document Increased RE Filter Change Frequency

1999-1300 RE Filter Analysis Results from Southwest Research Institute

1999-1614 LER 1998-009

1999-1098 Issues with DB-OP-01200 RCS Leakage Management

2000-0781 Boric Acid on RV Studs

2000-0782 RV Flange Boric Acid from Weep Holes

2000-0903 Two CRDM Flange Fasteners Fail Preservice Exam.

2000-0994 CRDM Flange F-10 Pitted

2000-0995 CRDM Flange D-10 Pitted

2000-1037 Reactor Head Inspection Indicates Boric Acid Accumulation

2000-1210 CRDM D-10 Out of Plum

2000-1547 Containment Cooler Plenum Pressure Dropped

00-4138 Increased Frequency of Containment Air Cooler Cleaning

01-0039 Step Drop in Containment Air Cooler Plenum Pressure

01-0487 Higher Containment Temperatures

01-0890 RCS Leakage Calculation Data Scatter

01-1110 RE4597BA Filter Change Occurring More Frequently

01-1822 Increasing Frequency of RE4597BA Filter Changeout

01-1857 RCS Leakage Anomalies

01-2012 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

01-2769 Containment Wide Range Radiation Element (RE2387) Spiking

01-2795 RE4597BA Alarm

01-2862 Potential Adverse Trend in Unidentified RCS Leakage

01-2936 Unable to Perform RE4597BA/BB Functional by the Technical Specification

01-3025 RCS Leakage

01-3411 Equipment Failure on Detector Saturation during RE4597BA Testing

02-00685 Boron Build Up on Reactor Vessel Head

02-00846 More Boron on Head than Expected

02-00891 Control Rod Drive Nozzle Crack Indication

02-00932 CRDM Nozzle Crack Indications

02-01053 Unexpected Tool Movement

4. Procedures

   DB-OP-01200 Reactor Coolant System Leakage Management (rev 0 thru 3)

   DB-OP-06900 Plant Heatup

   DB-PF-00204 ASME XI Pressure Testing

   DB-PF-03010 RCS Leakage and RCS Hydrostatic Test

   DB-PF-03065 Pressure and Augmented Leakage Test

   NG-EN-00324 Boric Acid Corrosion Control (rev 2)

5. Other Station Documents

   Davis Besse System Health Report, 4th Quarter 2001

   Request For Modification 94-0025 Install Service Structure Inspection Opening

   Inservice Inspection Plan (ISI Plan) Volume II Third Ten-Year Interval Pressure Test Program

   Inservice Inspection Plan (ISI Plan) Volume II Second Ten-Year Interval Pressure Test Program

   Relief Request RR-A3 Insulated ASME Class 1 and 2 Pressure Retaining Bolted Connections

   Relief Request RR-A10 ASME Class 1 and 2 Pressure Retaining Bolted Connections

   System Description:

- SD-022B Containment Air Cooling System and Recirculation System
- SD-39A Reactor Coolant System

Technical Specifications:

- 3/4.4.6.1 Reactor Coolant Leakage Detection Systems
- 3/4.4.6.2 Reactor Coolant System Operational Leakage
- 3/4.4.10 Structural Integrity ASME Code Class 1, 2, and 3 Components

Updated Safety Analysis Report Sections:

- 5.1 Reactor Coolant System summary Description
- 5.2 Integrity of Reactor Coolant Pressure Boundary (RCPB)
- 11.4.4.4.5 Containment Vessel Monitor
- Fig. 5.1-2 Functional Drawing Reactor Coolant System
- Fig. 5.1-3 Reactor Coolant System and Supporting Structures - Plan
- Fig. 5.1-4 Reactor Coolant System and Supporting Structures – Plan

RWP 2000-5132 Clean Boric Acid from Rx Head

8.2 Vendor References

1. B&WOG Integrated Response to NRC Generic Letter 97-01 Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations, BAW-2301, Framatome ANP Report, July 1997

2. Framatome ANP Report 51-5001951-01, Alloy 600 PWSCC Susceptibility Model, December 9, 1998 (Proprietary)
3. Oconee 1 RPV Head Nozzle Leaks presented by Dave Whitaker at EPRI Alloy ITG meeting January 19, 2001
4. Dominion Engineering, Inc. Calculation No. C-5509-00-6 Davis Besse CRDM Leak Rates using ANSYS Crack Opening Area (non-safety related), Revision 0 3/19/2002 (Proprietary)
5. Dominion Engineering, Inc. Calculation No. C-5509-00-7 Davis Besse CRDM Nozzle Crack Opening Displacement Analysis, Revision 0 3/19/2002 (Proprietary)
6. Dominion Engineering, Inc. Calculation No. C-5509-00-5 Leak Rate through Axial Crack in Davis Besse CRDMs (non-safety related), Revision 1 3/19/2002 (Proprietary)
7. BAW-10190P Safety Evaluation for B&W-Design Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking (Proprietary)
8. BAW-1019P Addendum 1 External Circumference Crack Growth Analysis for B&W Design Reactor Vessel head CRDM Nozzles (Proprietary)
9. BAW-1019P Addendum 2 Safety Evaluation for Control Rod Drive Mechanism Nozzle J-Groove Weld (Proprietary)
10. B&WOG Materials Committee Report 51-1201160-00 Alloy 600 SCC Susceptibility: Scoping Study of Components at Crystal River 3
11. B&W Report 51-1218440-00 Alloy PWSCC Time-To-Failure Models (Proprietary)
12. B&W Report 51-1219143-00 CRDM Nozzle Characterization (Proprietary)
13. Dominion Engineering, Inc. Calculation No. C-5509-00-7 Volume and Weight of Boric Acid Deposits on Vessel Head.
14. Framatome-ANP report #51-5018613-00, Davis-Besse Reactor Vessel Head Deposit Characterization results, June 2002
15. Framatome-ANP report #51-5018965-00, Davis-Besse Reactor Head Deposit Sample Characterization (Second Batch, Nozzle #2 Removal), July 2002
16. Framatome-ANP report #51-5018376-00, Davis-Besse CRDM Crack Profiles, May 13, 2002.

8.3 NRC References

1. GL 88-05 Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants
2. GL 97-01 Degradation of CRDM/CEDM Nozzle and Other Vessel Closure Head Penetrations
3. Regulatory Guide 1.45 Reactor Coolant Pressure Boundary Leakage Detection Systems
4. Bulletin 82-2 Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants
5. Bulletin 2001-01 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles
6. Bulleting 2002-01 Reactor pressure Vessel head Degradation and Reactor Coolant Pressure Boundary Integrity

7. IN 80-27 Degradation of Reactor Coolant Pump Studs
8. IN 82-6 Failure of Steam Generator Primary Side Manway Closure Studs
9. IN 86-108 Degradation of RCS Pressure Boundary Resulting From Boric Acid Corrosion
10. IN 86-108 Supplements 1 & 2 Degradation of RCS Pressure Boundary Resulting From Boric Acid Corrosion
11. IN 86-108 Supplement 3 Degradation of RCS Pressure Boundary Resulting From Boric Acid Corrosion
12. IN 90-10 Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600
13. IN 94-63 Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks
14. IN 96-11 Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations
15. IN 2001-5 Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3
16. IN 2000-17 Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V.C. Summer
17. IN 2000-17 Supplement 1 Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V.C. Summer
18. IN 2000-17 Supplement 2 Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V.C. Summer
19. IN 2002-11 Recent Experience with Degradation of Reactor Pressure Vessel Head
20. Safety Evaluation for Potential Reactor Vessel Head Adaptor Tube Cracking, November 19, 1993

8.4 INPO References

1. SOER 81-12 Reactor Coolant Pump Closure Stud Corrosion
2. SOER 84-5 Bolt Degradation or Failure in Nuclear Power Plants
3. SER 46-80 Reactor Coolant Pump Closure Stud Corrosion
4. SER 35-81 Corrosion of Reactor Coolant System Piping
5. SER 11-82 Reactor Coolant Pump Closure Flange Stud Corrosion
6. SER 57-83 Cracking in Stagnant Boric Acid Piping
7. SER 72-83 Damage to Carbon Steel Bolts and Studs on Valves in Small Diameter Piping Caused by Leakage of Borated Water
8. SER 32-84 Contamination of Reactor Coolant System by Magnetite and Sulfates
9. SER 41-85 Containment Spraying Events
10. SER 13-87 Reactor Vessel Stud Corrosion from Primary Coolant Leak
11. SER 31-87 Pressurizer Vessel Corrosion due to Pressurizer Heater Rupture
12. SER 35-87 Non-Isolable Reactor Coolant System Leak

13. SER 10-89 Reactor Coolant Pump Flange Leak from Loss of Bolt Preload. Bolts should be checked for preload
14. SER 90-2 Pressurizer Heater Sleeve Cracking
15. SER 20-93 Intergranular Stress Corrosion Cracking in Control Rod Drive Mechanism Penetrations
16. SER 4-01 Recent Events Involving Reactor Coolant System Leakage at Pressurized Water Reactors
17. SEN 6 Boric Acid Corrosion
18. SEN 18 Reactor Vessel Head Corrosion
19. SEN 190 Pressurizer Spray Valve Bonnet Nuts Dissolved by Boric Acid
20. SEN 216 Leakage from Reactor Vessel Nozzle-to-Hot Leg Weld
21. SEN 220 Pressure Boundary Leakage at Palisades. Palisades had a through-wall crack in a CRDM housing
22. O&MR 348 Failure of a Limitorque Operator Stem Nut

8.5 Industry References

1. PWSCC of Alloy 600 Materials in PWR Primary System Penetrations, EPRI TR-103696. (Proprietary)
2. EPRI Technical Report -104748 Boric Acid Corrosion Guidebook (Proprietary)
3. EPRI Technical Report -1000975 Boric Acid Corrosion Guidebook, Revision 1 (Proprietary)
4. EPRI Technical Report -103696 PWSCC of Alloy 600 Materials in PWR Primary System Penetrations (Proprietary)
5. MRP-44, Part 2, PWR Materials Reliability Program – Interim Alloy 600 Safety Assessments for US PWR Plants, Part 2: Reactor Vessel Top Head Penetrations (Proprietary)
6. EPRI NP-6301-D, Ductile Fracture Handbook
7. EPRI Technical Report -107621-R1, Steam Generator Integrity Assessment Guidelines: Revision 1 (Proprietary)
8. EPRI draft report NP-6864-L, PWR Steam Generator Tube Repair Limits: Technical Support Document for Expansion Zone PWSCC in Roll Transitions
9. MRP crack growth rate report (Proprietary)
10. EPRI NP-7094, Literature Survey of Cracking of Alloy 600 Penetrations
11. EPRI technical report 1006284, Report MRP-48, PWR Materials Reliability Program Response to NRC Bulletin 2001-01, August 2001 (Proprietary)

8.6 Other References

1. V.C. Summer Nuclear Station Root Cause Investigation "A" Hot Let Nozzle Weld Cracks
2. Dominion Engineering Meeting Presentation to NRC technical Staff, May 22, 2002

9.0 Personnel Interviews

9.1 Personnel Interviewed

Andrew Siemaszko, current Davis-Besse RCS System Engineer
Ed Chimahusky, former Davis-Besse RCS System Engineer
Dan Haley, former Davis-Besse RCS System Engineer
George Chung, current Davis-Besse Radiation Monitor System Engineer
Bob Hovland, former Davis-Besse Radiation Monitor System Engineer
Walt Molpus, current Davis-Besse Boric Acid Corrosion Control Program owner
Peter Mainhardt, performed Davis-Besse Reactor Vessel Head inspections
Jerry Lee, Davis-Besse Leak Program owner
Glenn McIntrye, former Davis-Besse Mechanical Systems Supervisor
Jim Marley, Davis-Besse System Engineering
Pete Seniuk, Davis-Besse ISI Pressure Test Engineer
Chuck Daft, Davis-Besse ISI Engineer
Mike Shepherd, Davis-Besse ISI Engineer
Prasoon Goyal, Davis-Besse B&WOG Material Committee representative
Ken Byrd, Davis-Besse Nuclear Engineering (PSA Engineer) Supervisor
Rich Edwards, Davis-Besse Chemistry Technologist
Bruce Geddes, Davis-Besse Containment Deconing
Mark Mclaughlin, Davis-Besse CRDM Project Manager
Charles (Steve) Steagall, Davis-Besse VT-2 Inspector
Richard Cockrell, Davis-Besse VT-2 Inspector
Chuck Ackerman, FENOC Quality Assurance Engineering Supervisor
Henry Stevens, FENOC Manager Quality Assurance
Dave Lockwood, Davis-Besse Manager Learning Organization and Regulatory Programs
Dave Geisen, Davis-Besse Design Basis Engineering Manager
Dave Eshelman, former Davis-Besse Plant Engineering Manager
Joe Rogers, Davis-Besse Outage Director
Scott Coakley, Davis-Besse Outage Director
Steve Moffitt, Davis-Besse Director Technical Services
John Messina, Davis-Besse Director Work Management
John Wood, FENOC Vice President Engineering Services
Jim Harris, Framatome 13R Reactor Services Lead
Fred Currence, Framatome 13R Reactor Services Lead
Mike Hacker, Framatome UT expert
Rich Garrison, Framatome CRDM Nozzle Inspection/Repair Manager
Ron Pillow, Framatome CRDM Component Engineer
Steve Fyfitch, Framatome Metallurgist
Cary Bowles. Framatome
Rod Emery, Oconee CRDM Engineer

9.2 Personnel Consulted

Steve Moffitt, Davis-Besse Director Technical Services
John Hickling, EPRI Materials expert
Kim Kietzman, EPRI UT expert
Chuck Welty, EPRI Director
Jeff Gorman, Dominion Engineering PhD Materials expert
Chuck Marks, Dominion Engineering PhD Chemistry expert
Matt Brown, Radiation Protection Servicemen

10.0 Methodologies Employed

Event & Causal Factors Charting
Procedure Review/Analysis
Difference Analysis
Barrier Analysis
Possible Cause Analysis
Structured Interviewing

Table 1. Nozzle 1 NDE Examination Results

FRAMATOME ANP

CRDM Nozzle Ultrasonic Examination Data Sheet

Customer: FENOC		**Plant:** Davis Besse	**Unit:** n/a		**Nozzle:** 1
Procedure: 54-ISI-100-08	CA: FRA-02-002, DB-02-012	**Nozzle Dimensions: (in.)**	**ID:** 2.765	**OD:** 4.06	**Thickness:** 0.649
Downhill Side of Nozzle (deg.): 183	**End of Noz. (in** 29.6	Probe Serial No.'s:	Ch 1	2078-01002-0L	Ch 6 21GB-01002-45L
Axial Scan Start: -6, 15.06	Stop: 360, 29.63	**Setup:** 1	Ch 2	21GF-01004-30L	Ch 7 21GC-01001-55L
Files: T2061_12.36.51			Ch 3	21GA-01004-45L	Ch 8 22CD-01001-65L
Circ. Scan Start: -5, 19.23	Stop: 360, 29.63	**Setup:** 2	Ch 4	2623-01002-60S	Ch 9 2624-01005-60S
Files: T2061_11.11.08			Ch 5	2623-01002-60S	Ch 10 2624-01005-60S

Flaw No.	Surface (ID/OD)	Depth to Flaw Tip	End Point 1 Min (in.)	End Point 1 Min (deg.)	End Point 2 Max (in.)	End Point 2 Max (deg.)	Axial Total (in.)	Adjusted Circ. Extent Min (deg.)	Adjusted Circ. Extent Max (deg.)	Adjusted Circ. Extent Total (in.)	Flaw Length (in.)	Flaw Angle (deg.)	Flaw TWD (in.)	Flaw Aspect Ratio	Flaw Orientation	Weld Location @ Flaw Min	Weld Location @ Flaw Max
1	OD	0.29	26.97	133	28.31	128	1.34	50.0	55.0	0.18	1.35	8	0.36	0.27	AXIAL	In Weld Region	
2	OD	0.24	26.63	115	28.29	113	1.66	68.0	70.0	0.07	1.66	2	0.41	0.24	AXIAL	In Weld Region	
3	OD	0.63	27.71	51	28.11	53	0.40	132.0	130.0	0.07	0.41	10	0.02	0.05	AXIAL	In Weld Region	
4	OD	TW	26.9	31	28.67	29	1.77	152.0	154.0	0.07	1.77	2	0.65	0.37	AXIAL	In Weld Region	
5																	
6	OD	0.04	27.1	334	28.8	334	1.70	209.0	209.0	0.00	1.70	0	0.61	0.36	AXIAL	In Weld Region	
7	OD	TW	25.95	285	29.43	291	3.48	258.0	252.0	0.21	3.49	3	0.65	0.19	AXIAL	In Weld Region	
8	OD	0.32	27.58	233	28.45	233	0.87	310.0	310.0	0.00	0.87	0	0.33	0.38	AXIAL	In Weld Region	
9	OD	0.28	27.6	202	28.35	202	0.75	341.0	341.0	0.00	0.75	0	0.37	0.49	AXIAL	In Weld Region	
10	OD	0.24	27.64	181	28.86	181	1.22	2.0	2.0	0.00	1.22	0	0.41	0.34	AXIAL	In Weld Region	
11																	
12																	
13																	
14																	
15																	
16																	
17																	
WELD PROFILE		Data Loc.	183	213	243	273	303	333	3	33	63	93	123	153	183	Degrees	
		Noz. Loc.	0	30	60	90	120	150	180	210	240	270	300	330	360	Degrees	
		Noz. End	29.60	29.60	29.60	29.60	29.60	29.60	29.60	29.60	29.60	29.60	29.60	29.60	29.60	Inches	
		MAX.	27.85	27.82	27.89	27.89	27.89	27.89	27.97	27.97	27.93	27.85	27.89	27.82	27.85	Inches	
		MIN.	26.55	26.55	26.67	26.71	26.59	26.40	26.40	26.40	26.44	26.59	26.59	26.59	26.55	Inches	

Notes: Adjusted Circ. Extent is relative to downhill side of nozzle;clockwise looking down. TWD is Through-Wall Dimension

Comments: Data was encoded with positive Theta going counterclockwise. Adjusted circ. positions have corrected the position to read clockwise looking down.

Flaw # 5 was identified as an axial flaw using the circ. blade probe but is not confirmed with the rotating UT. Therefore, flaw #5 is not relevant.

Analyzed by: K.C.Gebetsberger **Date:** 3/5/02 **Analyzed by:** M.G. Hacker **Date:** 3/5/02

Table 2. Nozzle 2 NDE Examination Results

FRAMATOME ANP

CRDM Nozzle Ultrasonic Examination Data Sheet

Customer:	FENOC	**Plant:**	Davis Besse	**Unit:**	N/A	**Nozzle:**	2
Procedure:	54-ISI-100-08 CA: FRA-02-002, DB-02-012	**Nozzle Dimensions: (in.)**		**ID:** 2.765	**OD:** 4.06	**Thicknes**	0.649
Downhill Side of Nozzle (deg.): 315	**End of Noz. (in.):** 30.78	Probe Serial No.'s:		Ch 1	2078-01002-0L	Ch 6	21GB-01002-45L
Axial Scan Start: -5, 16.1	Stop: 360, 30.77	**Setup:** 1		Ch 2	21GF-01004-30L	Ch 7	21GC-01001-55L
Files: T2061_09.12.19				Ch 3	21GA-01004-45L	Ch 8	22CD-01001-65L
Circ. Scan Start: 0, 18.95	Stop: 360, 29.52	**Setup:** 2		Ch 4	2623-01002-60S	Ch 9	2624-01005-60S
Files: T2061_07.25.10				Ch 5	2623-01002-60S	Ch 10	2624-01005-60S

Flaw No.	Surface (ID/OD)	Depth to Flaw Tip	End Point 1 Min (in.)	End Point 1 Min (deg.)	End Point 2 Max (in.)	End Point 2 Max (deg.)	Axial Total (in.)	Adjusted Circ. Extent Min (deg.)	Adjusted Circ. Extent Max (deg.)	Adjusted Circ. Extent Total (in.)	Flaw Length (in.)	Flaw Angle (deg.)	Flaw TWD (in.)	Flaw Aspect Ratio	Flaw Orientation	Weld Location @ Flaw Min	Weld Location @ Flaw Max
1	OD	0.236	27.46	291.0	29.51	275.0	2.05	24.0	40.0	-0.57	2.13	165	0.41	0.19	AXIAL	In Weld Region	
2	OD	TW	26.59	262.0	30.37	240.0	3.78	53.0	75.0	-0.78	3.86	168	0.65	0.17	AXIAL	In Weld Region	
3													0.65				
4	OD	TW	26.69	148.0	29.39	141.0	2.70	167.0	174.0	-0.25	2.71	175	0.65	0.24	AXIAL	In Weld Region	
5	OD	0.33	27.87	130.0	28.7	127.0	0.83	185.0	188.0	-0.11	0.84	173	0.32	0.38	AXIAL	In Weld Region	
6	OD	TW	26.8	67	29.36	78	2.56	248.0	237.0	0.39	2.59	9	0.65	0.25	AXIAL	In Weld Region	
7																	
8	OD	TW	26.35	32	30.16	61	3.81	283.0	254.0	1.03	3.95	15	0.65	0.16	AXIAL	In Weld Region	
9																	
10	OD	TW	27.39	7	30.35	26	2.96	308.0	289.0	0.67	3.04	13	0.65	0.21	AXIAL	In Weld Region	
11	OD	0.344	27.9	314	27.75	347	0.15	361.0	328.0	1.17	1.18	83	0.31	0.26	CIRC.	0.1	0.1
12	OD	0.572	29.02	320	29.6	327	0.58	5.0	12.0	0.25	0.63	23	0.08	0.12	AXIAL	In Weld Region	
13	OD	TW	26.6	259.0	29.81	258.1	3.21	101.9	101.0	-0.03	3.21	179	0.65	0.20	AXIAL	In Weld Region	
14																	
15																	
16																	
17																	

Revision 2 5/4/02

WELD PROFILE														
Data Loc.	315	345	15	45	75	105	135	165	195	225	255	285	315	Degrees
Noz. Loc.	0	30	60	90	120	150	180	210	240	270	300	330	360	Degrees
Noz. End	30.78	30.78	30.78	30.78	30.78	30.78	30.78	30.78	30.78	30.78	30.78	30.78	30.78	Inches
MAX.	29.17	29.09	29.02	28.84	28.61	28.49	28.46	28.49	28.76	28.92	29.04	29.14	29.17	Inches
MIN.	28.06	27.79	27.36	27.39	27.31	27.16	27.16	27.24	27.36	27.39	27.84	27.89	28.06	Inches

Notes: Adjusted Circ. Extent is relative to downhill side of nozzle;clockwise looking down. TWD is Through-Wall Dimension

Comments: Data was encoded with positive Theta going counterclockwise. Adjusted circ. positions have corrected the position to read clockwise looking down.

Flaws #3, 7,and 9 were identified as axial flaws using the circ. blade probe but are not confirmed with the rotating UT. Therefore, flaws #3, 7, and 9 are not relevant.

Analyzed by: K.C.Gebetsberger **Date:** 3/5/2002 **Analyzed by:** M.G. Hacker **Date:** 3/5/2002

Table 3. Nozzle 3 NDE Examination Results

FRAMATOME ANP

CRDM Nozzle Ultrasonic Examination Data Sheet

Customer:	FENOC		**Plant:** Davis Besse	**Unit:** n/a		**Nozzle:**	3
Procedure:	54-ISI-100-08	CA: FRA-02-002, DB-02-012	**Nozzle Dimensions: (in.)**	**ID:** 2.765	**OD:** 4.06	**Thickness:**	0.649
Downhill Side of Nozzle (deg.):	150	**End of Noz. (in** 30.75	Probe Serial No.'s:	Ch 1	2078-01002-0L	Ch 6	21GB-01002-45L
Axial Scan	Start: -5, 16	Stop: 360, 30.81	**Setup:** 1	Ch 2	21GF-01004-30L	Ch 7	21GC-01001-55L
Files:	T2061_15.39.37			Ch 3	21GA-01004-45L	Ch 8	22CD-01001-65L
Circ. Scan	Start: 6, 20.3	Stop: 360, 30.88	**Setup:** 2	Ch 4	2623-01002-60S	Ch 9	2624-01005-60S
Files:	T2061_14.09.39			Ch 5	2623-01002-60S	Ch 10	2624-01005-60S

Flaw No.	Surface (ID/OD)	Depth to Flaw Tip	End Point 1 Min (in.)	End Point 1 Min (deg.)	End Point 2 Max (in.)	End Point 2 Max (deg.)	Axial Total (in.)	Adjusted Circ. Extent Min (deg.)	Adjusted Circ. Extent Max (deg.)	Adjusted Circ. Extent Total (in.)	Flaw Length (in.)	Flaw Angle (deg.)	Flaw TWD (in.)	Flaw Aspect Ratio	Flaw Orientation	Weld Location @ Flaw Min	Weld Location @ Flaw Max
1	OD	TW	26.6	151.0	30.68	156.0	4.08	1.0	6.0	0.18	4.08	2	0.65	0.16	AXIAL	In Weld Region	
2																	
3	OD	0.234	28.07	275.0	29.19	280.0	1.12	125.0	130.0	0.18	1.13	9	0.42	0.37	AXIAL	In Weld Region	
4	OD	TW	26.07	319.0	29.89	330.0	3.82	169.0	180.0	0.39	3.84	6	0.65	0.17	AXIAL	In Weld Region	
5	OD	0.212	28.4	136.0	29.46	143.0	1.06	346.0	353.0	0.25	1.09	13	0.44	0.40	AXIAL	In Weld Region	
6																	
7																	
8																	
9																	
10																	
11																	
12																	
13																	
14																	
15																	
16																	
17																	

WELD PROFILE	Data Loc.	150	180	210	240	270	300	330	360	30	60	90	120	150	Degrees
	Noz. Loc.	0	30	60	90	120	150	180	210	240	270	300	330	360	Degrees
	Noz. End	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	Inches
	MAX	29.08	29.08	29.02	28.70	28.54	28.38	28.35	28.41	28.63	28.80	28.96	29.02	29.08	Inches
	MIN.	27.83	27.77	27.51	27.23	27.07	26.94	26.95	27.00	27.19	27.42	27.67	27.80	27.83	Inches

Notes: Adjusted Circ. Extent is relative to downhill side of nozzle;clockwise looking down. TWD is Through-Wall Dimension

Comments: These are axial flaws that extend from below the weld region into the weld region. They were also detected with the circ. blade probe.

Flaw # 2 was identified as an axial flaw using the circ. blade probe but is not confirmed with the rotating UT. Therefore, flaw #2 is not relevant.

Analyzed by: K.C. Gebetsberger **Date:** 3/5/02 **Analyzed by:** M.G. Hacker **Date:** 3/5/02

Table 4. Nozzle 5 NDE Examination Results

FRAMATOME ANP

CRDM Nozzle Ultrasonic Examination Data Sheet

Customer:	FENOC	**Plant:**	Davis Besse	**Unit:**	N/A	**Nozzle:** 5
Procedure:	54-ISI-100-08	CA: FRA-02-002, DB-02-012	**Nozzle Dimensions: (in.)**	**ID:** 2.765	**OD:** 4.06	**Thickness:** 0.649
Downhill Side of Nozzle (deg.):	320	**End of Noz. (in** 30.75	Probe Serial No.'s:	Ch 1 2078-01002-0L		Ch 6 21GB-01002-45L
Axial Scan	Start: -4, 16.11	Stop: 360, 30.78	**Setup:** 1	Ch 2 21GF-01004-30L		Ch 7 21GC-01001-55L
Files:	T2061_18.30.12			Ch 3 21GA-01004-45L		Ch 8 22CD-01001-65L
Circ. Scan	Start: -6, 19	Stop: 360, 29.41	**Setup:** 2	Ch 4 2623-01002-60S		Ch 9 2624-01005-60S
Files:	T2061_16.53.38			Ch 5 2623-01002-60S		Ch 10 2624-01005-60S

Flaw No.	Surface (ID/OD)	Depth to Flaw Tip	End Point 1 Min (in.)	End Point 1 Min (deg.)	End Point 2 Max (in.)	End Point 2 Max (deg.)	Axial Total (in.)	Adjusted Circ. Extent Min (deg.)	Adjusted Circ. Extent Max (deg.)	Adjusted Circ. Extent Total (in.)	Flaw Length (in.)	Flaw Angle (deg.)	Flaw TWD (in.)	Flaw Aspect Ratio	Flaw Orientation	Weld Location @ Flaw Min	Weld Location @ Flaw Max
1	OD	0.2	28.44	274.0	29.69	271.0	1.25	274.0	271.0	-0.11	1.25	5	0.45	0.36	AXIAL	In Weld Region	
2																	
3																	
4																	
5																	
6																	
7																	
8																	
9																	
10																	
11																	
12																	
13																	
14																	
15																	
16																	
17																	
WELD PROFILE		Data Loc.	320	350	20	50	80	110	140	170	200	230	260	290	320	Degrees	
		Noz. Loc.	0	30	60	90	120	150	180	210	240	270	300	330	360	Degrees	
		Noz. End	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	30.75	Inches	
		MAX	29.10	29.07	29.07	28.91	28.73	28.60	28.52	28.40	28.46	28.67	28.91	28.96	29.10	Inches	
		MIN.	27.90	27.89	27.89	27.68	27.39	27.21	27.13	27.10	27.10	27.21	27.68	27.91	27.90	Inches	

Notes: Adjusted Circ. Extent is relative to downhill side of nozzle;clockwise looking down. TWD is Through-Wall Dimension

Comments: This is an axial flaw that extends into the weld region. This flaw was also detected sing the circ. blade probe.

Analyzed by: K. C. Gebetsberger **Date:** 3/5/02 **Analyzed by:** M.G.Hacker **Date:** 3/5/02

Table 5. Nozzle 47 NDE Examination Results

FRAMATOME ANP

CRDM Nozzle Ultrasonic Examination Data Sheet

Customer: FENOC		**Plant:** Davis Besse	**Unit:** N/A		**Nozzle:** 47
Procedure: 54-ISI-100-08	CA: FRA-02-002, DB-02-012	**Nozzle Dimensions: (in.)**	**ID:** 2.765	**OD:** 4.06	**Thickness:** 0.649
Downhill Side of Nozzle (deg.): 143	**End of Noz. (in** 45.9	Probe Serial No.'s:	Ch 1	2078-01002-0L	Ch 6 21GB-01002-45L
Axial Scan Start: -6, 29.9	Stop: 360, 45.9	**Setup:** 1	Ch 2	21GF-01004-30L	Ch 7 21GC-01001-55L
Files: T2062_01.40.41			Ch 3	21GA-01004-45L	Ch 8 22CD-01001-65L
Circ. Scan Start: -6, 34	Stop: 360, 46	**Setup:** 2	Ch 4	2623-01002-60S	Ch 9 2624-01005-60S
Files: T2062_23.53.48			Ch 5	2623-01002-60S	Ch 10 2624-01005-60S

Flaw No.	Surface (ID/OD)	Depth to Flaw Tip	End Point 1		End Point 2		Axial	Adjusted Circ. Extent			Flaw	Flaw	Flaw	Flaw	Flaw	Weld Location @ Flaw	
			Min (in.)	Min (deg.)	Max (in.)	Max (deg.)	Total (in.)	Min (deg.)	Max (deg.)	Total (in.)	Length (in.)	Angle (deg.)	TWD (in.)	Aspect Ratio	Orientation	Min	Max
1																	
2																	
3	OD	0.06	43.23	181.0	45	202.0	1.77	38.0	59.0	0.74	1.92	23	0.59	0.31	AXIAL	In Weld Region	
4																	
5																	
6																	
7																	
8																	
9																	
10																	
11																	
12																	
13																	
14																	
15																	
16																	
17																	
WELD		Data Loc.	143	173	203	233	263	293	323	353	23	53	83	113	143	Degrees	
		Noz. Loc.	0	30	60	90	120	150	180	210	240	270	300	330	360	Degrees	
		Noz. End	45.90	45.90	45.90	45.90	45.90	45.90	45.90	45.90	45.90	45.90	45.90	45.90	45.90	Inches	
PROFILE		MAX	44.48	44.58	44.10	43.40	42.67	42.10	41.75	41.94	42.58	43.31	44.01	44.42	44.48	Inches	
		MIN.	43.10	42.96	42.42	41.49	40.54	39.62	39.39	39.49	40.19	41.40	42.38	42.96	43.10	Inches	

Notes: Adjusted Circ. Extent is relative to downhill side of nozzle;clockwise looking down. TWD is Through-Wall Dimension

Comments: Flaw #3 is an axial flaw that extends into the weld region. This flaw was also detected with the circ. blade probe.

Flaws #1 and #2 were identified with the circ. blade probe but were determined not to be valid detections with the rotating UT. Nozzle ovality in the location of these indications is the source of these false indications. Flaw #4 was detected with the rotating UT but it is located in the J-groove weld fillet and outside the nozzle wall and is therefore outside the scope of this procedure.

Analyzed by: K. C. Gebetsberger **Date:** 3/4/02 **Analyzed by:** M. G. Hacker **Date:** 3/4/02

Table 6. Comparison of Davis-Besse to Other B&W Design Plants

Parameter	Oconee 1	Oconee 2	Oconee 3	ANO-1	Davis-Besse	TMI-1	Crystal River 3
NSSS*	B&W	B&W	B&W	B&W	B&W	B&W	B&W
Material Supplier*	BWTP	BWTP	BWTP	BWTP	BWTP	BWTP	BWTP
Head Fabricator*	B&W	B&W	B&W	B&W	B&W	B&W	B&W
Design Nozzle Fit (mils)*	0.5 – 1.5	0.5 – 1.5	0.5 – 1.5	0.5 – 1.5	0.5 – 1.5	0.5 – 1.5	0.5 – 1.5
EFPYs Through Feb 2001*	20.4	20.3	20.1	8.0	14.7	16.8	14.9
Head Temp (°F)*	602	602	602	602	605	601	601
EFPYs Normalized to 600°F*	22.1	22.0	21.7	19.5	17.9	17.5	15.6
EFPYs to Reach Oconee 3*	-0.3	-0.2	0.0	2.1	3.1	4.1	5.9
Access Ports in Lower Shroud	Yes	Yes	Yes	No	No	Yes	Yes
Boric Acid on Head	Small Amount	Small Amount	Large Amount Prior to 2000	Some	Large Amount	Some	Some
Number of CRDM Nozzles	69	69	69	69	69	69	69
- With Leaks	1	4	14	1	3	5	1
- Leaks & Circ Cracks	0	1	4	0	1	0	1
- With Heat M3935	0	0	68	1	5	0	0
Number of T/C Nozzles	8	0	0	0	0	8	0
- With Leaks	5 confirmed	N/A	N/A	N/A	N/A	8	N/A
Counterbore at Bottom of CRDM Nozzles	Yes	Yes	Yes	Yes	No	Yes	Yes
As-Built Fit Range for Leaking Nozzles (mils)	Clearance	Clearance to 1.4 Interference	Clearance to 1.0 Interference	0.4 – 0.7	0.1 – 2.0		
Wastage at Leaks	No	No	No	No	Yes	No	No

* Data from MRP-48, PWR Materials Reliability Program – Response to NRC Bulletin 2001-01 (EFPY data as of February 2001).

Table 7. Nuclear Industry Experience Review Results

NRC Documents

Document	Davis-Besse Response/Actions	Comments
Bulletin 82-2, Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants. • Implement maintenance procedures for threaded fasteners. • Inspect and clean fasteners when removed. • List RCS closures that have leaked. • List where thread lubricants and Furminite was used on RCS fasteners.	• Maintenance procedures for threaded fasteners were written. • Inspection and cleaning of fasteners was added to the maintenance procedure. • Ten CRDM flanges and OTSG lower primary hand holes have leaked. • CRD reactor vessel nozzle bolts and OTSG manway & hold down bolts are lubricated. • One of the RCS cold leg thermowells was Furminited.	In 1987, an NRC inspection of the Bulletin concluded there were no violations or deviations.
GL 88-5, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants. The document requested assurances that Davis-Besse have a program to ensure that boric acid corrosion does not lead to degradation of the RCS boundary. The program should include: • Listing where small leaks could cause degradation, • Procedures for finding small leaks, • Evaluating the impact of leaks, & • Preventive actions for corrosion.	The Davis-Besse program consists of several programs and procedures. • Leakage Management Program, which identifies and the location of the leakage and evaluates the boric acid concern. • Shutdown procedure, which requires a walkdown of containment valves and a general containment walkdown. • ASME Section XI Inservice Pressure Test, which performs a visual inspection to look for discoloration. If boric acid residue is identified, find the source, determine the extent, and repair. • CRD Flanges are inspected each refueling. Gaskets are replaced on leaking joints. This will be incorporated into the PM program.	Although CRDM flanges are inspected, CRDM nozzles are not specifically listed. During an audit of the boric acid corrosion prevention program, the NRC found the program met the intent of the Generic Letter. Implementing procedures still need to be made effective. Engineers should be trained. Inspections should be documented.

Document	Davis-Besse Response/Actions	Comments
	• Periodic fastener inspection as a result of the IE Bulletin 82-2, Degradation of Threaded Fasteners in the RC Pressure Boundary of PWRs. • Limited Thermographic Inspections in containment to detect steam leaks as part of the current outage. • Live Load Packing of Valves to reduce stem leakage may be used if it proves a viable method. Davis-Besse will implement a Boric Acid Corrosion Program to include all the requirements of GL 88-5 in 1989.	
Generic Letter 97-1, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations. An integrated, long-term program, which includes periodic inspections and monitoring, is necessary. The following is requested: • Results of CRDM nozzle inspections. • Schedule for subsequent CRDM nozzle inspections. • The scope of subsequent inspections. • Or justify why no inspection is needed. • A description of resin bed intrusions.	The response is in B&WOG Topical Report, "B&WOG Integrated Response to Generic Letter 97-01: Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," BAW -2301. Inspections for B&W plants will be preformed based on susceptibility. There have been no resin bed intrusions at B&W plants. NEI proposed an integrated inspection program based on susceptibility.	Responses to requests for additional information were answered by NEI for the industry. The response emphasized that the integrated program is an ongoing program that will be implemented in conjunction with EPRI, the PWR Owners Groups, the participating utilities, and the Material Reliability Project's Subcommittee on Alloy 600.
IN 80-27, Degradation of Reactor Coolant Pump Studs. Several reactor coolant pump studs incurred boric acid wastage as a result of leaks in the pump flanges. If undetected,	An inspection of the Davis-Besse studs in 1980 revealed no corrosion in the studs for 3 of 4 RCPs. A small amount of rust and boric acid around the studs for 1 RCP was from an	Also described in SOER 81-12 and SER 46-80.

Document	Davis-Besse Response/Actions	Comments
corrosion of RCP studs could cause the loss of the RCS pressure boundary. To detect, supplemental visual examinations and instrumented leak detection are needed. Undetected wastage could occur in other components.	overhead valve leak, which was fixed previously. A work order was issued to clean the area. There is a drain between the inner and outer gaskets which goes to the containment sump, but there is no monitoring of the leakage and the drain valve is normally closed.	
IN 82-6, Failure of Steam Generator Primary Side Manway Closure Studs. There have been a significant number of failed or degraded bolts and studs due to stress corrosion cracking and corrosion wastage that are difficult to detect.	Response was deferred to the response to NRC Bulletin 82-2 Degradation of Threaded Fasteners in the RC Pressure Boundary of PWR plants.	
IN 86-108, Degradation of RCS Pressure Boundary Resulting From Boric Acid Corrosion. Boric acid from a leaking valve caused wastage of a carbon steel HPI line. The primary defense is to minimize leaks, detect and stop leaks soon after they start, and promptly clean up any boric acid residue. Detection of leaks will be enhanced by an evaluation of any iron oxide stains on insulation.	The Davis-Besse HPI line geometry is different. Provisions regarding iron oxide stains on RCS piping insulation will be included in the ASME Section XI Inservice Pressure Tests procedure.	The response is limited and fails to recognize the larger issue of boric acid corrosion.
IN 86-108 Supplements 1 & 2, Degradation of RCS Pressure Boundary Resulting From Boric Acid Corrosion. Supplement 1: Boric acid corrosion/wastage on the head of the Turkey Point 4 reactor and boric acid crystals in the CRDM cooling ducts. Small RCS leaks can concentrate the boric acid and	During shutdowns, a mode 3 containment walkdown will look for any buildup of boron on piping or valves and to notify engineering of any of any potential problem areas. An RCS leakage management policy maintains RCS leakage as low as possible	The mode 3 walkdowns cannot inspect the reactor head.

Document	Davis-Besse Response/Actions	Comments
rapidly corrode carbon steel. Supplement 2: Boric acid corrosion/wastage on the head of the Salem 2 reactor and failure of a shutdown cooling valve bolts due to boric acid corrosion. The INs recommended that inspection programs be reviewed to ensure adequate monitoring.	and identifies and evaluates corrosion concerns.	
IN 90-10, Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600. Plants should review their Inconel 600 applications and implement an augmented inspection program.	BWOG studied the problem in B&W Document 51-1201160-00. We expected the BWOG to recommend additional inspections. The study demonstrates that the issue of Inconel 600 applications is adequately reviewed and inspections are being formulated. Therefore, the intent of the IN is met.	This was evaluated along with SER 2-90 by RFA 90-831. However, the NRC made the issue much broader than INPO. We deferred our evaluation to the BWOG, which is summarized in the "Other Documents" below.
IN 86-108 Supplement 3, Degradation of RCS Pressure Boundary Resulting From Boric Acid Corrosion. Issued in 1995. Corrosion problems at Calvert Cliffs and TMI had earlier indication of leakage and in both cases, boric acid leakage was not immediately cleaned and stopped. The primary defense is minimize leakage, detect and stop leaks, & promptly clean the residue.	The Boric Acid Corrosion Control program addresses the issue.	The response just make the statement that the Boric Acid Corrosion Control program covers the concern but provides no basis for the conclusion.
IN 94-63, Boric Acid Corrosion of Charging Pump Casing Caused by Cladding Cracks. Although boric acid wastage occurs slowly, an attack can eventually lead to significant thinning of carbon steel cladding and possibly leakage. Corrosion of the base metal is easy to find though visual inspection.	This is not applicable to Davis-Besse since the Make-up Pumps and HPI pumps are solid stainless steel.	The Davis-Besse evaluation was narrowly focused on the charging pump and not on boric acid corrosion in general.

Document	Davis-Besse Response/Actions	Comments
IN 96-11, Ingress of Demineralizer Resins Increases Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations. EPRI is researching ways to mitigate PWSCC and developed a demonstration program to ensure that inspections performed on CRDM penetrations are highly reliable in detecting and determining the size of flaws. Resin intrusion into the RCS will cause circumferential Intergranular Stress Corrosion Cracking. There is a high probability that CRDM penetrations contain cracks caused by PWSCC.	The response deals with intrusion of demineralizer resins in the RCS. Davis-Besse has had no resin intrusion. PWSCC probability is low because of water chemistry and actions would be taken on high sulfate levels.	The Davis-Besse evaluation was narrowly focused on the resin intrusion and did not address PWSCC.
IN 2000-17, Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V.C. Summer. A crack was found on a weld on a hot leg pipe. Elevated leakage and radiation was not seen. It was found by discovering boric acid. When the root cause is determined, a supplement will be issued.	This is preliminary information and no action can be taken at this time. The information was adequately distributed for current needs. This information will be added to the final OE evaluation.	Although the IN only contained information and gave no recommendation on what could be done, it may have been more appropriate to have the system experts make that call. See the V.C. Summer Root Cause in the "Other Documents" section.
IN 2000-17 Supplement 1, Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V.C. Summer. A multi-disciplined team will conduct a root cause. A foreign plant also had crack indications in the hot leg. When the root cause is determined, another supplement will be issued.	This is preliminary information and no action can be taken at this time. The information was adequately distributed for current needs. This information will be added to the final OE evaluation.	Although the IN only contained information and gave no recommendation on what could be done, it may have been more appropriate to have the system experts make that call. See the V.C. Summer Root Cause in the "Other Documents" section.
IN 2000-17 Supplement 2, Crack in Weld Area of Reactor Coolant System Hot Leg Piping at V.C. Summer. The crack was	The issue is still under evaluation and we expect further information to be released by the NRC. The only action needed at this time	The OE program incorrectly assumed that more information would be issued. However, the V.C. Summer Root Cause

Document	Davis-Besse Response/Actions	Comments
caused by PWSCC. Extensive weld repairs were a contributing cause. The V.C. Summer root cause was thorough and concluded it was PWSCC. Welding met code requirements. Leak detection enhancements will be made. The following generic issues need to be addressed. • NDE failed to detect the cracks. • ASME code allows multiple weld repairs. • Weaknesses in leak detection systems. • Applicability of "Leak before break" analysis.	is information distribution. When the final document is evaluated, this information will be attached.	Evaluation was complete. Yet it wasn't obvious to the review committee that this supplement listed the generic causes. It may have been more appropriate to have the system experts review the information. There are several references to additional problems, but there was no effort to seek out the additional information. See the V.C. Summer Root Cause in the "Other Documents" section.
IN 2001-5, Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3.	Response was deferred to the response to NRC Bulletin 2001-1.	The response to the Information Notice failed to follow the OE program. See CR 2001-2997.

INPO SEE-IN Documents

Document	Davis-Besse Response/Actions	Comments
SOER 81-12, Reactor Coolant Pump Closure Stud Corrosion. The SOER noted that insulation reduces the likelihood of discovering leakage/boric acid deposits and the insulation may have caused retention of borated water and increased the possibility of corrosion. The SOER noted that the rate of corrosion increased when boric acid deposits are wetted and present inspection frequencies	The DB response said that the RCP studs were inspected in 1980 and no damage was found. Boric acid was found and cleaned. We have a procedure and PM to inspect the studs. Both perform a visual examination and generate a Material Deficiency if anything relevant is found.	This SOER was last reviewed in March 2001. The SOER and evaluation is very focused on RCP studs. However, it brings out the facts that boric acid corrosion can be rapid and insulation needs to be removed to find boric acid deposits.

Document	Davis-Besse Response/Actions	Comments
are not adequate for timely detection. Recommended a visual inspection of the RCP closure studs. Recommended removal of residual leakage and boron deposits from the closure flange area.	The response says that if boric acid deposits are found, areas will be inspected & deposits removed according to NG-EN-324.	Also described in IN 80-27 and SER 46-80.
SOER 84-5, Bolt Degradation or Failure in Nuclear Power Plants. The SOER noted that fastener failures are occurring due to boric acid corrosion and stress corrosion cracking. The SOER recommended that we ensure prompt repair of leaking joints with boric acid deposits.	Practices are in place to identify and fix leaks. We perform walkdowns in containment to find and fix leaks (if possible) to minimize boric acid damage. Work requests for boric acid leaks receive higher priority due to radiation and contamination corrosion concerns.	A Green SOER that is no on the INPO 97-10 list. This SOER was last reviewed in late 1987. The response many times cited routine inspections or walkdowns that we perform, but those can't identify leaks in containment. The response still didn't seem to recognize the importance of boric acid corrosion. The response says boric acid leaks are repaired because of radiation and contamination concerns, not because of corrosion concerns. Based on the lack of action to fix RC2, we did not promptly repair the leaking joint with boric acid deposits.
SER 46-80, Reactor Coolant Pump Closure Stud Corrosion. The SER noted that leaking gasketed joints (e.g., Control rod drives & reactor vessel head) might be affected by boric acid attack. Although closure studs are subject to inservice inspections, corrosion damage was not detected.	No specific DB response was found.	This issue was subsequently described in SOER 81-12. Also described in IN 80-27.
SER 35-81, Corrosion of Reactor Coolant System Piping. The SER says corrosive	No DB response was found.	

Document	Davis-Besse Response/Actions	Comments
attack could reduce primary boundary integrity. INPO will continue to evaluate this event.		
SER 11-82, Reactor Coolant Pump Closure Flange Stud Corrosion. The repeat of stud corrosion and the amount of corrosion re-inforces the importance of frequent visual inspections and removal of boric acid deposits - as described in SOER 81-12.	No DB response was found.	
SER 57-83, Cracking in Stagnant Boric Acid Piping. Many cracking incidents have occurred.	Seven line handwritten response saying boric acid piping is inspected in the ISI program and this hasn't happened here. The SER was distributed for information.	
SER 72-83, Damage to Carbon Steel Bolts and Studs on Valves in Small Diameter Piping Caused by Leakage of Borated Water. When scheduling maintenance, take boric acid corrosion rates into account. Ten year ISI may not be frequent enough.	The evaluation was deferred to SOER 84-5. The SER was distributed for information.	In previous responses, we've claimed that boric acid piping is inspected during by the ISI program, yet this has warned us that the ISI is not adequate to detect these problems.
SER 32-84, Contamination of Reactor Coolant System by Magnetite and Sulfates.	No DB response was found.	Although this discusses RCS leakage, this doesn't appear to provide any insight to this issue.
SER 41-85, Containment Spraying Events. Prompt clean up of boric acid reduces corrosion. Boric Acid solutions in insulation are hard to remove.	DB recognizes that prompt clean up is essential to ensuring the integrity of carbon steel. The ability to detect and clean up each boric acid spill will depend on the circumstances. An Erosion/corrosion program will find degradation.	The evaluation failed to address the problems with insulation. The erosion/corrosion program response has no bearing on the concern.
SER 13-87, Reactor Vessel Stud Corrosion	We inspect reactor head area by operations	The body of the SER was focused on

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from Primary Coolant Leak. Inspect reactor head for boron during all planned and unplanned outages. The 1 GPM T.S. won't detect small leaks.	walkdown during shutdowns. During startups, we inspect containment.	fasteners and said that no structural integrity was effected. This may have influenced the evaluators against concerns about what is happening in the service structure. Operations walkdowns would not be able to detect boric acid on the head. At best, this evaluation may have assumed that operations could see any boric acid draining down onto the reactor head studs. The evaluation failed to understand that a detailed internal inspection was needed. During the times cited in the evaluation, this could not have been done.
SER 31-87, Pressurizer Vessel Corrosion due to Pressurizer Heater Rupture. The SER noted that Boric Acid corroded a 1/2 inch diameter, 3/4 inch deep hole in the lower pressurizer head and could only be seen with the insulation removed. Boric acid corrosion causes damage and extends outages. Rates can be up to 1.65 inches per year. Small leaks can cause severe damage. Periodic inspections are needed to identify leaks. Sources of leaks need to be repaired.	Evaluation of boric acid damage was deferred to the evaluation of SER 13-87. Evaluation of inspection for boric acid was deferred to the evaluation of SER 13-87. Since maintenance will walk down and determine repairs, boric acid damage will be found and fixed.	The evaluation missed the point that the insulation needs to be removed to find the damage. There was no effort made to try to highlight this concern.
SER 35-87, Non-Isolable Reactor Coolant System Leak. Make sure that resistant material is used for valves. If a valve in the boric acid system fails, consider possible boric acid causes.	Spec M-452Q considers component specifications. Maintenance reports as found conditions to the plant engineers. They would recommend corrective actions.	The response was superficial and missed the point, but has little bearing on this issue.

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	The SER was distributed for information.	
SER 10-89, Reactor Coolant Pump Flange Leak from Loss of Bolt Preload. Bolts should be checked for preload.	Preload was checked due to other reasons earlier.	The focus and recommendations are on RCP stud tightness and not boric acid corrosion, which is referenced back to SOER 81-12 & SER 13-87.
SER 90-2, Pressurizer Heater Sleeve Cracking. Inspect Inconel 600 pressurizer heater sleeves for leakage.	The overall evaluation was deferred to the BWOG Material Committee "to monitor this issue to conclusion." The SER was distributed for information.	We were given the right answers, it's unknown if we recognized it and used it. This is a very interesting issue. NRC IN 90-10 was also issued on Inconel 600 Stress Corrosion Cracking and made much broader recommendations. The industry conducted studies on the problem. Based on the detail in related documentation, we seem to recognize the concern and we expended much effort in studying the problem. In memorandum NED 91-20038, we recognized that only a visual inspection can find a through wall crack. Boric acid is an indicator of a potential problem. It recommended that we inspect the CRDM tubes. Based on damage DB incurred in 6RFO, we understood the consequences of boric acid corrosion. See the BWOG safety evaluation, which is summarized in the "Other Documents" below.
SER 20-93, Intergranular Stress Corrosion Cracking in Control Rod Drive Mechanism Penetrations. The affected plants (in Europe) planned on inspected all head penetrations	Response deferred to BWOG. The conclusion said, "Based on the completed safety evaluation and the ongoing	The response documentation includes a BWOG Project Authorization Request for the Material Committee. Task 5.4 is for developing top-of-head inspection tooling for

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and installing new insulation to allow leak detection testing. The cracks are not significant to safety. Plants with similar head penetrations should review their testing and inspection programs.	industry effort, no further action with respect to this SER is deemed necessary."	CRDM nozzles. The task was planned for 1996. There seems to be a gap of SEE-IN documents addressing boric acid corrosion and stress corrosion cracking between 1990 and 2000 - as if both issues fell off the nuclear radar screen. This was the only SEE-IN document found in that time frame. See the BWOG safety evaluation, which is summarized in the "Other Documents" below.
SER 4-01, Recent Events Involving Reactor Coolant System Leakage at Pressurized Water Reactors. Detailed reactor inspections are important to identify boric acid. Of particular concern are areas covered by insulation or otherwise inaccessible. Undetected or uncorrected RCS leakage can result in reactor coolant system pressure-retaining component degradation from corrosion and wastage. RCS leakage can result in extended outages or substantial increases in personnel radiation exposure. Small leaks often are not detected by installed leak detection systems or RCS inventory balance calculations, emphasizing the need for thorough visual and other nondestructive examinations. Oconee modified the service structure and cleaned	NG-EN-00324, Boric Acid Corrosion Control, provides the required actions to identify, evaluate, and resolve boric acid leakage and corrosion. Any identified leakage is evaluated to determine corrective actions. For leakage that is not repaired, monitoring is specified. The specific locations include Control Rod Drive Flanges. Inservice inspection program will perform leakage inspections beneath the reactor vessel head insulation.	The response gave the impression that the program was comprehensive. There was one OERC member who did feel the response was not adequate, but backed off. The response did not raise the issues that are coming to light now that we were unable to inspect the center part of the head and there was boric acid there and that we had decided not to fix or clean those areas. The response did not give any hints that there were weaknesses.

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the head to allow easier detection. Although still in study, VC Summer is doing Noble Gas sampling.		
SEN 6, Boric Acid Corrosion.	Evaluation deferred to SER 13-87.	
SEN 18, Reactor Vessel Head Corrosion	Evaluation deferred to SOER 81-12.	
SEN 190, Pressurizer Spray Valve Bonnet Nuts Dissolved by Boric Acid.	No evaluation found. Distributed for information.	A Davis-Besse event.
SEN 216, Leakage from Reactor Vessel Nozzle-to-Hot Leg Weld.	OERC determined that the document only contained preliminary information and no action can be taken at this time. Distributed for information.	Although the SEN only contained information and gave no recommendation on what could be done, it may have been more appropriate to have the system experts make that call.
SEN 220, Pressure Boundary Leakage at Palisades. Palisades had a through-wall crack in a CRDM housing.	Deferred to SEN 4-01.	
O&MR 348, Failure of a Limitorque Operator Stem Nut	DB is in compliance with recommendations.	This does not seem to provide any value to this issue.