UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

March 15, 2001

EA-01-071

South Carolina Electric & Gas Company
ATTN: Mr. Stephen A. Byrne
Vice President, Nuclear Operations
Virgil C. Summer Nuclear Station
P. O. Box 88
Jenkinsville, SC 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION - NRC SPECIAL INSPECTION REPORT NO. 50-395/00-08, EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Byrne:

On February 15, 2001, the Nuclear Regulatory Commission (NRC) completed a special inspection at your Virgil C. Summer reactor facility. The enclosed report presents the results of that inspection which were discussed with you, and other members of your staff on February 15, 2001, in an exit meeting open for public observation, and in a telephone conversation with Mr. Gary Williams and other members of your staff on March 13, 2001.

The special inspection was initiated following your identification of a large quantity of boric acid residue on the containment floor during a containment inspection and an apparent through-wall crack in the “A” hot leg pipe to vessel weld. This was later confirmed to be a 2-1/2 inch axial crack with a 3/16 inch diameter exit point in the weld. The basis for the special inspection was to review the event, determine potential generic issues, and to determine the implications to the inservice inspection program. It included review of original fabrication records for the weld, the inspection history of the weld, the metallurgical examination and root cause evaluation, your extent of condition review and corrective actions, the weld repair, and your leak detection methods. The team examined special process procedures, observed activities, and interviewed personnel. The charter directing the team’s objectives is included in Enclosure 2.

Based on the results of this inspection, no performance deficiencies were identified. Your root cause analysis was thorough, well organized, and performed with personnel with appropriate expertise. You determined that extensive repairs to this weld during original plant construction in 1978 generated high residual tensile stresses, which contributed to primary water stress corrosion cracking. The team determined that Code requirements had been met throughout the history of this weld. The team’s identification of generic issues and inservice inspection implications will be documented in separate correspondence, with recommendations, to the Office of Nuclear Reactor Regulation.

The team determined that the small leak from the reactor coolant pressure boundary had been ongoing for several months. Because Technical Specifications (TS) require that with any reactor coolant pressure boundary leakage, the plant be placed in hot standby within six hours, the NRC concluded that a violation of TS occurred. This violation involved equipment failure

not avoidable by reasonable quality assurance measures or management controls and is considered to have resulted from matters not within your control. Based on this and in consultation with the Regional Administrator, Region II, the Director, Office of Enforcement, and the Deputy Executive Director for Reactor Programs, the NRC has determined that the exercise of enforcement discretion is warranted in accordance with Section VII.B.6 of the "General Statement of Policy and Procedure for NRC Enforcement Actions - May 1, 2000," NUREG-1600, as amended on November 3, 2000 (65 Federal Register 59274) (Enforcement Policy). Accordingly, a Notice of Violation will not be issued. Based on your letters dated December 29, 2000, January 9, 2001, and February 9, 2001, we understand that you intend to enhance your leak detection procedures (including boric acid walkdown procedures) and perform additional inspections of the reactor vessel nozzle-to-pipe welds during the next two refueling outages.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available **electronically** for public inspection in the NRC Public Document Room **or** from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

C. Casto, Director
Division of Reactor Safety

Docket No.: 50-395
License No.: NPF-12

Enclosures: 1. NRC Special Team Inspection Report 50-395/00-08
2. Special Inspection Charter Dated October 19, 2000

Attachments: 1. List of Documents Reviewed
2. List of Acronyms

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NAME	B. Crowley	E. Girard	W. Koo	S. Doctor	M. Lesser	K. Landis	O. DeMiranda
DATE	3/8/2001	3/8/2001	3/12/2001	3/8/2001	3/13/2001	3/14/2001	3/14/2001
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-395
License No.:	NPF-12
Report No.:	50-395/00-08
Licensee:	South Carolina Electric & Gas (SCE&G)
Facility:	Virgil C. Summer Nuclear Station
Location:	P. O. Box 88 Jenkinsville, SC 29065
Dates:	October 18, 2000 - February 15, 2001
Team Leader:	B. Crowley, Senior Reactor Inspector
Inspectors:	S. Doctor, Senior NDE Specialist, Pacific Northwest National Laboratory E. Girard, Senior Reactor Inspector W. Koo, Senior Material Engineer
Approved by:	M. S. Lesser, Chief, Maintenance Branch Division of Reactor Safety

Enclosure 1

SUMMARY OF FINDINGS

IR 05000395/00-08, October 18, 2000 - February 15, 2001; South Carolina Electric & Gas Company; Virgil C. Summer Nuclear Station; Special Team; “A” hot leg nozzle-to-pipe weld crack.

The team members included personnel from Region II and the Office of Nuclear Reactor Regulation and a specialist in nondestructive examination under contract with the Office of Nuclear Regulatory Research.

On October 7, 2000, at the beginning of refueling outage 12, plant personnel identified a large accumulation of boric acid residue in the area of the “A” reactor coolant (RC) system hot leg near the reactor vessel (RV). A potential leak was visually identified at the “A” loop hot leg nozzle-to-pipe field weld (the weld attaching the hot leg pipe to the reactor vessel nozzle). Trace amounts of boric acid buildup were identified on the pipe outside diameter (OD) in the weld area. Liquid penetrant (PT) inspection of the area in question identified a four-inch long circumferential indication. Further nondestructive examinations (NDE) determined that this was a surface indication and was not the source of the RC leak. A 2-1/2" long axial indication was identified at approximately nine degrees from the top of the pipe looking toward the RV. In addition, eddy current testing (ET) identified a number of small indications on the inside diameter (ID) of the weld, which were not identified by ultrasonic testing (UT). ET testing also identified similar small indications on the ID of four of the other five nozzle-to-pipe welds. The “A” hot leg weld was cut out and replaced. Metallurgical analysis of the weld showed the through-wall leak was from the 2-1/2" axial crack with a 3/16" exit point in the nozzle-to-pipe weld at approximately the nine degree location identified by NDE. The analysis also showed a number of the small ET indications to be shallow cracks. The cracking was determined to be primary water stress corrosion cracking (PWSCC). An integrity evaluation was performed to justify continued plant operations without removal of the indications in the other four welds. The destructively determined flaw sizes of the ET indications in the loop "A" hot leg nozzle-to-pipe weld were used to support the crack growth calculations of the ET indications found in the other nozzle-to-pipe welds. On February 20, 2001, the NRC staff issued a safety evaluation documenting review of the integrity evaluation. The staff's safety evaluation concluded that V. C. Summer nuclear plant could be safely operated for one fuel cycle (1.5 years) with the ET indications in the other nozzle-to-pipe welds not being removed.

Cornerstones: Initiating Events, Barrier Integrity

The team’s review of original weld fabrication records verified compliance with American Society of Mechanical Engineers (ASME) Code requirements for the original weld. The records were detailed and provided a good weld history. No Code compliance issues were identified. However, the records revealed extensive repairs to the ID of the “A” hot leg nozzle-to-pipe weld, which were determined by the root cause analysis to be a contributor to the crack by producing high residual tensile stresses at the ID of the weld. The radiographic (RT) film did not reveal any fabrication flaws that could have contributed to the through-wall leak.

The nozzle-to-pipe inspections conducted in 1980 for the preservice inspection (PSI) and in the 1987 and 1993 inservice inspections (ISI) met the applicable ASME Code requirements. These inspections used the state-of-the-art NDE technology that was available at that time. No flaws were detected that were unacceptable to the 1977 Edition of the ASME Code including the Summer of 1978 Addenda.

Overall, the team concluded that the examinations performed on the nozzle-to-pipe welds during the current outage were of high quality. The examination methods were successfully demonstrated on a mockup with qualified personnel and state-of-the-art NDE equipment and procedures. The use of redundant and complementary NDE techniques (visual, ultrasonic and eddy current) and the successful demonstrations of these techniques provided confidence that a sensitive inspection was conducted. The examinations exceeded the minimum requirements of the ASME Section XI Code and NRC Regulatory Guide 1.150.

The licencee's metallurgical evaluation of the cracked hot leg loop "A" nozzle weld, including the size measurements (length and depth) of the reported UT and ET indications by destructive examinations, was thorough. The licensee adequately characterized the failure mode to be primary water stress corrosion cracking (PWSCC).

The licensee's root cause analysis was thorough and well-organized and was performed utilizing personnel with appropriate expertise. The root causes and contributory factors for the leak and the extent of condition were adequately determined. Actions have been established to address the root causes, contributing factors, and the extent of condition. The team concluded that the licensee's assessment provided reasonable assurance that structural integrity of the RC system, from an impending gross failure standpoint, was maintained during past operation.

All welding and NDE activities for the new welds met Code requirements. The gas tungsten arc welding (GTAW) process with Alloy 52 material resulted in rejectable weld defects in the new nozzle-to-pipe weld. A number of repair attempts were required before successful repair using the shielded metal arc welding (SMAW) process with Alloy 152 welding material. The team agreed with the licensee's evaluation that the new weld, with different , more resistant material and less ID tensile stress, should be much more resistant to PWSCC than the old weld. However, based on the fact that PWSCC is not totally understood, the team concluded that further evaluations and inspections will be needed before it can be concluded that the new weld is totally immune to PWSCC.

The inspection team concluded that a comprehensive and effective inspection, that met or exceeded the requirements of the ASME Code, was conducted on the replacement nozzle-to-pipe dissimilar metal weld (DMW) and the stainless steel pipe-to-pipe weld.

The licensee's leak detection practices generally would not have been expected to identify the small leak on the "A" hot leg leak during plant operation. Although 0.3 gallons per minute (gpm) of unidentified reactor coolant leakage was present during the operating cycle, as determined by a periodic water inventory balance, this leak rate was not considered unusual and was well below the Technical Specification limit of 1 gpm. The licensee plans a number of enhancements to their RC system leakage detection practices.

In addition to the unidentified leakage limit, Technical Specifications (TS) do not allow any pressure boundary leakage and require shutdown within six hours. This leakage was pressure boundary leakage and existed for several months prior to its discovery and therefore constitutes a violation of the Technical Specifications. However, based on the team's conclusion that the violation was not avoidable by reasonable licensee quality assurance measures and management controls, the NRC is refraining from issuing enforcement action in accordance with section VII.B.6 of the NRC Enforcement Policy.

The boric acid corrosion inspections performed in the last two refueling outages (RO) (April 1999 and October 1997) were adequately performed, and the pre-entry radiological survey for RO-12 identified the large accumulation of boric acid crystals on the reactor building floor that led to the discovery of "A" hot leg RC leak. Since some details had been deleted from the program, the team concluded that the boric acid corrosion inspection program should be enhanced to improve guidelines for early detection of reactor coolant leakage and to expand the scope of inspection to include the welds that are susceptible to PWSCC. By letter dated December 29, 2000, the licensee stated that their boric acid inspection procedures will be enhanced to provide additional detail for the inspection and evaluation of RC system leakage, and that specific components and locations to be inspected will be listed and guidance provided on methodologies for evaluation.

Report Details

BACKGROUND

On October 7, 2000, at the beginning of refueling outage 12, plant personnel identified a large accumulation of boric acid residue in the area of “A” hot leg near the reactor vessel (RV). Boric acid deposits and RC pipe insulation were removed from the area of the suspected leak to allow for further inspection. On October 12, plant personnel visually identified a potential leak at the “A” loop hot leg nozzle-to-pipe field weld FW-23 (the weld attaching the hot leg pipe to the reactor vessel nozzle). Trace amounts of boric acid buildup were identified on the pipe OD in the weld area. Subsequent cleanup and PT inspection of the area in question identified a four inch long circumferential indication. Looking toward the RV, the indication was located approximately 17" counterclockwise from the top of the pipe. Video inspection of the inside of weld FW-23 from the RV using a pole-mounted camera identified suspected cracking on the inside of the nozzle-to-pipe weld near the nozzle side of the weld. On October 18, 2000, a special inspection was chartered to review and observe licensee actions to identify the cause of the weld leak, determine the root cause and extent of condition, and make necessary repairs. This report documents the results of these inspection activities.

Subsequent to initiation of the special inspection, the licensee found that the four inch circumferential PT indication was not the source of the RC leak. It was resolved to be a surface indication caused by boric acid attack (from the leak) at the fusion line between the nozzle base material and Inconel butter of the nozzle-to-pipe weld. The suspected cracking observed on the video was a result of shadows from the camera angle and not cracking. The actual leak was from a 2-1/2" axial crack with a 3/16" exit point in the nozzle-to-pipe weld. The crack was located at a different circumferential location from the PT indication.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Barrier Integrity

1R1 Review of Original Construction Records

a. Inspection Scope

The team reviewed the original construction weld records for all six reactor vessel nozzle-to-pipe field welds to verify that original welding and NDE, conducted in 1978, were performed in accordance with applicable ASME Code requirements, and to determine if the records contained evidence of fabrication flaws that could have contributed to a through-wall flaw. The records consisted of: Controlled Weld Joint Records, Liquid Penetrant Inspection Request and Reports, Radiographic Inspection Request and Reports, and Repair Instruction Sheets. These records for the original welds and repair welds were reviewed. In addition, a sample of welder qualification records and weld material certification records were reviewed. The team also reviewed the weld history summary developed by the licensee’s Root Cause Team. In addition, RT film (including repair, intermediate, and final) were reviewed for the “A” hot leg weld (FW-23), the “A” cold leg weld, and the “B” cold leg weld. The computerized enhanced RT images for the “A” hot leg weld produced by Electric Power Research Institute (EPRI) as part of the root cause analysis were also reviewed.

b. Observations and Findings

The nozzle-to-pipe field welds were dissimilar metal welds (DMWs) between the SA-508 low-alloy steel nozzle forging and the type 304 stainless steel pipe (hot legs) and elbows (cold legs). For consistency and simplicity, the welds will be referred to as nozzle-to-pipe welds for the remainder of the report. The end-prep of the SA-508 nozzle material was shop welded ("buttered") by the reactor vessel manufacturer before stress relief of the reactor vessel using the SMAW process and Alloy 182 Inconel weld material. The field welds between the "buttered" nozzles and the pipes (hot legs) and elbows (cold legs) were welded predominately with the GTAW process using Alloy 82 Inconel welding material. Most of the repair welding was done with the GTAW process using Alloy 82 Inconel material, and the records indicated some Alloy 182 Inconel was used in the repairs.

The records revealed that all of the nozzle-to-pipe field welds, except the "B" cold leg, had repairs. However, the "A" hot leg weld was the only weld with extensive repairs on the ID. Although the records were detailed and provided a good history of the sequence and number of repairs, it was not possible to determine the full extent of repairs in many cases. The "A" hot leg weld was the first RC system field weld to be made and extensive repairs were required. The records indicated that six repairs were made over a period of several months. The weld was started using the manual process with the root pass being welded with the GTAW process and several layers of SMAW welding over the root to obtain a thickness of approximately 3/4". The intermediate RT inspection at this level revealed weld defects essentially 360 degrees around the weld. Welding was discontinued for approximately four weeks while an automated GTAW welding procedure was prepared and qualified. The weld was repaired by: removing a portion of the defective material from the outside, welding a "bridge" weld over the remaining defective weld using the newly qualified automated process, removing the remaining defective weld material from the inside, and re-welding from the inside with the manual process using mostly the GTAW process with some SMAW process welding. The weld was then completed from the outside using the automated GTAW process.

The "A" hot leg weld had five additional repairs with some of the repairs extensive and extending through-wall. These repairs included extensive repair in the 7-10 degree circumferential location, the location of the through-wall leak and the 2-1/2" axial crack (described in detail below). In addition, the weld ID surface was ground extensively during the repair process.

The original construction RT film was in good condition and of very high quality with good sensitivity and revealed good quality welds. In addition to the required final weld film, all weld repair film had been retained. The film did not reveal any fabrication flaws that could have contributed to the through-wall leak.

c. Conclusions

Review of original weld fabrication records verified compliance with ASME Code requirements. The records were detailed and provided a good history of the sequence and number of repairs. However, it was not possible to determine the full extent of

repairs in many cases. The “A” hot leg weld (FW-23) was the only weld with extensive repairs on the ID. The RT film did not reveal any fabrication flaws that could have contributed to the through-wall leak.

1R2 Previous Inspection Results (PSI and ISI Inspections)

a. Inspection Scope

The team reviewed previous PSI and ISI records for the six reactor vessel nozzle-to-pipe welds to determine if previous ISI inspections met ASME Code requirements and to verify that any identified weld flaws were adequately dispositioned. This review included review of reports for the 1980 PSI and 1987 and 1993 ISIs. In addition, for the 1993 ISI, a detailed review of the raw data and processed data in B, C, and D scan images for the “A” hot leg weld and a sample of data for the other five welds was performed. The PSI in 1980 and the ISIs through 1993 were all conducted to the 1977 Edition of the ASME Boiler and Pressure Vessel Code including the Summer of 1978 Addenda.

b. Observations and Findings

The PSI baseline was performed in 1980. PT and visual inspections (VT) were performed from the OD. Westinghouse (now WesDyne International Inc.) performed a manual UT inspection from the OD and an automated UT inspection from the ID. In discussions with WesDyne personnel, they stated that the typical examinations were conducted using a 15" water path for the vessel and probably about 8" for the nozzle-to-pipe welds. Generally, transducers operating at 1 to 2.25 MHz were used that had a diameter of 1" to 1.5". Inspections were performed in four directions using inspection angles that were approximately 45 degrees. The inspection system was set up to scan and when a signal exceeded a set threshold level in a gate, the gate alarm would stop the scanner. Then the certified Level 2 examiner would jog the scanner around the indication and manually record information such as location, amplitude, transit time, and echodynamic properties. One recordable indication was reported in the “A” hot leg nozzle-to-pipe weld. The indication was located circumferentially at 272 degrees when looking out of the vessel and in a clockwise direction and was determined to be near surface but embedded (not surface connected) flaw. The indication was interpreted to be slag with a length and depth of 0.4". It was dispositioned to be acceptable according to the 1977 edition of the ASME Code with Summer of 1978 Addenda using Table IWB-3514-2.

An inspection was conducted in 1987, although the reason for this inspection was not clear. The ISI program would require an inspection of these welds every 10 years, and since the plant went operational on 01/01/84, no inspections were expected until the 1993-1994 time period. For the 1987 inspection, PT was performed on the OD of the nozzles. There was some VT performed on the ID. There was an inspection conducted with UT from the ID using an automated system by Combustion Engineering. These inspections were performed with 2.25 MHz, 65 degree and 1 MHz, 45 degree refracted longitudinal (RL) transducers A KB 6000 UT flaw detector was used for these inspections. There were several acceptable indications recorded but only one in the “A” hot leg nozzle-to-pipe weld. This indication was located in the DMW at 272 degrees and

had a ligament to the wetted ID surface of 0.85", a length of 0.5" and a through-wall of 0.29". This indication correlated well with the indication recorded in the 1980 PSI. This indication was considered to be planar and was accepted based on the same Code as used for the PSI.

The planned 10-year inservice inspection was performed during the 1993 refueling outage. Ebasco performed PT on the OD of the six RV nozzle-to-pipe welds. Some small 1/8" to 1/4" long indications were detected on the OD of the “A” hot leg nozzle-to-pipe weld with PT. These indications were located near 180 degrees from top dead center (TDC) and on the OD of the 508 forging as it transitions to a larger thickness. These indications were acceptable per ASME Code Section XI IWB- 3514.3.

Also, UT and VT inspections were performed by Westinghouse from the ID. The UT was a conventional near-surface examination using a 70 degree transducer from the ID and imaging conducted in accordance with the 1977 Edition of the ASME Section XI Code, Summer 1978 Addenda as augmented by NRC Regulatory Guide 1.150, Revision 1. These inspections were performed using the Ultrasonic Data Recording and Processing System (UDRPS) II data acquisition and analysis system. The inspections were based on amplitude response and for the nozzle-to-pipe welds were specified by the procedure to be 50% distance amplitude correction (DAC) recording criteria. The nozzle-to-pipe examinations were conducted from the ID using 70 degree dual longitudinal transducers scanning parallel and perpendicular to the weld seam. Transducers were used operating at 2 MHz and focused at a metal path of 30mm. These transducers were rigidly mounted in a 5-transducer sled that was contoured to the diameter of the nozzle bore so that all of the transducers were aligned to the scanning surface. The sled held four 70-degree transducers and a 0-degree transducer. Calibration was on side drilled holes at 0.25", 0.5" and 0.75" depth from the ID surface in a DMW calibration block. The UDRPS system was set-up to record all signals from 5% to 120% of DAC. The 70-degree transducers were used to focus on the inner 1/3 of the nozzle-to-pipe weld.

No indications were recorded in the hot leg and cold leg nozzle-to-pipe welds. WesDyne staff repeated the analysis of the 1993 inspection data on October 23-24, 2000, and reached the same conclusions. There were no indications that could be identified other than normal persistent patterns of interface response attributable to normal weld/interface metallurgy.

During early November 2000, team members conducted an independent review of all 1993 UT data for the “A” and “B” hot leg nozzle-to-pipe welds. This included the axial scans for circumferential flaws and the circumferential scans for axial flaws. In reviewing the axial scans, there were no signals that could be identified as originating from a flaw. The weld and the buttering produced high levels of noise, but these did not have the properties of flaws. The circumferential scans looking for axial flaws were also examined. For the “A” hot leg nozzle-to-pipe weld, the scan was clean with no apparent signals except in one location. This location was an area where there was an obvious decoupling of the transducer. Decoupling is very apparent in the image because one of two things always happens. Either the zone is not effectively inspected because in the image the acoustic noise level becomes very low relative to the areas adjacent to it or there are signals that are saturated caused by reverberations with the surface

discontinuity causing the decoupling or a combination of both conditions. The most important point is that this kind of condition is generally easy to recognize in the image. When it occurs, this zone is declared uninspected. For the “A” hot leg DMW, this occurred in only one location for about 2 inches out of a total of 107 inches. About 98% of the weld was successfully inspected but this uninspected zone unfortunately, was in the location of approximately 9 degrees where the inspection conducted in 2000 found the 2-1/2" long axial crack.

Only the inspection results documented in the summaries for the 1980 and 1987 inspections were reviewed by the team since the raw data was primarily tables of numbers containing the inspection results, which does not lend itself to independent re-interpretation. The inspections were conducted using an automated inspection tool and when a signal exceeded a gate the system stopped and the operator manually moved the tool and wrote down data on transducers location, metal path, and amplitude response and comments on observations. However, the inspection results from 1993 could be independently reviewed and this was performed by using the raw data, processed data and a computer work station. The reinterpretation did not find any new indications. Furthermore, the data showed that the circumferential scans (used to detect axial flaws) had good coupling except in the location where the through-wall axial flaw was located. The inspection met ASME Code requirements because more than 90% of the weld was fully inspected.

c. Conclusions

The inspections conducted in 1980 for the PSI and in the 1987 and 1993 ISIs met the applicable ASME Code requirements. These inspections used the state-of-the-art NDE technology that was available at that time. No flaws were detected that were unacceptable to the 1977 Edition of the ASME Code including the 1978 Addenda.

1R3 Review of Current Outage Pre-repair NDE

a. Inspection Scope

The inspection team reviewed/observed the NDE inspections conducted in the current outage on all six nozzle-to-pipe DMWs. This review included: review of the procedures employed; observing demonstrations of the NDE techniques on mock-ups containing calibration reflectors and laboratory-induced fatigue cracks; observing set-up and operation of the inspection equipment; review of inspection data during data collection; observing and reviewing analysis of all the NDE data, and review of the final reports. The UT inspections were performed to the 1989 Edition of the ASME Section XI Code and NRC Regulatory Guide 1.150 Revision 1.

b. Observations and Findings

The current outage pre-repair inspections included VT, ET and UT techniques. The team observed portions of the automated ID UT and ET examinations, including data analysis, of the flawed “A” hot leg weld and the other five reactor coolant nozzle-to-pipe welds. In addition to assessing the adequacy of these UT inspections to detect and to

size flaws, the team reviewed inspection procedures for these inspections. Prior to the inspections, both the UT and ET procedures were demonstrated on mock-ups provided from several sources. These demonstrations were conducted on weld profiles that had similarities to the welds in V. C. Summer using a number of laboratory-induced fatigue cracks in both the axial and circumferential directions. These demonstrations showed that sound energy was getting to the required inspection zones, that the sound energy was returning and could be used to detect and to size the flaws. This provided confidence that the procedures worked and sensitive inspections would be conducted and that if any structurally significant flaws existed in these dissimilar metal welds, they would probably be detected and sized.

The UT inspections were conducted from the ID of the weld by WesDyne, the vendor that conducted the ISI in 1993. There were a number of improvements for the UT inspections performed during the current inspection versus the ISI in 1993. The tool to deploy the end effector was a robotic arm that was controlled by a software system. The as-constructed vessel specifics were entered into the computer, and when the tool was deployed, the tool did a calibration to adjust the software to ensure that at all times the tool position was known to a high level of accuracy. This hardware system and software package was state-of-the-art and was observed to work very smoothly and reliably. The data acquisition and analysis system has evolved and is now called Paragon. This system is faster than the older version and offers improved data processing options. One of the more significant improvements in the system was the design of a new transducer sled. The new design employs articulating ball housings to achieve independent gimbaling for each transducer so that each transducer can accommodate both vertical and angular misalignments. This new design worked very well as evidenced by the data quality. The location in the “A” hot leg weld where the transducer sled was decoupled in the 1993 inspections (see section 1R2 above for details) was not a problem for the new sled. This does not mean that there were no problems. On one of the cold leg welds where there had apparently been an extensive amount of ID rework, it was observed that there were a number of locations where decoupling still occurred. The extent of the decoupling was quantified and reported in the final NDE report. The report states that for all three of the outlet (hot leg) nozzle-to-pipe welds, 100% good coupling was achieved for all four scan directions. However, for the inlet (cold leg) nozzle-to-pipe welds, 100% coverage was achieved for three directions and coverages of 88% for Loop A, 89% for Loop B and 94% for Loop C were achieved for axial scanning looking out of the vessel. This meets ASME Section XI Code requirements of greater than 90% overall coverage.

The ID inspections included the use of a color video camera mounted to the end effector deployed for conducting the UT examinations. The video camera had one light leading and one trailing the camera and the camera looked at an area 90 degrees in front of where the UT sled was positioned. The video camera could be zoomed and during the scanning several zoom levels were employed. The magnification was not quantified so these were un-calibrated visual inspections. Images from the camera showed the axial crack identified by UT and ET and in particular where there had been leakage of some kind of corrosion products from the crack. Overall, the video camera was very useful in providing guidance on the ID conditions of the welds and images of the axial crack that leaked.

In the UT axial scans inspecting for circumferentially oriented flaws, there were no significant UT signals that identified the presence of a flaw. In the circumferential scans for axial flaws there was a very large signal that occurred at about 9 degrees from top dead center (TDC) of the weld, clockwise when looking out of the vessel. This indication was detected with the first sled that contained dual 70-degree longitudinal transducers. These transducers can detect the presence of a flaw from about 0.05" - 0.1" below the inspection surface to a depth of about 1" - 1.25" below the inspection surface. The UT images showed that the signals were being returned all along the length of the flaw over the depth zone that the transducer covered. Furthermore the clockwise and the counterclockwise scans showed similar results. The next sled contained 37 degree dual longitudinal transducers, which produced responses from 1" below the ID surface all the way to the OD. The signals from these transducers showed responses over the entire length of the axial indication and the responses tended to show scattering from facets of the indication along its entire depth. Because of the corner trap effect, it was not possible to tell if the indication went through-wall or not, but it was apparent that the indication was nearly through-wall. The third and final sled contained two 45 degree dual longitudinal transducers looking circumferentially for axial flaws. These were operating at 4 MHz and were designed to provide information about the overlap zone around a depth of 1". It was recognized that the 4 MHz would be severely attenuated in the coarse grained weld material but it was hoped that the higher frequency, if it successfully penetrated the weld metal, would provide improved resolution information about any indications that were in its coverage zone. The noise level was very high for the 45 degree scans and the flaw indication identified with the 70 degree and 37 degree transducers was not seen.

Because initial PT testing indicated the potential of an OD circumferential crack, the UT transducers were selected based in part on optimizing inspection performance for circumferentially oriented cracks. The UT transducers and articulated ball housing had large footprints (1.76" by 1.76") which created coupling problems for some of the ID surface conditions. However, the UT detected the through-wall leaking axial crack with high signal to noise ratio and provided useful images showing the crack was nearly through-wall over most of it length.

The third sled also contained two ET plus point probes that were being used for the first time in the USA for this kind of application. ET is extremely sensitive to any surface breaking flaws. However, ET cannot indicate the depth of the flaws. The WesDyne staff thought that the ET tests might be vulnerable to false calls caused by unexpected changes in permeability and conductivity. Consequently, two ET test frequencies were used to assist in discriminating these conditions. In addition to identification of the near through-wall axial crack in the “A” hot leg weld, the ET examinations identified a number of other indications in the “A” hot leg weld and four of the other five nozzle-to-pipe welds. These indications all had very similar ET signal properties in all five welds except for the through-wall leaker. These indications were from 0.25" to about 1" in length. For this reason and because destructive examination of the “A” hot leg weld showed a number of the ET indications in the weld to be PWSCC, some of the ET indications in the other four welds are likely PWSCC also.

Although the use of ET for inspecting the inside of large diameter pipes had not been employed in the this country before, the ET results were considered to be quite good for the following reasons. This technology was successful in part because the ET probe is very small and did an excellent job of tracking and thus, maintaining good coupling for the irregular inside surface conditions in these DMWs. Also, ET is very sensitive to surface breaking flaws. There are currently no regulatory requirements for use of ET in this application, nor standards to apply.

There were some problems with decoupling of the UT probe because of adverse ID conditions (irregular surfaces due to ID grinding, counterbore, weld root and offset between the nozzle and the pipe) and the inability of the UT transducer housing to accommodate these conditions in the weld and buttering of the DMWs. As noted in the WesDyne final report, only one circumferential scan line would have detected many of the smaller cracks located in the weld/buttering area. The team noted that in order to better accommodate these kinds of conditions, smaller transducer and mounting assemblies would be needed to allow more effective tracking of the ID surface conditions and consequently provide a more reliable examination for small cracks.

c. Conclusions

Overall, the team considered examinations performed on the nozzle-to-pipe welds during the current outage were high quality. They were successfully demonstrated on a mockup (although the mockup did not simulate the adverse ID conditions on the V. C. Summer welds), performed with qualified personnel, and used state-of-the-art NDE equipment and procedures. The use of redundant and complementary NDE techniques (VT, UT and ET) and the successful demonstrations of these technologies provided confidence that a sensitive inspection was conducted. Destructive testing on the “A” hot leg weld supported this position. The examinations exceeded the minimum requirements of the ASME Section XI Code and NRC Regulatory Guide 1.150. UT identified an axial indication about 2-1/2" long, which was verified by ET and VT. ET inspection detected several additional small indications in the “A” hot leg weld and indications in four of the other nozzle-to-pipe welds, which were not identified by UT. An inspection conducted to minimum Code requirements (ultrasonic testing using the 70-degree longitudinal transducer) would have detected the through-wall leaking axial crack, but would not have detected other small cracks detected by non-Code required ET.

1R4 Metallurgical Evaluation

a. Inspection Scope

To support the root cause determination, the licensee removed a spool piece about 12 inches long from the reactor pressure vessel loop “A“ hot leg piping for metallurgical analysis. The spool piece contained the cracked nozzle-to-pipe weld, approximately eight inches of hot leg reactor coolant piping, and one inch of nozzle material. The team reviewed/observed the metallurgical analysis performed by Westinghouse of the flawed “A” hot leg nozzle-to-pipe weld to verify that the analysis was adequate to identify the cause and extent of cracking in the weld. The team reviewed the weld removal and

sectioning plan for the spool piece. The plan was designed to remove the cracked weld and capture all the ET indications for metallurgical investigations.

Team members observed in-process NDE and metallurgical analysis activities at the Westinghouse facility at Spartanburg, South Carolina and the hot cell laboratory at the Westinghouse Energy Center in Monroeville, PA. At the Energy Center team members were briefed by the Westinghouse personnel regarding the results and status of on-going examinations. At the Westinghouse hot cell laboratory, team members examined a number of metallurgical samples to verify the extent of cracking, microstructure, material chemistry and cracking morphology.

The team reviewed Westinghouse's initial hot cell investigation report, report up-dates and final report, WCAP-15616 Metallurgical Investigation of Cracking in the Reactor Vessel Alpha Loop Hot Leg Nozzle-to-pipe Weld at the V. C. Summer Nuclear Generating Station, to verify the analysis was adequate to accurately characterize the flaw.

b. Observations and Findings

The destructive examinations confirmed that the spool piece captured the main 2-1/2" axial crack as detected by UT, as well as other indications detected by ET testing (see paragraph 1R3). At the Westinghouse Spartanburg facility, the 12-inch long spool piece was decontaminated and reduced to about a six inch wide spool piece containing the affected weld for preliminary examinations. The six inch wide spool piece containing the weld and base material on both sides was sectioned into 14 segments for further metallurgical evaluation. A detailed metallurgical evaluation of the flawed weld was performed at the Westinghouse hot cell facility. The metallurgical evaluation was documented in Westinghouse WCAP-15616 and consisted of metallographic examinations, fractographic examinations, chemistry evaluations, hardness measurements and residual stress measurements.

The main 2-1/2" axial crack detected by the field UT examination was confirmed by destructive examination. The size of the axial crack was about 2.5 inches in length with a depth of about 2.5 inches. The through-wall portion of the axial crack was very small as it only broke through the outside surface at the "weep hole" location. The "weep hole" showed the characteristics of steam cutting and erosion. The main crack extended axially from the nozzle to butter (Alloy 182) interface, through the Alloy 82 weld and into the heat affected zone of the stainless steel piping. The fractographic examination of this crack showed features of multiple crack initiation sites. The inside diameter surface of the weld and adjacent material showed evidence of significant grinding and machining.

Additional indications were identified by ET examination. Some of the ET indications were confirmed as cracks and some could not be confirmed. One of the indications was confirmed to be a circumferential crack intersecting the main axial crack. Axial UT scans had suggested a circumferential component of the main axial crack. The circumferential crack had a length of about 2 inches and a depth of 0.2 inch and was located in the Alloy 182 clad adjacent to the stainless steel cladding with the depth limited to the thickness of the clad. The crack depth arrested at the interface of the clad

and the alloy steel nozzle. One end of this circumferential crack showed the appearance of turning into the axial orientation. Another ET indication was determined to be sub-surface, resulting from an iron rich inclusion.

In addition to the through-wall axial crack and the intersecting circumferential crack, four other ET indications were confirmed to be PWSCC by metallurgical examinations. The sizes (length and depth) of these crack indications were destructively measured. The UT had detected only the one through-wall axial crack with the suggestion of a circumferential component and missed the other four ET indications. The deepest crack identified by ET and missed by UT is an axial crack with a depth of 0.615 inch and a length of 0.75 inch. There are significant uncertainties associated with the crack length measured by ET. Based on the crack length determined by destructive examinations, the ET undersized the deep cracks with lengths over 0.5 inch by as much as 60%. However, ET oversized the shallow cracks with lengths 0.5 inch and less by as much as 66%.

The initial PT inspection performed in the field on the OD surface of the subject weld had identified a circumferential indication about four inches in length. This indication was located at the fusion line between the weld butter and the nozzle base material. Subsequent PT inspection revealed similar circumferential PT indications in other circumferential locations around the weld at the butter to nozzle base material fusion line. Metallurgical investigation in the hot cell laboratory confirmed that the observed PT indications were not cracks, but associated with a shallow lap at the interface of the weld butter and nozzle base material and boric acid (from the leak) attacking the low alloy steel at the interface.

Based on the cracking morphology, the observed cracking was determined to be PWSCC. The cracking was characterized by inter-dendritic and serrated cracking with localized branching.

Residual stress measurements by displacement measurement technique were performed during the first axial cut of the spool piece. The results showed the presence of a small tensile hoop stress, approximately 3 thousand pounds per square inch, in the weld. Local residual stresses were measured by the hole drilling technique using strain gages. The results showed that significant tensile residual stresses were present in the axial and hoop directions.

c. Conclusions

Based on the review as discussed above, the team concluded that the licencee's metallurgical evaluation of the cracked hot leg loop "A" nozzle-to-pipe weld, including the size measurements (length and depth) of the reported UT and ET indications by destructive examinations, was thorough. The licensee adequately characterized the failure mode to be PWSCC.

1R5 Root Cause and Extent of Condition

a. Inspection Scope

The team conducted a review to verify that the licensee's root cause investigation adequately determined the cause of the leak in the "A" hot leg nozzle-to-pipe weld and to verify that the extent of condition was adequately addressed. This review included information from the historical welding and NDE records, current repair welding and NDE activities, metallurgical examinations, etc., as described in the other sections of this report. In addition, the review included attendance at root cause investigation team meetings and discussions with members of the root cause investigation team. Documents reviewed included Root Cause Report C-00-1392, dated February 15, 2001, computer files of water chemistry data for the past operating cycle, and corrective action documents tracking current and future actions addressing the extent of condition.

b. Observations and Findings

Root Cause Investigation

The licensee's approach in determining the root cause of the leak involved identifying potential failure modes and then eliminating the failure modes by evaluating the supporting and refuting evidence. The root cause investigation was performed by a 10-person team: two licensee personnel, one as team lead and the other in a support function; two root cause analysis specialists who acted as mentors; and six industry experts in areas that included welding, nondestructive examination, failure analysis, structural analysis, and mechanical engineering.

The licensee's metallurgical examinations determined that the crack that caused the leak in the "A" hot leg nozzle-to-pipe weld was the result PWSCC. The conclusion was based on the crack morphology and the lack of any contaminants. The team concurred with this determination, based on their review of the photographs and data provided from the examinations.

The licensee's root cause investigation team identified 37 potential failure modes that could lead to the cracking and eliminated all but 7 through their evaluations. From the remaining seven failure modes, the licensee's team identified two root causes and three contributing factors for the cracking, which are summarized as follows:

Root Causes

- Extensive repairs during completion of the original "A" hot leg nozzle-to-pipe weld (weld repairs and grinding performed during construction were the only source available to provide the high stresses required to produce primary water stress corrosion cracking)

- The applicable welding codes, standards, and welding processes did not recognize or require consideration of the high residual stresses caused by multiple weld repairs and the associated grinding (there were and are no requirements to preclude repairs that can lead to the high residual stresses)

Contributing Factors

- hot cracking (micro-cracking that occurs during weld metal solidification or reheating and may not be detectable by NDE) in the Inconel weld material may have exacerbated the flaw growth (the metallurgical examination observed occasional hot cracking present in the progression of the crack)

- NDEs did not detect the flaws in the weld prior to the leak (although code requirement were met, surface contour, surface roughness, and UT detector parameters inhibited the ability to detect PWSCC)

- The difficulty of field welding versus shop welding with Inconel material and the fact that automatic welding operators were initially qualified for use on the construction welds using stainless steel materials instead of Inconel. (This is allowed by the Code, however this could have contributed to the need for extensive weld repairs, believed to have induced high residual stresses

Corrective action recommendations addressing the root cause and contributing factors were documented in the final investigation report dated February 15, 2001, and were tracked in corrective action documents such as PIP 0-C-00-1392.

The team found that the above causes and contributing factors were adequately supported by the licensee's design information, metallurgical examination data, historical and current NDE data, and the requirements applicable to the welding. The team noted that, since the mechanism that results in primary water stress corrosion cracking is not fully understood, there may be other factors that contributed to the initiation and progression of the cracking identified in the "A" hot leg nozzle-to-pipe weld.

Extent of Condition

The extent of the cracking in the "A" hot leg nozzle-to-pipe weld and potential cracking at other similar locations was established primarily through the metallurgical examination and NDE discussed in other sections of this report. The cracking in the "A" hot leg weld was repaired by removal and replacement of the weld. Flaw indications were found by ET inspection in four of the other hot and cold leg nozzle-to-pipe welds but were determined by analysis to be acceptable for continued operation. The destructively determined flaw sizes of the ET indications in loop "A" hot leg nozzle-to-pipe weld were used to support the crack growth calculations of the ET indications found in the hot leg loops "B" and "C" nozzle-to-pipe welds. Westinghouse performed an integrity evaluation for the licensee in WCAP-15615 to justify continued operation of V C Summer nuclear plant. The NRC staff reviewed the licensee's integrity evaluation and performed an independent bounding flaw growth calculation; the staff's review and analysis were documented in a safety evaluation dated February 20, 2001. The staff's safety evaluation concluded that V C Summer nuclear plant can be safely operated for one fuel

cycle (1.5 years) with the ET indications at the other nozzle-to-pipe welds not being removed.

As stated in licensee letter dated January 9, 2001, the licensee plans to re-inspect the more susceptible “B” and “C” hot leg welds at refueling outage 13, scheduled for the spring of 2002, using the best available UT inspection method. The letter further states that, in refueling outage 14, the licensee plans to perform the full 10-year ISI, again using the best available, approved methodology and techniques. This will include inspection of all hot leg and cold leg nozzle-to-pipe welds.

In addition to the assessment for future operations, the licensee performed an analysis to demonstrate a large margin existed between the as-found crack in the “A” hot leg nozzle-to-pipe weld and the critical flaw length (CFL). CFL is the length of the through-wall crack that would result in either slot or guillotine break of the pipe (depending on the orientation of the flaw), upon application of extreme loading conditions postulated for normal operation and accidents. The analysis was documented in the RV Nozzle Weld Repair Restart Report and Westinghouse WCAP-15615 and included calculation of the “best estimate” critical flaw length (CFL) and primary coolant leak rates associated with through-wall flaws in the “A” hot leg weld. The analysis found the CFL to be in excess of 30" axially and that a through-wall flaw length of 2.2" axially would result in a leak of 1 gpm. A leak rate of 1 gpm would be recognized by the plant operators and TS would require a shutdown. Based on the analysis and the actual length of the crack (2-1/2") versus the CFL, the licensee concluded the structural integrity of the RC system was maintained during all Technical Specification modes of operation, and that it would have been maintained under seismic, off-normal, or accident conditions.

Based on the large CFL and the fact that the crack in the “A” hot leg was axial with the length confined to the width of the weld (PWSCC could not grow out of the Alloy 182 weld material and heat effected zone), the team concluded that gross failure of the pipe would not have occurred. Also, based on the analysis showing that an axial through-wall crack of 2.2" would result in a leak of 1 gpm, had the full length of the 2-1/2" crack (as opposed to the 3/16" exit point) extended through-wall, the leak would have exceeded the TS limit of 1 gpm of unidentified leakage causing a plant shutdown. Therefore, the team found the licensee’s assessment provided reasonable assurance that structural integrity of the RC system, from an impending gross failure standpoint, was maintained during past operation.

c. <u>Conclusions</u>

The licensee’s root cause analysis was thorough and well-organized and was performed utilizing personnel with appropriate expertise. The root causes and contributory factors for the leak and the extent of condition were adequately determined. Actions have been established to address the root causes, contributing factors, and the extent of condition. The staff's safety evaluation concluded that V C Summer nuclear plant can be safely operated for one fuel cycle (1.5 years) with the ET indications in the other nozzle-to-pipe welds. The team found the licensee’s assessment provided reasonable assurance that structural integrity of the RC system, from an impending gross failure standpoint, was maintained during past operation.

1R6 Repair and Inspection Activities for New Pipe Spool Welds

a. Inspection Scope

The repair of the "A" hot leg nozzle-to-pipe weld consisted of complete removal and replacement of a pipe spool containing the cracked weld. The pipe spool was approximately one foot in length and contained the cracked weld, approximately one inch of nozzle base material, and approximately eight inches of attached stainless steel RC pipe. The pipe spool was replaced with a pipe spool approximately one foot in length. The team monitored the repair process, including observation of in-process welding and NDE activities, review of personnel (welder and NDE) qualification records, review of welding procedure specifications and qualification records, review of NDE procedures, and review of material and equipment certification records for the replacement pipe spool to verify compliance with ASME Code requirements. The applicable Codes for the repair were:

Welding - ASME Boiler & Pressure Vessel (B&PV) Code Section XI, 1989 Edition, no Addenda
ASME B&BP Code Section III, 1971 Edition, S73 Addenda
ASME Code Case N-432, Repair Welding Using Automatic Gas Tungsten-Arc Welding (GTAW) Temperbead Technique
ASME Code Case 2142-1, F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Material
ASME Code Case 2143-1, F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode

Temperbead Butter Repair - ASME B&PV Code, Section XI, 1992 Edition, no Addenda

NDE - ASME B&PV Code Section III, 1992 Edition, no Addenda

The following specific activities were reviewed/observed:

- PT examination of the nozzle "A" surface prior to "buttering" with Inconel weld material
- the start of the Inconel weld "buttering" process (including temperbead preheat temperature)
- PT and UT of the Inconel "buttering" deposited on reactor vessel nozzle "A"
- PT examination of the weld preparation of the installed "A" loop pipe
- preheat and post-weld heat treatment records for the temperbead weld butter process
- in-process welding activities for the Inconel nozzle-to-pipe weld and the stainless steel pipe-to-pipe weld
- final NDE (PT&RT) for the Inconel nozzle-to-pipe weld and the stainless steel pipe-to-pipe weld
- in-process welding activities for repair of the Inconel nozzle-to-pipe weld, including repair to the nozzle weld butter
- in-process and final NDE (RT&UT) for repair of the Inconel nozzle-to-pipe weld, including repair to the nozzle weld butter

The applicable welding and NDE procedures reviewed are listed in the Attachment 1 to this report.

b. Observations and Findings

As noted above, the licensee's repair plan for the crack in the "A" hot leg nozzle-to-pipe weld consisted of complete removal and replacement of the weld. This replacement process required weld buttering the nozzle end-prep with Inconel material and installation of a new stainless steel pipe spool. To complete the nozzle-to-pipe weld, the pipe end-prep was also weld buttered with Inconel resulting in the final weld being an Inconel to Inconel weld. The weld at the other end of the pipe spool was a stainless steel to stainless steel pipe weld. The weld design for both welds was a "narrow groove" joint design with the weld width of only about 5/8". The automatic GTAW process was used with Alloy 52 welding material used for the nozzle-to-pipe weld. Alloy 52 welding material for the GTAW welding process and Alloy 152 welding material for the SMAW process were replacement materials developed for better resistance to stress corrosion cracking.

Problems were encountered in welding the root pass of the nozzle-to-pipe weld. However, repairs were made to the root pass, including grinding irregular areas on the ID, and the welds were completed. The PT inspection of both original replacement welds was acceptable. Also, the RT inspection of the stainless to stainless weld was acceptable. However, RT inspection identified significant rejectable lack of fusion defects, as well as incomplete penetration of the root, in a large part of the circumference of the Inconel nozzle-to-pipe weld. The incomplete penetration was removed by grinding on the ID of the weld. No repair welding was performed on the ID. A 360° repair groove, centered on the nozzle sidewall of the original weld, was machined from the OD surface to a depth of 1-3/4". The repair groove was approximately twice the width of the original narrow groove configuration. The repair was made with the automatic GTAW process and RT inspections performed at ½" increments identified acceptable porosity and some lack of penetration still present on the ID of the weld. Additional grinding was performed on the ID to clear up the lack of penetration indications. Upon completion of the repair, RT inspection resulted in six defective areas of lack of fusion and/or porosity.

The licensee decided to repair these six areas by localized excavations and manual GTAW repair. Excavations 1 and 2 were relatively short cavities and were successfully completed using the manual GTAW process. However, repeated attempts at repair of cavity 3 (21" cavity) and cavities 4, 5, and 6 (which eventually resulted in a single cavity from 5:30 to 9:00 o'clock facing the vessel) were unsuccessful with the GTAW process. The welding process was switched to the SMAW process with Alloy 152 material and the two cavities were successfully completed. Although a short section of cavity 4, 5, and 6 was ground through to the ID, all repairs were made from OD of the weld with only grinding on the ID. The team found that repair welding and inspection of the repairs met Code requirements.

Based on the difficulty with producing acceptable welds with the Alloy 52 welding material, the licensee and their contractor investigated the reasons for the excessive lack of fusion and porosity produced in the weld. The investigation found that the higher

amounts of aluminum and titanium used as deoxidizers in the Alloy 52, and the affinity for nitrogen and oxygen of these two elements, resulted in higher than expected concentrations of oxides and nitrides. These non-metallic constituents resulted in what are believed to be fine stringers which RT examination identified as rejectable indications. It is thought that when welding over these deposits, oxygen and nitrogen can dissociate from the oxides and nitrides and produce porosity. The investigation found that industry experience with thick welds having these problems is just beginning to be recognized. Again, for the weld in question, all rejectable indications were removed and repaired to Code requirements.

Based on the root cause of the original crack being partially attributed to high residual stresses because of extensive repairs, the team questioned the effect on residual stresses and PWSCC of the multiple repairs on the new Inconel nozzle-to-pipe weld. This was evaluated by the licensee and their contractors with the conclusion that even though ID tensile stresses would be present in the area of local repairs, the stresses would be less severe than those in the original weld that cracked. They further concluded that, considering the improved resistance to cracking provided by the use of the more resistant weld material, Alloys 52 and 152, the margin to cracking should be more than sufficient to provide the necessary cracking resistance for component serviceability. One report concluded that the PWSCC phenomenon that cracked the original weld is avoided. This conclusion was based on the use of the Alloy 52/152 welding material and the lower ID tensile stresses. The team pointed out that since the phenomenon of PWSCC is not totally understood, further studies developing more quantitative data will be needed before the new weld can be considered immune to PWSCC.

c. Conclusions

The team found that all welding and NDE activities met Code requirements. However, for the nozzle-to-pipe DMW, the GTAW welding process with Alloy 52 material resulted in rejectable weld defects and a number of repair attempts before successful repair using the SMAW process with Alloy 152 welding material. The team agreed with the licensee's evaluation that the new weld with different, more resistant material and less ID tensile stress should be much more resistant to PWSCC. However, based on the fact that PWSCC is not totally understood, the team considered the further evaluations and inspections will be needed before it can be concluded that the new weld is immune to PWSCC.

1R7 Preservice Inspection (PSI) For New Pipe Spool Welds

a. Inspection Scope

The team reviewed/observed the PSI NDE inspections conducted on the replacement spool piece welds which included the pipe-to-pipe stainless steel weld and the nozzle-to-pipe DMW. This review included: review of the procedures employed; observation of PT inspections; observation of demonstrations of the UT techniques on mockups containing calibration reflectors and laboratory induced fatigue cracks; observation of the set up and operation of the UT inspection equipment; review of UT inspection data during data

collection; observation of UT data analysis; observing and reviewing all of the NDE data and review of the final NDE reports. The applicable Code for the PSI was ASME Boiler and Pressure Vessel Code Section XI, 1989 Edition with no Addenda.

b. Observations and Findings

The manual UT inspections conducted on the stainless steel pipe-to-pipe weld were performed using procedures and personnel qualified to the Performance Demonstration Initiative (PDI) Program to meet the requirements of ASME Section XI, Appendix VIII, 1995 Code Edition including the 1996 Addenda and as modified by Revision 1, Change 1, of the PDI Program Description. There was no need for demonstration of this procedure because the PDI program has been previously reviewed and endorsed by the NRC. No indications were detected when this weld was inspected. Based on observation of the inspections and because the requirements in Appendix VIII provide confidence that effective and reliable inspections would be performed, the inspection team concluded that the inspections were of high quality and acceptable.

The inspection of the nozzle-to-pipe DMW was conducted to a procedure in accordance with IWA-2240 for alternative examinations of the 1989 Edition of ASME Section XI Code. Inspections of the DMW were more challenging than the stainless steel pipe-to-pipe inspections. There is no PDI program implemented at this time for these welds. Therefore, the inspection procedure for the DMW was demonstrated on a mock up provided by the EPRI that contained fatigue cracks. This demonstration showed that performing an automated inspection of the DMW from the OD surface of the pipe and nozzle can detect the fatigue cracks in the mockup that are circumferentially oriented with good sensitivity and acceptable signal to noise ratio. However, the automated inspection was not able to reliably detect the small axially oriented cracks. Because of the ability to skew the UT transducer, manual inspection provides better detection performance than the automated procedure. Thus, the automated inspection was augmented with a manual inspection for axially oriented flaws. Since DMWs are coarse grained material, mostly longitudinal wave transducers were used operating at 1 MHz.

Because of geometrical constraints on the nozzle side of the weld, several transducers were required to provide the needed coverage of the inspection volume. The inspection exceeded the ASME Section XI Code requirement of 90% coverage with an aggregate of 98.6% coverage of required volume.

The inspections of the DMW produced only one indication that required analysis to disposition. This indication did not possess the UT signal properties of the fatigue cracks that had been demonstrated using the EPRI DMW mockup. The analysis concluded that the indication was a result of acoustic beam redirection in the coarse grained Inconel weld and the ground counterbore region and therefore acceptable. The inspection team agreed that this was the most likely interpretation of the UT signals.

The observed PT inspections were well performed in accordance with approved procedures by qualified NDE personnel in accordance with ASME Code requirements.

In addition to the PSI reviews, the team reviewed the ASME Section III radiographs for the nozzle-to-pipe DMW weld and the stainless steel pipe-to-pipe weld. The radiographs exhibited good resolution and sensitivity that met ASME Section III Code requirements. The team did not identify any rejectable indications, which agreed with the licensee's interpretation.

c. Conclusions

The inspection team concluded that a comprehensive and effective inspection that met or exceeded the minimum requirements of the ASME Code was conducted on both the nozzle-to-pipe DMW and the stainless steel pipe-to-pipe weld.

1R8 Leak Detection

a. Inspection Scope

The team reviewed the licensee's leak detection procedures and leakage detection capability to determine if current leak detection practices should have identified the "A" hot leg leak during plant operation. The review included:

- descriptions of the leak detection practices and requirements provided in the Final Safety Analysis Report (FSAR) and Technical Specifications (TS)
- recorded leakage data
- calibration procedures and data
- documented examples of previously identified leaks
- discussions with cognizant plant personnel
- photographs and videotapes of areas examined for evidence of the "A" hot leg leak, including the incore instrument sump

The procedures, recorded data, and examples of previously identified leaks that were reviewed are listed in the Attachment 1 to this report.

b. Observations and Findings

The licensee measured a maximum of 0.3 gpm of unidentified reactor coolant leakage during the operating cycle through a water inventory balance. This leakage may have included reactor coolant from the through-wall crack in the "A" hot leg. The unidentified leak rate at the beginning of the cycle was approximately 0.1 gpm, increased to the 0.3 gpm value during the first three months of the cycle, and remained relatively constant thereafter. Prior to shutdown there was no reason to suspect that this leakage was due to a crack in the reactor coolant pressure boundary or to investigate the possible source(s) of the leakage. The unidentified leak rate was not unusual and well below the TS limit of 1 gpm.

Six leak detection practices were identified in the FSAR as having the capability to detect small (1 gpm or less) reactor coolant pressure boundary leaks. These practices are listed below, followed by the team's findings regarding their capabilities:

- reactor coolant system water inventory balance monitoring
- reactor building (RB) sump inventory monitoring
- RB Cooling Unit (RBCU) condensate flow rate
- RB atmosphere particulate radioactivity monitoring
- RB atmosphere gaseous radioactivity monitoring
- incore instrument sump level monitoring

The reactor coolant system water inventory balance was capable of detecting leaks of about 0.25 gpm and possibly less. The inventory balance was recorded and monitored at least every 72 hours for compliance with TS 4.4.6.2.1, which required verification that unidentified reactor coolant system leakage did not exceed 1 gpm.

The sump level monitoring was capable of detecting leaks of about 0.25 gpm and possibly less. The results were trended on the plant computer and were assessed every 12 hours for compliance with a TS 4.4.6.2.1, which required verification that unidentified reactor coolant system leakage did not exceed 1 gpm. The sump leak rate was set to alarm at less than the 1 gpm limit.

RBCU condensate flow rate was not recorded or trended but was set to alarm at 0.5 gpm. The team considered the flow rate capable of detecting leakage of about 0.5 gpm. Technical Specification 4.4.6.2.1 required the use of the RBCU in verifying that unidentified reactor coolant system leakage did not exceed 1 gpm.

Neither the RB atmosphere particulate radioactivity monitoring nor the RB atmosphere gaseous radioactivity monitoring provided the leak detection capability originally intended, the capability to detect unidentified reactor coolant leakage at the 1 gpm level within 1 hour. The radiation protection manager stated that this was due to the relative absence of reactor fuel leaks, resulting in low concentrations of radioactive material in the reactor coolant. The radiation alarms were set at twice background, and not based upon leak rate. The team was made aware that the NRC has taken a position on this issue with another plant with a similar condition as documented in internal correspondence dated July 30, 1998, Safety Assessment of Region II Concerns Regarding Discrepancies of Containment Radiation Monitor Sensitivities at St. Lucie Units 1 and 2, and Turkey Point Units 3 and 4. The NRC reviewed this issue and determined that although the system is not consistent with the recommendations in Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, sensitivity response time discrepancies of the monitors is of low safety significance because of other methods to detect leakage. Additionally the NRC has ongoing efforts to establish guidance in leak-before-break technology, which is expected to result in a review of Regulatory Guide 1.45 for updates to address inconsistencies in assumptions regarding reactor coolant radioactivity concentrations and radiation monitor sensitivity response times.

The incore instrument sump could potentially receive reactor water leaked from cracks in the reactor coolant pressure boundary near the reactor vessel. Had it been of sufficient size the “A” hot leg leak might have partially accumulated in this sump. A videotaped licensee inspection showed that there was no water in the sump as a result of the “A” hot leg leak.

The licensee did not use containment pressure, temperature, or humidity measurements in detection of reactor coolant leaks. The team reviewed containment pressure and temperature data recorded by the licensee and found that it would only be useful in detecting gross leaks.

In response to NRC questions documented in a December 22, 2000 letter, the licensee's response (documented in letters dated December 29, 2000, and February 9, 2001) detailed the following enhancements planned for their leak detection procedures:

- noble gas sampling and analysis to provide additional verification of RC system integrity

- calculation of RC system water inventory balance daily (when plant conditions allow collection of accurate and meaningful data), retention of this information for trending, and investigation at or before 0.8 gpm

- addition of a main control board annunciator to alarm at a sump in-leakage of 0.75 gpm, such that operators will be alerted prior to the leakage exceeding TS limits

- enhancement of boric acid inspections

Although it is not possible to determine when the "A" hot leg nozzle-to-pipe weld leak began, it existed for at least several months while the reactor was at full power. This is based upon the quantity of boric acid residue, estimated to be in excess of 100 pounds, and the size of the leak path. Technical Specification Limiting Condition for Operation (LCO) 3.4.6.2.a requires that reactor coolant system leakage shall be limited to "No PRESSURE BOUNDARY LEAKAGE," when in Modes 1, 2, 3 and 4. The associated action statement requires that with any pressure boundary leakage, be in Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours. The boric acid residue (>100 pounds) was discovered at 9:30 a.m. on 10/7 (plant in Mode 3) and the unit was taken from Mode 4 to Mode 5 at 4:40 a.m. on 10/8. Thus, the action statement to be out of Modes 1, 2, 3 and 4 was met from the time the boric acid residue was discovered (also note that the leakage was not determined to be pressure boundary leakage until 10/12). The pressure boundary leakage existed for greater than the required action to be in Hot Standby within 6 hours and this constitutes a violation of the TS.

However, based on the fact that: (1) there was no previous history of PWSCC in RC loop welds; (2) applicable construction and inspection ASME Code requirements were complied with; and (3) with the installed leakage detection equipment, the licensee could not have reasonably known that pressure boundary leakage occurred during the operating cycle; the licensee could not have reasonably been expected to identify the crack or pressure boundary leakage. Therefore, since the violation resulted from matters not within the licensee's control, such as equipment failure not avoidable by reasonable quality assurance measures or management controls, discretion is exercised in accordance with section VII.B.6 of the NRC Enforcement Policy and a notice of violation will not be issued.

c. Conclusions

The team concluded that the licensee's leak detection practices would not have been expected to identify the "A" hot leg leak during plant operation. Although the licensee measured 0.3 gpm of unidentified reactor coolant leakage during the operating cycle through a water inventory balance, some of which may have been due to the through-wall crack , there was no reason to suspect that this leakage was due to a crack in the reactor coolant pressure boundary or to investigate the possible source(s) of the leakage. The leak rate was well below the TS limit of 1 gpm for unidentified leakage. The TS limit of no pressure boundary leakage was violated. However, based on the fact that the violation resulted from matters not within the licensee's control, discretion was used and no notice of violation will be issued. The licensee plans a number of enhancements to their RC system leakage detection practices.

1R9 Boric Acid Corrosion Inspections (Boron Walkdown Inspection)

a. Inspection Scope

The licensee performs a boric acid walkdown each outage when the containment is opened. The team reviewed the licensee's plan/procedure and records for these walkdowns to determine their scope and thoroughness. The review included the scope of the program as defined in Preventive Test Procedure PTP-151.001, Revision 2, and the results of the inspections performed during the last two refueling outages (RO-10: October 1997, RO-11: April 1999). The team also reviewed procedure HPP-402, Revision 10, "Radiological Survey Requirements and Controls for Reactor Building and Incore Pit Entries," pertaining to the boric acid leakage inspection performed during pre-entry survey for the current outage (RO-12). Samples of the pre-entry radiological survey results documented in the records of procedure HPP-401, "Issue, Termination, and Use of RWPs and SRWPs," were also reviewed. This survey identified the large accumulation of boric acid residue on the reactor building floor that led to the discovery of "A" hot leg reactor coolant leak.

b. Observations and Findings

Generic Letter (GL) 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," provides guidelines to licensees of PWR plants for implementing a program to identify small reactor coolant leakage (smaller than technical specification limits) to prevent boric acid buildup on the reactor coolant pressure boundary components. The GL was issued due to concern that the buildup of boric acid on carbon steel piping and components would cause corrosion degradation of carbon steel materials. The licensee responded to the GL by letter CGE-89-519 dated February 10, 1989, and subsequently implemented a program in PTP-151.001. The program as defined in PTP-151.001 consisted of two phases of inspections. The first phase was a walkdown inspection for any evidence of leakage without removal of protective covers, shields, or insulation prior to the inspection. This inspection was to be started with reactor coolant system pressure at 2235 psig and completed prior to depressurizing below 350 psig. The first phase of inspection consists of inspection of the following components and their adjacent areas for boric acid residues: (1) reactor

vessel head area including head ventilation ductwork, (2) reactor coolant pumps, (3) steam generator manways, (4) pressurizer spray valves, (5) pressurizer safety valves, (6) pressurizer manways and (7) general areas of all elevations. The second phase of inspection was to be performed after cool down and depressurization and after removal of insulation which would prevent close inspection. This second phase of inspection was focused on the locations of mechanical connections for any indication of boric acid induced corrosion and consisted of the inspection of the following seventeen components and their individual parts: (1) two pressurizer spray valves, (2) three reactor coolant pumps, (3) reactor vessel, (4) three steam generators, (5) pressurizer assembly, (6) three pressurizer safety valves, (7) two reactor vessel head vent isolation valves and (8) two reactor vessel head vent valves.

The first phase of the boric acid walkdown inspections in RO-10 and RO-11 identified boric acid residue at a number of locations. Most of this residue was due to leakage from flanges, seals, manways, fittings, packing and test connections of various components. Maintenance work requests (MWRs) were written for each observed condition to further evaluate the significance of any corrosion and implement appropriate corrective actions. The second phase of inspection per PTP-151.001 was performed using Surveillance Test Procedure STP-250.001," Reactor Coolant System Leak Test." The results of the second inspection performed in RO-10 and RO-11 were all satisfactory with no boric acid residue identified on any of the components inspected.

The team reviewed the pre-entry radiological survey records (HPP-401) for the current refueling outage (RO-12) and verified that the survey had identified the large accumulation of boron acid residue on the reactor building floor in areas adjacent to RC system "A" hot leg nozzle weld. However, the records did not show any significant increase in the background radiation and the contamination measured on smear samples taken in this area was not high.

Based on discussions with licensee personnel, the team found that the licensee had deleted the PTP-151.001 program in May 2000 and incorporated the first phase of the boric acid leakage inspection requirements into Health Physics Procedure HPP-402. The second phase of the program continued to be inspected through the Surveillance Test Procedure STP-0250.001. However, the team noted that the first phase detailed inspection guidelines from Procedure PTP-150.001 were not completely incorporated into HPP-402. The lack of inspection guidelines could impact the capability of the program to identify the components contaminated with boric acid residue.

This review found that the boric acid corrosion inspections performed in the last two refueling outages (April 1999 and October 1997) were adequately performed and the scope of the licensee's inspection program met the intent of GL 88-05. However, with the finding of PWSCC in a reactor vessel nozzle-to-pipe weld and the lack of detailed inspection guidelines in the existing program, as discussed above, the team considered that the boric acid corrosion inspection program should be enhanced to improve guidelines for early detection of reactor coolant leakage and to expand the scope of inspection to include the welds that are susceptible to PWSCC. In response to questions by the NRC, the licensee stated in their letter dated December 29, 2000, that their boric acid inspection procedures will be enhanced to provide additional detail for the inspection and evaluation of RCS leakage, and that specific components and

locations to be inspected will be listed and guidance provided on methodologies for evaluation.

c. Conclusions

The boric acid corrosion inspections performed in the last two refueling outages (April 1999 and October 1997) were adequately performed, and the pre-entry radiological survey for RO-12 identified the large accumulation of boric acid crystals on the reactor building floor that led to the discovery of “A” hot leg reactor coolant leak. However, since some details had been deleted from the program, the team considered that the boric acid corrosion inspection program should be enhanced to improve guidelines for early detection of reactor coolant leakage and to expand the scope of inspection to include the welds that are susceptible to PWSCC. By letter dated December 29, 2000, the licensee stated that their boric acid inspection procedures will be enhanced to provide additional detail for the inspection and evaluation of RC leakage, and that specific components and locations to be inspected will be listed and guidance provided on methodologies for evaluation.

4OA3 Event Followup

(Closed) LER No. 2000-008-00, Reactor Coolant System Pressure Boundary Degradation

a. Inspection Scope

On October 7, 2000, the licensee identified an accumulation of boric acid residue near the “A” loop of the RV. Subsequent investigation found a through-wall leak at the “A” hot leg RV nozzle-to-pipe weld resulting from PWSCC in the weld. The team reviewed circumstances related to this event. The team reviewed the licensee’s corrective actions, including: the root cause analysis, metallurgical analysis, original construction records, previous inspection records, leak detection records, and repair activities to verify the appropriate corrective actions were taken.

b. Findings and Observations

On October 7, 2000, plant personnel identified an accumulation of approximately 100 pounds of boric acid in the “A” loop hot leg area of the Reactor Vessel (RV). Boric acid and insulation was removed from the area of the suspected leak path to allow for further inspection. Subsequent licensee inspections and analysis determined that the leak was from a through-wall crack in the “A” loop hot leg nozzle-to-pipe weld. The corrective actions included a root cause evaluation, repair of the weld, and restart justification. Repair was accomplished by complete removal and replacement of the weld. The root cause analysis determined that the leak was the result of PWSCC.

A special inspection team was chartered and performed a detailed inspection of activities associated with analysis and repair of the reactor coolant leak. The team inspection activities are detailed in the above sections of this report.

c. Conclusions

The licensee's root cause analysis, extent of condition determinations, and corrective actions were appropriate. No performance deficiencies were identified. This LER is closed.

4OA6 Management Meetings

Exit Meeting Summary

The team presented the inspection results to Mr. S. Byrne and other members of the licensee's staff on February 15, 2001. The inspection results were further discussed in a telephone conversation on March 13, 2001, between Mr. Richard Emch, Acting Deputy Director, Division of Reactor Safety, Region II, and other RII staff members including the team leader, and Mr. Gary Williams and members of the licensee staff. The licensee acknowledged the findings presented.

The team asked the licensee whether any of the material examined during the inspection should be considered proprietary. Although proprietary information was reviewed, no proprietary information is included in the report.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

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F. Bacon, Manager, Chemistry Services
L. Blue, Manager, Health Physics and Radwaste
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B. Williams, General Manager, Nuclear Plant Operations
G. Williams, Manager, Maintenance Services

Contractors and Others

W. Alford, WesDyne Field Lead
W. Bamford, Westinghouse, Fellow Engineer
C. Cheezem, DESI - Business and Technical Support Manager
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J. Derrico, G. A. Grace, Inc. - Senior Consultant
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R. Emch, Acting DRS Deputy Director
M. King, Resident Inspector
M. Widmann, Senior Resident Inspector
K. Wichman, NRR

LIST OF DOCUMENTS REVIEWED

Licensee Event Report (LER 2000-008-00) Reactor Coolant System Pressure Boundary Degradation dated November 10, 2000

Preventive Test Procedure PTP-151.001, Revision 2, “Inspection for Boric Acid Corrosion.”

Boric acid corrosion inspection (PTP-151.001) results for refueling outages #10 (October 1997) and #11 (October 1997).

Health Physics Procedure HPP-402 “Radiological Survey Requirements and Controls for Procedure HPP-401 “Issue, Termination, and Use of RWPs and SRWPs,” initial Entry Survey results for Refueling outages #11 and #12.

Generic Letter (GL) 88-05 “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” dated March 17, 1988.

Letter from D. A. Naoman (SCE&G) to USNRC Document Control Desk, dated June 3 1988, providing South Carolina Electric & Gas Company (SCE&G) response to GL 88-05.

Procedure OAP-106.1, Operating Logs, Revision 8 (reviewed regarding monitoring RB sump)

Framatome Welding Procedure Specification 55-WP8/8/F6NG3-05, Revision 05, Machine-Gas Tungsten Arc Welding (GTAW)

Framatome Qualification Record PQ7032-02

Framatome Welding Procedure Specification 55-WP43/43/F43NG3-00, Revision 00, Machine-Gas Tungsten Arc Welding (GTAW)

Framatome Qualification Record 55-PQ7065-02

Framatome Welding Procedure Specification 55-WP3/F43TB2-06, Revision 06, SMAW Temperbead Welding

Framatome Qualification Record 55-PQ7000-04

Framatome Welding Procedure Specification 55-WP43/43/F43AW-03, Revision 03, Manual Shielded Metal Arc Welding

Framatome Qualification Record 55-PQ7072-01

Framatome Narrow Groove Gas Tungsten Arc Welding (NG-GTAW) V. C. Summer “A” Hot Leg Repair Operating Instruction

Framatome General Welding Procedure - 1 (GWP-1), Revision 04

Framatome Welding Control Procedure - 3 (WCP-3), Revision 05

Framatome Welding Procedure Specification 55-WP3/F43TB3-06, Machine Temperbead GTAW, Approved 8/26/99

Framatome Welding Procedure Specification 55-WP8/F43AW3-00, Machine Gas Tungsten Arc Welding, Approved 11/10/00

Framatome Procedure 54-PT-6-07, Visible Solvent Removable Liquid Penetrant Examination Procedure, Revision 07 with Change Authorization 001

Framatome Procedure 54-ISI-141-00, Procedure for Ultrasonic Examination of Weld Buttering, Approved 11/13/00

Initial report of preliminary results of Westinghouse's Investigation of V. C. Summer alpha loop hot leg nozzle-to-pipe weld cracking, dated November 28, 2000.

Updated Westinghouse investigation reports dated December 8, 2000 and December 19, 2000.

Final Westinghouse investigation report WCAP-15616, Revision 0, "Metallurgical Investigation of Cracking in the Reactor Vessel Alpha Loop Hot Leg Nozzle-to-pipe Weld at the V. C. Summer Nuclear Generating Station," January 2001.

WCAP 15615, "Integrity Evaluation for Future Operation: Virgil C. Summer Nuclear Plant Reactor Vessel Nozzle-to-pipe Weld Regions," Revision 0, dated December 2000 and Revision 1, dated December 26, 2000.

V. C. Summer Annunciator Response Procedure ARP-001-XCP-606, Rev. 5, Setpoints:
- RBCU 1A/2A Drain Flow High, 0.5 gpm
- RB Sump High Level, 405'
- RB Sump > 10 gpm Leakage
- Incore Sump > 1 gpm Leakage

V. C. Summer Annunciator Response Procedure ARP-001-XCP-607, Rev. 5, Change A, Setpoints:
- RBCU 1B/2B Drain Flow High, 0.5 gpm
- RB Sump High Level, 405'
- RB Sump > 10 gpm Leakage

V. C. Summer Annunciator Response Procedure ARP-001-XCP-615, Rev. 5, Setpoint:
- RB Sump calculated fifteen minute running average, 1 gpm

V. C. Summer Annunciator Response Procedure ARP-001-XCP-631, Rev. 5, Change A, Setpoint:
- Incore Sump Level High > 387'5"

V. C. Summer Procedure HPP-904, Use of the Radiation Monitoring System, Rev. 9, Change A

V. C. Summer Procedure STP-114.001, Operational Leakage to Reactor Coolant Pump Seals, Rev. 5

V. C. Summer Procedure STP-114.002, Operational Leakage Test, Rev. 10

V. C. Summer Procedure STP-114.003, RCS Leak Detection Setpoint Determination, Rev. 1

Licensee Event Report 94-006, Reactor Coolant Pressure Boundary Leakage

Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage, May 1973

Design Basis Document; Drains, Sumps, and Leak Detection (ND); Rev. 2

RB atmosphere gaseous and particulate radioactivity plot for 11/1/1997 - 10/14/2000

V. C. Summer Procedure STP-0360.034, Operational Test for Gas and Particulate, performed 12/4/00

V. C. Summer Procedure STP-0360.033, Loop Calibration, gas and particulate channels, performed 8/6/00

RB sump level leak rate plot for 11/1/1997 - 10/6/2000

V. C. Summer Procedure STP-0342.003, RB Cooling Unit Condensate Flow Determination, performed 10/9/00

PIP 0-C-00-0666, RB sump alarm at 1 gpm - temperature change increased condensation rate

PIP 0-C-99-1046, Reactor coolant inventory balance detected about 0.2 gpm vent valve leak

PIP 0-C-99-1254, Leakage into RB sump increased by over 0.1 gpm

V. C. Summer Procedure PTP-175.001, RCS Supplementary Leakage Assessment, Rev. 0

Reactor Coolant System unidentified leak rate plot for 5/1/97 - 10/1/00

Dynamic Test of Reactor Coolant System Leakage Program, test report dated August 2, 1989

V. C. Summer Procedure ICP0300.075, Calibration, incore sump leakage alarm level switch, performed 11/97

Containment temperature plot for 5/1/99 - 10/6/00

Containment pressure plot for 5/1/99 - 10/6/00

V. C. Summer Procedure STP0303.001, Reactor containment pressure XTMR test, performed 9/00

V. C. Summer Procedure STP0303.002, Reactor containment pressure XTMR test, performed 12/00

V. C. Summer Procedure STP0303.003, Reactor containment pressure XTMR test, performed 9/00

V. C. Summer Procedure STP0304.001, Reactor containment pressure XTMR test, performed 9/00

V. C. Summer Procedure STP0395.016, Reactor containment pressure XTMR calibration, performed 10/00

V. C. Summer Procedure STP0395.017, Reactor containment pressure XTMR calibration, performed 10/00

V. C. Summer Procedure STP0395.018, Reactor containment pressure XTMR calibration, performed 10/00

V. C. Summer Procedure STP0395.019, Reactor containment pressure XTMR calibration, performed 10/00

V. C. Summer Procedure STP0345.068, Loop Calibration - RB ambient temperature element, performed 6/00

V. C. Summer Procedure STP0300.007, RB sump A level transmitter calibration, performed 10/21/00

V. C. Summer Procedure STP0300.008, RB sump B level transmitter calibration, performed 10/21/00

PIP 0-C-00-1392, Crack in the weld between hot leg piping and reactor vessel nozzle

PIP 0-C-99-1310, AC power was inadvertently removed from ILS01973 and ILS01974

Letter from V. C. Summer to NRC dated January 9, 2001, Response to NRC Questions dated December 28, 2000

Letter from V. C. Summer to NRC dated January 16, 2001, Response to NRC Questions dated November 3, 2000

Letter from V. C. Summer to NRC dated January 9, 2001, Response to NRC Questions dated October 23, 2000

Letter from V. C. Summer to NRC dated December 29, 2000, Response to NRC Questions dated December 22, 2000

Root Cause Report C-00-1392, V. C. Summer Nuclear Station Root Cause Investigation - “A” Hot Leg Nozzle Weld Cracks, February 15, 2001 (Final Report)

Framatome Procedure Number 54-ISI-131-01, Revision 1, Ultrasonic Examination of Dissimilar Metal Piping Welds

Framatome Procedure Number 54-ISI-170-01, Revision 1, Automated Ultrasonic Examination of Dissimilar Metal Piping Welds

Framatome Procedure Number 54-ISI-141-00. Revision 0, Procedure for the Ultrasonic Examination of Weld Buttering

Framatome Procedure Number 54-ISI-112-9, Revision 9, Ultrasonic Examination for Thickness Measurement Using Pulse-Echo Techniques

Framatome Procedure Number 54-ISI-836-03, Revision 3, Procedure for the Ultrasonic Examination of Austenitic Piping Welds

Framatome Procedure Number 54-ISI-240-39, Revision 39, Visible Solvent Removable Liquid Penetrant Examination Procedure

Framatome Procedure Number 54-ISI-6-07, Revision 7, Visible Solvent Removable Liquid Penetrant Examination Procedure

SPEC Procedure Number SC-111, Revision 7, Radiographic Testing

V. C. Summer Report C-00-1392, Revision Initial Report, Root Cause Investigation “A” Hot Leg Nozzle Weld Cracks

V. C. Summer Letter Report RC-00-0377, Revision 0, Response to NRC Questions Dated December 22, 2000

V. C. Summer Letter Report RC-01-0010, Revision 0, Response to NRC Questions Dated December 28, 2000

WesDyne Final Report, Examination Report of the V. C. Summer Unit #1 Reactor Vessel Nozzle-to-pipe Welds for South Carolina Electric & Gas

WesDyne International Final Report Summary, Revision 0 (dated November 15, 2000), Reactor Vessel Outlet Nozzle-to-pipe and Inlet Nozzle to Elbow Welds

WesDyne International Eddy Current Examination Summary Report, Revision 2 (dated December 5, 2000), Reactor Vessel Outlet Nozzle-to-pipe and Inlet Nozzle to Elbow Welds

Westinghouse Report WCAP-15615, Revision 0, Integrity Evaluation for Future Operation Virgil C. Summer Nuclear Plant: Reactor Vessel Nozzle-to-pipe Weld Regions

EPRI Report PWRMRP-21, Revision 0, Crack Growth of Alloy 182 Weld Metal in PWR Environments

Westinghouse Field Service Procedure CGE-ISI-207-ET, Revision 0, Eddy Current Examination of Reactor Vessel Nozzle to Safe End Weld Inside Surfaces for V. C. Summer

Westinghouse NDE Procedure CGE-ISI-254, Revision 1 and Field Revision 01, Remote Inservice Examination of Reactor Vessel for V. C. Summer

WesDyne Program Plan, Revision 0, V. C. Summer Nuclear Power Plant Year 2000 Reactor Vessel Examination Program Plan

WesDyne Technical Procedure WCAL-002, Revision 1, Pulser/Receiver Linearity Procedure

Westinghouse Report MEM-MNA-520, Revision 0, Design Specification for Surry Units 1 and 2 RV Nozzle Safe-End Mock-Up and RV Safe-End Block Modification

Westinghouse Certification Package MB-11184-D, Revision 001, Certification Package of Materials for Surry Safe End Mock-Up

WesDyne Report, Revision 0, Results of Demonstration of Surface Equivalency Technique for Surry Reactor Vessel Inlet Nozzle to Safe Ends

Root Cause Weld History Summary

V. C. Summer RV Outlet Nozzle Weld Repair Restart Report

LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers
B&PV	Boiler & Pressure Vessel
CFL	Critical Flaw Length
DAC	Distance Amplitude Correction
DMW	Dissimilar Metal Weld
EPRI	Electric Power Research Institute
ET	Eddy Current Testing
FSAR	Final Safety Analysis Report
GL	Generic Letter
gpm	Gallon Per Minute
GTAW	Gas Tungsten Arc Welding
HPP	Health Physics Procedure
ID	Inside Diameter
ISI	Inservice Inspection
LCO	Limiting Condition for Operation
MHz	Megahertz
MWR	Maintenance Work Request
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
OD	Outside Diameter
PDI	Performance Demonstration Initiative
PSI	Preservice Inspection
psig	Pounds per Square Inch Gauge
PWSCC	Primary Water Stress Corrosion Cracking
PT	Liquid Penetrant Testing
PTP	Preventive Test Procedure
PWR	Pressurized Water Reactor
RB	Reactor Building
RC	Reactor Coolant
RBCU	Reactor Building Cooling Unit
RL	Refracted Longitudinal
RO	Refueling Outage
RT	Radiographic Testing
RV	Reactor Vessel
RWP	Radiation Work Permit
SMAW	Shielded Metal Arc Welding
SRWP	Standing Radiation Work Permit
STP	Surveillance Test Procedure
TDC	Top Dead Center
TS	Technical Specification
UT	Ultrasonic Testing
UDRPS	Ultrasonic Data Recording and Processing System
VT	Visual Inspection