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Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
RESPONSE TO INITIAL QUESTIONS DATED
OCTOBER 23, 2000
MSP 00-0244

South Carolina Electric & Gas Company (SCE&G) submits the attached responses to the initial questions asked by the NRC during a telephone conference on October 13, 2000, as followed up with a letter dated October 23, 2000. These questions pertain to the cracked weld in the A loop of the Reactor Coolant System.

Should you have any questions, please call Mr. Phil Rose at (803) 345-4052.

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Very truly yours,

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Responses to NRC Questions on

A loop hot leg crack

Dated 10/23/2000

1. Based on the estimated flaw size, what would be the predicted magnitude of the leakage?

Response

SCE&G is not aware of any known analytical methodology that could be used to calculate, based on as found crack configuration, the expected leak rate. However, estimates from indirect means place the leak at approximately 0.12 – 0.15 gpm. In addition, other information exists applicable to the issue of leak rates.

Prior to flaw characterization, a preliminary safety evaluation was conducted by Westinghouse to demonstrate that operation with this crack was not detrimental to the health and safety of the public. Per this safety evaluation, a 2.3 inch through-wall circumferential crack in an Alloy 82/182 (Inconel 600 material) weld would be expected to permit a leak rate of about 1.0 gpm. The same size leak in a stainless steel weld would be expected from a 1.9 inch crack. The critical length crack developed through fracture analysis for Alloy 82/182 is greater than 43 inches. This means that a pressure boundary leak would be easily detected before the crack length approached that which presents an immediate safety concern.

Additional NDE completed on November 6, 2000 determined that the through wall crack was oriented axially rather than circumferentially as initially suspected. An axially oriented crack in an Alloy 82/182 weld that is 2.2 inches at the leaking surface would be expected to permit a leak rate of about 1.0 gpm. The critical length crack developed though fracture analysis for Alloy 82/182 for an axial crack is approximately 35 inches.

Through destructive examination, the axial crack has been determined to be approximately 2.5 inches at the inside diameter, with a 3/16 inch opening on the outside diameter at approximately 7 degrees off top dead center. This defect is contained exclusively in the Inconel weld material, in the weld buildup (buttering), closure weld and nozzle cladding. A circumferential indication has been identified as a 1.6 inch crack that intersects the axial crack in the nozzle Inconel cladding. This crack progressed only through the Inconel cladding up to the carbon steel interface.

2. Describe the leak detection procedures and containment leak detection capability at Summer, and discuss why the leak was not discovered during plant operation.

Response

The existing design of leak detection systems at V. C. Summer fully satisfies the performance requirements of Regulatory Guide 1.45 (May 1973). The leak detection hardware is supported by visual inspections and engineering oversight of plant operation, including the data from leak detection systems and visual inspections.

Plant procedures and programs have been developed to provide testing requirements of the installed equipment and provide guidance on responding to abnormal indications of leakage. The Surveillance Test Procedures (STP) include procedures to check for leakage (STP 114.002) and are performed at a frequency to ensure that leakage approaching the Technical Specification (TS) limit of 1.0 gpm will be discovered. STP 114.002 uses the plant computer to calculate the leak rate based on water inventory balances with an action level of 0.8 gpm. This surveillance is actually performed once a day with the TS requirement of once per 72 hours. Should the surveillance results indicate leakage of 0.8 gpm, direction is provided to perform a more detailed calculation (PTP 175.001) to quantify the identified and unidentified leakage.

Annunciator Response Procedures (ARP) are used by the operations staff to provide guidance whenever a main control board (MCB) indication alarms. Direction is provided to verify the alarm condition and perform predetermined corrective actions.

Operators are trained on the use of ARPs and required responses. When an annunciator alarm is received, the procedure is reviewed to determine probable causes and course of action. These procedures are kept within reach at the MCB and are used during simulator training.

The capability of the leakage detection system has been repeatedly demonstrated over the plant's operating history. The following examples are provided as illustration:

a. On December 8, 1982, operators noted frequent automatic makeup to the RCS. An inventory balance was performed and unidentified leakage was found to be 1.56 gpm. Operators promptly investigated and discovered that a drain valve on a reactor coolant pump seal injection filter (outside containment, non-RCS) was leaking approximately 0.7 gpm.

b. On October 7, 1986, the reactor was shut down because the operators could not accurately verify the amount of unidentified leakage. Over a period of three weeks, identified leakage had increased slowly to 3.91 gpm. During performance of the required inventory balance, the Reactor Coolant Drain Tank (RCDT) relief valve opened and released an unknown volume of water to the Reactor Building Leak Detection Sump. Since the RCDT level is an input to the leak rate calculation, the actual leak rate could not be determined. Upon repair of the relief valve, unidentified leakage was 0.0 gpm.

c. On December 5, 1994, RCS unidentified leakage was assessed to be 0.8 gpm. Operators initiated an investigation and concluded that 0.2 gpm was attributed to an unidentified source inside the reactor building (RB). A visual inspection was conducted inside the RB and a leaking pressure boundary weld on a reactor coolant pump seal injection line was located. The plant was placed in cold shutdown to repair the weld.

d. On October 7, 2000, during visual inspections following shutdown for RF-12, plant personnel identified an accumulation of boric acid residue under the loop A hot leg. Further investigation resulted in the identification of pressure boundary leakage on October 12, 2000. Throughout the preceding cycle, unidentified leakage trended between 0.2 and 0.3 gpm.

The activity in the Reactor Coolant System (RCS) during the last cycle was much lower than in previous cycles and may have been the primary reason why the leak was not detected by the installed leak detection systems. The total estimated leak rate (unidentified) from all sources in the RB trended between 0.2 and 0.3 gpm. The installed systems are adequate to detect a 1.0 gpm leak within 1 hour, per Regulatory Guide 1.45, and the leak rate from the crack was simply too small to discriminate it as a discrete leak location.

3. When was the subject weld last inspected and what were the results of that inspection? Also, discuss the inspections of the reactor pressure vessel welds. Describe the inspection technique(s) utilized, and the qualifications of those techniques and personnel, if any. Provide your estimate of the probability of detection for a 4-inch crack that is 25-percent, 50-percent, and 100-percent through-wall.

Response

The subject weld was last inspected in the spring of 1993 during the 10-year inservice inspection (ISI) inspection. The examination scope included all accessible vessel shell welds, nozzle welds and nozzle-to-pipe welds. No indications were recorded in the inlet and outlet nozzle-to-pipe welds. A total of 64 indications were recorded in the vessel shell welds. The recordings made were primarily small embedded fabrication type flaws.

Inlet and outlet nozzle-to-pipe welds were examined from within the nozzle bores using 70 degree L dual element transducers applied four-directionally in scans performed normal to and parallel to the weld seam. The transducers were RTD construction, 2.0 MHz ((10x18 mm), FS 30 mm (metal path)). The transducers were arranged in a 5-pocket sled having a curved surface conforming to the nozzle bore. Recording criteria for the nozzle-to-pipe welds was specified by procedure at 50% distance amplitude curve (DAC). All examination data was recorded using the UDRPS II data acquisition system.

Calibration for the nozzle-to-pipe examinations was conducted on calibration block CGE-RV-4, a bi-metallic welded calibration block with near surface side drilled holes installed in both joining materials (clad carbon steel and wrought stainless steel) at depths of .25", .5", and .75". Acquisition parameters were established as follows:

Digitization rate	:	10 MHz
Inter Pulse Period	:	0.096 (distance between recorded A-scans in the processed data display)
Examination depth	:	2.25" (Code volume depth is 0.91")

The entire dynamic display range of the ultrasonic instrument was recorded by the UDRPS system (from approximately 5% full screen to 120% full screen). Even with the interpretation and investigation level at 50%, the entire lower amplitude response range was captured and displayed down to the baseline in the processed data.

A review of the nozzle to pipe examination data was performed on October 23-24, 2000. The review consisted of analyzing the full digitized UDRPS II data, which was evaluated by interpreting individual scan lines (At 1° increment or 0.25" between scan lines). It was determined that no indications could be identified other than normal persistent patterns of interface response attributable to normal weld/interface metallurgy.

The examination of the reactor pressure vessel welds used single element 45 and 60 degree shear wave RTD transducers, having an element size of 30x25 mm at 2.0 MHz. For under-clad detection and comprehensive examination of the first 5 inches, RTD 70 degree L dual element 2 (15x25 mm) transducers at 2.0 MHz were used. Scanning speed was 3 inches per second. All shell welds were interrogated four-directionally and scanning was conducted normal and parallel to the weld seam.

Personnel involved in reviewing the 1993 examination data were Level III analysts certified in accordance with ASME Section XI requirements. Both analysts had previous experience in automated data acquisition and analysis. The technique qualification was established based on the acceptable methodology at the time (1993). The validation of acquisition and procedure parameters by way of the dynamic calibration verification using the acquisition system was performed in accordance with the requirements of Regulatory Guide 1.150. The acquisition parameters chosen for the examination, were developed by WesDyne through experience with other qualifications performed with the same data acquisition system.

WesDyne, who performed both the 1993 and the 2000 UT examinations, stated the probability of detection for a 4 inch crack in the nozzle-to-pipe weld, having 25%, 50% and 100% through-wall cracking is estimated to be at least 90%. This is based on the acquisition parameters of the 1993 examination and the equipment in the examination system. The estimate is made with the understanding that historically good detection rates with the 70L twin crystal transducer have been demonstrated in manual and automated trials where the scanning surface permits reasonably good transducer contact as the scan line approaches the defect. Testing performed in 1993 has determined that detection of a flaw is beam direction dependent and that protrusions into the nozzle bore (such as root protrusions) did not preclude detection.

After additional review, it was discovered that one location near 10 degrees from top dead center experienced significant detector lift-off. At the time, this was not considered significant, as only about 2% of the Code required weld volume was missed. The lift-off was caused by surface irregularities, which did not allow the rigid sled and transducers constant contact with the surface. This surface condition was later confirmed during hot cell examination.

Based on the root-cause assessment that PWSCC was involved and the aggressive nature of PWSCC growth, it is unlikely that the crack existed in 1993.

4. Based on your review of the construction radiograph (RT) films for the subject (and other, as appropriate) welds, describe the results. In retrospect, based on this re-review of the RT films, could construction-related defects have been overlooked? Have the RT films been digitized and enhanced? Please explain.

Response

The original construction radiograph films were reviewed (interpreted) by four Level III certified individuals (one from the RT vendor and 3 from the prime contractor – Fluor Daniel) who concurred that there were no Code rejectable indications in the subject weld.

After the discovery of a crack on October 12, 2000, the radiographs were scanned by four Level III certified individuals (one utility and three industry experts). Without performing a detailed re-interpretation of the radiographs for the A hot leg nozzle-to-pipe weld, they concluded that there did not appear to be anything that looked like a rejectable indication. SCE&G does not feel, based on the number of qualified personnel who have reviewed these radiographs, that any significant flaws in the subject weld could have been overlooked.

The A hot leg nozzle-to-pipe weld radiographs have been digitized. This process did not yield any additional information.

5. What is the design cumulative usage factor for the subject weld?

Response

The design Cumulative Usage Factor (CUF) for the subject weld is 0.52.

6. Describe, in detail, the inspection plans for the subject (and other, as appropriate) welds. Include in this description, as a minimum, what is to be inspected, scope expansion, and the inspection techniques to be used, including how the techniques and examiners are to be qualified, the establishment of inspection uncertainties from tooling positioning, and of measurement errors associated with the transducers.

Response

Pre repair inspection:

The inspection plan for the reactor vessel nozzle-to-pipe welds included automated ultrasonic and eddy-current techniques on the inside diameter (nozzle bore). Examination end effectors were delivered to the inspection surface by the SUPREEM 6-axis robot. Ultrasonic acquisition was performed by the WesDyne Paragon system. Eddy current (ET) data was acquired and analyzed using the Amdata Intraspect system.

The ET was performed as a complementary NDE technique using a transmit/receive or compensating probe with digitization and display software by the Intraspect system. Calibration, demonstration, acquisition, and analysis were performed by certified and qualified personnel. These are ASNT Level III certificate holders and have current Appendix VIII automated certification attachments for Supplements 3, 4, and 6.

The ultrasonic techniques involved the use of three beam angles and three separate end-effector combinations to examine the full volume around the nozzle-to-pipe weld. A 70 degree refracted longitudinal (RL) dual element 2.00 MHz, 20 mm focal distance (metal path) transducer was applied to the nozzle-to-pipe weld examination volume four-directionally. A 45 degree RL dual element 4 MHz, 25 mm (metal path) transducer was applied to the nozzle-to-pipe weld examination volume in at least two directions (circumferentially) for the sizing of cracks tips from a depth of about 4 mm to about 30 mm. A 37 degree RL dual element 1.0 MHz, 2.74 inch OD focus transducer was applied to the nozzle-to-pipe weld volume from a depth of about 1 inch to the OD surface and was intended to support detection and flaw tip sizing.

These transducers were calibrated on the existing calibration block, CGE-RV-4 and performance was verified on the Appendix VIII, Supplement 10 specimen supplied by EPRI. Examination sensitivity was adjusted as necessary to provide high signal to noise ratio on the flawed specimen. Calibration, demonstration, acquisition, and analysis were performed by certified and qualified personnel.

Two spring loaded ET probe assemblies were used. These are identical plus point coils, permanent magnets in X-rotation orientation. The primary probe is sensitive to axial and circumferential flaws. The secondary probe was rotated 45 degrees for better sensitivity to off axis (e.g., 45 degree) indications. Operating frequencies were 250 and 150 KHz. Scanning was performed with an Amdata 5090 scanner at four inches per second using 0.125 inch increments. Acquisition was performed using the Intraspect EC Data Acquisition system at a rate of 40 samples per inch along the scan axis.

The tooling positioning accuracy in terms of position accuracy and repeatability was previously demonstrated in nozzle bore applications for the Sizewell B project during November 1998. The accuracy was shown to be within 6 mm with a typical safe-end transducer package attached on a scramble/return test. It is anticipated that sweep to sweep repeatability on a given flaw target with the SUPREEM/Paragon combination will be around 6 mm maximum.

These inspections were performed on all six reactor vessel nozzles to quantify the extent of condition and assist in planning the scope of the repair effort. The A loop hot leg nozzle-to-pipe weld was determined to be the only weld with recordable UT indications.

7. By letter dated June 22, 1992, SCE&G requested to eliminate from the Summer design basis the dynamic effects of postulated pipe ruptures in the reactor coolant piping, based on a leak-before-break analysis. The staff granted this request later that year. Based on the cracking found to date, describe the implications of the missed leak on the bases of this request and approval, and what, if any, revisions to the present design basis should be made.

Response

The original basis for Leak-Before-Break was developed in the early 1980's. Shortly after the concept was originally accepted by the NRC, the Boiling Water Reactor recirculation piping stress corrosion cracking issue erupted, and several instances of circumferential cracking were found. This led the NRC to the proviso that Leak-Before-Break should not be applied to cases where such cracks could exist. The requirement was generalized to prohibit application in any case where SCC could occur.

At V. C. Summer the cracking was found to be axially oriented and limited in length to the Alloy 82/182 weld, which is a length of about 2.5 inches. The likelihood of a longer crack is extremely small, because the cracking cannot extend into either the low alloy steel nozzle or the stainless steel piping beyond the heat affected zones by a stress corrosion mechanism.

Representative finite element analysis confirms that hoop stresses (causing axial cracks) are larger than axial stresses. This would tend to indicate that crack growth in the axial direction would dominate.

The future operation of V. C. Summer will be limited to ensure that any potential flaws will remain acceptable to the requirements of the ASME Code, Section XI. Therefore, the basic principles of Leak-Before-Break remain valid; that is, detectable leakage will occur prior to pipe rupture.

Per a safety evaluation, dated January 11, 1993, the NRC approved the use at V. C. Summer of WCAP 13206, Leak-Before-Break methodology. This safety evaluation states that operating experience has demonstrated that stress corrosion is not a failure mechanism for Class 1 piping in Westinghouse plants. Since the determination that the crack in the A hot leg is propagated by this mechanism, V. C. Summer has concluded that we are not in verbatim compliance with the conclusion in the Safety Evaluation.

This condition is documented in our corrective action program. Based on the evaluations performed for V. C. Summer in WCAP 15615, SCE&G has confidence that the structural integrity of the piping will remain intact such that Leak-Before-Break should remain applicable. The dimensions of a through-wall crack that will leak 10 gpm (a readily detectable amount) is much less than the size of a crack that would cause structural failure of the pipe (Critical Flaw Length).

8. Describe your schedule and scope for performing a root cause determination. As this is a unique situation, discuss your plans for removal of the crack, or portions thereof, for destructive metallurgical examination, and the metallurgical and fractographic analyses you are planning to perform to determine the failure mode. Provide a comparison of the ultrasonic examination performed on the in-situ weld and the destructive metallurgical examination to be performed.

Response

A root cause team was formed consisting of plant individuals and industry experts. They developed a matrix of all possible failure mechanisms and have completed the collection and evaluation of information to support or refute each failure mechanism.

The entire weld in question and a short piece of adjoining piping were removed and sent for destructive and non-destructive testing to firmly establish the cracking mechanism.

The initial testing performed on this defective weld included PT, UT, RT, and ET to establish the extent, location and distribution of cracking in the weld. The ID and OD surfaces were examined by visual and low powered light microscopy. Detailed surface examinations were performed prior to, and following sectioning by light optical and scanning electron microscopy techniques. These results were recorded on digital photographic recordings.

A detailed test plan was finalized once the results of the surface examinations and NDE results were reviewed. The plan included metallographic examinations, fractographic examinations, chemical evaluation of surface deposits, and mechanical property measurements. Specimens have been preserved for further testing as determined necessary.

A preliminary determination was provided by Westinghouse which included stress corrosion as a significant contributor. The interim root cause has been determined to be:

a. Extensive repairs on the VC Summer reactor vessel "A" hot leg nozzle to pipe weld created high welding residual stresses in a material (Alloy 182/Alloy 82) and in an environment (nuclear primary water) known to cause primary water stress corrosion cracking (PWSCC).

b. Neither the codes, standards nor the welding process recognized or required consideration of the cumulative effect of multiple repair welding and weld grinding in the creation of high residual stresses.

Additional Contributing Factors:

a. The formation of hot cracking through the use of weld materials (Alloy 182/Alloy 82) that may exacerbate this type of defect.

b. Lack of NDE detection of the flaw

- Failure to recognize the contribution of surface contour, surface roughness, and detector physical parameters on the qualification of the NDE process for this type of flaw.

- Change to ultrasonic inspection process (angles/volumes at 10 year ISI). The pre-service and 1987 in-service inspections both included UT performed with multiple angles. The 1993 in-service inspection was performed using only the 70º angle transducer.

c. The use of a field weld versus shop weld with Alloys 182 and 82.

- Automatic welding was initially qualified at VC Summer for use on these welds using stainless steel materials.

- There are no code limits or guidance on the number and extent of welding and grinding repairs using these materials.

A final report will be generated to fully document the results.

9. Describe in detail the schedule and scope for performing repairs for the subject weld.

Response

The pre-existing field configuration is illustrated in Figure 2. The repair plan included the removal of the defective weld along with all of the nozzle butter and approximately 8 inches of the pipe. A stainless steel pipe spool piece was welded back in to restore the integrity of the RCS (Figure 3).

The Code of record for inservice inspection is the ASME Code, Section XI, 1989 Edition (no addenda). The construction Code of record for the A hot leg piping is the ASME Code, Section III, 1971 Edition (Winter addenda). The construction Code of record for the reactor vessel is the ASME Code, Section III, 1971 Edition (no addenda).

A relief request for the use of Alloy 52/152 weld filler materials and the associated Code Cases was submitted and subsequently approved. This allowed the use of Alloy 52/152 (Inconel 690) material in lieu of Alloy 82/182 and the appropriate procedures and qualifications for the new weld material.

The entire weld was removed and made available for failure analysis to support the root-cause evaluation. Ultrasonic examination was performed to determine the cut line on the reactor vessel side of the weld and assure complete flaw removal. Approximately 12 inches from this cut line (away from the vessel) a second cut was made to allow access to the nozzle. Both the A hot leg piping and the A steam generator were restrained to prevent any undesired movement.

The nozzle was built up with Alloy 52 to prepare for the nozzle-to-spool weld. The stainless steel cladding on the inside of the nozzle was removed for a distance of 1/2 inch from the nozzle end. The cladding was replaced with Alloy 52. Both activities were completed using machine gas tungsten arc welding (GTAW) and using the temper bead process.

Machine GTAW was used for temper bead welding per ASME Section XI Code Case N-432. The NRC approved Code Case, an alternative to ASME Code, Section III, Subsection NB, allowed the post-weld heat treatments to be performed at a reduced temperature range. Required pre-heats and soaks were performed in accordance with ASME Code requirements. The weld build up on the nozzle was inspected using the appropriate NDE techniques, machined into the narrow groove weld prep configuration, and inspected again.

The replacement spool piece was purchased from another utility which met ASME Code, Section III, 1974 Edition (Summer 1975) requirements. Code reconciliation was performed per the requirements of ASME Section XI. The replacement pipe was certified except for the hydrostatic test, which was performed prior to installation at V. C. Summer in accordance with ASME Code, Section III, NB-6000 requirements. The test pressure was 3106 psig (1.25 times the system design pressure).

The nozzle end of the replacement spool piece was buttered using the GTAW process and Alloy 52 filler material. This activity eliminated the need to perform a dissimilar metal weld in the field during installation

After the weld build up was complete, a groove was machined on the inside diameter of the spool piece, centered on the heat-affected zone under the buildup. An Alloy 52 weld overlay of this groove was performed as an engineered improvement to enhance the corrosion resistance of the spool piece.

The spool piece was machined on both ends to the narrow groove weld design, inspected per Code requirements and installed into the piping. This installation involved two narrow groove closing weld operations from the outside diameter. The nozzle-to-spool piece weld employed machine GTAW and Alloy 52 filler materials. The pipe-to spool piece weld employed machine GTAW and SFA 5.9 ER 308/308L (stainless steel) filler materials. ASME Code, Section III, qualified welding procedures were utilized, and radiography of the stainless steel weld was found to be acceptable.

Following radiographic exams on the Inconel weld that identified the presence of lack of fusion and porosity, the Inconel narrow groove weld was excavated to a depth of about 0.50 inches, effectively removing about 1.75 inches of material from the outer diameter. Grinding was also performed from the inside diameter to eliminate indications in the root area. After grinding was completed, “flapping” (the use of fine grit emery paper) was performed to condition the surface and minimize the residual stress imparted by the grinding operation. Once the indications were removed, welding re-commenced with radiography being performed approximately every 1/2 inch of the replacement weld until completion.

The NDE procedures for the repair were required to meet the 1992 Edition of Section III NB, since ASME Code Case N-416-1 was applied as part of the weld repair.

Following completion of the welds, RT will be performed with the source on the inside of the pipe and PT will be performed per Code requirements on the outside diameter of the weld. The volumetric examination for the completed weld will satisfy the ASME Code, Section III requirements. The examination performed to satisfy ASME Code, Section XI pre-service baseline requirements will be UT performed from the outside diameter after the Section III testing is complete.

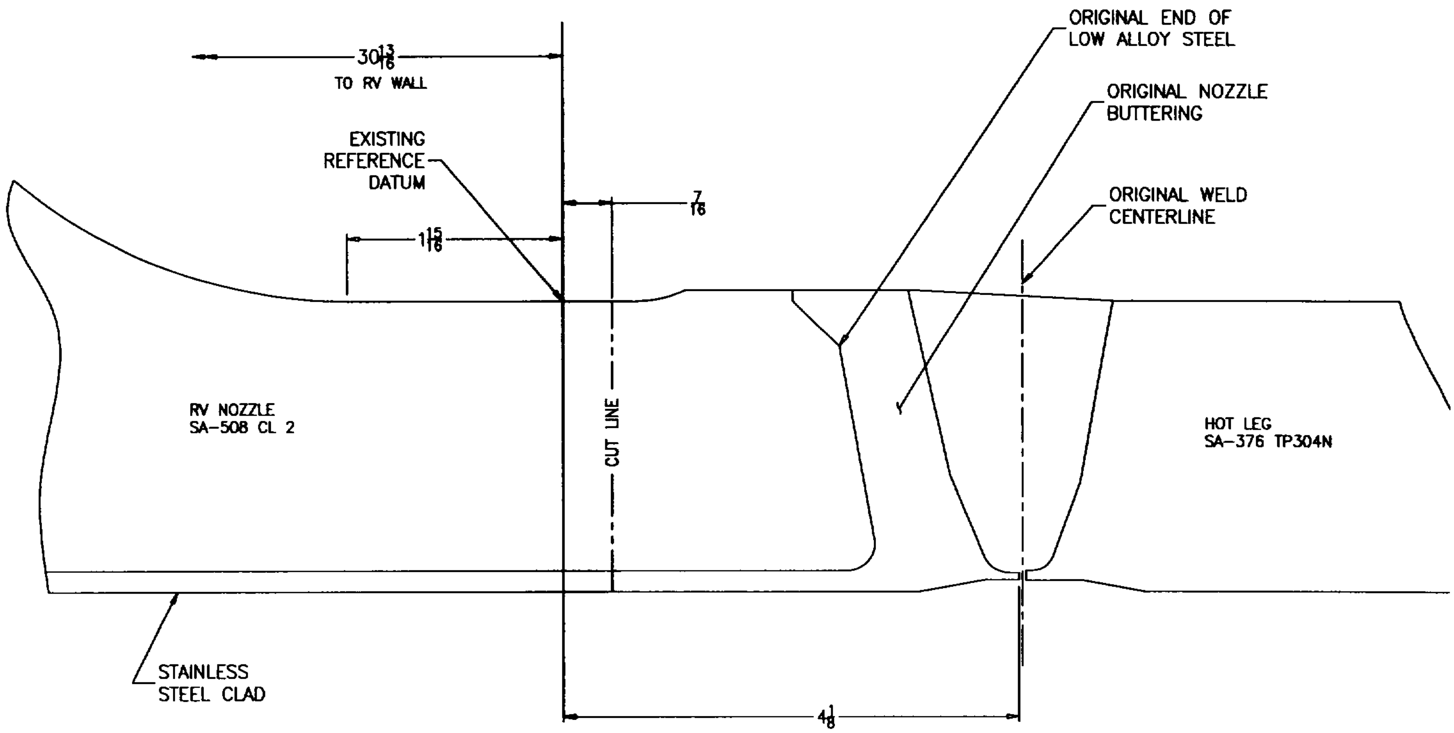

FIGURE 2
VC SUMMER NUCLEAR STATION
"A" LOOP HOT LEG NOZZLE
EXISTING WELD DESIGN

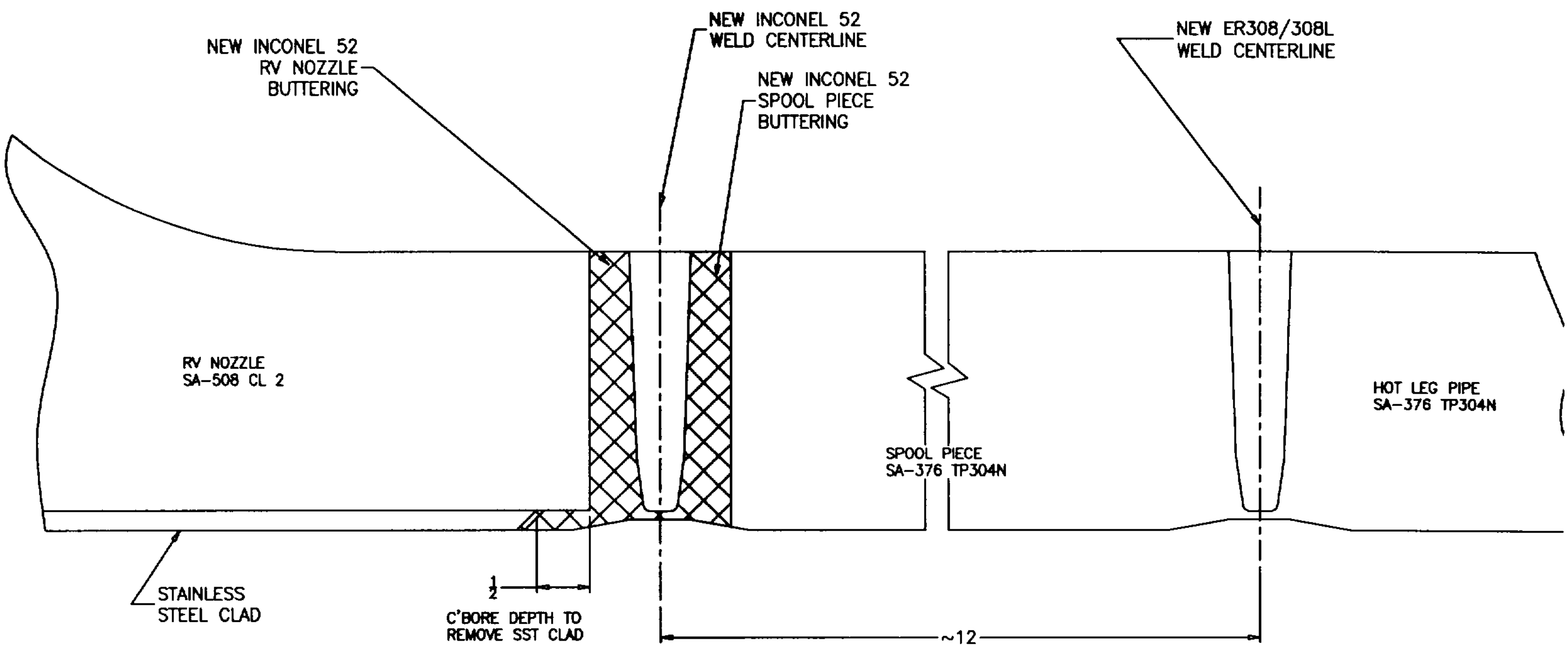

FIGURE 3
VC SUMMER NUCLEAR STATION
"A" LOOP HOT LEG NOZZLE
NEW WELD DESIGN