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SCE&G®
A SCANA COMPANY

April 3, 2002
RC-02-0055

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. G. E. Edison

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
RESPONSE TO NRC BULLETIN 2002-01
REACTOR PRESSURE VESSEL HEAD DEGRADATION AND
REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY

Reference:
1. SCE&G Letter to NRC (Document Control Desk), RC-01-0155, August 31, 2001; Response to NRC Bulletin 2001-01
2. PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48), EPRI, Palo Alto, CA: 2001. 1006284.

The U. S. Nuclear Regulatory Commission (NRC) issued NRC Bulletin 2002-01 to: (1) require that pressurized water reactor (PWR) utilities provide information related to the integrity of the reactor coolant pressure boundary including the reactor pressure vessel head and the extent to which inspections have been undertaken to satisfy applicable regulatory requirements, and (2) the basis for concluding that plants satisfy applicable regulatory requirements related to the structural integrity of the reactor coolant pressure boundary and future inspections will ensure continued compliance with applicable regulatory requirements, and (3) require that all addressees provide to the NRC a written response in accordance with the provisions of 10 CFR 50.54(f) if they are unable to provide the information or they can not meet the requested completion dates.

South Carolina Electric & Gas Company (SCE&G) acting for itself and as agent for South Carolina Public Service Authority, hereby submits the attached in response to the bulletin.

These statements and matters set forth herein are true and correct to the best of my knowledge, information, and belief.

Should you have questions, please call Mr. James Turkett at (803) 345-4047 or Mr. Charles Rice at (803) 345-4491.

Very truly yours,

Gregory H Halnon for SAB

Stephen A. Byrne

JT/CHR/SAB
Attachment

c: N. O. Lorick
N. S. Carns
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NSRC
RTS (0-C-02-0703)
File (815.02)
DMS (RC-02-0055)

STATE OF SOUTH CAROLINA :
: **TO WIT** :
COUNTY OF FAIRFIELD :

I hereby certify that on the 3rd day of April 2002, before me, the subscriber, a Notary Public of the State of South Carolina personally appeared Gregory H. Halnon, being duly sworn, and states that he has signature authority for the Senior Vice President, Nuclear Operations of the South Carolina Electric & Gas Company, a corporation of the State of South Carolina, that he provides the foregoing response for the purposes therein set forth, that the statements made are true and correct to the best of his knowledge, information, and belief, and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal

James W. Turkett, Jr.
Notary Public

My Commission Expires

October 2, 2010
Date

South Carolina Electric & Gas Company (SCE&G) submits the following response to NRC Bulletin 2002-01, *REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY.* This response utilizes, in part, plant information provided in MRP-48, *PWR Materials Reliability Program Response to NRC Bulletin 2001-01.* MRP-48 provides the latest plant rankings for all 69 domestic operating PWRs based on the time-at-temperature model.

VCSNS has been ranked for the potential for primary water stress corrosion cracking (PWSCC) of the reactor pressure vessel (RPV) top head nozzles using the time-at-temperature model and plant-specific input data reported in MRP-48. Using the criteria stated in NRC Bulletin 2001-01, VCSNS falls into the NRC category of plants considered to have a low susceptibility (i.e., greater than 30 Effective Full Power Years (EFPY)) to PWSCC of the RPV top head nozzles. The susceptibility ranking performed by the EPRI Material Reliability Project provides good results in predicting those plants susceptible to PWSCC of the CRDM Penetrations. Vessel head penetration inspections to comply with Generic Letter (GL) 97-01 are currently planned for VCSNS refueling outage 17 (Spring 2008).

1. <u>15 Day Response Required Information</u>

   A. <u>A summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at V. C. Summer Nuclear Station (VCSNS).</u>

   The reactor vessel head inspection program is conducted in accordance with the applicable in-service inspection requirements specified in Section XI of the 1989 edition of the ASME Boiler and Pressure Vessel Code. Prior to January 1, 1994, the 1977 edition Summer 78 addenda were utilized.

   All ASME class 1 components, piping and pressure retaining boundary category B-P receive an inspection prior to plant startup following each refueling outage utilizing an ASME visual examination VT-2 during a system leak test. VT-2 is an ASME B&PVC visual examination to locate evidence of pressure boundary leakage during a system pressure test. At the time of the inspection, the plant has been pressurized at nominal operating pressure for at least 4 hours with all of the insulation on. For this inspection, IWA-5242(a) requires that insulation be removed from pressure retaining bolted connections. VCSNS was granted relief by TAC M98145 (9/22/97) for the bolted joint inspection to be performed with the Reactor Coolant System cold (< 200° F) and depressurized followed by the system pressure test.

In addition to the inspection activities required by the Code, a visual examination of the reactor coolant system is conducted in accordance with Generic Letter (GL) 88-05 at each refueling outage to identify any boric acid leakage. These examinations include the reactor vessel closure stud bolts, closure nuts and washers, the reactor vessel head, the reactor vessel head vent isolation valves and the reactor vessel head vent valves.

The GL 88-05 and ASME Code walk down inspections, which include the reactor vessel head, are performed with insulation in-place. A 100% bare metal inspection of the reactor vessel head at VCSNS has not been performed since the plant started operation.

Necessary maintenance of pressure retaining components is conducted under 10 CFR 50, Appendix B station programs.

B. <u>Evaluation of the ability of VCSNS inspection and maintenance programs to identify degradation of the reactor pressure vessel head including, thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis-Besse.</u>

V. C. Summer utilizes a defense-in-depth approach to ensure that there is no degradation of the reactor vessel head. These defenses consist of a series of activities beginning at the start of each refueling outage, and ending just prior to startup for the subsequent operating cycles. Procedures require that leakage sources be identified; the extent of leakage determined; corrosion if any be evaluated; boric acid cleaned up; and leakages be repaired and/or evaluated. The success of this approach was demonstrated when a very small reactor coolant pressure boundary leak was discovered during refuel 12. Even this very small leak produced a large amount of boric acid residue, which was easily detectable. The type of leakage that is likely to produce a wet corrosion environment would show up as boric acid residue on the reactor vessel head closure bolts, reactor vessel head flange, conoseal bolting, and reactor head vent flange bolting.

The initial inspection, conducted shortly after plant shutdown, is to identify any evidence of leakage during the previous operating cycle. Evidence of boric acid residue is documented and dispositioned in the station's Corrective Action Program. After plant cooldown, walkdowns of systems containing borated water are performed for GL 88-05 and ASME bolted joint inspections [IWA 52423(a)].

Finally, reactor coolant system pressure boundary system leak tests are performed during each refueling outage just prior to startup. This test is performed after the reactor coolant system has been pressurized for a minimum of four hours. This ensures that the type of leakage that is likely to produce a wet corrosion environment will become evident. Inspections are performed above and below the reactor vessel.

Leakage monitoring is performed while the plant is in operation. Any evidence of leakage in excess of established limits, or significant changes in measured leak rates are investigated.

In conclusion, the current and future inspection programs together with the low susceptibility of the CRDM penetrations to PWSCC provide a reasonable assurance that there is no unacceptable degradation of the reactor vessel head and that degradation will be detected prior to it becoming unacceptable.

C. Description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in response 1.A that could have led to degradation and the corrective actions taken to address such conditions.

Prior to 1991, the station maintenance work request program was used to identify and correct component leakage around the reactor vessel head. A review of all maintenance work requests related to the reactor pressure vessel head was performed.

As noted in GL 88-05, various plants experienced leakage at the fittings used to seal the incore thermocouple extension wiring at the top of the conoseal assembly on the reactor head. VCSNS had experienced this after reconnecting the fittings following the second and third refueling outages. The leakage was detected and corrected prior to startup for each event. The magnitude of the refuel 2 leakage was considered small. This was supported by the amount of leakage found in refuel 3. A discussion with the mechanic who performed the repairs in refuel 3 indicated that there was minor boric acid buildup around the flanges that did not come in contact with the reactor vessel head or insulation. The connections were modified to reduce the possibility of leakage from these locations. No further leaks have occurred at these locations.

Since 1991, inspections have been performed utilizing plant procedures developed to implement GL 88-05 inspections and ASME Code (VT-2, VT-3) inspections. These inspections have identified valve packing and gasket leaks. All leaks were addressed in accordance with plant procedures and were either repaired or evaluated for future action. Repair documents were referenced on test data sheets. Evaluations were included in the test package documentation. No leakage around the head or head degradation has been identified through these inspections.

Based on the maintenance and repair history at VCSNS, SCE&G believes that there is no boric acid build up under the insulation on the head surface.

D. <u>Description of VCSNS schedule, plans, and basis for future inspections of the reactor pressure vessel head and penetration nozzles to include the inspection method(s), scope, frequency, qualification requirements, and acceptance criteria.</u>

Visual examinations of the reactor vessel head area, as required by the ASME code and GL 88-05, are scheduled to be conducted each refueling outage. The entire scope of the code required reactor vessel inservice inspection examination is scheduled to be performed during refuel 14. This includes category B-N-1, B-O, B-G-1, and B-G-2. Inspections after refuel 14 will be driven by the ASME code edition and addenda requirements applicable to the next interval, as well as any pertinent regulatory commitments.

Based on our review of maintenance history and the results of the small conoseal leaks found in Refueling outages 2 and 3 with subsequent action for boric acid clean up, SCE&G does not believe that accumulation exists under the head insulation. However, SCE&G feels that it is prudent to apply the operating experience from Davis-Besse and to take a proactive position to address the event. Therefore, an inspection will be performed in the upcoming refueling outage in April 2002. The inspection will utilize a remote optical device under the insulation and will cover all accessible areas of the bare metal on the head. Individuals who are qualified to the ASME code VT-3 inspection criteria will perform the inspection. Evaluation of any detectable degradation will be in accordance with the applicable ASME Code requirements and station procedures.

This approach is appropriate since 1) VC Summer has been determined to have a low susceptibility to primary water stress corrosion cracking (PWSCC) and 2) there is no evidence of any boric acid leakage coming into direct contact with the reactor vessel head or insulation during the life of the plant.

Vessel head penetration inspections to comply with GL 97-01 are currently planned for VCSNS refueling outage 17 (Spring 2008).

E. Conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met. Basis for concluding that the inspections discussed in response 1.D will provide reasonable assurance that these regulatory requirements will continue to be met. (1) If evaluation does not support the conclusion, discuss plans for plant shutdown and inspection, or (2) If evaluation supports the conclusion, provide basis for concluding that all regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

SCE&G supports continued operation of VCSNS based on the conclusion that actions addressed by the following programs provide reasonable assurance that current regulatory requirements are being met and will continue to be met.

Appendix A to 10 CFR Part 50 provides the general design criteria (GDC) for Nuclear Power plants or, as appropriate, similar requirements in the licensing basis for a reactor facility, the requirements of 10 CFR 50.55a, and the quality assurance criteria of Appendix B to 10 CFR Part 50 provide the bases and requirements for NRC staff assessment of the potential for and consequences of degradation of the reactor coolant pressure boundary.

The VCSNS Final Safety Analysis Report (FSAR) discusses the extent to which the design criteria for the plant structures, systems and components important to safety meet the NRC "General Design Criteria for Nuclear Power Plants" specified in Appendix A to 10 CFR Part 50. The VCSNS is designed, constructed and operated to comply with SCE&G's understanding of the intent of the NRC's "General Design Criteria for Nuclear Power Plants".

Detailed evaluations of compliance with the various General Design Criteria are incorporated in applicable sections of the FSAR. The details for GDC 14, "Reactor Coolant Pressure Boundary" are contained in FSAR sections 3.6, 3.7, 3.9, 5.2 and 5.4. The details for GDC 30, "Quality of Reactor Coolant Pressure Boundary" are contained in FSAR section 5.0. As indicated in FSAR section 3.1.2.4, the reactor coolant pressure boundary components are designed, fabricated, inspected and tested in accordance with the ASME Code, Section III. The details for GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary" are contained in FSAR section 3.1.2.4 and chapter 5.0. The details for GDC 32, " Inspection of Reactor Coolant

Pressure Boundary" are contained in FSAR section 3.1.2.4 and chapter 5.2. According to the FSAR, provisions have been made in the Reactor Coolant system design for adequate inspection, testing and surveillance during the service lifetime.

FSAR section 5.7, "In-service Inspection Program (Including Preservice Inspection)" defines the applicable codes and standards used at VCSNS for in-service and preservice inspections. These inspection programs are administered in accordance with the applicable codes and standards as addressed in 10 CFR 50.55a.

The Quality Assurance program, which supports implementation of the above regulatory requirements, for the VCSNS is defined in FSAR section 17.0. Section 17.0 indicates that SCE&G recognizes the need for a comprehensive, formalized, documented method of specifying and verifying that the VCSNS station design, procurement, construction, modification, maintenance, in-service inspection, and operation has been and will continue to be accomplished without undue risk to the health and safety of the public. Cognizant of this responsibility SCE&G has established and will execute an effective Quality Assurance program for the VCSNS which meets the requirements of Title 10 CFR Part 50 Appendix B. This QA program provides the necessary systematic activity and administrative control to provide checks to assure activities which affect quality and safety-related functions during design, procurement, construction, modification, maintenance, in-service inspection, and operation are performed in accordance with established procedures.

FSAR section 17.1 defines the Quality Assurance requirements during the design, construction and major modifications.

FSAR section 17.2 defines the Quality Assurance requirements during the Operations phase. Section 17.2 is designed to assure VCSNS safe operation, to assure the installed quality of the station is maintained throughout the life of the plant, and to satisfy the quality assurance requirements of 10 CFR 50, Appendix B, and those quality related regulatory guides as described in Appendix 3A. Regulatory Guide 1.33, Revision 2, recommends compliance with the stipulations of ANSI N18.7-1976. Compliance with these requirements constitutes administrative controls for the operation of nuclear power plants in a manner consistent with the applicable criteria for quality assurance.

Section 17.2 is written so as to address each of the 18 criteria of 10 CFR 50, Appendix B, and describes the action taken by organization units and individuals within SCE&G to assure the safe operation and the installed quality of the plant are maintained throughout the operational life of the plant. The station Operational Quality Assurance Plan (OQAP) is a sub-tier document to FSAR section 17.2 and provides additional detail regarding the implementation of each of the 10 CFR 50 Appendix B criteria.

The regulatory requirements specified in the FSAR are incorporated into various implementing procedures at VCSNS. These implementing procedures are governed by 10 CFR 50 Appendix B Criteria V, "Instructions, Procedures and Drawings". Criterion V states, in part, that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. As stated in NRC Bulletin 2002-01, visual and volumetric examinations of the reactor coolant pressure boundary are activities that should be documented in accordance with these requirements. These examination activities are procedurally controlled. These procedures ensure that inservice testing is conducted in an effective and consistent manner that complies with the plant operating license, procedures, management directives, Technical Specifications and applicable regulations and standards.

The procedure review and approval cycle provides reasonable assurance that procedures contain appropriate qualitative and quantitative acceptance criteria.

Criterion IX (Control of Special Processes) of 10 CFR 50 Appendix B states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. As stated in NRC Bulletin 2002-01, special requirements for visual examination and/or ultrasonic testing would generally require the use of qualified visual and ultrasonic testing methods. Such methods are ones that a plant specific analysis has demonstrated would result in reliable detection of degradation prior to a loss of specified reactor coolant pressure boundary margins of safety. The qualification requirements for personnel performing special processes, including nondestructive testing, are defined in Quality Systems Procedures and the Nuclear Training Manual.

Criterion XVI (Corrective Action) of Appendix B to 10 CFR 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. As stated in

NRC Bulletin 2002-01, these actions should include proactive inspections and repair of degraded portions of the reactor coolant pressure boundary.

Administrative and implementing procedures have been developed to identify deficiencies and provide resolution including the performance of root cause evaluations when necessary.

If contractor or vendor personnel conduct any of the above activities, then contractors or vendors are determined to be qualified through audits performed by the Procurement Quality organization as defined in applicable Engineering Services procedures. Either Quality Systems or Procurement Quality, and the applicable on site organization review any Special Process Procedures. These special process procedures are then released for contractor/vendor use by the "Release to Work" process as defined in the FSAR and implementing procedures. This release to work process requires that procedures are reviewed and approved, personnel certifications are up to date and applicable, and that measuring and testing equipment is suitable and within the current calibration period.

In addition, there is Quality Systems oversight of activities associated with the above. This oversight function is defined in FSAR section 17.2.18, "Audits". The oversight function consists of audit and surveillances. Audits are conducted in accordance with ANSI 45.2.12, "Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants" and are conducted by personnel qualified per ANSI 45.2.23, " Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants".

The VCSNS Technical Specifications do not permit operation with Reactor Coolant System pressure boundary leakage.

GL 88-05 is implemented as discussed in item "A" above.

All of the above actions provide reasonable assurance that current regulatory requirements are being met and will continue to be met.

Reporting of Future Inspection Results

VCSNS will provide the information requested in Item 2 of NRC Bulletin 2002-01 within 30 days after plant restart following the next refueling outage, which is currently scheduled to begin in April 2002.

Information required by Item 3 for the 60 day response will be submitted by May 17, 2002.