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Gary J. Taylor
Vice President
Nuclear Operations

July 30, 1997
RC-97-0150

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

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
RESPONSE TO GENERIC LETTER 97-01

References: (1) G.J. Taylor Letter to Document Control Desk, RC-97-0087, May 1, 1997.
(2) Westinghouse WCAP-13565, Revision 1, "Alloy 600 Reactor Vessel Adapter Tube Cracking Safety Evaluation", February 1993 (Proprietary), submitted to the NRC by the Westinghouse Owner's Group (WOG) through Nuclear Management and Resource Council (NUMARC) on June 16, 1993.
(3) Westinghouse WCAP-14901, "Background and Methodology for Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Westinghouse Owner's Group", (Proprietary), to be submitted to the NRC by the WOG.

Attachments: (1) NRC letter from W. T. Russel to W. Rasin, NUMARC, dated 11/19/93 containing "Safety Evaluation for Potential Reactor Vessel Head Adaptor Tube Cracking".
(2) Westinghouse WCAP-14932 (Proprietary) and WCAP-14555 (Non-Proprietary), "Probabilistic and Economic Evaluation of Reactor Vessel Closure Head Penetration Integrity for Virgil C. Summer Nuclear Station".

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Pursuant to 10CFR50.54(f), the Nuclear Regulatory Commission (NRC) issued Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations", requesting that, within 120 days from the issuance of the generic letter, licensees provide the following:

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1) (1.1) a description of all inspections of control rod drive mechanism (CRDM) nozzle and other vessel head penetrations (VHPs) performed prior to the generic letter (GL), if any, and the results obtained;

(1.2) a plan to periodically inspect the CRDM nozzles and other VHPs, including scope of the inspections and the technical basis for the schedule;

(1.3) if a plan is not to be developed, provide analysis to support why an inspection plan is not necessary,

(1.4) the technical basis and analysis that supports the selected course of action as listed in either 1.2 or 1.3 above, should include relevant data and/or tests used to develop crack initiation and growth models, the methods and data used to validate these models, the plant-specific inputs to these models, and how these models substantiate the susceptibility evaluation

and

2) a description of any resin bead intrusions, as described in IN 96-11, that have exceeded the current Electric Power Research Institute (EPRI) Pressurized Water Reactor (PWR) chemistry guidelines recommendations

South Carolina Electric and Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby submits the requested 120 day response to Generic Letter 97-01 as it applies to the Virgil C. Summer Nuclear Station (VCSNS).

Prior to issuance of the GL, VCSNS has worked with the Westinghouse Owner's Group (WOG), the Electric Power Research Institute (EPRI) and the Nuclear Energy Institute (NEI) to understand the operational experience, identify techn cal issues, cause factors, relative importance, and solutions in regards to the degradat on of CRDM nozzles and other VHPs. One of the comprehensive tasks was the development of safety evaluations that characterized the initiation of damage, propagation and consequences. The safety evaluations, as submitted by NUMARC (now NEI) for WOG, are contained in Reference (2) and are applicable to VCSNS. The NRC reviewed these safety evaluations and issued a safety evaluation report (SER) to NEI on November 19, 1993. A copy of the SER is enclosed as Attachment (1). The WOG safety evaluation and the associated SER establish the basis for VCSNS continued operation.

Requested information Item 1.1:
"A description of all inspections of CRDM nozzle and other VHPs performed to the date of this generic letter, including the results of these inspections."
"Footnote: Those licensees that have previously submitted the requested information need not resubmit it, but may instead reference the appropriate correspondence in their response to this Generic Letter."

Response:
For VCSNS, no inspections have been performed for the ISI Section XI program. System leakage tests are regularly performed as part of refueling maintenance and operation activities with no visual indications of leakage to date.

A summary of VHP inspections compiled for all WOG plants is contained in Section 1.3 of Reference (2).

Requested information Item 1.2:
"If a plan has been developed to periodically inspect the CRDM nozzle and other VHPs:

A. Provide the schedule for first, and subsequent, inspections of the CRDM nozzle and other VHPs, including the technical basis for this schedule."

Response:
VCSNS is a participant in the WOG/NEI reactor pressure vessel head penetration integrated inspection program. This integrated program includes volumetric inspection of head penetrations that have been performed (refer to Reference (2), Section 1.0) and additional volumetric inspections that will be performed. Present plans call for two Combustion Engineering-design plants and two Babcock & Wilcox-design plants to be inspected over the next three years. Additionally, Westinghouse-design plants are likely to be added to the list over the next few months, as an integrated industry inspection plan is formulated. VCSNS believes that the number of plants that have or will be inspected is sufficient to demonstrate the adequacy of the WOG/NEI integrated inspection program. The need and schedule for an inspection for VCSNS will be based upon an evaluation of the inspection results from the integrated inspection program.

In addition, the results of a plant specific probabilistic analysis has indicated that there is no immediate need for an inspection at VCSNS. As will be explained in the response to Item 1.4 below, the probability for a penetration failure (a crack of 75% wall thickness) is very small in the near future. Based on the technical information to date, it is not until VCSNS has operated for about 186,500 hours of critical operation (186,500 hours corresponds to approximately 25 years of plant operation and occurs after refuel outage 17) that the probability begins to rise. Based on this information, VCSNS has concluded that an inspection prior to this time is not warranted.

Therefore, as part of the plan for inspections during the time period leading up to refuel outage 17, VCSNS will monitor the inspection results from the integrated inspection program and evaluate the accumulation of industry knowledge of the causes for the degradation of CRDM nozzles and other VHPs. Based on this on-going evaluation of inspection data and the accumulation of technical information, VCSNS will determine the need for and the appropriate time to perform any necessary inspections.

The technical basis for this plan is contained in the response to Item 1.4 below.

<u>Requested information Item 1.2 - continued:</u>

B. "Provide the scope for the CRDM nozzle and other VHP inspections, including the total number of penetrations (and how many will be inspected), which penetrations have thermal sleeves, which are spares, and which are instrument or other penetrations."

<u>Response</u>

There are sixty-five (65) penetrations in the head of the VCSNS reactor vessel. These penetrations are arranged in four rings around the reactor vessel head as shown on page 4-1 of Attachment (2). The scope of a planned inspection would be to include the twenty-four (24) penetrations in the outer most ring. These penetrations are numbered forty-two (42) through sixty-five (65) and are tabulated in Table 4-1 of Attachment (2). For each penetration, this table gives a description of the penetration use, identifies those with thermal sleeves and provides the material heat number.

<u>Requested information Item 1.3:</u>

"If a plan has <u>not</u> been developed to periodically inspect the CRDM nozzle and other VHPs, provide the analysis that supports why no augmented inspection is necessary."

<u>Response:</u>

As stated in the response to Item 1.2, a plan has been developed for the inspection of the CRDM nozzles and other VHPs.

<u>Requested information Item 1.4:</u>

"In light of the degradation of CRDM nozzle and other VHPs described above, provide the analysis that supports the selected course of action as listed in either 1.2 or 1.3, above. In particular, provide a description of all relevant data and/or tests used to develop crack initiation and crack growth models, the methods and data used to validate these models, the plant-specific inputs to these models, and how these models substantiate the susceptibility evaluation. Also, if an integrated industry inspection program is being relied on, provide a detailed description of this program."

Response:

VCSNS is a participant in the WOG/NEI reactor pressure vessel head penetration integrated inspection program. Within this program, WOG has developed the analysis to support the VCSNS action listed as a response to Item 1.2. The technical basis for this analysis is contained in Reference (2). The following describes the contents of Reference (2) pertinent to this response:

- Section 2.0 contains the crack growth model used in the VCSNS analysis and the basis for it.
- Section 3.0 contains the crack initiation model used in the VCSNS analysis and the basis for it.
- Section 4.0 contains a technical description of the probabilistic model used in the VCSNS analysis.

An in-depth probabilistic analysis has been completed for all the VCSNS reactor vessel closure head penetrations using the state-of-the-art methods of Reference (2). The inputs used in the VCSNS analysis are contained in Attachment (2). Pertinent inputs of the analysis are listed in Table 4-2 of Attachment (2).

The results of the WOG/NEI probabilistic analysis for VCSNS are contained in Attachment (2). The results are summarized in Tables 4-3 and 4-5 of Attachment (2). The results of the analysis show that the probability of a flaw initiating in at least one of the sixty-five penetrations and reaching 75% of the wall thickness in forty years was calculated to be 33.5%. For a flaw initiating in any of the eight worst susceptible penetrations and reaching 75% of the wall thickness in forty years, the analysis calculated a probability of only 2.6%. (Reference (3) notes the calculated probability of a flaw initiation in the worst susceptible penetration at D. C. Cook and reaching 43% of the wall thickness after 11.6 years to be 38.1%)

VCSNS has concluded that an inspection prior to 186,500 hours of critical operation is not warranted. This conclusion is based on the following:

- The results of the probabilistic analysis for individual penetrations is summarized in Table 4-3 of Attachment (2). Prior to twenty years of service, the probability of failure (75% wall thickness) is small; 2.1% for at least one of the sixty-five penetrations, 0.2% for any of the eight worst susceptibie penetrations, less than 0.1% for any of the sixteen other penetrations and approximately zero for any of the remaining forty-one penetrations. Based on these small values, inspections prior to this time would not be warranted.

- Failure probability with time is shown in Figure 4-2 of Attachment (2). A study of this graph concludes that the probability for failure (75% through wall thickness) remains small for the first twenty-five years of critical operation; less than 5.9% for at least one of the sixty-five penetration and less than 0.45% for any in the worst susceptible group of eight penetrations. After this time period, the probability begins to increase rapidly for the following years; to a value of 33.5% at forty years for at least one penetration and to a value of 2.6% at forty years for any in the worst susceptible group of penetrations. This indicates that after the twenty-fifth year of operation would be the most appropriate time for inspections.

VCSNS recognizes that a significant amount of industry study continues on the issue of degradation in CRDM nozzles and other VHPs. In addition, VCSNS will continue to be a member in the WOG integrated inspection program. Through these activities, knowledge of the technical mechanisms and the probability for occurrence of this issue will increase significantly over the next few years. Therefore, as part of the inspection plan, during the time period leading up to the refuel outage 17, VCSNS will monitor the accumulation of industry knowledge of vessel head penetration degradation. This plan will allow VCSNS to determine the need for and the appropriate time to perform any necessary inspections. This plan does not constitute a commitment to perform inspections.

<u>Requested information Items 2., and 2.1 through 2.6:</u>

"2. Provide a description of any resin bead intrusions, as described in IN 96-11, that have exceeded the current EPRI PWR Primary Water Chemistry Guidelines recommendations for primary water sulfate levels, including the following information:

2.1 Were the intrusions cation, anion, or mixed bed?

2.2 What were the duration of these intrusions?

2.3 Does the plant's RCS water chemistry Technical Specifications follow the EPRI guidelines?

2.4 Identify any RCS chemistry excursions that exceed the plant administrative limits for the following species: sulfates, chlorides or fluorides, oxygen, boron, and lithium.

2.5 Identify any conductivity excursions which may be indicative of resin intrusions. Provide a technical assessment of each excursion and any follow-up action.

2.6 Provide an assessment of the potential for any of these intrusions to result in a significant increase in the probability for IGA of VHPs and any associated plan for inspections.

Response:
VCSNS has reviewed the plant historical records to determine if any incident of resin ingress similar to those which occurred in 1980 and 1981 at the Jose Cabrera (Zorita) plant has occurred at VCSNS. This data search is structured to identify all resin intrusion events into the primary coolant system with a magnitude greater than 1 ft^3 (30 liters). The threshold of 1 ft^3 was chosen as a conservative lower bound since it represents less than 15% of the estimated volume of resin released into the reactor coolant system during the two events at Jose Cabrera.

For the period of the plant operation prior to the routine analysis for sulfate in reactor coolant, the data search was based on a review of the plant's reactor coolant chemistry records relative to specific conductance of the reactor coolant. An elevation of a 28 microS/cm increment in specific conductance was the value used as an indicator of cation resin ingress equivalent to a volume of 1 ft^3.

Routine analysis for sulfate in the reactor coolant was performed for plant operation from January 1, 1988 to the present. A sulfate concentration in the range of 15 to 17 ppm peak concentration was used as the indicator of cation resin ingress. This concentration is approximately equivalent to a volume of 1 ft^3.

The data evaluation performed by VCSNS did not indicate a resin ingress that exceeded the threshold quantities. Had either specific conductance or sulfate increases indicated resin ingress to the magnitude of the threshold quantity identified above, additional data evaluation would have been conducted to look for a corresponding depression in pH or elevation in lithium as corroborating information of the incident. In the case of the use of sulfate data as the indicator, specific conductance would also have been included as confirmatory data had a significant in-leakage event been identified.

It is unnecessary to review plant records for boron, chlorides, fluorides and oxygen because these species are not viewed as valid indicators of cation resin ingress and degradation within the primary coolant system of a PWR. Borate, chloride and fluoride anions could be associated with the anion portion of mixed bed resin (cation plus anion); however, if mixed bed resin leakage to the RCS occurred, the cation portion of the resin would contain the sulfate indicator described above. Detectable dissolved oxygen in reactor coolant, during power operation with appropriate hydrogen overpressure on the volume control tank and specified residual dissolved hydrogen in the reactor coolant, could not occur and, therefore, could not be associated with resin in-leakage.

VCSNS has followed the EPRI PWR Primary Water Chemistry Guidelines in principle since Revision 1 to these guidelines was issued. Subsequent revisions to the guidelines have been reviewed and adopted as they were deemed applicable to the operation of VCSNS.

Enclosed are:

1. WCAP-14932 "Probabilistic and Economic Evaluation of Reactor Vessel Closure Head Penetration Integrity", July, 1997 (Proprietary)

2. WCAP-14955 "Probabilistic and Economic Evaluation of Reactor Vessel Closure Head Penetration Integrity", July, 1997 (Non-proprietary)

Also enclosed are a Westinghouse authorization letter, CAW-97-1146 accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Corporation, it is accompanied by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-97-1146 and should be addressed to N.J. Liparulo, Manager of Equipment Design and Regulatory Engineering, Westinghouse Electric Corporation, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

I declare that these statements and matters set forth herein are true and correct to the best of my knowledge, information, and belief.

Should you have any questions, please call Mr. Charles Barbier at (803) 345-4019 or Mr. Jim Turkett at (803) 345-4047 at your convenience.

Very Truly Yours,

Gary J. Taylor

Gary J. Taylor

JWT/ JT/jt

Attachments

c: J. L. Skolds (w/o Attachments)
W. F. Conway (w/o Attachments)
R. R. Mahan (w/o Attachments)
R. J. White (w/o Attachments)
L. A. Reyes
A. R. Johnson
NRC Resident Inspector
J. B. Knotts, Jr. (w/o Attachments)
C. C. Barbier (w/o Attachments)
W. F. Bacon (w/o Attachments)
NSRC
DMS (RC-97-0150)
RTS (LTR 970001)
File (815.14)

STATE OF SOUTH CAROLINA :

: TO WIT :

COUNTY OF FAIRFIELD :

I hereby certify that on the 30th day of July 1997, before me, the subscriber, a Notary Public of the State of South Carolina, personally appeared Gary J. Taylor, the Vice President, Nuclear Operations of the South Carolina Electric and 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

Notary Public

My Commission Expires 7-26-2005

NRC - NEI

W.T. Russel to W. Rasin

November 19, 1993

12 Pages Total

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 19, 1993

William Rasin, Vice President
Director of ~~the~~ Technical Division
Nuclear Management and Resources Council
1776 Eye Street, N.W.
Suite 300
Washington, D.C. 20006-3706

Dear Mr. Rasin:

The attached safety evaluation was prepared by the Materials and Chemical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, on the NUMARC submittal of June 16, 1993, addressing the Alloy 600 Control Rod Drive Mechanism (CRDM)/Control Element Drive Mechanism (CEDM) pressurized water reactor vessel head penetration cracking issue. This submittal addressed stress analyses, crack growth analyses, leakage assessments, and wastage assessments for potential cracking of the inside diameter of CRDM/CEDM nozzles. Based on the overseas inspection findings and the review of your analyses, the staff has concluded that there is no immediate safety concern for cracking of the CRDM/CEDM penetrations. This finding is predicated on the performance of the visual inspection activities requested in Generic Letter 88-05. Also, special nondestructive examinations are scheduled to commence in the Spring of 1994 to confirm your safety analyses for each PWR owners group.

Your submittals for each PWR type did not address the Bugey-3 flaw that was oriented approximately 30° off the vertical axis nor a circumferential, J-groove flaw discovered at Ringhals. Preliminary information supplied to the staff by Swedish authorities indicates that the J-groove flaw may be associated with a fabrication defect. We are continuing to work with the Swedish authorities to confirm this. From the information available to us today, neither of these flaws would pose a threat to the integrity of the CRDM penetrations. It is our understanding that you are also reviewing these flaws and you will provide your assessment as to their significance and origin. NRC will issue a supplemental safety evaluation after reviewing your supplemental assessment.

The staff agrees that there are no unreviewed safety questions associated with CRDM/CEDM penetration cracking. The staff agrees that the flaw predictions based upon penetration stress analyses are in qualitative agreement with inspection findings. However, the stress analyses do not address stresses from possible straightening of CRDM penetration tubes during fabrication. These stresses, if large, could result in circumferential flaw orientations. The staff requests that you also address this issue in your supplemental assessment. Based upon information received from overseas regulatory authorities, your analyses, and staff reviews, the staff believes that catastrophic failure of a penetration is extremely unlikely. Rather, a flaw would leak before it reached the critical flaw size and would be detected during periodic surveillance walkdowns for boric acid leakage pursuant to Generic Letter 88-05. However, the staff recommends that you consider

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enhanced leakage detection by visually examining the reactor vessel head until either inspections have been completed showing absence of cracking or on-line leakage detection is installed in the head area. The staff requests that you also address the issue of enhanced leakage detection in your supplemental assessment.

The NRC staff has reviewed your July 30, 1993 submittal, which proposed flaw acceptance criteria to be used in dispositioning any flaws found during CRDM/CEDM inspections. The staff finds the proposed flaw acceptance criteria acceptable for axial cracks because the criteria conform to the American Society of Mechanical Engineers (ASME) Section XI criteria. The staff determined that flaws that are primarily axial (less than 45° from the axial direction) should be treated as axial cracks as indicated in Figure 1(b), (d), and (f) of your July 30, 1993 letter. Flaws more than 45° from the axial direction should be treated as circumferential flaws. However, based upon information submitted to date and the more serious safety consequences of circumferential flaws, the staff does not agree with your proposed criteria for circumferential flaws. Circumferential flaws which a licensee proposes to leave in service without repair, should be reviewed by the staff on a case-by-case basis.

Sincerely,

Original signed by

William T. Russell, Associate Director
for Inspection & Technical Assessment
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

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SAFETY EVALUATION
FOR
POTENTIAL REACTOR VESSEL HEAD ADAPTOR TUBE CRACKING

1.0 INTRODUCTION

Primary water stress corrosion cracking (PWSCC) of Alloy 600 was identified as an emerging issue by the NRC staff to the NRC Commission following a 1989 leakage from an Alloy 600 pressurizer heater sleeve penetration at Calvert Cliffs Unit 2, a Combustion Engineering designed pressurized water reactor (PWR). Several instances of PWSCC of Alloy 600 pressurizer instrument nozzles had been reported to the NRC between the time period of 1986 to the present on domestic and foreign pressurized water reactors (PWR). The licensee at Arkansas Nuclear Operations, Unit 1, a Babcock & Wilcox (B&W) designed PWR, reported a leaking pressurizer instrument nozzle in 1990, after 16 years of operation. Westinghouse PWR's do not use Alloy 600 for penetrations or nozzles in the pressurizers.

According to the information provided to the staff by NUMARC at a public meeting held on July 5, 1993, a leak was discovered in an Alloy 600 control rod drive mechanism (CRDM) adaptor tube penetration during a hydrostatic test at the Bugey 3 plant in France in 1991 after 12 years of operation. A visual examination of the CRDM adaptor tube penetration indicated the presence of axial flaws in the inside diameter (ID) of the CRDM adaptor tube penetration. The remaining 65 CRDM adaptor tube penetrations were examined at Bugey 3 and 2 additional CRDM adaptor tube penetrations contained axial cracks on the ID of the CRDM adaptor tube penetrations. An examination of 24 CRDM adaptor tube penetrations at Bugey 4 revealed axial ID cracks in 8 CRDM adaptor tube penetrations. CRDM adaptor tube penetrations have been examined at 37 nuclear power plants in France, Sweden, Switzerland, Japan, and Belgium and 59 of the 1,850 penetrations have revealed short, axial crack indications.

The primary safety concern associated with stress corrosion cracking in Alloy 600 in CRDM penetrations is the potential for circumferential cracks. Extensive circumferential cracking could lead to the ejection of a CRDM resulting in an unisolable rupture in the primary coolant system. As indicated above, the inspections to date have identified short axial cracks. However, two other inspection findings are of particular interest. First, the CRDM penetration that leaked during hydrostatic testing at Bugey-3 was removed and examined metallurgically during December 1992. A secondary crack that was 0.120 inches long and 0.090 inches deep at about 30 degrees to the axial direction was observed on this CRDM. Second, in early in 1993, a J-groove weld at the Ringhals plant in Sweden was discovered to contain a circumferential crack. Preliminary indications are that this flaw is a fabrication defect. Additional work is in progress by the staff at the Swedish Nuclear Power Inspectorate to confirm this.

The Westinghouse CRDM adaptor tube penetrations are similar in design to the European PWR's and use Alloy 600 for the penetrations. The NRC staff met with the WOG on January 7, 1992 to discuss the experience at the Bugey 3 plant and the relationship of the French design of the CRDM adaptor tube penetrations to the design of domestic Westinghouse plants. The WOG informed the NRC staff that a program had been initiated in December 1991 to: (1) determine the root cause of the CRDM penetration cracking; (2) analyze the stress distributions in the CRDM penetrations of a typical domestic plant; (3) compare the design and operational characteristics of domestic and French plants to determine the likelihood for cracking; and (4) identify the need for additional efforts. The NRC staff also met with the Combustion Engineering Owners Group (CEOG) and the Babcock & Wilcox Owners Group (B&WOG) to discuss the PWSCC of CRDM adaptor tube penetrations. The Nuclear Management and Resources Council (NUMARC) coordinated the PWR Owners' Group efforts on this subject.

On June 16, 1993, NUMARC submitted safety assessments to the NRC from WOG, CEOG, and B&WOG for review by the NRC staff. These safety assessments present stress analyses, crack growth analyses, leakage analyses, and wastage assessments for flaws initiating on the ID of CRDM adaptor tube penetrations. NRC requested additional information on the safety assessments by letter dated September 2, 1993. NUMARC submitted the response to NRC on September 22, 1993. The safety assessments submitted to the NRC did not address the secondary flaw observed at the Bugey-3 plant that was oriented approximately 30° from the longitudinal axis of the penetration nor the apparent fabrication flaw at the Ringhals plant. Neither of these flaws posed a threat to the integrity of the CRDM penetrations. However, NUMARC has committed to submit a safety assessment relevant to this type of cracking. After this safety assessment has been reviewed by NRC, a supplement to this SER will be issued.

2.0 STAFF EVALUATION

2.1 WOG WCAP-13565, ALLOY 600 REACTOR VESSEL HEAD ADAPTOR TUBE CRACKING SAFETY EVALUATION

The WOG submitted the, "Alloy 600 Reactor Vessel Head Adaptor Tube Safety Evaluation," through NUMARC on June 16, 1993. The safety evaluation addresses the following elements:

1. A summary of the stress analysis focusing on the type (orientation) of cracking that may be expected in the Alloy 600 material, and the stresses necessary for flaw propagation;

2. A summary of the flaw propagation analysis along with the background of the flaw prediction method;

3. An assessment of the WOG plants with respect to penetration flaw indication data from plant inspections at Ringhals, Beznau, and various Electricite de France plants, in which the key parameters for cracking are compared to WOG plants;

4. A leakage assessment summarizing leak rate vs. flaw size, and postulating leaks for WOG plants for which leakage considerations may apply; and,

5. A vessel head wastage assessment including the process that leads to wastage and an estimate of the allowable wastage.

2.1.1 REGULATORY BASIS AND DETERMINATION OF UNREVIEWED SAFETY QUESTIONS

The WOG prepared safety evaluation addresses the potential for cracking and the ramifications of such cracking of the reactor vessel head adaptor tubes at Westinghouse designed NSSS plants. The WOG compared the results of this safety evaluation to the criteria in the Title 10, Code of Federal Regulations, Section 50.59 (10 CFR 50.59). The WOG concluded that an unreviewed safety question did not exist. Its evaluation considered the following:

1. Continued plant operation will not increase the probability of an accident previously evaluated in the FSAR.

2. The consequences of an accident previously evaluated in the FSAR are not increased due to continued plant operation.

3. Continued plant operation will not create the possibility of an accident which is different than any already evaluated in the FSAR.

4. Continued plant operation will not increase the probability of a malfunction of equipment important to safety.

5. Continued plant operation will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR.

6. Continued plant operation will not create the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR.

7. The evaluation for the effects of continued plant operation with potentially cracked reactor vessel head adapters has taken into account the applicable technical specifications.

2.1.2 STAFF'S EVALUATION OF THE REGULATORY BASIS AND DETERMINATION OF UNREVIEWED SAFETY QUESTIONS

The staff agrees that no unreviewed safety question exists, provided only axial flaws are found. Those axial flaws would be expected to be short, and they would most probably leak noticeably prior to the flaw size reaching unstable dimensions. The existence of any unexpected leaks would not adversely affect plant operation, or accident/transient response. No significant equipment degradation would be expected. Details of the staff's evaluation that led to the above conclusions is discussed in the following sections.

2.1.3 PENETRATION STRESS ANALYSIS

The WOG conducted an elastic-plastic, finite element analysis of a 4-loop WOG plant vessel head penetrations. The WOG concluded that the 4-loop WOG plant is bounding since prior analyses showed that the operating and residual stresses are higher on a 4-loop plant than on 2 or 3-loop plants on the outermost penetrations. Three penetration locations were modeled, the center location, the outermost location, and the location next to the outermost location. The stress history was simulated by using a load sequence of the thermal load from the first welding pass, the thermal load from the second weld pass, the fabrication shop cold hydrotest, the field cold hydrotest, and the steady state operational loading.

The highest stresses are found in the zone around the weld and are the highest in the penetration farthest from the center of the vessel (peripheral penetrations). The highest stresses on that penetration are on the side of the penetration nearest to the center of the vessel (centerside) and on the side of the penetration farthest from the center of the vessel (hillside). Also, the stresses are the highest below the weld and decrease significantly above the weld. The ratio of peak hoop stress to axial stress at the same location at the outermost penetrations was about 1.4 compared to a value of about 1.6 estimated based on the degree of ovaling measured on actual penetrations. The ratio of hoop stress to axial stress was about the same for center penetrations as for peripheral penetrations (1.6 for center penetrations compared to 1.4 for peripheral penetrations); however, the magnitude of the stresses at the peripheral penetrations was higher. The analysis indicates that axial flaws would be more likely than circumferential flaws, flaws are more likely below the weld than above the weld, and that axial flaws would appear at locations in the penetrations where they have been found in service.

2.1.4 STAFF EVALUATION OF THE PENETRATION STRESS ANALYSIS

The staff is in agreement with the results of the WOG stress analysis that predicts that the cracking will be predominately axial. These results are in qualitative agreement with field inspection findings. However, the WOG did not address the effects of possible straightening of the CRDM penetration tubes during fabrication. Such straightening operations could significantly alter the residual stress fields within the penetration tubes. Results of inspections to date have not identified any problems directly related to this process; however, the staff requests that NUMARC address this issue for all three owners groups' plants.

2.1.5 CRACK GROWTH ANALYSIS: FLAW TOLERANCE

The WOG crack growth analysis was based on the assumptions that the flaw would be caused by primary water stress corrosion cracking, and that the crack growth is controlled by the hoop stress. The maximum principal stress will be oriented at a slight angle to the hoop stress and flaws would be expected to be perpendicular to the maximum principal stress. However, all of the flaws found in service with two exceptions have been axially located. Hence, the WOG used the hoop stress as an approximation of the maximum principal stress. The outer- most penetration for a 4-loop Westinghouse plant was selected for analysis since this location experiences the highest stresses. The highest stress was located along the inner surface just below the center side of the weld. The calculated hoop stress through the wall of the penetration was used for flaw growth calculations. The flaw growth data were obtained from steam generator field experience and laboratory data.

Based on the stress fields that exist in the CRDM penetrations, any flaw growth that occurs is expected to be predominately axial in nature. Furthermore, the growth of any flaws inclined from the vertical would be limited in length due to the nature of the existing stresses. These conclusions are consistent with the inspection results described above. Accordingly, there is no significant potential for failure of a penetration by ejection of the CRDM sleeve. With regard to axial cracking, WOG has concluded that the critical flaw length for an axial flaw for Alloy 600 is sufficiently long that leakage would occur and be detected during surveillance walkdowns as required by GL 88-05. Therefore, the consequences of cracking in the penetration sleeve are limited to the affects of leakage as discussed below.

The flaw growth analysis showed that under the most severe conditions of metallurgical microstructure, peak hoop stress, and operating temperature, it would take about five years for a flaw to grow through wall. Under the same conditions, it would take an additional 10 years for a through-wall flaw to grow 1 ¼ inches above the weld on the lower hillside of the outermost head penetrations (Figure 3.2-2) and about the same time to grow two inches above the J-groove weld on the center side of the outermost penetrations (Figure 3.2-3). The flaw growth analysis indicates that through wall flaws would essentially arrest before growing a maximum of two inches above the weld. These flaws would be constrained within the head and could not significantly open thus limiting the amount of leakage that could occur.

2.1.6 STAFF EVALUATION OF THE CRACK GROWTH ANALYSIS

The WOG stated that the crack growth analysis is in general agreement with the inspection findings. The crack growth rate data used in this analysis was limited, but the results predicted using these flaw growth data bound the results of the inspections. Crack growth rates are difficult to determine precisely; however, the assumed growth rates compare well with inspection data available to date and the large margins that exist in the analyses will account for any possibly higher growth rates. There are large margins of safety in the analyses and the CRDM penetrations are constructed of inherently tough material with a critical flaw size of approximately 13 inches in the free span above the reactor vessel shell. Therefore, the staff concludes that catastrophic failure of a penetration is extremely unlikely because a flaw would be detected during boric acid leakage surveillance walkdowns before it reached the critical flaw size.

2.1.7 ASSESSMENT OF WOG PLANTS

The WOG compared the Ringhals and Beznau plants to the domestic Westinghouse plants and developed a model for the relative susceptibility to PWSCC. The WOG considered residual and operating stresses in the penetrations, the environment, material condition, operating temperature, and time-of-operation at temperature, and pressure. Based on this evaluation, the WOG has evaluated domestic WOG PWR's with regard to their degree of susceptibility. Based on what WOG considers to be conservative assumptions, the Ringhals plants envelope 45 domestic plants. None of these plants are expected to have any flaws other than some short, shallow, axial flaws. Nine additional WOG plants are not enveloped by the Ringhals plants. Based on the stresses, operating temperatures, hours of operation, and the flaw growth curves provided in the WOG safety assessment, the WOG does not expect any CRDM penetration axial flaws to reach a length in excess of 1 inch before about the middle of 1995.

2.1.8 STAFF EVALUATION OF THE WOG ASSESSMENT

The susceptibility model developed by the WOG considers the appropriate parameters affecting IGSCC and should provide a reasonable ranking of plant susceptibilities. In addition, this evaluation indicates that it is unlikely that U.S. plants should exhibit any cracking significantly worse than that found in European plants.

2.1.9 LEAK RATE CALCULATIONS

The leak rates were calculated based on the assumption that the leak rate will be controlled by the flow rate through the flaw in the head penetration or by the flow through the penetration annulus, whichever is smaller. WOG estimates the maximum leak rate would be 0.7 gpm for a 2 inch long flaw and an annular clearance of 0.003 inches. Leakage above 1.0 gpm is detectable in domestic WOG plants according to WOG. Growth of an axial flaw outside of the part contained within the reactor head will result in leakage greater than 1.0 gpm prior to reaching the critical flaw size. The WOG stated that an axial flaw would remain stable for growth up to 13 inches above the reactor vessel head.

2.1.10 STAFFS EVALUATION OF THE WOG LEAK RATE CALCULATIONS

The staff agrees with the WOG assumptions about leakage and concludes, that based on existing leakage monitoring requirements, there is reasonable assurance that leakage in excess of the 1.0 gpm technical specification limit would be detected prior to any unstable extension of the flaw.

2.1.11 REACTOR VESSEL HEAD WASTAGE ASSESSMENTS

This section assesses the potential wastage of the reactor vessel head due to leakage of primary coolant through the CRDM penetrations. This assessment is based on wastage data from previous Westinghouse experiments and from the results of a penetration mockup test conducted by the Combustion Engineering Owners Group (CEOG).

This analysis assumed that coolant escaping from the penetration would flash to steam leaving boric acid crystals behind. WOG assumed that crystals would accumulate on the vessel head but would cause minimal corrosion while the reactor was operating. The head temperature would be about 500°F during operation and significant wastage of the reactor head by the boric acid crystals would not be expected. Dry boric acid crystals do not cause corrosion. Wastage would only occur during outages when the head temperature is below 212°F.

The CEOG provided all of the PWR owners groups with the results of pressurizer penetration mockup test results. The WOG examination of the CEOG mockup test results showed that the maximum penetration rate at the deepest pit was 2.15 inches/year while the average penetration rate was 0.0835 inches/year. The maximum total metal loss rate or wastage volume was 1.07 in^3/year, and the greatest damage occurred where the leakage left the annulus. The WOG considered the maximum wastage would be 6.4 in^3 of vessel head material. The assumptions made were that any leakage over 1.0 gpm can be detected so only leak rates between 0.0 and 1.0 gpm were considered. The WOG analyzed the situation using finite element analyses for a 2 loop, 3 loop, and 4 loop reactor vessel head where a 1.0 gpm leak went undetected for 6 years and concluded that the ASME code minimum wall thickness requirement would be satisfied and that the stresses remain within the ASME code allowable stresses.

2.1.12 THE STAFF'S EVALUATION OF THE REACTOR VESSEL HEAD WASTAGE ASSESSMENTS

The assumption used in the WOG corrosion assessment are based on experimental data and should provide a reasonable estimate of potential wastage of the reactor vessel head. Based on these evaluations, there would be significant time between initiating a leak and experiencing wastage that would reduce the structural integrity margins of the reactor vessel head to below acceptable levels. Considering the length of time involved, there is reasonable assurance that leakage, manifested by the accumulation of moderate amounts of boric acid crystals would be detected during a surveillance walkdown in accordance with GL 88-05.

3.0 CEOG SAFETY EVALUATION

The CEOG safety evaluation is essentially the same as the WOG safety evaluation. The CEOG plants run at a slightly higher temperature than the European plants that have experienced cracking, have greater hillside angles, and have been in operation longer than many of the European plants. The CEOG indicated that all of these factors would increase the probability of cracking for the CEOG plants. However, the CEOG plants have significantly less weld metal in the J-groove welds and the CEOG stated that this would significantly reduce the residual welding-induced stresses and would reduce the probability of PWSCC. CEOG concluded that any PWSCC that formed would be short, axial flaws.

The CEOG states that they can detect a 0.12 gpm leak in the primary coolant system. CEOG also states that the boric acid accumulation as a result of a 0.12 gpm leak would not result in wall thinning below the code allowables in less than 8.8 years compared to 6 years for WOG plants and that surveillance walkdowns would detect boric acid crystals long before the 8.8 years.

3.1 STAFF EVALUATION OF THE CEOG SAFETY EVALUATION

The staff has concluded that the potential for PWSCC of CRDM/CEDM for CEOG plants does not create an immediate safety issue as long as the surveillance walkdowns required by GL 88-05 continue and corrective action is instituted when leaks are discovered. The CEOG analyses indicating that the stresses would favor development of axial rather than circumferential cracks and that significant time would be required to reduce the wall thickness of the vessel head to below the ASME code allowables demonstrates that an immediate safety concern does not exist.

4.0 B&WOG SAFETY EVALUATION

The B&WOG safety evaluation was essentially the same as the WOG and CEOG safety evaluations. The B&WOG analysis indicates that B&WOG plants have essentially the same susceptibility to PWSCC as the European plants based on operating temperature, residual stresses, and operational life. The B&WOG predicts short, axial flaws on the peripheral locations based on the results of finite element analyses. The B&EOG estimates that it would take 10 years from the time a flaw initiates on the inside diameter of a CRDM penetration until a leak appears. Once a leak starts, B&WOG concluded that it would take 6 years before enough corrosion would occur to reduce the wall thickness of the reactor vessel head to below ASME code minimums, and that this amount of leakage would be detected during surveillance walkdowns.

4.1 STAFF EVALUATION OF THE B&WOG SAFETY EVALUATION

The staff has concluded that the potential for PWSCC of CRDM for B&WOG plants does not create an immediate safety issue as long as the surveillance walkdowns required continue and as long as any leakage is corrected. The B&WOG analyses, indicating that the stresses would favor development of axial rather than circumferential cracks and that significant time would be required to reduce the wall thickness of the vessel head to below the ASME code allowables, demonstrates that an immediate safety concern does not exist.

5.0 PROPOSED FLAW ACCEPTANCE CRITERIA

On July 30, 1993, NUMARC submitted the proposed flaw acceptance criteria for flaws identified during inservice inspection of reactor vessel upper head penetrations to the NRC for review. These criteria were developed by utility technical staffs and the domestic PWR vendors. NUMARC proposes that axial flaws are permitted through-wall below the J-groove weld and 75 percent through-wall above the weld. There is no limit on the length of the flaws. NUMARC proposes that circumferential flaws through-wall and 75 percent around the penetration be allowed below the J-groove weld and that circumferential flaws above the weld could be 75 percent through-wall and 50 percent around the penetration. Proximity rules found in ASME Section XI, Figure IWA 3400-1 are proposed for determining the effective length of multiple flaws in one location. NUMARC proposes that the flaws be characterized by length and preferably depth. NUMARC proposes that if only the length is characterized, that the depth be assumed to be one half of the length based on inspection findings to date.

5.1 STAFF EVALUATION OF THE PROPOSED FLAW ACCEPTANCE CRITERIA

The staff finds the proposed flaw acceptance criteria acceptable for axial flaws because the criteria conform to the American Society of Mechanical Engineers (ASME) Section XI criteria. The assumption that flaw depth is one half the flaw length for flaws whose depth cannot be determined will limit the flaw length to 1.5 times the thickness of the penetration sleeve. However, it is expected that reasonable attempts will be made to determine flaw depths. Flaws found through inservice inspection (ISI) that are primarily axial (less than 45° from the axial direction) will be treated as axial flaws as indicated in Figure 1(b), (d), and (f) of NUMARC'S July 30, 1993 letter. Flaws more than 45° from the axial direction are considered to be circumferential flaws. Based upon information submitted to date and the more serious safety consequences of circumferential flaws, the staff has concluded that criteria for circumferential flaws should not be pre-approved. Detection of such flaws would be contrary to inspection results to date and to the conclusion of the Owners Groups evaluations. The curcumstances associated with such a flaw would have to be well understood. Therefore, any circumferential flaws found through ISI, which a licensee proposes to leave in service without repair, will be reviewed on a case-by-case basis by the staff.

6.0 LEAKAGE MONITORING

NUMARC, through the owners groups' reports, determined that any leakage in excess of 1 gpm would be detected prior to any unstable extension of axial flaws. Also, leakage at less than 1 gpm would be detectable over time based on boric acid buildup as noted during periodic surveillance walkdowns. Although NUMARC has proposed, and the staff agrees, that low level leakage will not cause a significant safety issue to result, the staff determined that NUMARC should consider methods for detecting smaller leaks to provide defense-in-depth to account for any potential uncertainty in its analyses. The reported leak rate at Bugey 3 was about 0.003 gpm and was detected using acoustic monitoring techniques during the performance of a hydrostatic test. The staff does not think that it is necessary to detect a 0.003 gpm leak, but does think that permitting leakage just below 1.0 gpm as currently proposed may be undesirable. Leakage of this magnitude would produce significant deposits (thousands of pounds/year) of boric acid on the reactor vessel head. Further, most facilities' technical specifications state that no pressure boundary leakage is permitted. The staff notes that small leaks resulting from flaws which progressed through-wall just prior to a refueling outage would be difficult to detect while the thermal insulation is installed. Although running for an additional cycle with that undetected leak would not result in a significant safety issue, the NUMARC should consider proposing a method for detecting leaks that are significantly less than 1.0 gpm, such as the installation of on-line monitoring equipment.

7.0 CONCLUSIONS

Based on review of the NUMARC submittal and the overseas inspection results, the staff concludes that the CRDM/CEDM cracking at the reactor vessel heads is not a significant safety issue at this time as long as the surveillance walkdowns in accordance with GL 88-05 continue. The staff agrees with the NUMARC's determination that there are no unreviewed safety questions associated with stress corrosion cracking of CRDM penetrations. However, new information and events may require a reassessment of the safety significance. Furthermore, there is a need to verify the conclusions of the NUMARC's safety evaluations. Therefore, nondestructive examinations should be performed to ensure there is no unexpected cracking in domestic PWRs. These examinations do not have to be conducted immediately since only short, shallow, axial flaws are likely to be present in the CRDM penetrations. The industry has committed to conduct inspections at three units in 1994. They are:

(a) Point Beach Unit 1 in the Spring of 1994,
(b) D.C. Cook Unit 2 in the third quarter of 1994,
(c) Oconee Unit 2 in September 1994.

As the surveillance walkdowns proposed by NUMARC are not intended for detecting small leaks, it is conceivable that some affected PWRs could potentially operate with small undetected leakage at CRDM/CEDM penetrations. In this regard, the staff believes it is prudent for NUMARC to consider the implementation of an enhanced leakage detection method for detecting small leaks during plant operation.

The staff found NUMARC's flaw acceptance criteria acceptable for axial flaws but NRC review and approval of the disposition of any circumferential flaws will be required.

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